Technical Report

TR-10-63

RD&D Programme 2010

Programme for research, development and demonstration of methods for the management and disposal of nuclear waste

September 2010

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Preface

SKB, Svensk Kärnbränslehantering AB (the Swedish Nuclear Fuel and Waste Management Co), which is owned by the companies that operate the Swedish nuclear power plants, has been assigned the task of managing and disposing of the radioactive waste and the spent nuclear fuel from the reactors. The Nuclear Activities Act requires a programme for the comprehensive research and development that is needed to manage and dispose of the waste in a safe manner and to decommission and dismantle the nuclear power plants. SKB is now presenting RD&D Programme 2010 in fulfilment of these requirements.

After more than 30 years of research and development regarding final disposal of spent nuclear fuel, an application under the Nuclear Activities Act for final disposal of spent nuclear fuel and an application under the Environmental Code for the KBS-3 system are now being finalized. The site investigations are completed, and SKB selected Forsmark as the site for the final repository in June 2009. In 2006 we published the SR-Can report in which we presented an initial analysis of the long-term safety of a final repository at the Forsmark and Laxemar sites, based on data from the initial site investigation phase. The SR-Can report gave the responsible authorities an opportunity to review the methodology for the safety assessment. SKB has now taken care of the comments made in the review. The safety assessment for the final repository in Forsmark will be an important supporting document for the applications. This RD&D Programme describes how we have handled the review comments made by SSM on RD&D Programme 2007.

SKB now has a sufficient scientific and technical knowledge base to submit the applications. The methodology for assessment of long-term safety is highly advanced. Qualified safety assessments will continue to be an integral part of the execution of the Nuclear Fuel Programme. In order to gain a deeper understanding and reduce the uncertainties in the assessment, we will in our upcoming research programme concentrate on a few key issues such as corrosion of copper and processes that may affect the buffer material.

Our technology development has now come so far that we believe we have feasible technical solutions for the different parts of the final repository. By means of demonstrations at our laboratories, we have shown on a full scale that we can manage the different steps from fabricating and depositing canisters to closing and sealing tunnels. We believe the time has now come to proceed to the next stage in the Nuclear Fuel Programme and, after a due licensing process, commence the actual construction of the facilities in the KBS-3 system. The plan for implementing the work of building the system has previously been described in our RD&D Programmes. When SKB submits the applications, this implementation plan will become a licensing issue that will be handled within the licensing process.

In its decision regarding RD&D Programme 2007, the Government called for a supplementary account concerning planning of parts in the LILW Programme (low- and intermediate-level radioactive waste). Supplementary accounts were presented in March 2009, and the programme is further elaborated on in this RD&D Programme.

SKB plans to submit an application for a licence to build a final repository for long-lived operational and decommissioning waste in around 2030.

The Government's decision that the operation of the two reactors at Barsebäck should cease on 31 May 2005 gave added impetus to the question of decommissioning of nuclear power plants. In its review and evaluation of RD&D Programme 2004, SKI stated that SKB should intensify the work on decommissioning issues and investigate the shortest time required for the start of a licensing process for the disposal of decommissioning waste. SKB has since initiated a project aimed at extending SFR in order to dispose of both operational and decommissioning waste there. The current operating licence for SFR includes only operational waste. Rock investigations have been carried out and the work of designing the extension is under way. We plan to submit applications under the Nuclear Activities Act and under the Environmental Code for the extended facility in 2013 and put the facility into operation in 2020.

We are now at a point where we are about to start putting the results of many years of research, development and demonstration to practical use in industrial processes in new facilities. SKB's organization will be adapted to meet the requirements made by this on new competence, competence transfer and refined work methodology with a focus on quality and safety issues. Maintain and enhance society's confidence and active support will be crucial for the continued work and requires a high measure of persistence and continuity in our work.

Stockholm in September 2010
Svensk Kärnbränslehantering AB

Claes Thegerström	Tommy Hedman
President	Head of Technology Department

Summary

RD&D Programme 2010 presents SKB's plans for research, development and demonstration during the period 2011–2016. SKB's activities are divided into two main areas: the programme for low- and intermediate-level waste (the LILW Programme) and the Nuclear Fuel Programme. Operation of the existing facilities takes place within the Operational Process.

RD&D Programme 2010 consists of five parts:

Part I	Overall plan of action
Part II	The LILW Programme
Part III	The Nuclear Fuel Programme
Part IV	Research for assessment of long-term safety
Part V	Social science research

RD&D Programme 2007 was mainly focused on development of technology to realize the final repository for spent nuclear fuel. The efforts described were aimed at gaining a greater knowledge of long-term safety and compiling technical supporting documentation for applications under the Nuclear Activities Act for the final repository for spent nuclear fuel and under the Environmental Code for the final repository system. Many important results from these efforts are reported in this programme. The integrated account of the results will be presented in applications submitted in early 2011.

The regulatory review of RD&D Programme 2007 and its supplement called for clarifications of plans and programmes for the final repository for short-lived radioactive waste, SFR, and the final repository for long-lived waste, SFL. This RD&D Programme describes these plans more clearly.

A summary of the contents of each part follows below.

Part I Overall plan of action

SKB's plan of action describes the principles for the management of radioactive waste that serve as a basis for the design of our facilities. The planning of the different parts of the nuclear waste system is guided by various factors, for example the properties of the spent nuclear fuel, the quantities and types of waste and the operating times for the reactors.

Site investigations and preliminary design work for the extension of SFR are being pursued within the LILW Programme. The extension is needed to make room for short-lived decommissioning waste from the power plants and other nuclear facilities. According to our plans, the long-lived waste from decommissioning will be disposed of in SFL.

In June 2009, SKB chose Forsmark as the site for the Spent Fuel Repository. Work is under way to finalize the supporting documentation for the licence applications for construction of the KBS-3 system. They will be submitted to SSM at the beginning of 2011. Our plans call for us to begin construction of the Spent Fuel Repository and the encapsulation plant in 2015 and 2016, respectively, and to commence trial operation of the Spent Fuel Repository and Clink, which will be the name of the facility when Clab and the encapsulation plant have been integrated.

Our research and development laboratories – the Äspö Hard Rock Laboratory (HRL), the Canister Laboratory and the Bentonite Laboratory – will continue to play important roles in our research and development work, for training personnel and for demonstrating technology.

SKB's planning is based on 50 years' operation of the reactors in Forsmark and Ringhals and 60 years' operation of the reactors in Oskarshamn. These times are important factors in our planning and entail that the repository is projected to be closed and sealed in about 75 years. The planning premises will in all likelihood change. SKB takes this into account and therefore plans for a certain measure of flexibility in the design of facilities and systems.

If current reactors are replaced with new ones, this will substantially alter the premises for SKB's planning. We present an analysis of what this would entail with respect to waste volumes, the need for interim storage capacity and consequences for the size of the final repositories. New reactors entail longer operating times and extension of some of the facilities.

Part II The LILW Programme

The LILW Programme includes all low- and intermediate-level waste that will be disposed of in SKB's facilities, including decommissioning and dismantling of the Swedish reactors and SKB's facilities. Besides the waste from the nuclear power plants, waste from Studsvik Nuclear AB, AB SVAFO and Westinghouse Electric Sweden is also included.

Plan of action for execution of the LILW Programme

SKB plans to extend SFR to be able to receive decommissioning waste and additional operational waste resulting from prolonged planned operating times for the nuclear power plants. The extension of SFR is being designed to receive all additional short-lived low- and intermediate-level operational waste and all short-lived decommissioning waste expected to result from the dismantling of today's nuclear power plants, including the Ågesta Reactor and the research reactor at Studsvik. Decommissioning waste from Clink and additional waste from SVAFO and Studsvik – which also includes waste from hospitals, research and industry – is also being taken into account in the design of SFR.

We plan to extend SFR in stages in such a way that there will always be room for decommissioning waste from the nuclear power plants, in accordance with the nuclear power companies' decommissioning plans.

A large portion of the long-lived waste from the nuclear power plants arises when they are dismantled, but long-lived waste also arises during operation when the reactors' internal components are replaced. Studsvik Nuclear AB and AB SVAFO also have long-lived low- and intermediate-level waste originating from research and development. SKB plans to interim-store long-lived waste in SFR. We estimate that an application for a licence to build the final repository for long-lived waste, SFL, can be submitted in about 2030. A number of important milestones must be passed before then, such as selection of repository concept and site, investigations, evaluation of long-term safety, preparation of applications, etc. We will also study the possibility of reconditioning the waste for more efficient final disposal. When it comes to selection of a site for SFL, we will need to conduct studies that take into account both results from site investigations conducted by SKB in Oskarshamn and Forsmark and the need to select a different site. In the course of this work we will make judgements of what requirements the repository concept we have chosen imposes on the rock and its properties. Since these efforts lie far in the future, we do not present a detailed timetable.

Milestones in the LILW Programme

Milestones for management of short-lived operational and decommissioning waste:

- Site investigations for extension of SFR are concluded in 2011.
- Applications for extension of SFR are submitted in 2013.
- Extension of SFR starts in 2017.
- Application for routine operation of extended SFR is submitted in 2020.

Milestones for management of long-lived operational and decommissioning waste:

- Application for licensing of waste transport containers, ATB 1T, is submitted in 2012.
- Interim storage of core components in SFR commences in 2020.
- Safety assessment for SFL is presented in 2016.
- Application to take SFL into operation is submitted in 2030.

Research and technology development for the LILW Programme

SFR

Planning of the final design of the extension of SFR is being carried out for repository parts for both low-level waste and intermediate-level waste. We will study how the barriers in the extended repository parts should be designed. The principle is that the low-level waste is placed in disposal chambers of the same type as BLA, rock vault for low-level waste. Barriers that limit the groundwater flow are needed for the intermediate-level waste. Strategies will be devised for closure of both the existing part of SFR and the extended part. This work will serve as a basis for the safety assessment completed in 2013.

Final repository for long-lived waste, SFL

During the coming three-year period, we will focus the development work on conducting a concept study for the final repository for long-lived waste. The ultimate purpose of the study is to select possible concepts which can then be evaluated in the safety assessment that will be presented in 2016. SKB has begun the research and development work for SFL within the following areas:

- Age-related changes in cementitious materials.
- Corrosion of metals in a repository environment.
- Degradation of organic waste in a cement environment.
- Gas permeability of concrete and cement.

Responsibility, planning and technology for decommissioning and dismantling of nuclear facilities

Planning of decommissioning and execution of dismantling will be carried out in cooperation between the nuclear power companies and SKB. The main responsibility lies with the licensee of the nuclear power reactor, who is responsible for planning, licensing and execution of dismantling, as well as for treatment of the waste. SKB participates in the work of developing general methods and procedures for the dismantling work, radioactivity monitoring and classification of waste. We also keep track of international developments in the decommissioning field. Coordination between the nuclear power companies and SKB takes place in the industry-wide Decommissioning Group.

Decommissioning of the Ågesta Reactor and management of waste from decommissioning are financed in accordance with the Studsvik Act (1988:1597). Vattenfall holds the nuclear licence for the activities at Ågesta. In the winter of 2009/2010, Vattenfall and SVAFO applied for permission for SVAFO to take over the nuclear licence from Vattenfall.

This RD&D Programme contains summaries of the nuclear power companies' decommissioning plans (including Ågesta). The section also describes the plans for the decommissioning of SKB's nuclear facilities.

Barsebäck

Decommissioning of the reactors in Barsebäck is planned to start in 2020. Barsebäck Kraft AB will conduct studies and analyze and draw lessons from international experience in the area.

Forsmark

Forsmarks Kraftgrupp AB's goal for decommissioning is to restore the facility so it can be released for unrestricted use. FKA plans to update its decommissioning plan during 2010. The need for updating depends mainly on the conclusions in the ongoing unit-specific decommissioning studies.

Oskarshamn

The objective for OKG Aktiebolag is to bring the facility to a state where its buildings and land can be released for unrestricted use. Different decommissioning alternatives have been studied, but the choice of strategy will not be made until closer to the final shutdown of the facilities. When ongoing unit-specific decommissioning studies for the Oskarshamn plants are completed, they will serve as a basis for updating the decommissioning plan, which is planned for 2011.

Ringhals

Ringhals AB's goal for decommissioning is to remove radioactive material and restore the facility so it can be released for unrestricted use. The strategy for dismantling entails that most of the work is done five years after final shutdown. It is further assumed that the nearby unit is out of service. Ringhals AB's ongoing unit-specific decommissioning studies will comprise the basis for an update of the preliminary decommissioning plan during 2010–2011.

Ågesta

The Ågesta plant is currently in service operation. Service operation is planned to continue until dismantling starts, no earlier than 2020 when the environmental permit expires. A preliminary decommissioning plan has been prepared for the plant.

Ågesta's decommissioning study will be updated and will be ready during 2011, along with an application for clearance of areas outside the containment in Ågesta.

Part III The Nuclear Fuel Programme

The planning for construction and operation of the Spent Fuel Repository will be described in the account that is provided in the applications under the Nuclear Activities Act and the Environmental Code in early 2011. A full account of technology development needs will be presented in the supporting material for the applications.

Main phases and timetable

The Nuclear Fuel Programme includes licensing, design, construction and commissioning of the encapsulation plant and the final repository for spent nuclear fuel.

Important premises for both the applications and the continued work are:

- A total of about 6,000 canisters will be managed and disposed of. This corresponds to 50 years' operation of the reactors in Forsmark and Ringhals and 60 years' operation of the reactors in Oskarshamn.
- In routine operation the deposition rate is 150 canisters per year. The system is designed for a maximum deposition capacity of 200 canisters per year.
- The chosen reference design is KBS-3 with vertical deposition in a final repository at a depth of about 500 metres.
- The encapsulation plant will be built adjacent to the interim storage facility Clab, and the two facilities will be operated as one integrated facility called Clink.
- The Spent Fuel Repository will be located at Forsmark and its layout will be adapted to the bedrock and other conditions on the site.
- Operation of the system will start as soon as possible, but with realistic timetables for the licensing process, technology development, construction and commissioning.

An application under the Nuclear Activities Act for a licence to build the encapsulation plant and own and operate it as an integrated facility with Clab, called Clink, was submitted in 2006. A supplement to the application was submitted in 2009. Operation of Clab will continue throughout the construction of the encapsulation plant. The physical interconnection of the facilities will take place when the encapsulation plant is finished and necessary alterations have been made in Clab.

With the decision to locate the Spent Fuel Repository in Forsmark, planning of the construction and operation of the repository has entered a new phase. Technology development is being concentrated on finding solutions that meet requirements on safety and functionality for the conditions that prevail in Forsmark. In parallel with the licensing process, SKB will proceed with the design of the Spent Fuel Repository. The construction phase will begin when SKB has obtained all licences and conditions needed to start construction of the final repository. Extensive rock investigations (detailed characterization) will be performed as a part of the design and construction process. Commissioning of the final repository's subsystems will take place as the systems are built and installed. In connection with commissioning the systems will be tested, first separately and then gradually more interconnected. As the different parts of the facility are being commissioned, the operating organization will be assembled and trained. Running-in of technology and organization will be concluded with integrated testing of the whole facility under realistic conditions. The overall goal of construction and commissioning is that we can apply for a licence to commission the whole final repository. In order for this to be possible the following must have been achieved:

- The safety analysis report has been updated in the manner required for trial operation.
- The final repository has been built and commissioned. The facility documentation must also be complete and the operating organization and administrative procedures must be in place and run-in.

Technology development

Technology development has now come to the point that a reference design for the Spent Fuel Repository is finalized. Continued technology development is needed as we proceed from schematic solutions to solutions that are tailored to an industrialized process with established requirements on quality, cost and time.

The remaining development work requires extensive technical resources. SKB's own laboratories – the Äspö HRL, the Canister Laboratory and the Bentonite Laboratory – are built and equipped for full-scale tests, demonstrations and dress rehearsals. Other facilities, such as Posiva's Onkalo facility in Finland and our underground facilities and laboratories in Europe, will also be valuable for our development work.

SKB applies systematic requirements management to control technology development. This means that decisions to modify solutions in relation to the current reference design or reference activity are made in a systematic and controlled fashion. The development work in the different production lines for fuel, canister, buffer, backfill, closure and rock proceeds according to a delivery control model for technology development. The results of the technology development within the production lines serve as a basis for the safety analysis report for the final repository and the encapsulation plant. The current status of and need for technology development within the different production lines is described below.

The fuel line

All currently known fuel types in the Swedish nuclear power programme have been analyzed to ensure that the canister remains subcritical under all circumstances. SKB will study how fuel that does not satisfy the criticality conditions should be managed. This only applies to possible future PWR fuel with high enrichments and low burnups.

SKB has followed the development of calculation programs for decay heat since the mid-1990s and has developed methods for supplementary measurements. Calorimetric measurements have been performed on fuel assemblies for accurate determination of decay heat. Work is under way to develop a faster method for decay heat measurements.

Other areas that will be studied are how long storage times affect the fuel and methods for drying fuel.

The canister line

SKB has chosen reference methods for fabrication of the canister's components and for welding and sealing. We have continued the work of specifying design premises for the canister. Detailed specifications and design loads have been determined for the canister in the Spent Fuel Repository.

Continued efforts will focus on completing detailed design for the canister and on the encapsulation process with nuclearization of welding and nondestructive testing. Furthermore, we will continue our work with development of machines for handling the canisters in the repository.

The buffer line

SKB has conducted extensive tests of the technology for fabrication and installation of buffer. Large-scale trials with the buffer protection, which is supposed to prevent excessively rapid saturation of the buffer, have been carried out at the Äspö HRL. Work on developing a method for compressing buffer blocks has continued and some fifteen full-scale buffer blocks have been pressed.

Studies of how the buffer is affected by inflowing water have been conducted in the Bentonite Laboratory. Phenomena we have studied are erosion of the bentonite, heave of buffer blocks and buildup of water pressure in the buffer.

Continued efforts will focus on erosion of buffer and completing the design of the buffer within the chosen reference design. The Prototype Repository that was installed in 2001 will be retrieved in 2011, giving us an opportunity to study how the integrated system of canister, buffer and backfill has evolved. A quantitative model will be developed to describe the evolution of density and swelling pressure for the integrated system of buffer, backfill and plug for the period between installation and repository closure.

The backfilling line

Development of a backfilling concept with natural swelling clay has gone on for a long time. Technology and methods for preparing the clay prior to pressing of blocks are known and proven, as is the method of fabricating blocks by uniaxial pressing.

The concept for installing buffer blocks will be further developed, and handling and installation will be developed and tested both in the Bentonite Laboratory and under more realistic conditions at the Äspö HRL. We will also work to complete design of the plug that will be used in the deposition tunnels.

The closure line

SKB plans to study alternative design concepts for repository closure during the next three years.

The rock line

The rock line includes detailed characterization, design, construction and maintenance of the Spent Fuel Repository's underground openings. The development work spans a wide field and is concerned with methods for investigation, characterization and rock construction, including rock sealing and support measures, as well as development of special equipment. The selection of Forsmark as the site for the Spent Fuel Repository has great significance for the rock line's technology development programme.

SKB has conducted extensive investigations of the emergence of an excavation-damaged zone. The experience gained from these investigations is very positive with respect to the requirements made by the backfill on the tunnel contour, as well as requirements for fulfilling long-term safety. Further development is needed to fulfil these requirements in routine production. We will also continue our work with further development of methods for verification of excavation damages.

Methods for investigations, interpretation of results and modelling have been developed over a long period of time. SKB has a broad knowledge base for future detailed characterization. A programme for these investigations will be presented in connection with the application.

A comprehensive full-scale test has been carried out to demonstrate the technology for sealing of the rock by grouting. The project has yielded valuable knowledge on how the properties of the grouting material affect the grouting process and its result. We now have a good basis for testing different models for inflows and sealing in order to understand and predict these processes. This understanding will then be translated into methods for investigation and construction, as well as development of equipment and material.

KBS-3H

SKB and Posiva are studying whether horizontal deposition can constitute an alternative to vertical deposition. The goal is to develop the technology for KBS-3H to the point that it becomes possible to demonstrate the technology on a full scale at a later stage and to permit comparison with vertical deposition. The programme for the next few years includes the following main activities:

- Design of a KBS-3H repository.
- Demonstration in the Äspö HRL.
- Studies of key issues relating to long-term safety.

Part IV Research for assessment of long-term safety

The research in support of the assessments of the long-term safety of the Spent Fuel Repository and the extended SFR is described in RD&D Programme 2010. Much of the research that relates to the Spent Fuel Repository and SFR is also relevant to the research that will be conducted for the safety assessment for SFL.

The main purpose of the research that is being pursued to gain a greater understanding of the long-term safety of the final repository for spent nuclear fuel has been to gather material as a basis for the safety assessment of a repository in Forsmark, SR-Site. The research is focused on processes in the engineered and natural barriers included in the repository concept.

Certain research areas are overarching and are being handled jointly for all repository systems. These areas are: general methods for safety assessment, climate evolution, geosphere and surface ecosystems. Future climate change may entail glaciation and permafrost. These two phenomena have a great impact on the environment around a final repository. The climate can therefore indirectly affect the barriers in a repository and thereby the outcome of a safety assessment.

A large number of processes in the rock influence the outcome of a safety assessment. They include: fracturing, groundwater flow, hydrochemistry and earthquakes. Radionuclide transport and retention in the rock are included in the modelling of these processes.

Site data and models of the ecosystem in Forsmark serve as a basis for the research on surface ecosystems. The research includes work with numerical models for dose calculations.

Methodology development for assessment of long-term safety for SFR is based on the methodology developed for the Spent Fuel Repository.

Research linked to a particular repository system

The research that specifically relates to the long-term safety assessment for the Spent Fuel Repository is being pursued within the areas Fuel, Canister, Buffer and Backfill.

Research for SFR and the safety assessment for the extended repository is being pursued within the areas Short-Lived Low- and Intermediate-Level Waste and Engineered Barriers for SFR.

The Spent Fuel Repository

Our knowledge level regarding the long-term safety of a final repository in Forsmark is now deemed to be so high that it is possible to bound the importance of identified uncertainties. The research programme is continuing, however, so that we can gather further knowledge and quantify remaining uncertainties.

The properties of the spent fuel and the processes that occur if the fuel comes into contact with water comprise a considerable portion of the background material for the safety assessment. Some of these processes are strongly linked to the initial state (type of fuel, burnup, etc.).

The ability of the canister to isolate the fuel is vital, and the research is focused on the processes that can be expected to occur after deposition. Important processes are corrosion and mechanical loadings.

All processes in the buffer after deposition – for example water uptake and swelling, or freezing and erosion – are important for the outcome of the safety assessment. Many processes in the backfill are virtually identical to those that occur in the buffer.

SFR

Work is under way on the safety assessment for the extended SFR. The processes that are dealt with are specific for this particular type of waste, and the research is focused on corrosion and degradation of organic compounds in the waste.

The existing engineered barriers in SFR and those that are planned in the extension are affected to a high degree by processes that occur in cement and concrete, which is reflected in the research programme. Research on the processes that occur in the clay barriers that are used in SFR (the silo buffer) is presented together with research on buffer and backfill.

Other methods

SKB continues to follow developments in partitioning and transmutation (P&T) as well as deposition of spent nuclear fuel in deep boreholes.

Part V Social science research

SKB has been conducting and funding research in the social sciences since 2004. The research results have contributed to a deeper understanding of historical, financial and public opinion aspects. The societal research has thereby contributed to increasing the general knowledge base and has also proved useful in our practical work.

When SKB has submitted the applications under the Nuclear Activities Act and the Environmental Code, they will be handled within the framework of the democratic system, locally and nationally. SKB would therefore like to be able to present to decision-makers and the general public material independent of the applications that can shed light on important societal aspects.

The fourth and fifth calls for proposals for funded research were issued in the spring of 2008 and 2009, guided by the viewpoints on the programme that had been received from municipalities, regulatory authorities and reviewing bodies. The following projects were granted funds:

- Industrial organization of the final repository – pitfall or logical consequence? (Uppsala University).
- Core issues of democracy – A study of how opinions and global changes influence decision processes around the final disposal of nuclear waste (Mid Sweden University).
- The comparative time perspective of the nuclear waste (Royal Institute of Technology).

New and supplementary questions may arise during the examination of SKB’s licence applications. It is our ambition that the research programme for the period 2004–2011 should satisfy the needs that exist to shed light on different societal aspects.

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Part I

Overall plan of action

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1 Management of radioactive waste and spent nuclear fuel

The Swedish power industry has been generating electricity by means of nuclear power for about 40 years. A large part of the system that is needed to manage and dispose of the waste from operation of the reactors has been built up during this time. The system consists of the interim storage facility for spent nuclear fuel (Clab), the final repository for short-lived radioactive waste (SFR), plus the ship m/s Sigyn and casks for transport.

What remains to be done is to build and commission the system of facilities, the KBS-3 system, needed for final disposal. This includes extending Clab with a plant for encapsulation of the spent nuclear fuel, casks for transport of spent fuel canisters, and building a final repository where the canisters will be deposited. For the low- and intermediate-level waste it will be necessary to add an extension to SFR, build a repository for long-lived radioactive waste (SFL) and casks for shipments of long-lived waste. SKB's plan of action describes the overall plans for realizing the remaining parts of the waste system in such a manner that man and the environment are protected – today and in the future.

Furthermore, existing systems and facilities (Clab, SFR and transportation system) must be constantly maintained and modernized, particularly in view of the plans for extended operating times for the Swedish reactors and thereby for the nuclear waste system.

1.1 Premises

1.1.1 Relevant regulatory framework and SKB's mission

In Sweden, the holder of a licence to own a nuclear facility is responsible for managing and disposing of the radioactive waste and the spent nuclear fuel in a safe manner. The obligations with regard to management and final disposal of radioactive waste are regulated in the Nuclear Activities Act (1984:3), the Radiation Protection Act (1988:220), the Act (2006:647) on Financial Measures for the Management of Residual Products from Nuclear Activities, as well as in related ordinances and certain permits and guidelines issued by the Government.

These provisions entail that the nuclear power companies, i.e. the licensees for the nuclear power plants, are responsible for all the measures that are needed to dispose of the radioactive waste in a safe manner. This includes both safely managing and disposing of nuclear waste and spent nuclear fuel arising from operation of the plants, and safely decommissioning and dismantling the plants at the end of their service life. Furthermore, they shall conduct the comprehensive research and development work needed to fulfil these obligations.

The nuclear power companies have given the Swedish Nuclear Fuel and Waste Management Co, SKB, responsibility for nuclear waste management from the time the waste leaves the nuclear power plants. This is why SKB owns and operates the facilities for today's waste management, see Section 1.2. It is also SKB that directly or indirectly conducts the research and development work for final disposal of the waste. The Swedish Radiation Safety Authority (SSM) controls SKB's activities.

The holder of a licence to own or operate a nuclear reactor shall, in consultation with other reactor owners, draw up a programme for the research and development activities and other measures needed to manage and dispose of the nuclear waste and the spent nuclear fuel in a safe manner and to decommission and dismantle the nuclear power plants. Such a programme (RD&D Programme) shall be submitted every three years to SSM. The programme is reviewed and evaluated by the Authority after extensive circulation for comment and by the Swedish National Council for Nuclear Waste, which submits its viewpoints to the Government. The Government then decides whether or not to approve the programme. It is SKB that, on behalf of the nuclear power companies, prepares the RD&D Programmes. The RD&D Programmes presented so far by SKB are described briefly in Section 1.3.

The nuclear power companies are obliged to pay the costs of the measures needed to manage and dispose of the nuclear waste and the spent nuclear fuel and to decommission the facilities. The costs for management and disposal of the operational waste are paid directly, while the financing of the rest of the nuclear waste programme is based on the payment of fees to a special fund, the Nuclear Waste Fund, according to rules in the Financing Act and the Financing Ordinance. On behalf of the nuclear power companies, SKB prepares a cost calculation every three years. SSM reviews SKB's calculation and makes recommendations for fees and guarantees. The size of the fees and guarantees is determined by the Government (with the exception of the guarantee made for Barsebäck, which is determined by SSM). The nuclear power companies pay the fees to the Nuclear Waste Fund, whose assets are, in accordance with Government regulations, deposited in an interest-bearing account with the National Debt Office or in debt instruments issued by the state. The Fund's investment options were broadened in 2009. The Fund's assets can now also be invested in certain debt instruments not issued by the state. The securities in question are covered mortgage bonds.

At the end of 2009 there was about SEK 42 billion in the nuclear power companies' shares of the Nuclear Waste Fund (market value). In addition, some SEK 30 billion (current price level) has been spent in the creation and operation of today's system and for the research and development work. During 2010 and 2011, the average fee is about one öre (100 öre = 1 Swedish krona) per kilowatt-hour of electricity produced for the nuclear power plants that are in operation. The fee for Barsebäck Kraft AB is SEK 247 million for each of these years.

Besides paying fees, the nuclear power companies' parent companies pledge guarantees to cover the fees that have not yet been paid. For the reactors that are in operation, a guarantee is also pledged for the eventuality that the Fund proves insufficient due to unplanned events. The Government can also order a nuclear power company to pay, in addition to the nuclear waste fee, a risk fee and to require the company to identify one or more owner companies that undertake to fulfil the company's payment obligation.

Besides management of the radioactive waste from the nuclear power plants, SKB is also responsible for the management of certain other radioactive waste. This includes low- and intermediate-level waste from medical care, industry and research, which is treated and packaged at Studsvik, waste from handling at Studsvik, and waste from decommissioning of facilities at Studsvik. Waste from Westinghouse's facilities in Västerås and from uranium mining in Ranstad is planned to be managed by Studsvik Nuclear AB or AB SVAFO[1] and is therefore included in the waste coming from there. An agreement exists between Studsvik and SKB concerning the waste managed by Studsvik.

AB SVAFO manages older waste and fuel from the Studsvik area, as well as from decommissioning of the Ågesta reactor and site remediation of the mining operation at Ranstad. According to an agreement with SVAFO, SKB has undertaken to make room for final disposal of radioactive operational and decommissioning waste as well as nuclear fuel from SVAFO.

1.1.2 Fundamental principles

The management of radioactive substances is regulated by legislation. The focus of the work has furthermore been determined by a long series of political decisions and statements, which can be summarized in the following points:

- The waste from the Swedish nuclear power plants will be disposed of within the country's borders.
- Sweden will not dispose of waste from other countries.
- The spent nuclear fuel will not be reprocessed.

SKB plans for a geological disposal of the long-lived nuclear waste and the spent nuclear fuel. Geological disposal was discussed as early as the 1950s. Other more or less unrealistic strategies have also been studied, such as launching the fuel into space, disposing of it beneath the seabed or burying it in the continental ice sheet. Most countries are agreed today that geological disposal is a solution that satisfies all requirements on safe final disposal and feasibility.

[1] AB SVAFO, which is owned by Ringhals AB, Forsmarks Kraftgrupp AB, OKG Aktiebolag and Barsebäck Kraft AB, is a member of the Vattenfall Group.

The following principles underlie the design of SKB's final repository for the spent nuclear fuel and the long-lived radioactive waste:

- The repositories shall be located in a long-term stable geological environment.
- The repositories shall be situated in bedrock that can be assumed to be of no economic interest to future generations.
- The safety of the repositories shall be based on multiple barriers.
- Engineered barriers shall primarily consist of naturally occurring materials that are long-term stable in the repository environment.
- The barriers shall work passively, i.e. without human intervention and without input of energy or materials.
- The repositories shall be designed in such a manner that they do not need to be monitored after closure.

The final repository for short-lived radioactive waste, SFR, which is in operation, is also based on these principles.

The multiple barrier principle is a fundamental and internationally accepted safety principle for final disposal. It entails that the safety of a final repository shall be based on multiple barriers whose purpose is to contain, prevent or retard the dispersion of the radioactive substances in the waste. Which barriers or barrier functions are needed in a final repository is largely dependent on the content of radioactive substances and their half-lives.

The above principles, along with a number of other considerations, such as that it purely technically must be possible to construct the repository, have led SKB to choose the KBS-3 system for final disposal of spent nuclear fuel. In the comparative system analysis /1-1/ carried out by SKB in 2000, the KBS-3 method was judged to be the most advantageous method. In the analysis, SKB compared different methods for disposing of the spent nuclear fuel. The evaluation was done against stipulated requirements, both overall societal requirements and environmental, safety and radiation protection requirements.

The selection of suitable sites for the repositories should take place in steps with a focus on the safety and environmental aspects and by means of an open democratic process in cooperation with national authorities and the concerned municipalities. This principle has guided SKB during the site selection process which led to SKB's selection in 2009 of Forsmark as the site for the Spent Fuel Repository. At the end the choice was between Forsmark in Östhammar Municipality and Laxemar in Oskarshamn Municipality. A decisive factor in the selection of Forsmark was that the prospects of achieving long-term safe disposal were judged to be better there.

1.1.3 The radioactive waste and the spent nuclear fuel

The planning for the management and disposal of the radioactive waste and the spent nuclear fuel is guided by various factors. The properties of the waste are among the most important of these factors. The waste is divided into categories according to its level of radioactivity (very low-, low-, intermediate- and high-level) as well as the life of the radioactivity (short- or long-lived waste). The level of radioactivity determines how the waste is handled. The intermediate-level waste and the spent nuclear fuel, which is high-level, require radiation-shielded handling, while the very low-level and the low-level waste can be handled without radiation shielding. The design of final disposal is largely determined by whether the waste is short-lived or long-lived.

How much waste arises and when it arises are also important premises in the planning of the waste system. The waste quantities are dependent on how long the reactors are assumed to remain in operation. The assumed operating times of the reactors directly influence the capacity and operating times of the different waste facilities. The long-term planning for the waste system is based on the plans for existing reactors in operation. These plans call for 50 years' operation of the reactors at Forsmark and Ringhals and 60 years' operation of the reactors at Oskarshamn. The quantities to which this scenario gives rise are reported below.

Very low-level waste

Very low-level waste is always categorized as short-lived and arises during both operation and decommissioning of the nuclear power plants. Today this waste is deposited in surface repositories. The waste consists primarily of spent filters, replaced components, used protective clothing and trash such as plastic, paper, cables, etc. Near-surface repositories are operated today by the waste producers.

Low- and intermediate-level waste

The low- and intermediate-level waste is divided into short-lived and long-lived waste. Short-lived waste contains a significant quantity of radionuclides with a half-life of no more than 31 years and only a limited quantity of radionuclides with a longer half-life.[2] Long-lived waste contains significant quantities of radionuclides with long half-lives.

Low- and intermediate-level waste arises during both operation and decommissioning of nuclear facilities. The operational waste consists of, for example, spent filters, replaced components and used protective clothing. The decommissioning waste consists of, among other things, scrap metal and building materials.

Most of the low- and intermediate-level waste comes from the nuclear power plants. Other waste comes from Clab and Clink as well as from Studsvik and SVAFO. According to current estimates, SKB plans to dispose of a total of approximately 200,000 m^3 of short-lived radioactive waste. Ongoing decommissioning studies indicate that the waste volumes may be considerably less.

The long-lived waste from the nuclear power plants consists of used core components and control rods. The long-lived nuclides are formed from stable elements in, for example, steel when they are exposed to strong neutron radiation from the reactor core.

The total quantity of long-lived waste is estimated at about 10,000 m^3, about half of which comes from the nuclear power plants. The other half comes from Studsvik and SVAFO. SKB plans to dispose of the long-lived waste in SFL, a final repository for long-lived waste.

Spent nuclear fuel

The spent nuclear fuel is long-lived. It comprises a small fraction of the total quantity of waste to be disposed of. The fuel contains by far most of the radioactivity, both short- and long-lived. Spent nuclear fuel is high-level and requires radiation shielding in conjunction with all handling, storage and final disposal. Final disposal is planned to take place in the Spent Fuel Repository.

The spent fuel generates heat even after it has been removed from the reactor (decay heat). Because of the decay heat, the fuel must be cooled to avoid overheating. The amount of decay heat depends on the fuel's burnup, i.e. the quantity of energy that has been extracted from the fuel. Burnup is specified in megawatt-days per kilogram of uranium (MWd/kgU). Due to technical advances and changes in the operation of the reactors, fuel burnup has increased steadily since the reactors were put into operation. The reason for these changes is to optimize the operating economy of the reactors. The nuclear power companies plan to further increase fuel burnup. It is important in the planning work to clarify the consequences of higher burnup for all parts of the KBS-3 system.

According to the planning premises, the total quantity of spent nuclear fuel to be disposed of will comprise about 6,000 canisters. One canister contains about 2 tonnes of fuel. The quantity of spent nuclear fuel is given as the quantity of uranium that was originally present in the fuel.

In addition to all spent fuel from today's Swedish nuclear power plants, the quantity of spent nuclear fuel to be deposited in the Spent Fuel Repository also includes fuel from the Ågesta reactor, fuel residues from testing programmes at Studsvik and MOX fuel (Mixed Oxide Fuel). These fuel types comprise a very small fraction of the total quantity of spent nuclear fuel. Approximately 20 tonnes of

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[2] Short-lived waste is defined according to the IAEA's Radioactive Waste Management Glossary, 2003 Edition as "waste that does not contain significant levels of radionuclides with half-lives greater than 30 years". SKB uses the same definition but with 31 years to include cesium-137, which is used as a key nuclide to estimate the content of other radionuclides.

spent nuclear fuel from Ågesta and approximately two tonnes of spent nuclear fuel from Studsvik's investigation activities are being interim-stored today in Clab.

23 tonnes of MOX fuel obtained from Germany in exchange for the fuel that was sent to France (La Hague) for reprocessing at an early stage is also being stored in Clab. Sweden has also sent spent fuel for reprocessing to Sellafield in England. During reprocessing, uranium, plutonium and waste products are separated. Uranium and plutonium can be reused in new fuel, MOX fuel. Eighty such fuel assemblies have been fabricated and will be used in the Oskarshamn nuclear power plant.

1.2 Description of the waste system

1.2.1 The Swedish system

Figure 1-1 provides an overview of the system for management and disposal of Sweden's radioactive waste. The illustration shows the flow from the waste producers via interim storage and treatment plants to different types of final repositories. Solid lines represent transport flows of radioactive waste to existing or planned facilities. Dashed lines represent alternative handling pathways.

The Swedish system can be divided into two main parts: the system for management of very low-level and low- and intermediate-level waste, and the system for management of the spent nuclear fuel (the KBS-3 system). The facilities in the former system are operated by both SKB and the waste producers. The programme for realizing SKB's planned facilities in this system is designated the LILW Programme (low- and intermediate-level waste). The LILW Programme is also intended to coordinate planning and technology matters for the decommissioning of the nuclear power reactors and SKB's nuclear facilities. All facilities in the KBS-3 system will be operated by SKB. The programme for realizing the future facilities in this system is designated the Nuclear Fuel Programme.

SKB is responsible for the transportation system, which is common for the LILW and Nuclear Fuel Programmes. The shipments go by sea, since all nuclear power plants and nuclear waste facilities are situated on the coast. The transportation system consists of the specially built ship m/s Sigyn, different types of transport containers and casks, and special vehicles for loading and unloading. M/s Sigyn was built in 1982, and since then the transportation system has been progressively expanded and augmented. For added seaworthiness, the ship has a double bottom and a double hull, providing extra high buoyancy. The double hull also protects the cargo in the event of collision or grounding. Normally, the ship, which is operated by a contractor, makes between 30 and 40 trips per year between the nuclear power plants, Studsvik, SFR and Clab. SKB plans to replace m/s Sigyn with a new ship.

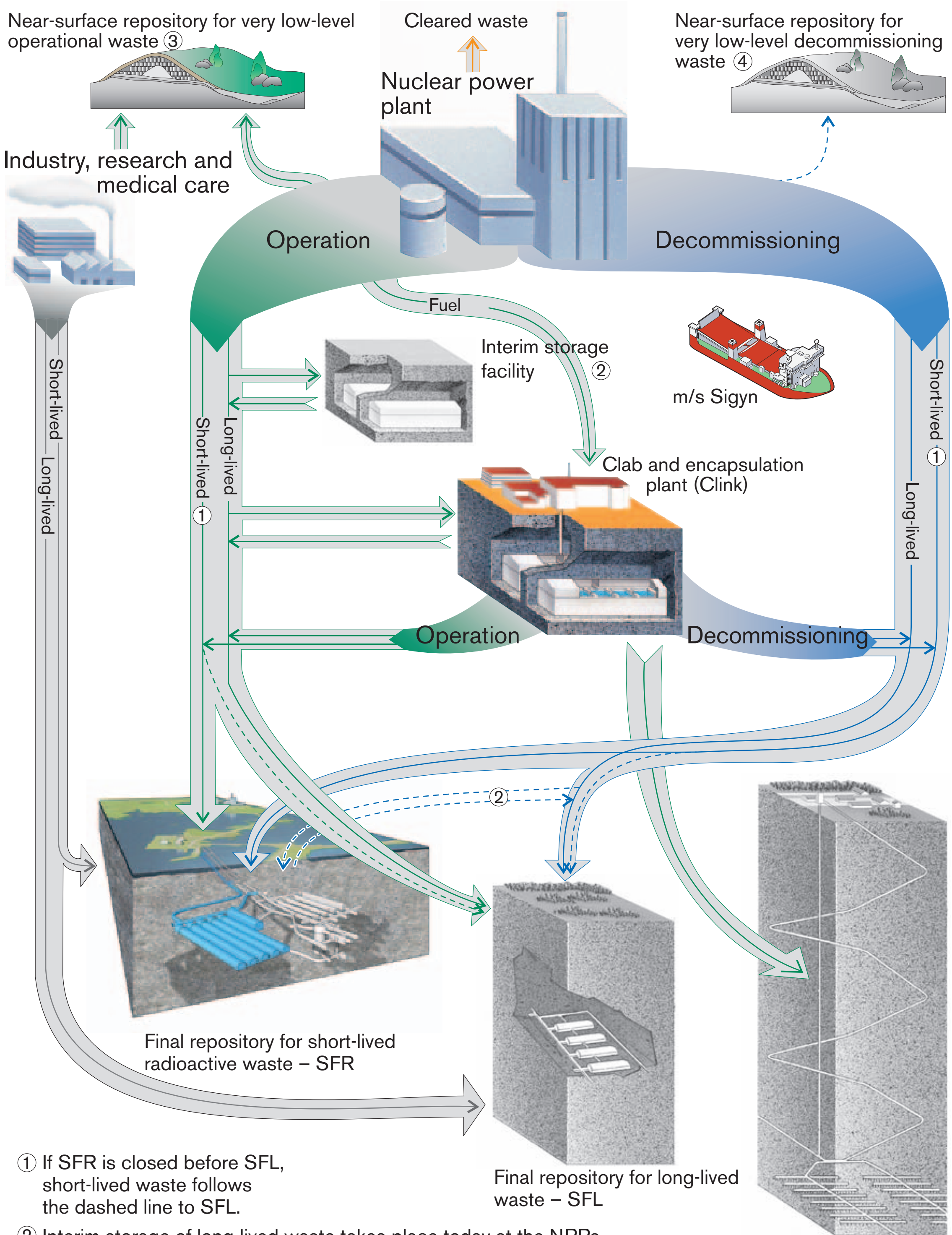
The next two sections describe the facilities that are needed in the two systems. Then SKB's facilities for research, development and demonstration are described.

1.2.2 Facilities in the system for very low-level and low- and intermediate-level waste

In order to be able to manage and dispose of very low-level and low- and intermediate-level waste in a safe manner, facilities for interim storage, treatment and final disposal are needed, along with a system for transportation. Both short-lived and long-lived waste are managed within the system. The facilities are operated both by SKB and the waste producers.

SKB's final repository for short-lived radioactive waste, SFR, has been in operation since 1988. The nuclear power companies, SVAFO and Studsvik operate local treatment plants, interim storage facilities and surface repositories for short-lived waste.

Today long-lived operational waste is interim-stored at the nuclear power plants, Clab and SVAFO's interim storage facility in Studsvik. SKB plans to dispose of the long-lived waste in SFL (final repository for long-lived waste).



① If SFR is closed before SFL, short-lived waste follows the dashed line to SFL.

② Interim storage of long-lived waste takes place today at the NPPs, in Clab and at Studsvik. Interim storage of long-lived waste may also occur in SFR.

③ There are near-surface repositories at the nuclear power plants in Forsmark, Oskarshamn and Ringhals. There is also a similar near-surface repository at Studsvik, to which some waste from industry, research and medical care is sent.

④ Possible alternative for very low-level decommmissioning waste. A decision has not yet been made on this.

Figure 1-1. *System for management and disposal of Sweden's radioactive waste and spent nuclear fuel.*

Facilities for treatment of waste

There are treatment plants for very low-level and low- and intermediate-level waste at the nuclear power plants and Studsvik. Here the waste is treated and packaged so that it meets the requirements for disposal in SFR or surface repositories. The purpose of the treatment may be volume reduction, concentration of the radioactivity or solidification.

Near-surface repositories

There are surface repositories for very low-level operational waste on the industrial areas at the Forsmark, Oskarshamn and Ringhals nuclear power plants and in Studsvik. Before they were put into operation, this waste was disposed of in SFR.

After roughly 50 years the radioactivity in the waste will have declined to the same level as in the natural surroundings and the waste will no longer be dangerous from a radiation viewpoint.

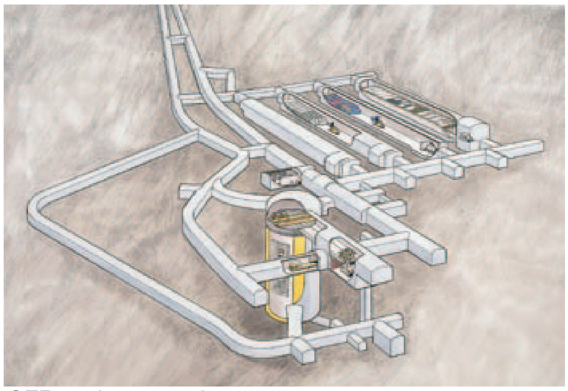
Final repository for short-lived radioactive waste, SFR

SFR is located at the Forsmark nuclear power plant, see Figure 1-2. The repository is situated beneath the Baltic Sea, covered by about 60 metres of rock. Two one kilometre long access tunnels lead from the harbour in Forsmark to the repository area.

The disposal chambers consist today of four 160-metre-long rock caverns of different kinds, plus a 70-metre-high cavern in which a concrete silo has been built. The facility's total storage capacity is 63,000 m^3. One of the four rock caverns contains low-level waste enclosed in standard ISO containers. The waste in this rock cavern can be handled without radiation shielding. Three of the caverns receive intermediate-level waste, which requires radiation shielding. The concrete silo is also intended for intermediate-level waste. The silo will contain most of the radioactive substances in SFR.

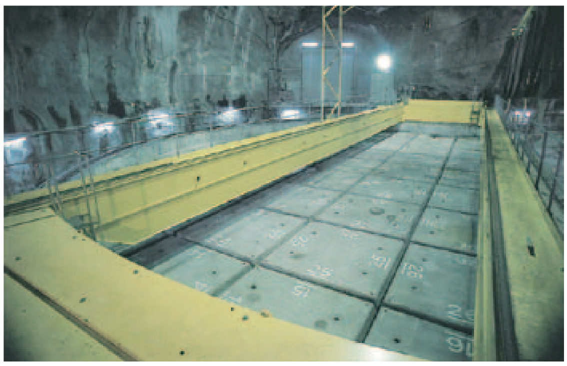
Today only operational waste from the nuclear power plants, Clab, Studsvik and SVAFO is disposed of in SFR. Approximately 33,300 m^3 of waste had been deposited in SFR by the end of 2009.

Aerial view of the surface part



SFR under ground

Rock vault for ILW



View over top of silo

Figure 1-2. *Final repository for short-lived radioactive waste, SFR.*

In order to be able to dispose of all additional short-lived operational and decommissioning waste, SKB plans an extension of SFR.

Interim storage facilities for long-lived waste

Today most of the long-lived waste is interim-stored in storage pools at the power plants and in Clab. In addition, OKG Aktiebolag uses a rock cavern on the Simpevarp Peninsula (BFA) for dry interim storage of operational waste. The operating licence is held by OKG Aktiebolag, but BFA is licensed for interim storage of core components from all Swedish nuclear power plants. Forsmarks Kraftgrupp AB also has a dry interim storage facility on its power plant site.

SKB wants to have the option of storing long-lived waste in the extended SFR and is therefore investigating this possibility.

Besides waste from the power plants, there is long-lived waste from research. This waste belongs to SVAFO and Studsvik and is interim-stored in the facility at Studsvik.

Final repository for long-lived waste, SFL

SKB plans to dispose of the long-lived waste in a facility that is similar to SFR, but that will probably be located at a greater depth. The repository will be the last facility to be put into operation. A repository concept will be developed during the period up to 2013. Siting of the repository is also an open question today.

The volume of SFL will be relatively small compared with SKB's other final repositories. The total storage volume is estimated at 10,000 m^3.

Transportation system for low- and intermediate-level waste

The transportation system consists of the ship m/s Sigyn, special vehicles and different types of transport casks and containers.

Short-lived waste is shipped today from the nuclear power plants and Clab to SFR. Low-level waste does not need any radiation shielding. It can therefore be transported in standard ISO containers. Intermediate-level waste, on the other hand, requires radiation shielding, and most is embedded in concrete or bitumen at the nuclear power plants. The waste is then shipped in transport containers (ATB) with 7–20 centimetre thick walls of steel, depending on how radioactive it is, see Figure 1-3.

Figure 1-3. *Transport container for short-lived radioactive waste (ATB).*

Today long-lived waste is shipped from the nuclear power plants to Clab. The waste consists of control rods from boiling water reactors and replaced core components. The waste is shipped in a transport cask (TK) with approximately 30 cm thick walls of steel, see Figure 1-4. The cask is adapted to Clab's storage canisters. A new waste transport container (ATB 1T) is being developed for shipping long-lived waste in BFA tanks intended for dry interim storage.

M/s Sigyn and the vehicles are also used for shipments of spent nuclear fuel in the KBS-3 system.

1.2.3 Facilities in the KBS-3 system

SKB's interim storage facility for spent nuclear fuel, Clab, has been in operation since 1985. SKB plans to extend Clab with a section for encapsulation of the nuclear fuel and to build a final repository, the Spent Fuel Repository, in Forsmark. In addition to these facilities, we plan to build a canister factory for assembly of the copper canisters in Oskarshamn.

Central interim storage facility for spent nuclear fuel, Clab

Clab is located at the nuclear power plants in Oskarshamn. The spent nuclear fuel is stored in the facility's water pools, see Figure 1-5. Clab consists of a receiving section at ground level and a storage section more than 30 metres below the ground surface. In the receiving section, the transport casks with spent nuclear fuel are received and unloaded under water. The fuel is then placed in storage canisters. Two types of storage canisters are used: normal and compact.

The actual storage chamber consists of two rock caverns spaced at a distance of about 40 metres and connected by a water-filled transport channel. Each rock cavern is approximately 120 metres long and contains four pools and one reserve pool. The water in the pools serves both as a radiation shield and a cooling medium. The top edge of the fuel is eight metres below the water surface. The radiation level at the edge of the pool is so low that the personnel can stand there without restrictions.

At year-end 2009 there were 5,050 tonnes of fuel (counted as original quantity of uranium) in the facility. SKB has a licence to store 8,000 tonnes of fuel in the facility, but if a switch is made to using only compact storage canisters in the pools the storage capacity can be increased to 10,000 tonnes of fuel.

Central facility for interim storage and encapsulation of the spent nuclear fuel, Clink

Before the spent fuel is disposed of it will be encapsulated in copper canisters. SKB plans to encapsulate the fuel in a new facility section adjacent to Clab. When this encapsulation section has been connected with Clab, the two facility sections will be operated as an integrated facility, Clink.

Figure 1-4. *Transport cask for core components (TK).*

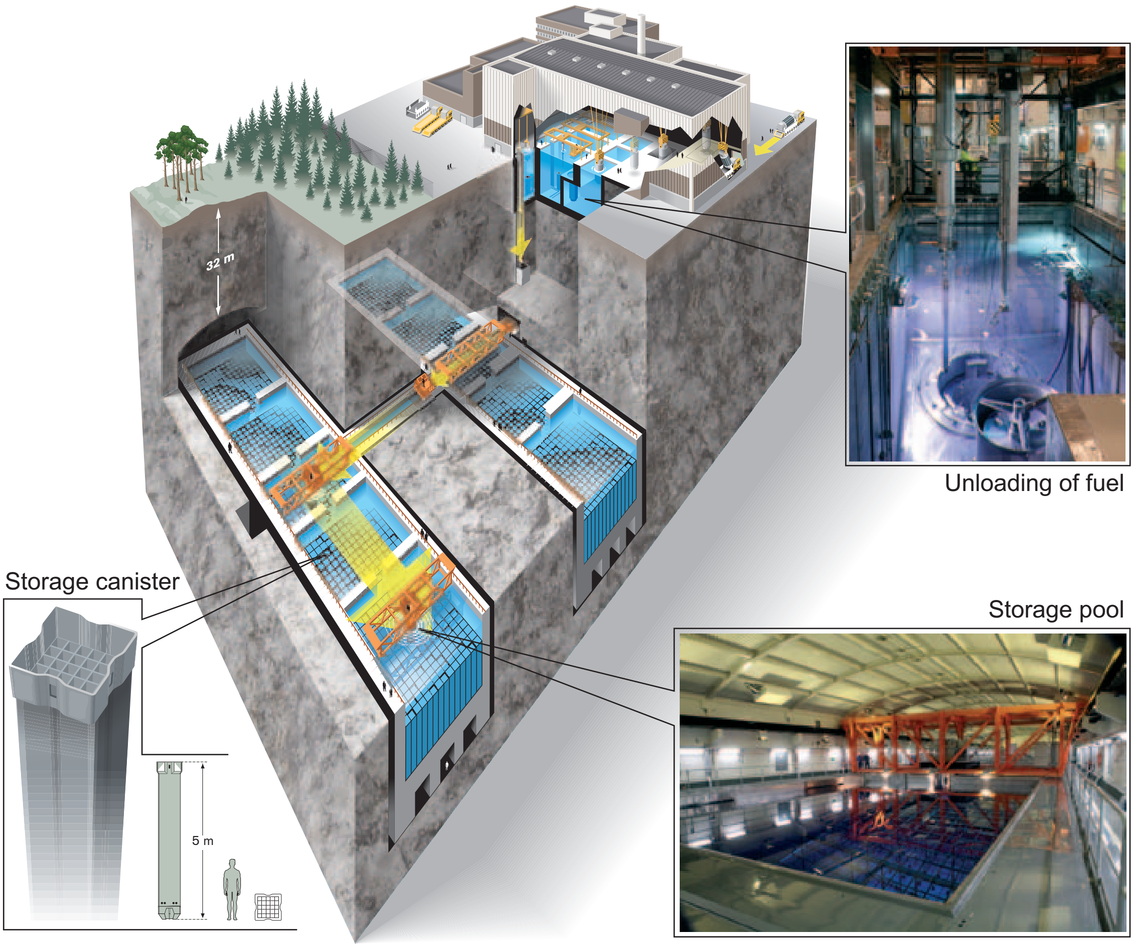
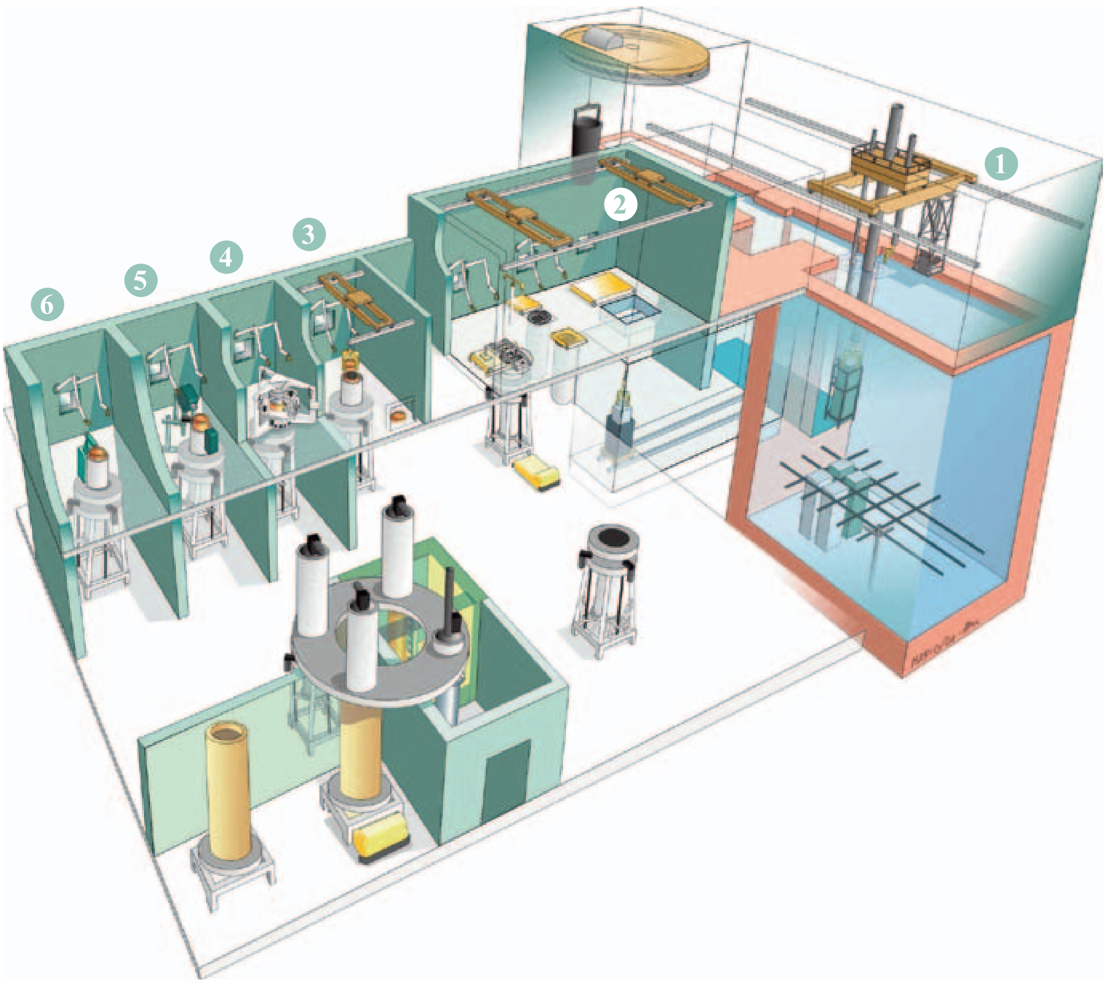


***Figure 1-5.** The central interim storage facility for spent nuclear fuel, Clab.*

The canister that will be used consists of a copper shell to protect against corrosion and an insert of nodular iron to withstand mechanical loads, see Figure 1-6. There are two types of inserts, one that holds twelve assemblies from boiling water reactors (BWRs) and one that holds four assemblies from pressurized water reactors (PWRs). There are other fuel types to be disposed of as well, see Section 1.1.3. They can be placed in one of the two insert types. Plans call for SKB to build a canister factory for assembly and fine machining of the canister's different components in Oskarshamn. The canister factory will not be a nuclear facility.

There will be a number of stations for different work operations in Clink, see Figure 1-7. All fuel handling is remote-controlled. The encapsulation process begins with conveyance of the fuel from the underground storage pools to pools in the new section. The fuel assemblies to be placed together in a canister are selected in such a way that the total decay heat in the canister will not be too great. The fuel is then dried in a radiation-shielded handling cell and lifted over to the canister. The air in the canister is replaced with argon before the canister is sealed. Sealing of the copper canister is done by friction stir welding. The quality of the weld is inspected, and if the weld is approved the canister is taken to the machining station, where excess material is machined off. Finally, a new quality inspection of the weld is performed. If necessary, the canister is cleaned before being placed in a special transport cask for transport to the Spent Fuel Repository.

Figure 1-6. *Copper canister with nodular iron insert (the inset photo shows the copper lid).*



1 = Handling pool. The fuel is moved over from a storage canister to a transfer canister.
2 = Handling cell. The fuel is dried and lifted over to the canister.
3 = Atmosphere change. The air in the canister is replaced with argon.
4 = Sealing. Welding by means of friction stir welding.
5 = Nondestructive testing. Inspection of the welds.
6 = Machining.

Figure 1-7. *The work stations in the encapsulation section of Clink.*

Spent Fuel Repository

The work of finding a suitable site for a final repository for spent nuclear fuel has been under way for several decades. At the end of the site selection process, the choice stood between Forsmark in Östhammar Municipality and Laxemar in Oskarshamn Municipality. After evaluations of the site investigations, SKB selected Forsmark as the site for the final repository for spent nuclear fuel in 2009.

The final repository will consist of a surface part and an underground part, see Figure 1-8. The underground part consists of a central area and a number of deposition areas plus connections to the surface part in the form of a ramp for vehicle transport and shafts for elevators and ventilation. The deposition areas, which together comprise the repository area, will be located about 470 metres below ground level and consist of a large number of deposition tunnels with bored deposition holes in the bottom of the tunnels. The positioning of the deposition tunnels, as well as the spacing between the deposition holes, is determined on the basis of the properties of the rock, for example fractures and thermal properties. The above-ground facility consists of operations area, rock heap, ventilation stations and storeroom.

When the canisters arrive at the Spent Fuel Repository, they are transloaded to a specially built transport vehicle which carries the canisters down to the deposition level via a ramp. There the canisters are transloaded to the deposition machine to be transported out to the deposition area and finally deposited. After the canisters have been emplaced in the deposition holes, surrounded by bentonite clay, the tunnel is backfilled with swelling clay. Other openings are also backfilled when all fuel has been deposited.

The KBS-3 method entails that the canisters can be emplaced either vertically (KBS-3V) or horizontally (KBS-3H). The reference design of the KBS-3 method is based on vertical deposition, but SKB is also investigating the possibility of changing to horizontal deposition later on. The development work being done on horizontal deposition shows that the method is interesting and promising, but more research and development is required before it can be considered available.

Transportation system for spent nuclear fuel

The transportation system consists of the ship m/s Sigyn, special vehicles and different types of transport casks/containers.

Today the spent fuel is shipped from the nuclear power plants to Clab in transport casks (TB) with roughly 30 centimetre thick steel walls, see Figure 1-9. These casks have cooling fins to remove the decay heat generated by the fuel.

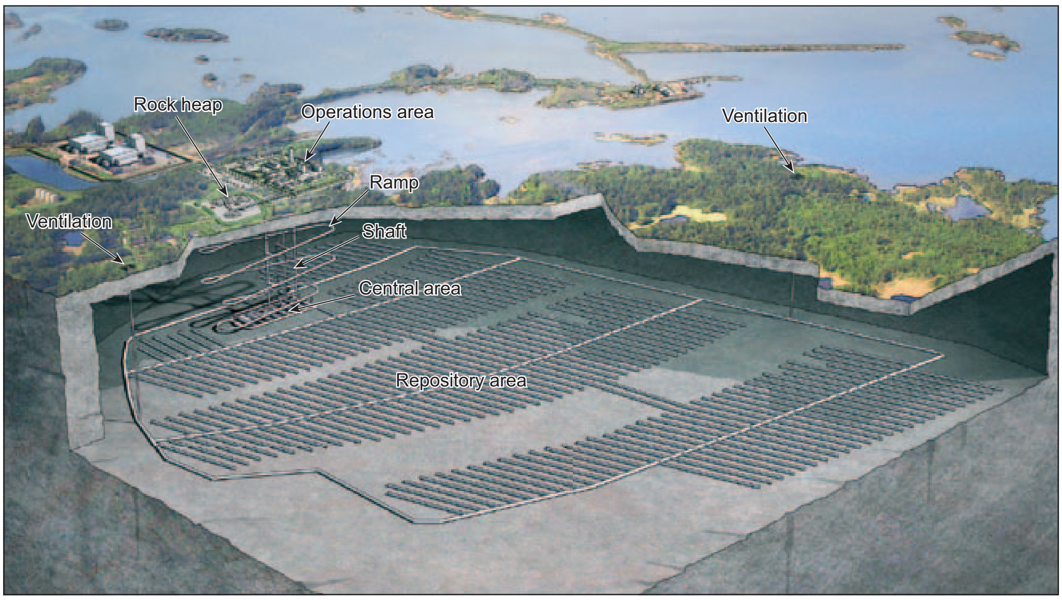


Figure 1-8. *Spent Fuel Repository in Forsmark.*

Figure 1-9. *Transport cask for fuel (TB).*

A new type of transport cask (KTB) will be developed for shipping spent nuclear fuel from Clink to the Spent Fuel Repository.

M/s Sigyn and the vehicles are also used for shipments of low- and intermediate-level waste within the LILW system.

1.2.4 Facilities for research, development and demonstration

Much of the research and development for encapsulation and final disposal of spent nuclear fuel needs to be done in a realistic setting and on a full scale. SKB has three laboratories for this purpose: The Äspö HRL, the Canister Laboratory and the Bentonite Laboratory. There we carry out research and development projects, mainly for the barriers in the KBS-3 repository. The results of the experiments and the projects in the three laboratories serve as a basis for the design of the Spent Fuel Repository and the encapsulation plant as well as the safety assessments that need to be done.

The laboratories also provide an opportunity to demonstrate that the barriers can be fabricated and installed with the quality required to meet the requirements on long-term safety. This is an important part of the work of showing that the initial state of the final repository, which constitutes the basis for the assessment of long-term safety, can be achieved.

The laboratories have so far played a vital role in the development of the KBS-3 method and in the work of providing information on the advances that are constantly being made.

Äspö Hard Rock Laboratory

The Äspö Hard Rock Laboratory (HRL), which was built during the period 1990–1995, is situated on Äspö north of the Oskarshamn Nuclear Power Plant. The underground laboratory consists of a tunnel from the Simpevarp Peninsula, where the Oskarshamn nuclear power plant is located, to the southern part of Äspö. On Äspö the main tunnel descends in two spiral turns to a depth of 460 metres. The various experiments are conducted in niches in the short tunnels that branch out from the main tunnel. An illustration of the HRL is shown in Figure 1-10.

The laboratory is used to investigate how the barriers in the final repository for spent nuclear fuel (canister, buffer, backfill, closure and rock) prevent the radionuclides in the fuel from reaching the ground surface. The laboratory is continuing the work previously conducted in the Stripa mine.

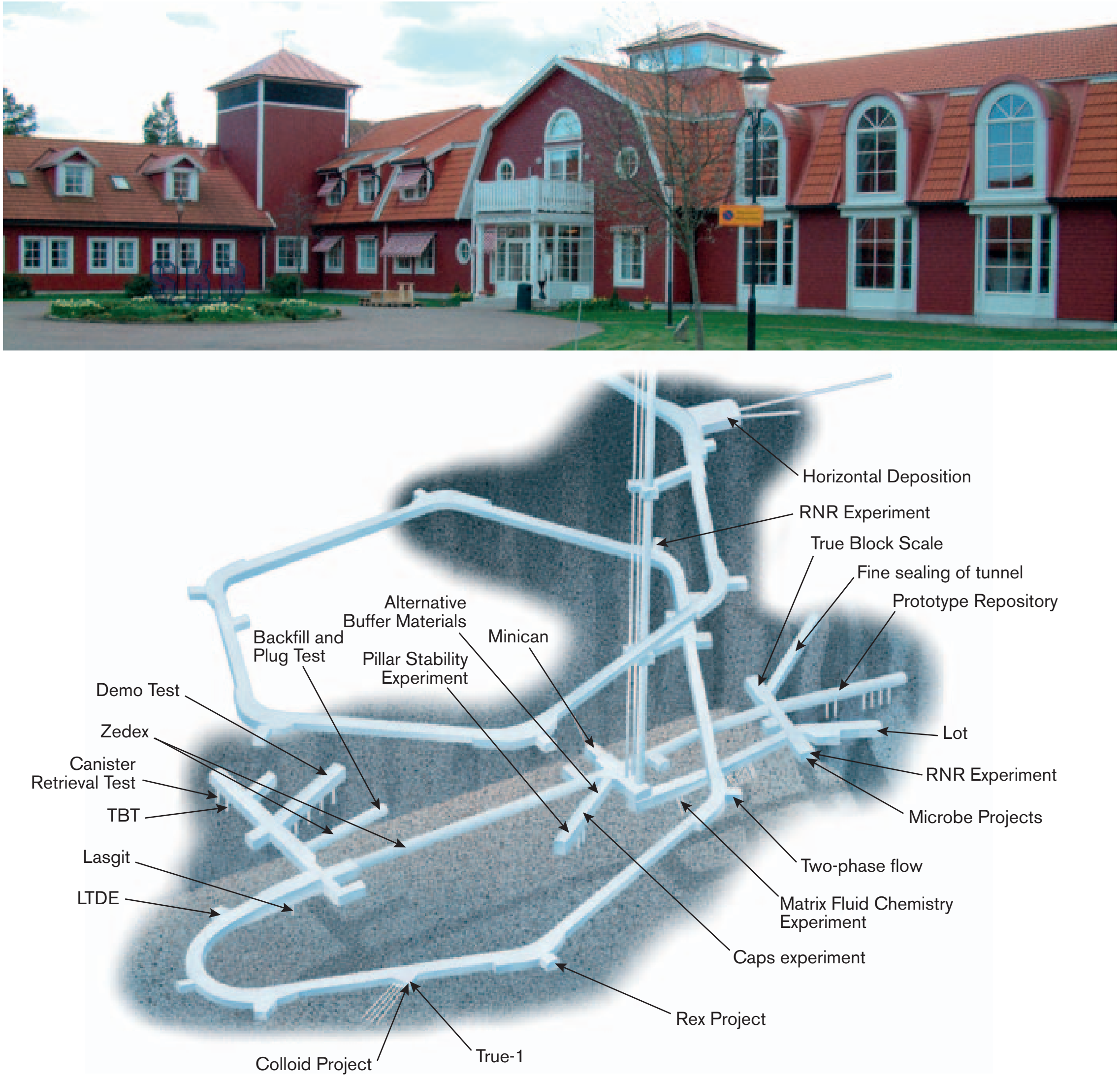


Figure 1-10. *The Äspö HRL.*

Another important purpose is to develop and demonstrate methods for building and operating the Spent Fuel Repository. All of the KBS-3 method's subsystems are available here for demonstration in a realistic setting.

In the future, the facility will be used to train the personnel who will work in the Spent Fuel Repository. SKB therefore expects the laboratory to be in operation at least until the Spent Fuel Repository is put into operation.

Research experiments similar to those being conducted today for the Spent Fuel Repository may be needed for the coming safety assessments of SFR and SFL. Uncertainties in the underlying data for the assessments show which issues may be urgent. We foresee that the parts of the facility nearer the surface will be suitable for experiments for SFR, while the deeper parts will be suitable for SFL.

Many different countries and organizations are participating in the experiments being conducted at the Äspö HRL. In different forms and project groups we are working with sister organizations, research institutes and universities in Canada, the Czech Republic, Finland, France, Germany, Japan and Switzerland. The international contacts are important for being able to compare different methods for calculation and analysis, as well as for a thorough discussion and evaluation of the results. The cooperation also gives us an opportunity to engage the foremost experts in different fields.

Canister Laboratory

The Canister Laboratory, situated in the harbour area at Oskarshamn, was built during the period 1996–1998. One of the shipyard's old welding halls has been converted for use in the development of the sealing technology for copper canisters. It is mainly equipment for welding of copper lids and bottoms and for nondestructive testing of the welds and the different parts of the canister that is developed there. But equipment and systems for handling spent nuclear fuel and canisters are also tested and developed in the laboratory. The facility will also be used for training of personnel in preparation for the commissioning of the encapsulation line in Clink. The Canister Laboratory is therefore planned to be in use until encapsulation of the spent nuclear fuel commences.

There are stations in the Canister Laboratory for testing different welding techniques and different methods for nondestructive testing. The goal is to develop methods that meet the stipulated quality requirements and have sufficiently high reliability to be used in Clink. The most important items of equipment in the laboratory are a friction welder, an electron beam welder, and equipment for radiographic and ultrasonic testing. Figure 1-11 shows the equipment for friction stir welding.

Bentonite Laboratory

SKB has been conducting research and development in the Bentonite Laboratory in Oskarshamn since 2007, see Figure 1-12. The facility is situated adjacent to the Äspö HRL and supplements the experiments being conducted there.

The Spent Fuel Repository's long-term safety is based on multiple barriers that are supposed to prevent radionuclides released from the canister from reaching the ground surface. One of the barriers is the swelling clay, bentonite, that will surround the canister. Bentonite-like clay will also be used to backfill the tunnels in the repository. In the Bentonite Laboratory we test the properties of the bentonite by, for example, simulating water conditions in a controlled manner. Here SKB is also developing methods for backfilling the repository's tunnels and building plugs to seal the deposition tunnels.

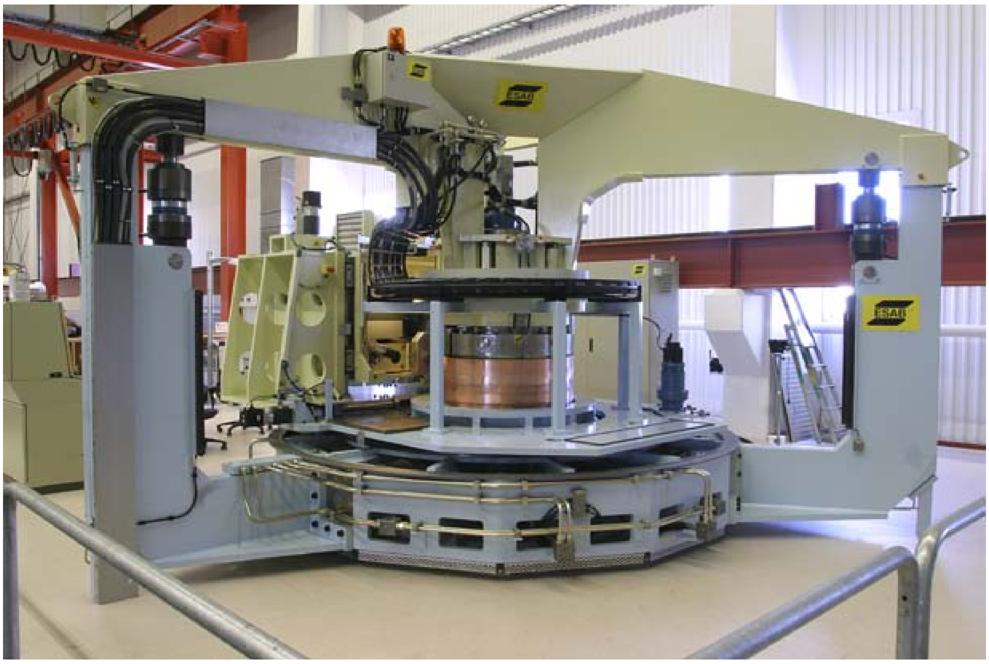


Figure 1-11. *The Canister Laboratory's equipment for development of friction stir welding.*

Figure 1-12. The Bentonite Laboratory.

1.3 Programme for research, development and demonstration

The Nuclear Activities Act regulates the periodicity and scope of the RD&D Programme. The programme shall contain an overview of all measures that may be needed to manage the radioactive waste and shall describe in greater detail the measures intended to be adopted within a timespan of at least six years. The Swedish Radiation Safety Authority reviews and evaluates the programme with regard to planned research and development activities, research results, alternative management and disposal methods and planned measures. After extensive circulation for comment, SSM turns the matter over to the Government, which makes a decision regarding approval of the programme and any guidelines for continued activities.

Development of the KBS-3 method for final disposal of spent nuclear fuel has been going on since the late 1970s. The method was presented in 1983 in a report that served as a basis for the applications for licences to commission the most recently built nuclear power reactors. After the new Nuclear Activities Act had entered into force in February 1984, the applications were supplemented with SKB's RD&D Programme 84, which thereby became a supporting document. In June 1984, the Government granted the nuclear power companies a fuelling permit for the Forsmark 3 and Oskarshamn 3 reactors. In its decision, the Government stated that the KBS-3 method "in its entirety has been found essentially acceptable with regard to safety and radiation protection." The KBS-3 method has since served as a basis for SKB's programmes for research, development and demonstration. SKB has also followed the development of other methods and has on a number of occasions evaluated them in relation to the KBS-3 method.

The focus of the RD&D Programmes has varied through the years, depending on where the emphasis has been in SKB's activities. A brief summary of the RD&D Programmes presented by SKB so far is given below. All the programmes described have been circulated for comment and subsequently approved by the Government.

RD&D Programme 84

As an appendix to the applications for charging permits for the Forsmark 3 and Oskarshamn 3 reactors, SKB submitted its first research programme under the Nuclear Activities Act, RD&D Programme 84, which above all focused on the KBS-3 method. The detailed repository layout and site selection required more research and development. The regulatory authorities commented on the programme and accepted it with a few minor remarks.

RD&D Programme 86

We submitted the first complete research programme under the new Nuclear Activities Act, R&D-Programme 86, in 1986. The guidelines which SKB's research has followed since KBS-3 were summed up and elaborated on as follows:

- The radioactive waste products will be disposed of in Sweden.
- The spent nuclear fuel will be interim-stored and disposed of without reprocessing.
- The waste problem should essentially be solved by the generation that utilizes the electricity from the nuclear power plants.
- The activities must be conducted openly and transparently.

In accordance with the requirements of the Nuclear Activities Act, SKB pointed out the importance of alternative studies and discussed other methods that could be regarded as alternatives to the KBS-3 method in a background report to the programme.

SKB proposed that a new underground laboratory should be built in undisturbed rock in order to make it possible to continue to study the geological and hydrogeological properties of the rock and nuclide transport on a real scale. The research activities at the Stripa Mine were planned to be concluded.

Most of the reviewing bodies found the programme to be well balanced and in compliance with the requirements of the Nuclear Activities Act. The Swedish Nuclear Power Inspectorate (SKI) said that disposal of long-lived waste at great depth (several hundred metres or more) in continental geological formations is the only method deemed to be available and feasible in Sweden within the foreseeable future.

RD&D Programme 89

SKB noted in RD&D Programme 89 that the KBS-3 method has been deemed by the regulatory authorities and the Government to be acceptable in terms of safety and radiation protection. It is therefore a reference alternative for further studies of other ways to dispose of the spent nuclear fuel. A comparison was made between KBS-3 and an alternative design for geological disposal called WP-Cave. SKB judged that it is more difficult to show that the WP-Cave method is safe in the long term, and the studies of WP-Cave as an integral system were therefore concluded.

In the programme, SKB provided information on its plans for a safety assessment, SKB 91. The reason was the need to evaluate what variations in geological conditions mean for the performance and safety of the final repository.

Preliminary investigations to locate a hard rock laboratory in the Simpevarp area were conducted and showed that good prospects existed for locating such a facility on Äspö, north of Simpevarp.

In its decision regarding RD&D Programme 89, the Government found that the research work ought to include an account and a follow-up of alternative management and disposal methods. A binding commitment should not be made to a method until a complete picture was obtained of safety and radiation protection aspects.

A point of departure for the continued RD&D activities should be that a final repository for nuclear waste and spent nuclear fuel could be put into operation stepwise.

RD&D Programme 92

The nuclear waste programme was concretized in RD&D Programme 92 by SKB's presentation of a plan to realize the deep geological disposal of encapsulated spent fuel. This was the start of the process of siting a final repository. The aim for the encapsulation plant was that it should be built at Clab. The proposal from the National Board for Spent Nuclear Fuel, SKN, that the repository should be built stepwise was taken up and included in the programme, in accordance with the Government's wishes. Demonstration deposition of about 400 canisters was planned in an initial stage. After this it will be possible to continue along the chosen path or retrieve the fuel.

Important background reports to the programme were the safety assessment SKB 91 /1-2/ and the PASS report /1-3/, which compared different encapsulation methods and final disposal methods (KBS-3, deep boreholes, long tunnels, medium-long tunnels). The PASS report recommended keeping KBS-3 as the reference system with a copper canister and a steel insert.

In its evaluation of the programme, the Swedish Nuclear Power Inspectorate, SKI, approved focusing the continued RD&D work on a KBS-3-type method. SKI observed that there is no method that appears to be significantly better from a safety viewpoint and that can be realized in Sweden without significantly extending the time frame compared to SKB's plans.

The Government's decision called for a supplementary account to SKI. SKB was supposed to supplement RD&D Programme 92 by describing:

- the criteria and methods that can form a basis for the selection of sites suitable for a final repository,
- a programme for description of design premises for an encapsulation plant and a final repository,
- a programme for the safety assessments which SKB intends to prepare,
- an analysis of how different measures and decisions influence later decisions in the final repository programme.

Supplement to RD&D Programme 92

SKB submitted the requested supplement in August 1994. In its subsequent decision, the Government made it clear that a licence application for a final repository should contain comparative assessments based on site-specific feasibility studies on between five and ten sites in the country and that site investigations should be conducted on at least two sites. The reasons for the selection of these sites should be given. The siting factors and criteria stipulated by SKB should, in the Government's opinion, serve as a starting point for the continued siting work.

The Government was also of the opinion that SKB, in order to provide background and premises for the siting work, should present its general siting studies and site-specific feasibility studies in an integrated account in future RD&D Programmes.

RD&D Programme 95

The emphasis in RD&D Programme 95 was on how SKB planned to execute the development projects (encapsulation, final repository) that are required to initiate deposition of encapsulated fuel. The programme also included the supportive research and development work needed for the projects as well as a follow-up of and research on alternative methods. The following comprised important background documents for the programme:

- The reports on the feasibility studies in Storuman and Malå.
- General siting study 95 – A nationwide survey of conditions and background for the siting work /1-4/.
- A template for safety reports, SR 95 /1-5/.

In its review of RD&D Programme 95, SKI said that considerable progress had been made since RD&D Programme 92. Newly developed methodology now needed to be applied and evaluated. Previous assessments of important safety factors must be reconciled with new knowledge and modifications of the repository system. Furthermore, SKI stressed that, as a basis for future decisions on the final choice of system solution, the zero alternative in particular should be presented as a reference for the deposition alternative.

RD&D Programme 98

In RD&D Programme 98, SKB presented detailed material on the points highlighted by the Government in its treatment of RD&D Programme 95, i.e. alternative solutions to the KBS-3 method, system analysis of the entire final repository system, siting data and site selection criteria.

Regarding alternative solutions, the report contained a broad account of both alternative methods and variants of the KBS-3 method. The long-term safety of the final repository was dealt with and a coming report on a safety assessment (SR 97) was announced. SKB also described the work that was planned prior to coming site investigations up to a decision in 2001. The reference canister with a five centimetre thick copper shell and a cast iron insert was also specified in the programme.

In its decision on RD&D Programme 98, the Government requested that SKB submit a supplementary account regarding alternative methods, background material for site selection, and a programme for the site investigations.

Supplement to RD&D Programme 98

In November 1999, SKB presented the safety assessment SR 97 /1-6/. SKI and SSI observed in their joint review that it contained the constituents required for a comprehensive analysis of safety and radiation protection. The authorities also said that the KBS-3 method was a good basis for SKB's upcoming site investigations and the continued development of the engineered barriers.

In December 2000, SKB submitted the supplementary accounts which the Government requested in its decision on RD&D Programme 98 regarding alternative methods, material for site selection, and programme for the site investigations. The account (often referred to as RD&D-K) contained an exhaustive summary and evaluation of the comprehensive siting data collected by SKB over the years. In the programme, SKB gave an account of its selection of sites for site investigations and a general programme for the site investigation phase. The Government's decision gave the go-ahead for SKB to continue the work according to the account in RD&D-K. The Government had no objections to SKB's initiation of site investigations within the three areas Simpevarp, Forsmark and Tierp north. The Government also deemed that the KBS-3 method should be used as a planning premise for the site investigations.

RD&D Programme 2001

RD&D Programme 2001 concentrated on questions related to the safety assessment and its research and technology development needs. Questions concerning siting of the encapsulation plant and the final repository were not taken up, since SKB was at this time waiting for the Government's decision regarding the supplement to RD&D Programme 98. The programme took as its starting point the regulatory requirements on long-term safety and linked this to the development of methodology for the safety assessment and to the research on the long-term processes in the repository. RD&D Programme 2001 also included accounts of the state-of-the-art regarding partitioning and transmutation of spent nuclear fuel and deposition in deep boreholes.

The programme also included a timetable where the starting point was that site investigations for a final repository for spent nuclear fuel should be initiated in 2002. The plan entailed that regular operation (now called routine operation) can be commenced in the early 2020s before the storage pools in Clab are full. This will avoid the necessity of further extension. According to the plans, the encapsulation plant will be ready for operation one year before the final repository. The timetable also included conducting safety assessments of the final repository, based on data from the site investigation phase.

In their comments on RD&D Programme 2001, SKI and SSI called for a clearer account of the planning for the remainder of the nuclear waste programme. The Government then requested a plan of action in connection with the approval of the research programme.

RD&D Programme 2004

RD&D Programme 2004 was mainly concerned with the development of fabrication and sealing of canisters for final disposal of spent fuel. The reason was that SKB was supposed to submit an application for a licence under the Nuclear Activities Act for an encapsulation plant during the coming programme period.

SKB presented the plan of action that was called for in the review of RD&D Programme 2001. The plan was divided into two parts: The KBS-3 system (the Nuclear Fuel Programme) and the waste system for low- and intermediate-level waste (LILW). The emphasis in the work during the programme period 2004–2009 was on the Nuclear Fuel Programme. The plan entailed that SKB should gather the material that was needed to submit the applications for the encapsulation plant and the final repository. In its review statement on RD&D Programme 2004, SKI stated that they thought that SKB's plan of action was incomplete and needed to be structured better. SKI called for a more detailed account of the content of the background material which SKB intended to submit on the different reporting occasions.

RD&D Programme 2007

RD&D Programme 2007 established the next big objective: to submit an application under the Nuclear Activities Act for the final repository for spent nuclear fuel and an application under the Environmental Code for the entire KBS-3 system. The plans for these parts of the Nuclear Fuel Programme were presented in a structured manner based on the stepwise execution of the programme described in SKB's plan of action.

The programme described the technology development for the different subsystems in the final repository. Their properties are determined by the design premises and specifications that have been formulated, but also by the production and inspection methods that are used. Six production lines were defined for the descriptions: the rock line, the buffer line, the fuel line, the canister line, the backfilling line and the closure line. In addition, retrieval of canisters and a variant of the repository design with horizontal deposition (KBS-3H) were described.

RD&D Programme 2007 also addressed the long-term safety assessment, with reference to the SR-Can safety assessment that was submitted to the regulatory authorities in November 2006. Furthermore, the programme described research activities within the social sciences field, the programme for management and disposal of low- and intermediate-level waste (LILW) from the nuclear power plants and decommissioning of nuclear facilities.

In its decision, the Government called for supplementary accounts of the final repository for long-lived low- and intermediate-level waste (SFL), the final repository for short-lived low- and intermediate-level waste (SFR), decommissioning and alternative methods for disposal of spent nuclear fuel.

Supplement to RD&D Programme 2007

In March 2009, SKB submitted the requested supplementary account of the LILW Programme and decommissioning, as well as the state-of-the-art regarding alternative final disposal methods.

In its review, SSM found that SKB's account in certain areas did not fully meet the Authority's expectations, but SSM found that the supplement to RD&D Programme 2007 was satisfactory without any additional measures. The Government concurred in SSM's conclusion and observed that the Authority had initiated a dialogue with SKB and the licensees and assumed that SKB would heed the viewpoints of the regulatory authorities in RD&D Programme 2010.

1.4 Competence and organization

SKB faces new challenges as the planned facilities are successively realized. An important part of meeting these challenges is to continuously develop our competence and organization. Historically, SKB has been a management organization with a number of experts assigned to lead and manage the work. This has gradually changed towards a broader staffing policy, and we now operate the facilities in-house and have more staff specialists and administrators.

SKB took over the operation of Clab in January 2007 and the operation of SFR in July 2009. These facilities were formerly operated under contract by OKG Aktiebolag and Forsmarks Kraftgrupp AB, respectively. The takeover is a natural step in SKB's development, where the emphasis in the activities is gradually shifting from research and development to operation of nuclear facilities. The experience we have gained from the construction and operation of Clab and SFR is important in the planning of the future facilities.

A new Department of Nuclear Safety was formed in conjunction with the takeover of Clab. The department has an independent role within SKB with responsibility for reviewing SKB's activities (independent review in accordance with nuclear activities legislation) and for development in the field of safety and radiation protection.

SKB has conducted a study aimed at proposing measures and a plan of action for development of the organization during the period up until when the Spent Fuel Repository and Clink are put into operation. The study was conducted in two stages. In the first stage, which was carried out during 2005–2006, two programmes were established: The LILW Programme and the Nuclear Fuel Programme. The activities required to realize the future facilities are being carried out within these programmes.

The second stage in the organizational study was carried out during 2007–2008, when more detailed proposals for how SKB should be organized were put forward. Most of the proposals have since gradually been implemented. Among other things, in 2009 SKB created a department for the management of low- and intermediate-level waste. A gradual accumulation of competence within the department SFR for executing the extension of SFL and the planning for SFL is taking place.

As another result of the study, competence is being accumulated in civil engineering, design engineering and interdisciplinary engineering. Previously this competence has been brought in from the outside, mainly from OKG. Competence accumulation will initially take place within the Operations Department. The purpose is that this competence will eventually be used throughout SKB.

SKB is strengthening its organization in other competence areas as well such as procurement, project management and information management and is working actively to establish guidelines for how to use consultants and key competencies as a part of our competency management.

SKB's management system comprises an important tool that is needed for us to be able to build and operate the planned facilities. The management system is certified to the quality management and environmental management standards ISO 9001 and ISO 14001. Furthermore, the management system lives up to all the requirements made in SSM's regulations for operating nuclear activities. The management system is being developed progressively as the activities change.

1.5 Requirements management

One of the prerequisites for building and operating the facilities is that they satisfy the requirements made by SKB, the regulatory authorities and other stakeholders. To ensure this, SKB applies systematic requirements management /1-7/. The methodology was originally developed for the design of the Spent Fuel Repository and is now also being used for the extension of SFR. An overall purpose is to facilitate system understanding by putting the details of the design in their context and deriving them from stipulated requirements. Another important purpose is to make background data and reasons for the design of the final repository traceable.

The methodology entails documenting the design premises in a database and structuring them in levels that correspond to different degrees of detail in the design. Each level can be seen as a specification of the repositories and their design. At the uppermost level, the problem to be solved is specified and fundamental requirements and principles for the design of the final repository are stipulated. At the next two levels, levels two and three, the functions and properties which the whole final repository and its parts must possess to solve the problem and comply with the principles are specified. Principles to be considered in the design work are also defined here, for example application of proven and reliable technology and limitation of environmental impact. At the next two levels, detailed design premises and reference design are specified. The final reference design will comprise a basis for fabrication and inspection specifications.

By relating each detailed design premise to a more general one, the requirements management system shows how SKB has taken into account the overall statutory requirements and translated them into the practical design.

Requirements management also includes decision and review procedures for design premises and reference design. Procedures for how the tool should be used are devised within the framework of SKB's management system.

In practice, the design premises are found in different types of documents, for example they are included in the safety analysis report and specifications for fabrication. The documents provide background and reasons for the design premises, while the system for requirements management keeps track of dependencies between the different parts of the repository and makes the development of the system traceable. The system should also show how compliance with the design premises is to be verified.

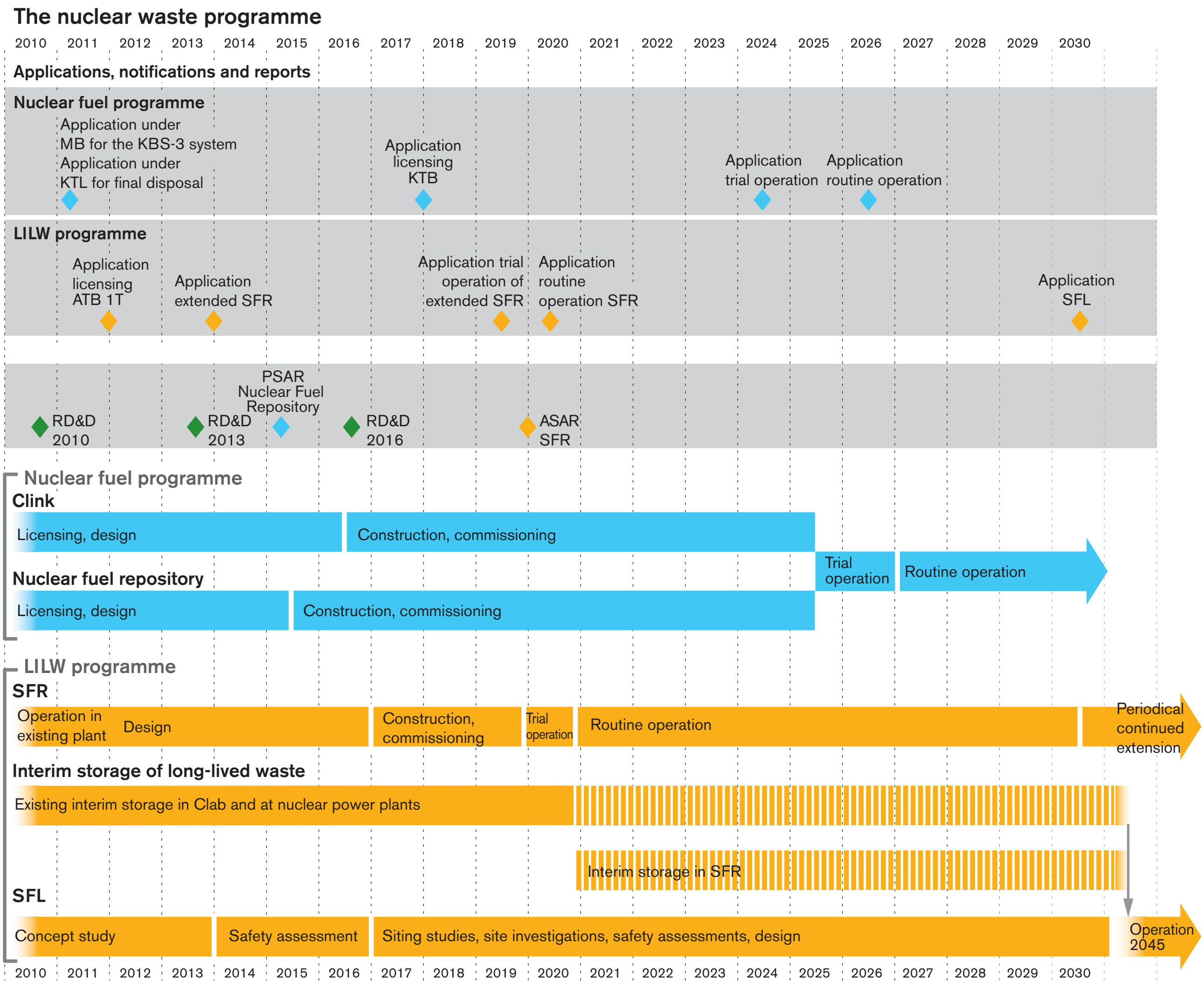
2 Overall plan of action

This chapter gives a general picture of SKB's planning to construct and commission new facilities and facility sections at existing facilities. More detailed plans of action for the LILW Programme and the Nuclear Fuel Programme are presented in Part II and Part III. The planning for analyses of the repositories' long-term safety and scientific research for both programmes is presented in Part IV. The planning for the social science research is presented in Part V.

2.1 Main timetable

Figure 2-1 shows the overall timetable for the entire nuclear waste programme. The plan briefly indicates the measures that are needed to execute the programme and when SKB plans to submit applications and other statutory reports, as well as periodic safety analysis reports.

The overall planning for the LILW and Nuclear Fuel Programmes is presented in the following sections.



ATB 1T Transport cask for long-lived waste
Clink Facility for central interim storage and encapsulation of spent nuclear fuel
RD&D Programme for research, development and demonstration
KTB Transport cask for canisters with spent nuclear fuel
KTL Nuclear Activities Act
MB Environmental Code
PSAR Preliminary Safety Analysis Report
ASAR As-operated Safety Analysis Report
SFL Final repository for long-lived waste
SFR Final repository for short-lived radioactive waste

Figure 2-1. *SKB's main timetable for execution of the LILW and Nuclear Fuel Programmes.*

2.2 LILW Programme

2.2.1 Current situation

The current situation in the LILW Programme can be summarized in the following points:

- Preliminary design and site investigations are under way for the extension of SFR. The site investigations including subsequent analysis work will be concluded during the first half of 2011.
- A study is being conducted to investigate the consequences of an interim storage of long-lived waste in the extension of SFR.
- SKB, together with the nuclear power companies, is conducting unit-specific decommissioning studies. The studies will provide better estimates of the waste quantities that arise in connection with decommissioning.
- SKB has carried out a feasibility study to investigate the alternative of disposing of very low-level decommissioning waste in surface repositories. The study will be updated with the results of the decommissioning studies.

2.2.2 Planning

Short-lived waste

SKB plans to extend SFR for disposal of the short-lived decommissioning waste and the additional quantity of short-lived operational waste that results from planned longer operating times of the nuclear power plants. In calculating capacity we also take into account waste from SVAFO and Studsvik, which includes waste from hospitals, research and industry.

SKB plans to extend SFR in stages to retain flexibility with regard to the necessary disposal volume. We plan to submit the applications under the Nuclear Activities Act and the Environmental Code at the end of 2013. They will include both the existing facility and the completed extension. In connection with the applications for extension, SKB will apply for a licence to dispose of both operational and decommissioning waste in the entire facility. Today the operating licence only includes operational waste.

According to the plans, construction of the first stage can begin in early 2017 and routine operation can start at the end of 2020. Routine operation is preceded by about one year's trial operation. Both trial operation and routine operation require licences from SSM. SFR can then be extended as the need arises.

SKB is considering the possibility of disposing of very low-level waste from the decommissioning of the nuclear power plants in surface repositories. A feasibility study of the possibilities and consequences of this has been completed. The study will be updated in 2011 with the results that emerge from the ongoing decommissioning studies. The nuclear power companies and SKB will then be able to make a joint policy decision.

Long-lived waste

The long-lived waste that arises during operation and decommissioning of the nuclear power plants and other nuclear facilities is planned to be disposed of in SFL. Before SFL has been commissioned, the long-lived waste must be interim-stored. Today long-lived waste is interim-stored at the nuclear power plants, in Clab and in facilities at Studsvik. SKB is currently investigating the possibility of arranging an interim storage facility in the coming extension of SFR. If long-lived waste is to be interim-stored in SFR, an application for interim storage will be made in conjunction with the application for extension.

Long-lived operational waste is currently kept at the Barsebäck plant. This waste will be removed from the facility and taken to an interim storage facility before dismantling of the nuclear power plant begins. According to the plans, the control rods will be taken to Clab, while Barsebäck Kraft AB is exploring various alternatives for interim storage for the remaining long-lived operational waste. Possible alternatives could be Clab, BFA (rock cavern for waste) on the Simpevarp Peninsula in Oskarshamn, or in an interim storage facility on the Barsebäck plant site. If a future interim storage of long-lived waste in SFR becomes reality, this and other waste stored on the power plant sites can eventually be transferred to SFR.

SKB is developing a new container, ATB 1T, for shipping long-lived waste. The container will be licensed by French regulatory authorities. The licence will then be validated by SSM.

Interim storage of control rods takes place today in pools in Clab. SKB is considering dry interim storage of these rods instead. A project to investigate this possibility will start within the coming three-year period. The purpose is to free storage capacity in Clab for the spent fuel.

According to current plans, we will be able to submit our applications for SFL in around 2030. Many milestones have to be passed before then. The most important ones are: selection of repository concept and site plus investigations and evaluation of repository safety, both long-term (post-closure) and during operation (pre-closure). The time from applications to commissioning of the repository is more uncertain, since it involves activities in a relatively distant future. At the present time we estimate that SFL can be put into operation in around 2045.

Even if the commissioning of SFL lies far in the future, the development work needs to begin. Work has begun on a feasibility study of possible repository concepts. The selected concept will then be further refined and serve as a basis for the assessment of long-term safety which SKB plans to complete at the end of 2016. Siting of the repository is still an open question. The site selection process will proceed in parallel with the development of the repository concept. SKB plans to begin the site investigations in 2020. The investigations will last for about six years. Before applications for SFL can be submitted, a preliminary design of the facility must also be produced.

2.3 Nuclear Fuel Programme

The programme for realizing the encapsulation plant and the final repository for spent nuclear fuel is called the Nuclear Fuel Programme.

2.3.1 Current situation

The current situation in the Nuclear Fuel Programme can be summarized in the following points:

- The site investigations have been concluded.
- In 2009 SKB selected Forsmark as the site for the final repository for spent nuclear fuel.
- Applications under the Nuclear Activities Act for final disposal of spent nuclear fuel and under the Environmental Code for the KBS-3 system will be submitted in early 2011.
- The Spent Fuel Repository and Encapsulation Plant projects have been initiated. The main goal of the projects is to design, build, staff and commission the respective facilities.
- Within the Spent Fuel Repository project, initial design is under way along with site activities such as establishment of a site office and further geoscientific investigations.
- Within the Encapsulation Plant project, preparations are under way to carry out system and detailed design.

2.3.2 Planning

SKB has now come so far in the development of the KBS-3 system for final disposal of the spent nuclear fuel and the assessments of long-term safety that we can show that the system has good prospects of fulfilling the stipulated requirements. SKB will shortly submit an application under the Nuclear Activities Act for the final repository and an application under the Environmental Code for the KBS-3 system. The environmental impact statement appended to both applications replaces the one submitted in 2006 with the application for the encapsulation plant under the Nuclear Activities Act.

Now that the applications are imminent, the planning of the construction and operation of the facilities for encapsulation and final disposal has progressed from being a development issue to being a licensing issue. The implementation plan should therefore be handled within the licensing process. Development of the technology needed for the two facilities will continue to be handled within the RD&D process, however. An overview of the planning is shown below to provide an overall picture of the Nuclear Fuel Programme.

The timetable for the Nuclear Fuel Programme after the applications have been submitted is divided into the following phases: an initial design and licensing period, thereafter construction and commissioning followed by operation with deposition of spent nuclear fuel.

Current planning for the Nuclear Fuel Programme is based on the assumption that licensing will take about five years. Before construction of the Spent Fuel Repository may begin, a preliminary safety analysis report (PSAR) must also be submitted and approved by SSM. SKB therefore expects to begin construction of the encapsulation plant and the Spent Fuel Repository in 2015. During the period up to the start of construction, the organizations for the Spent Fuel Repository and Encapsulation Plant projects will be built up. System and detailed design of the two plants will be carried out during this period. In addition, procurements will be carried out.

Construction of the facilities will gradually segue into commissioning as the subsystems are completed and can be tested. The testing will conclude with integrated testing of each facility and finally of the entire KBS-3 system. The encapsulation plant will be interconnected with Clab. Then integrated testing of the combined facility, Clink, will be carried out. Before trial operation of Clink and the Spent Fuel Repository can begin, the safety analysis reports for the different facilities will be updated so that they reflect the facilities as they are built. Trial operation begins when a license has been obtained from SSM and continues until routine operation can commence. SKB estimates that trial operation at the Spent Fuel Repository and Clink can begin in 2025.

Before shipments of encapsulated spent nuclear fuel and trial operation begin, SKB will develop and license a new cask for shipments from Clink to the Spent Fuel Repository, called KTB.

Based on the experience gained during trial operation, the safety analysis reports and the safety-related technical specifications will be supplemented. They will then be included as supporting material in SKB's applications for licences to commence routine operation of the facilities. Routine operation will continue until closure of the final repository can commence.

2.3.3 Closure and provision for retrieval

The Spent Fuel Repository is designed in such a manner that it is possible to retrieve deposited canisters during deposition. However, no special measures are taken to facilitate retrieval of the canisters after closure of the repository.

During the operating phase, individual canisters may need to be taken up out of the deposition hole if something unforeseen happens during deposition. SKB has demonstrated that this can be done in a safe manner in the Äspö HRL. After the Spent Fuel Repository has been closed, more labour is required to carry out a retrieval because large quantities of backfill and tunnel plugs must be removed.

In Sweden there is no formal requirement that it should be possible to retrieve deposited canisters after closure of the Spent Fuel Repository. Nor is it the intention with the KBS-3 method that deposited canisters should be retrieved, but the design of the repository is such that if future generations should wish to retrieve the fuel, this is fully possible (albeit resource-consuming).

The Spent Fuel Repository is designed in such a manner that its safety is not dependent on post-closure monitoring. Once the repository has been closed, SKB will have satisfied the statutory requirements on safe final disposal of the spent nuclear fuel. The question of ultimate responsibility for final disposal, i.e. long-term post-closure responsibility for the final repository for spent nuclear fuel, is currently being studied by the Inquiry on Coordinated Regulation in the Nuclear Safety and Radiation Protection Field. The chief investigator has been instructed to consider the need for and the possible design of a legal regulation of the long-term responsibility for the final repository for spent nuclear fuel.

3 Flexibility in the face of changed premises

Planning of the LILW- and Nuclear Fuel Programmes is based on the strategic assumptions that are judged to be most realistic today. The current time horizon is about 75 years, so we have to assume that changes will occur in the planning premises and that the current assumptions for the planning may be re-evaluated. Furthermore, the progress of both the LILW Programme and the Nuclear Fuel Programme is dependent on external decisions, which means that SKB does not have full control over the implementation of the programmes.

This necessitates flexibility to adjust to changes in the premises for the programmes. It is often possible to adjust to such changes by means of minor modifications of the programmes without major changes in the long-term timetable. But there is of course always a limit where the changes become so great that they require substantial and far-reaching adjustments, for example additional facilities or facility sections, changes in the layout of a final repository, shorter or longer licensing review processes than previously estimated, and so forth.

The activities permit relatively great flexibility with respect to altered premises. Examples of the flexibility of the programmes (and limitations on this flexibility) in response to a number of specific changes in the premises are given below.

3.1 Operating times of the nuclear power reactors

SKB's planning is based on 50 years' operation of the reactors in Forsmark and Ringhals and 60 years' operation of the reactors in Oskarshamn. Any changes in these assumed operating times entail changes in the planning premises for the LILW and Nuclear Fuel Programmes.

Extended estimated operating times for the reactors entail a larger quantity of both operational waste and spent nuclear fuel, requiring an increase in the capacity of the repository systems. Extended operating times also mean that SKB's facilities will be utilized for a longer time.

For the Nuclear Fuel Programme, current planning premises entail final deposition of 6,000 canisters, with a deposition rate of about 150 canisters per year. Extending the operating time of one reactor by one year entails that 10–30 more tonnes of spent nuclear fuel must be interim-stored and 5–15 more canisters must be disposed of (the volume depends on e.g. the size of the reactor and the operating time during the year). The need for interim storage of additional spent fuel arises at the end of the currently planned operating time of 50 or 60 years, and by then fuel will have been outloaded from Clab, making room for this additional fuel. It is assumed that today's planned final disposal capacity can be increased by better utilization of the deposition areas and by making use of unutilized areas at the selected repository depth. It is also possible that rock volumes outside of those deemed suitable today may be suitable for deposition.

Extended operating time also means more short-lived operational waste to dispose of. Planning of a gradual extension of SFR is currently under way. The design capacity is based on current operating plans for the nuclear power reactors, but with a certain uncertainty interval that could be utilized if extra capacity is needed. If the extended operating times should be even longer, giving rise to increased space requirements in SFR, this extra space could probably be provided by further extension of the repository area.

An extended operating time also postpones the date when the reactor will be shut down and the decommissioning process can begin. If the operating time for an individual reactor is extended, this means that the need for deposition of decommissioning waste is postponed for this particular reactor, without affecting the planning for the other reactors. If the reactors' operating times are extended in general, there will be a general delay of decommissionings and the need for disposal volume for the decommissioning waste. This in turn leads to delayed closure of the final repositories. This could apply to the Spent Fuel Repository, the final repository for short-lived radioactive waste (SFR) and the final repository long-lived radioactive waste (SFL).

According to SKB's plans, SFL will be put into operation in around 2045. In the work with SFL, it is possible to make allowance for changed premises since the activities in question are so far in the future.

Conversely, a shortening of the estimated operating times for certain reactors would instead entail reduced production of operational waste and spent nuclear fuel and therefore lead to a reduced space requirement in the repository system. All facilities for management and disposal of nuclear waste and spent nuclear fuel will be needed, however. The number of deposition positions in the Spent Fuel Repository can then be reduced. If SFR has already been built out to its full size in accordance with today's forecast volumes, a shortened operating time of the nuclear power reactors will probably mean that the facility will not be fully utilized.

A shortened operating time further entails that the total nuclear waste programme can be concluded earlier. If the reactors' operating times are shortened in general, decommissioning and disposal activities will have to commence earlier. SFR can then be closed earlier than currently planned. The scope of these reschedulings depends on how many reactors are affected and how great the reductions of the operating times are.

The operating scenarios on which the planning is based also include assumptions regarding the future power level in the reactors, the fuel's burnup and future modernizations of the nuclear power plants. SKB's current plans are based on the changes currently planned by the nuclear power companies. This entails an increased future disposal need, which is now being incorporated in the relevant plans. Input data for future planning is obtained every year from the nuclear power companies. Any further power increases may lead to an increased disposal need.

3.2 New nuclear power reactors

Given the possibility of generation changes in the nuclear power stock, today's operating scenario may be significantly altered. The total volume of waste may be much greater than has been assumed. It is not possible today to predict the size of the waste volumes that would result from generation changes. This depends above all on how future economic considerations affect the willingness to invest in new nuclear power. The construction of a new reactor can have a similar effect on the planning as was reported above for extended operating times. Replacing today's reactors with new reactors will more likely lead to a need for a completely new nuclear waste programme, possibly with a partially new nuclear waste system.

Current plans call for today's ten reactors to be taken out of service during the period 2025–2045. If new reactors are instead phased in at that time, the new scenario entails a total operating time of up to 130 years, assuming the new reactors are utilized for 60 years.

It can be assumed that the new reactors will belong to what is usually called the third generation of nuclear reactors. They are in many respects different and more technically advanced than today's reactors. But it is still a question of boiling or pressurized water reactors with the same fundamental technology as today's reactors, and they produce the same kinds of waste products. The relative composition of the waste may be different than today's. The radionuclide content of the spent nuclear fuel depends on, for example, what type of fuel is used, the operating conditions and the fuel's decay period. The composition of the fuel and the distribution between short- and long-lived radionuclides is determined by the fuel's burnup and the so called specific power.

The volume of radioactive operational waste will probably be lower per kilowatt-hour of electricity produced than that produced by the first- and second-generation reactors. The quantity of operational waste from the Swedish reactors is already much smaller in relation to the electricity produced than had been expected when the reactors were put into operation. The reason is that the waste is treated and compacted so that it takes less space.

It should theoretically be possible to use the waste system currently under construction for the additional nuclear waste and spent nuclear fuel from the new reactors as well. Many of the system's facilities must undergo extensive renovation during the long operating time that is assumed, and the final repositories for spent nuclear fuel and other radioactive waste must be extended.

It is assumed that the spent nuclear fuel from the new reactors will be interim-stored for about 30 years before it is encapsulated and taken to the Spent Fuel Repository. This means that the existing interim storage facility, Clab, will be used for a period extending into the 22nd century. Provided that deposition of the spent nuclear fuel can be commenced during the 2020s as planned and can then proceed at a rate of about 150 canisters per year, Clab will also be able to be used for the spent fuel from the new reactors. Clab will naturally have to undergo extensive renovations during this long period (as is now being done with the pools in Clab's first rock cavern), but the interim storage facility should be able to be used in the future as well. The same applies to the encapsulation plant, which is planned to be integrated with Clab. It should also be possible to continue operating the planned canister factory without major changes.

Two examples of what the generation change at the Swedish reactors might look like and how it might affect SKB's facilities in the future are given below.

In the first example, the net capacity of the replacement reactors is assumed to be the same as that of today's reactors. For this a repository area of the same size as the one planned for the spent nuclear fuel from today's reactors is needed. It would probably then be most advantageous in terms of both safety and economy to site the new repository areas adjacent to the Spent Fuel Repository in Forsmark. The advantages lie in good knowledge of the bedrock around the repository, reducing the need for new investigations, and in the fact that surface facilities, ramp and shafts can be used for the additional repository area as well. In order to meet the increased deposition need, SKB can investigate the possibility of building an additional level in the Spent Fuel Repository. This permits the upper level to be closed after the first generation of nuclear power plants, at which point operation of the lower level commences.

Furthermore, this scenario entails an additional need of repository capacity for operational waste of roughly the same scope as SFR with the currently planned extension. The additional repository chambers that will be needed can be built adjacent to SFR, provided the bedrock is considered suitable, or as a separate facility on another site.

A need will also arise for an additional extension of the final repository for long-lived waste, SFL. The construction of SFL would need to be commenced largely in accordance with current plans to dispose of waste from today's reactors.

In the second example, it is assumed that today's reactors are replaced by new reactors with a much greater aggregate net capacity. This would permit greater electricity production, and therefore also production of more radioactive waste and spent nuclear fuel. If today's waste facilities were to be used, extensions will be necessary. For example, greater storage capacity would probably be needed in Clab, and with time increased capacity in the encapsulation plant. Naturally, larger repository areas than currently planned would also be required. It is possible they can be built adjacent to the currently planned Spent Fuel Repository, but further investigations are required to determine this. The alternative is to take a new final repository into service.

In general, this suggests that an expanded operating scenario, with a higher electricity production capacity than today's, would warrant greater changes in today's waste system than a more limited scenario. For example, construction of a new facility can take the place of an extension, if maintenance of the older facility can be assumed to become costly with time. Thorough evaluations based on good knowledge of the new planning premises are required to determine this.

3.3 Commissioning of the Spent Fuel Repository and Clink

SKB plans to commence trial operation of the Spent Fuel Repository and Clink in 2025. Clab currently has a licence to store 8,000 tonnes of fuel (counted as amount of uranium). According to today's forecasts, this amount will be reached in around 2023. This means that SKB needs to take steps to increase Clab's licensed storage capacity. At about the same time, all the storage positions will be filled. It is therefore also necessary to take steps to utilize the existing storage space more efficiently.

Two types of storage canisters are used in Clab for fuel: normal and compact storage canisters. Using only compact storage canisters would increase the physical storage capacity in Clab and delay the time when Clab is full until about 2029.

Besides spent nuclear fuel, long-lived nuclear waste is also stored in Clab in the form of core components and control rods. The control rods from the pressurized water reactors are integrated in the fuel, while the control rods from the boiling water reactors are stored in storage canisters that hold nine control rods. In the late 1990s, SKB investigated the possibility of compacting the BWR control rods. This would reduce the space needed by half, which would delay the time when Clab becomes full by two years. If the control rods are compacted and all spent fuel is stored in compact storage canisters, Clab will be full in around 2031.

Another alternative could be to find other solutions for interim storage of the core components and the control rods. SKB plans to start a project aimed at investigating this within the next three years. One possible solution is to switch to dry interim storage. If an alternative solution for interim storage of all core components and control rods can be found and if only compact storage canisters are used for the spent nuclear fuel, all storage positions in Clab will be filled by 2037.

If it should prove necessary, it is also possible to extend Clab with a third rock cavern with storage pools. But this would only be considered in the event of a very great delay of the Nuclear Fuel Programme. An extension would probably be done in a similar manner as the extension of Clab stage 2, which was put into operation in 2008. If Clab is extended by a third rock cavern, the facility could receive fuel for another 20 to 25 years.

3.4 Commissioning of the extended SFR

The extension of SFR is planned so that decommissioning waste from Barsebäck, Ågesta and Studsvik can begin to be deposited in 2020. To achieve this goal, SKB plans to submit applications at the end of 2013. Construction is planned to start at the beginning of 2017.

The storage volume available today in SFR is not sufficient to accommodate the short-lived waste from the decommissioning of Barsebäck without leading to a shortage of space for the operational waste. SKB has explained this in the supplement to RD&D Programme 2007. In its comments on the supplement, SSM has indicated that the background material presented by SKB is adequate and acceptable as a basis for the continued planning.

A delay of the commissioning of the extended SFR would entail a corresponding delay of the decommissioning of Barsebäck and Ågesta. But a delay would not have any negative radiological consequences.

The deposition of operational waste could also be affected by a delay of the extension of SFR. The repository part expected to be fully utilized first is BLA (rock cavern for low-level waste). In the event of a delay, the low-level operational waste may need to be interim-stored, for example on the power plant sites. The margins for other repository parts are relatively great.

Part II

The LILW programme

4 Plan of action

5 Management of short-lived low- and intermediate-level waste

6 Management of long-lived low- and intermediate-level waste

7 Responsibility, planning and technology for decommissioning and dismantling of nuclear facilities

4 Plan of action

This part of the RD&D Programme describes the planning for the LILW Programme, i.e. the programme for disposal of the low- and intermediate-level waste, including planning and technology for decommissioning of nuclear facilities. The management of very low-level waste is also described in the event it affects the plans for the LILW Programme. The planning is presented in general terms in Chapter 4. The conclusions from the plan of action in RD&D Programme 2007 are briefly recapitulated, along with the regulatory authorities' review comments, plus the conclusions from and review comments on the supplement to RD&D 2007. A general timetable for the LILW programme is also shown. Based on the timetable, the general planning is then presented with milestones. More detailed plans and programmes for management of short-lived and long-lived low- and intermediate-level waste and decommissioning of nuclear facilities are presented in Chapters 5 to 7. The scientific research being conducted within the framework of the LILW Programme is described in Part IV.

Conclusions in RD&D 2007 and the supplement to RD&D 2007 and their review

The supplement to RD&D 2007 presented the planned Swedish system for management of low- and intermediate-level waste and a general timetable for the entire life of the LILW Programme, where the connection to future decommissioning of the nuclear power reactors was illustrated. The overall planning for interim storage and final disposal of low- and intermediate-level waste was also described there, including decommissioning of the facilities. Important milestones for the LILW Programme during the coming decade were indicated, such as applications, notifications, site investigations and site selection.

In the supplement to RD&D 2007, SKB explained that the applications for an extension of SFR will include the entire additional extension, but that the plans are not final regarding whether the extension will be built in one or two stages. The extended repository is planned to be commissioned in 2020. SKB also mentioned the possibility of disposing of very low-level decommissioning waste in surface repositories instead of in SFR.

The long-lived waste arising from operation, modernization and decommissioning of the nuclear power plants is planned to be interim-stored until a final repository for long-lived waste, SFL, has been established. The supplement to RD&D 2007 mentioned both the existing rock cavern for waste on the Simpevarp Peninsula (BFA) and a future rock cavern in SFR as possible interim storage facilities.

Alternative dates for commissioning of SFL and general reasons for these alternatives were discussed. SKB intended to provide a better estimate of waste quantities and when the long-lived waste arises in RD&D Programme 2010.

In addition to its responsibility for transport, interim storage and final disposal of the radioactive waste from the nuclear power plants, SKB also has a number of agreements or declarations of intent with other actors such as Studsvik and SVAFO regarding management and disposal of their radioactive waste. SKB intends to re-examine and compile its pre-existing undertakings and update them where necessary.

SSM notes that SKB is discussing the possibility of extending SFR in one or two stages. SSM expects a slightly more in-depth analysis of the various alternatives in the account in RD&D Programme 2010. SSM expects a clearer account of the flexibility permitted by an extension in two stages, in view of the fact that the reference scenario for decommissioning and dismantling of the nuclear power plants may change with time. SSM expects the discussion of different alternative courses of action for management and disposal of the long-lived low- and intermediate-level waste to be continued in RD&D Programme 2010. In SSM's opinion, the justification presented so far by SKB for postponing the completion of SFL until 2045 is not sufficient. The reason for SSM's scepticism is that a considerable portion of the waste already exists or will be produced before 2045.

4.1 The LILW programme

The purpose of the LILW programme is to manage and dispose of the low- and intermediate-level waste from the Swedish nuclear power plants in a safe manner and to coordinate planning and technology matters for decommissioning. Even though the main purpose of the LILW programme is to manage and dispose of low- and intermediate-level waste, the programme also covers issues associated with the very low-level waste. The very low-level waste is mainly managed today by the nuclear power companies. The radioactive waste that is managed within the LILW programme is defined in Section 1.1.3. A description of the Swedish system for management and disposal of nuclear waste and the facilities that exist and are planned for managing low- and intermediate-level waste is provided in Section 1.2.2.

4.2 Planning

Figure 4-1 shows the overall timetable and milestones for the LILW programme. The dashed bars indicate both uncertainties and flexibility in the planning. The more detailed planning extends over a period of about ten years and is presented in Chapters 5 to 7.

The milestones in Figure 4-1 are divided into two categories: applications, notifications and statutory reports, plus important milestones for construction and commissioning of planned facilities.

In order to illustrate the connection between the LILW programme's and the nuclear power companies' plans, the nuclear power companies' current planning for decommissioning of the reactors is also shown in the figure. SKB's and the nuclear power companies' planning is based on operation of the reactors in Forsmark and Ringhals for 50 years and of the reactors in Oskarshamn for 60 years. Decommissioning of a unit is not begun until adjacent units with common buildings and/or systems have been taken out of service. Decommissioning can begin at the earliest about two years after a unit has been finally shut down, and dismantling of radioactive parts is estimated to take five years.

SKB's overall planning for management and disposal of low- and intermediate-level waste is presented in the following sections.

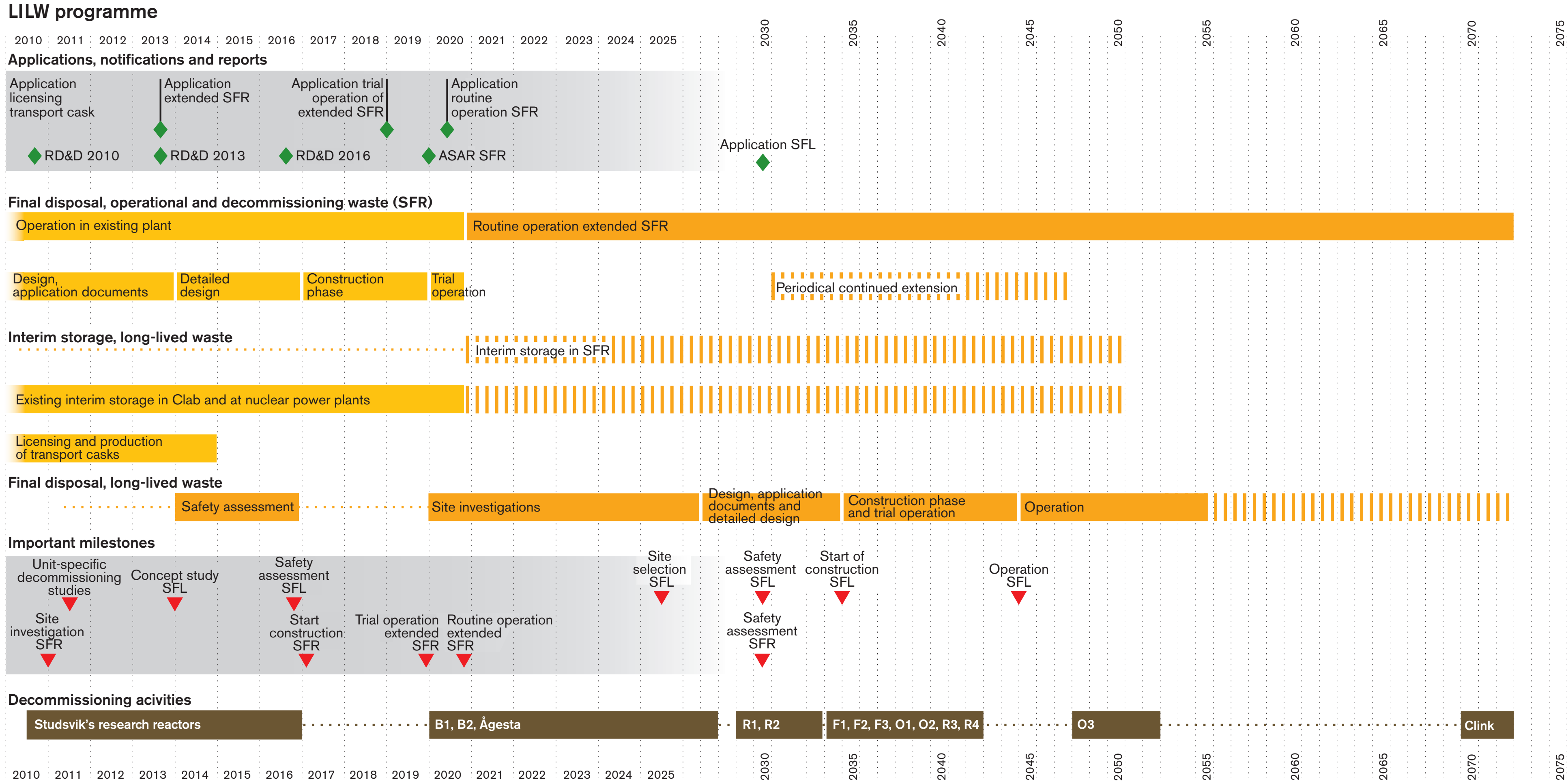
4.2.1 Planning for short-lived operational and decommissioning waste

SKB is planning to dispose of radioactive waste arising during operation and decommissioning of the Swedish nuclear power plants, including the reactor in Ågesta and the research reactors at Studsvik, as well as waste from SKB's own activities. In addition, SKB is planning to dispose of short-lived radioactive waste from medical care, research and industry. Before this waste can be disposed of in SFR, it must be conditioned and packaged in an appropriate manner. There are treatment plants in Studsvik where this is done. SKB has also undertaken to dispose of historic waste from SVAFO.

SKB is planning to extend SFR and the extension is being designed to accommodate additional short-lived low- and intermediate-level operational waste resulting from longer operating times of the nuclear power plants, plus short-lived decommissioning waste. Furthermore, we are planning for possible interim storage of long-lived waste from the nuclear power plants in SFR.

We are planning to carry out the extension of SFR in stages so that there is always room available for decommissioning waste from the nuclear power plants in accordance with the nuclear power companies' decommissioning plans. SKB judges that sufficient time margins exist, from the time a reactor is shut down until decommissioning waste is delivered for disposal, to allow extension of SFR with the corresponding disposal volume. This judgement assumes that no further licensing is required than that which is currently planned.

The advantage of extension in stages is that it reduces the risk that the repository will be overbuilt. The estimates of the waste volumes arising from decommissioning are uncertain, as are the estimates of the waste volumes that may be deposited in surface repositories instead of in SFR. By extending SFR in stages, SKB can gain experience from more decommissioning projects and the operation of the first stage, and gain a better idea of the waste volume that can be deposited in surface repositories, before the repository is finished. Before a decision is made on whether to build in stages, legal and economic aspects must be studied as well as consequences of the operating activities in SFR.



Clab	Central interim storage facility for spent nuclear fuel	ASAR	As-operated Safety Analysis Report	B1	Barsebäck 1	F1	Forsmark 1	O1	Oskarshamn 1	R1	Ringhals 1
Clink	Facility for central interim storage and encapsulation of spent nuclear fuel	SFL	Final repository for low- and intermediate-level long-lived waste	B2	Barsebäck 2	F2	Forsmark 2	O2	Oskarshamn 2	R2	Ringhals 2
RD&D	Research, development and demonstration	SFR	Final repository for radioactive operational waste			F3	Forsmark 3	O3	Oskarshamn 3	R3	Ringhals 3
										R4	Ringhals 4

Figure 4-1. *Timetable for the LILW programme. Hatched bars mark uncertainties or flexibility in the planning, for example the time for closure of SFL, as well as further extension of SFR.*

The first stage is planned to be in operation in 2020 and is designed to accommodate at least:

- Increased quantity of operational waste due to major rebuilds and extended planned operating time of the nuclear power plants.
- Decommissioning waste from Barsebäck 1 and Barsebäck 2, the research reactors at Studsvik, the Ågesta reactor and Ringhals 1 and 2.
- The possibility to dispose of large components such as intact reactor pressure vessels.
- Interim storage of long-lived waste from the nuclear power plants.

In conjunction with the applications to extend SFR, SKB is planning to apply for a licence to dispose of both operational and decommissioning waste in the entire facility. This will permit more optimal control of the waste streams in SFR. The applications for the extension will include the existing facility and the whole additional extension. The consequences of interim storage of long-lived waste for existing operating activities and technical requirements on equipment will be explored before we design the extension of SFR. In order to determine the consequences for operating activities, an event analysis and risk analysis for handling of BFA tanks in SFR will be performed. An application for interim storage of long-lived waste is planned to be submitted in conjunction with the applications for extension.

SKB will furthermore apply for a new radiation protection condition in conjunction with the applications for extension of SFR. The radiation protection condition limits the quantity of radioactivity and material that may be deposited in SFR and is currently based on waste forecasts for operational waste. The new radiation protection condition will include decommissioning waste and increased quantities of operational waste and will be based on the realistic waste forecasts that will be evaluated in the safety assessment done in preparation for the application for extension of SFR.

The extension of SFR is being preliminarily designed to include very low-level decommissioning waste. This waste may be emplaced in surface repositories instead of in SFR. SKB has conducted a feasibility study to shed light on the possibilities and consequences of disposing of some of the decommissioning waste in surface repositories, but further study is needed before a decision can be made, see Section 5.5. The volume of decommissioning waste that may be deposited in surface repositories will be estimated based on unit-specific decommissioning studies completed during 2011. A decision to dispose of large volumes of low-level decommissioning waste in surface repositories would reduce the requisite disposal volume in SFR.

SFR is planned to be closed when Clink has been decommissioned, see Figure 4-1.

4.2.2 Planning for long-lived operational and decommissioning waste

A large portion of the long-lived waste from the nuclear power plants arises when they are dismantled, but long-lived waste also arises during operation when the reactors' internal components are replaced. Furthermore, Studsvik and SVAFO have long-lived low- and intermediate-level waste originating from industry, research and development. Some of this is older waste from the time of development of the Swedish nuclear power programme. SVAFO works regularly with supplementary waste characterization for the purpose of distinguishing and sorting waste to be disposed of in SFL and SFR. An overview of the waste quantities and when the long-lived waste arises is provided in Section 6.1.

Interim storage

Today long-lived low- and intermediate-level waste is interim-stored at the nuclear power plants, in Clab and in facilities at Studsvik. Long-lived operational waste from Barsebäck may also continue to be interim-stored at Clab, but the possibility of interim storage in BFA or on the Barsebäck plant site is being explored.

In the ongoing project for extension of SFR, the technical feasibility and economic and radiological defensibility of establishing an interim storage facility for long-lived waste in the extended SFR are being studied. The decision regarding interim storage in SFR will be made before the design of the extension is finalized. SKB's plan is to interim-store long-lived waste in SFR and eventually transfer the long-lived waste that is currently being interim-stored on the nuclear power plant sites to the extended SFR.

A new waste transport cask, called ATB 1T, will be developed to transport the long-lived low- and intermediate-level waste. Delivery of the waste transport cask is expected in 2015. Equipment for handling of the waste at the nuclear power plants was put into service at the Forsmark nuclear power plant in the spring of 2010. In addition, SKB intends to relicense an existing transport cask (ATB 8K) for transporting residual waste from, for example, segmentation of the long-lived waste.

Interim storage of control rods takes place today in Clab. We will investigate different alternatives for interim storage of control rods during the coming three-year period. The purpose is to determine whether we can increase the interim storage capacity for fuel in Clab.

Final repository

SKB plans to apply for a licence to build a final repository for long-lived waste in around 2030 and to commission it in around 2045.

Before we can submit applications, several important milestones must be passed, such as selection of repository concept and site, investigations, assessment of long-term safety, preparation of applications, etc. A realistic expectation is that we can submit an application around 2030, provided we can find a repository site about which SKB previously has good knowledge. The time from applications to commissioning of the repository is more uncertain, since it involves activities in a relatively distant future.

With start of construction after 2030, most of the power plants' long-lived waste will be interim-stored. SKB does not regard extension of SFL in stages as an option, since there is only a limited volume of long-lived waste. The long-lived waste is estimated to correspond to a disposal volume of about 10,000 m^3, based on the assumptions in Section 6.1. In view of this, SKB finds that interim storage of all long-lived waste is more economical than hastening the extension of SFL after 2030, which would entail a long operating period.

We also judge that there will be radiological advantages to commissioning the repository in accordance with SKB's timetable. For example, it is an advantage if the waste has decayed for a long time before it is handled or reconditioned.

Closure of SFL is planned when all interim-stored long-lived waste and the long-lived waste from decommissioning of the last nuclear power plant has been deposited there. Before closure, SKB must ensure that the waste from decommissioning of Clink is suitable for SFR and does not have to be disposed of in SFL. In Figure 4-1, this uncertainty in the timetable has been illustrated with hatched bars.

4.3 Milestones

This section contains a brief description of the milestones shown in the timetable for the LILW Programme, Figure 4-1. The subheadings represent either one milestone or several combined milestones. Milestones that are closer in time are described in greater detail than those further off in time.

4.3.1 Milestones for short-lived operational and decommissioning waste

Site investigations for extension of SFR

The site investigations in Forsmark in preparation for the extension of SFR, with test drilling and other investigations of the rock with associated analyses, were begun in 2008 and are planned to be concluded during the first half of 2011. The test drilling has been concluded and the results are being analyzed. Preliminary assessments show that there are rock volumes in the investigation area that are suitable for an extension of SFR.

Application for extension of SFR under the Nuclear Activities Act

According to the Nuclear Activities Act, a Government licence is required to extend SFR. The application will cover the entire operation, i.e. the existing facility and all stages of the extension. In connection with the application for extension, SKB plans to apply for a licence to dispose of both operational and decommissioning waste in the entire facility. In the application, SKB will present the technical supporting material that is required to determine whether the existing and extended facility meets the requirements made under the Nuclear Activities Act and the Radiation Protection Act with associated regulations and ordinances. In the application, SKB will propose that conditions for future expansion stages be established by SSM and the County Administrative Board.

It has not yet been determined which supporting documentation will be appended to the application, but a preliminary safety analysis report (PSAR) of pre-closure (operational) safety and post-closure (long-term) safety will be included. The application will also include an environmental impact statement.

If long-lived waste is to be interim-stored in SFR, an application for interim storage will be made in conjunction with the application for extension.

Application for extension of SFR under the Environmental Code

Prior to extension, SKB will apply for a permit from the environmental court under Chapter 9 of the Environmental Code (environmentally hazardous activities) for the entire SFR facility. Under the current terms of the Environmental Code, it is probably not possible to license the extension alone ("add-on licence"); rather, the whole SFR will probably have to be relicensed. The extension of SFR will also require a licence under Chapter 11 of the Environmental Code (water operations). A prerequisite for granting a permit under the Environmental Code is that the activity in question does not conflict with the relevant detailed development plan or area regulations.

The application will cover the entire operation, i.e. the existing facility and all stages of the extension. In the application, SKB will propose that conditions for future expansion stages be established by SSM and the County Administrative Board.

If long-lived waste is to be interim-stored in SFR, an application for interim storage will be made in conjunction with the application for extension.

Start of construction for extension of SFR

When all necessary licences have been obtained, the extension of SFR can begin. SKB must consider the consequences of the conditions in the licences before the start of construction and adapt the planning accordingly.

Licence application for trial operation and routine operation of extended SFR

When the facility has been built and systems and processes function as intended, SKB will submit an application for a licence to commence trial operation. The application will contain an updated safety analysis report (SAR) with updated assessments of pre-closure safety and post-closure safety, plus updated safety-related technical specifications. The purpose of trial operation is to gather experience in preparation for routine operation. The safety analysis report and safety-related technical specifications will be supplemented and included as supporting material for the application to begin routine operation of the facility.

4.3.2 Milestones for long-lived operational and decommissioning waste

Application for licensing of transport cask ATB 1T

At the start of 2012, SKB plans to apply to French authorities for licensing of a transport cask (ATB 1T) for transport of BFA tanks containing long-lived low- and intermediate-level waste. Validation of this licence will be performed by SSM.

Development of handling equipment for core components

Common equipment for the nuclear power plants for handling of core components has been delivered to FKA and was commissioned in the spring of 2010. Work on inspection and design documentation is under way and will be finished during 2010.

Study and tendering work is under way for transport containers for contaminated handling equipment for core components. The containers do not need to be licensed and are planned to be delivered in 2011.

Interim storage of long-lived waste in SFR

We plan to commence interim storage in SFR of long-lived waste from the nuclear power plants when routine operation of the extended facility begins.

Concept study and safety assessment for SFL

An account of different repository concepts for SFL, including a qualitative assessment of their long-term safety function, will be presented in 2013. The goal of the study is to choose one or a couple of repository concepts to proceed with. Together with the results of other SFL work, the study will serve as a basis for continued efforts to compile supporting material for the safety assessment planned for 2016. A more detailed plan for the work leading up to this safety assessment is presented in Chapter 6.

Site selection for SFL

Based on the results of the assessment of long-term safety that is planned for 2016, preliminary requirements can be made on the site for the final repository. The continued research and safety assessment work will probably lead to modifications of these requirements before they are used to evaluate a candidate repository site. SKB's aim is primarily to evaluate the sites about which we already have good knowledge, and we intend to evaluate in a siting study whether it is warranted to conduct investigations on another site as well. Investigations of the properties of the rock, similar to those carried out for the Spent Fuel Repository and the extension of SFR, will be carried out on the site or sites identified in the siting study. After completed site investigations, the investigated site or sites will be evaluated, and only then will a decision be made on the siting of SFL.

Applications for construction, trial operation and routine operation of SFL

The process of establishing SFL will in many respects resemble the processes being planned and carried out in the Spent Fuel Project and the SFR Extension Project. Applications will be prepared under the Nuclear Activities Act and the Environmental Code. These applications will be preceded by a project that is reminiscent of the Spent Fuel Project and the SFR Extension Project and that includes site investigations, technology development, assessment of long-term safety, design, etc.

When all necessary licences have been obtained, the facility can be built. When the facility has been built and systems and processes function as intended, SKB will submit an application for a licence to commence trial operation. After the safety analysis report and the safety-related technical specifications have been supplemented, SKB will apply for a licence to commence routine operation of the facility.

4.3.3 Other important milestones

Unit-specific decommissioning studies

SKB is carrying out decommissioning studies together with the nuclear power companies to accumulate a more detailed body of data for estimating waste volumes, material quantities, activity quantities and decommissioning costs for the nuclear power plants. Studies will be conducted for each individual reactor unit and for each nuclear power plant. The studies will be adapted to the specific conditions prevailing at the site in question with regard to the physical layout of power plant buildings and with regard to the nuclear power companies' plans to decommission their own nuclear power plants.

The results of the studies will serve as a basis for designing the capacity of future repositories for decommissioning waste and for the safety assessments required in the licensing process.

The technical portions of the decommissioning studies with waste volumes and activity content are planned to be completed during the first quarter of 2011, and cost estimates for decommissioning and dismantling are planned to be presented in 2012.

As new knowledge is acquired in the decommissioning field, the decommissioning studies for the nuclear facilities will be updated. Decommissioning studies and decommissioning plans will thereby be progressively improved in an iterative process.

5 Management of short-lived low- and intermediate-level waste

This chapter presents SKB's plans and programme for extension and operation of SFR and describes the management of short-lived low- and intermediate-level operational and decommissioning waste. Section 5.1 deals with waste management, including a description of the technology development being conducted by the nuclear power companies at Clab and by Studsvik Nuclear AB and AB SVAFO. Section 5.2 gives an account of the operation of SFR and current and planned development work at the facility, while Section 5.3 describes the current situation and planned activities for the extension of SFR. Section 5.4 gives an account of the technology development that is planned for SFR, while Section 5.5 presents the work with surface repositories.

5.1 Waste management at waste suppliers

Conclusions in RD&D 2007 and the supplement to RD&D 2007 and their review

RD&D Programme 2007 included an overall description of the origin, quantities and types of the low- and intermediate-level waste.

SSM stressed that RD&D Programme 2010 should include an overview of the measures that need to be implemented in the final repository programme.

In order for the account in RD&D Programme 2010 to comply with Section 12 of the Nuclear Activities Act, a general description must also be provided of the measures that lie outside of SKB's undertaking vis-à-vis the power companies.

5.1.1 Origin and characterization of short-lived low- and intermediate-level waste

In order to be able to manage and store radioactive waste, it is necessary to categorize it based on the half-lives and activity contents of the nuclides. Based on the half-lives of the nuclides, the waste is categorized as either short-lived or long-lived radioactive waste, see Section 1.1.3. Short-lived waste contains a significant quantity of radionuclides with a half-life of no more than 31[3] years and only a limited quantity of radionuclides with a longer half-life. Long-lived waste contains significant quantities of radionuclides with long half-lives.

SKB has, together with the waste suppliers, worked out procedures for optimum management of the short-lived radioactive waste. Based on the waste suppliers' waste plans, deposition rules and acceptance criteria have been developed for waste to different parts of SFR, as described in Section 5.2.1. In order for a given waste type to be deposited in SFR, there must be an SSM-approved type description[4] for the waste.

Programme

In recent years the nuclear power plants have worked to minimize releases of radioactivity to the environs by evaporating floor drainage and process water with very low activity, which was previously released. Evaporating water containing radioactive substances gives rise to evaporator concentrates that must be disposed of. Different methods for treating this waste are being tested at the nuclear power plants, see Section 5.1.2.

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[3] Short-lived waste is defined according to the IAEA's "Radioactive Waste Management Glossary, 2003 Edition" as "waste that does not contain significant levels of radionuclides with half-lives greater than 30 years". SKB uses the same definition but with 31 years to include cesium-137, which is used as a key nuclide to estimate the content of other radionuclides.

[4] Type description – Refers to a safety report for the waste type in question. A type description establishes the packaging, waste category and treatment form for the waste.

In addition to sampling of carbon-14 and the research on carbon-14 presented in Part IV, a development project is under way to establish methods for measurement of nickel-63 in water from reactor and fuel pool water cleanup. One method being studied is chemical separation by means of ion exchange chromatography followed by measurement of activity for nickel-63. By measurement of nickel-63, nickel-59 can be indirectly determined. The sampling methodology follows the established procedure for sampling of transuranics and strontium-90 in the cleanup circuits at the nuclear power plants.

5.1.2 Conditioning of short-lived low- and intermediate-level waste

Before the short-lived low- and intermediate-level waste is transported to the final repository, it is treated at the nuclear power plants or at Studsvik's waste treatment unit. Besides packaging the waste, other purposes of waste treatment can be to reduce its volume, concentrate its activity or modify its physical and chemical properties.

The conventional method that is already established at the nuclear power plants for conditioning evaporator concentrates is solidification with bitumen or cement. Other wet waste is also treated by solidification with cement or bitumen or by dewatering. Solidification produces a waste form that is stable and more resistant to leaching. Scrap waste is treated by decontamination and/or melting, simplifying further handling and enabling the material to be cleared in some cases.

UV radiation can be used to break down organic compounds in the waste. This technique is used in the chemical decontamination method called CORD (Chemical Oxidation Reduction Decontamination), where an organic acid is used. To reduce the quantity of organic compounds in the decontamination solution, it is treated in a subsequent UV treatment step. Since the solution has to be disposed of, the reduction is favourable for the long-term safety of the repository.

Programme

The nuclear power plants have different development projects aimed at management of additional waste streams with evaporator concentrates and in some cases sludge.

At the nuclear power plants in Forsmark and Ringhals, two different methods are being tested for treating evaporator concentrates to minimize the quantity of organic compounds and salts in the conditioned waste packages.

At the nuclear power plant in Ringhals, a method is being tested for electrochemical degradation of organic compounds in evaporator concentrates. The residue after electrochemical treatment of the waste streams in question is planned to be conditioned with concrete. The goal is that the waste can be deposited in SFR after this process.

Treatment of evaporator concentrates by plasma arc combustion is being tested at the nuclear power plant in Forsmark. In this process, organic compounds are broken down and salts are vitrified at the high temperature. The residue is a vitrified waste, which would have little impact with respect to salt and complexing agents in a final repository. The project is in its final phase, and a decision regarding introduction and trial operation will be made shortly.

At the initiative of OKG and Studsvik, a project has been initiated for the purpose of developing a method for volume reduction of components with a large disposal volume in relation to their material content, mainly steam separators. By compacting or fragmenting the components, more material can be packed into each package, reducing the space requirement in the repositories. The results have so far been successful, and a decision has been made to carry out a broader study aimed at investigating whether, and if so how, this method can be developed and applied to other components besides steam separators.

5.2 SFR – final repository for short-lived radioactive waste

Conclusions in RD&D 2007 and the supplement to RD&D 2007 and their review

In its decision regarding RD&D Programme 2007, the Government stipulated that SKB should supplement the account in RD&D Programme 2007 with clearer plans and programmes for extension and operation of SFR plus a preliminary account of disposal of operational and decommissioning waste in SFR. The account was to be designed so that it would enable SSM to determine whether the account in RD&D Programme 2010 would be of sufficient scope.

The Government also stipulated that the account in RD&D Programme 2007 should be supplemented with a document describing, in quantitative terms, the possibilities and difficulties of disposing of decommissioning waste in the existing SFR.

In the supplement to RD&D Programme 2007, SKB described the possibilities and difficulties of commencing disposal of decommissioning waste in the existing SFR at different points in time. SSM judged that the account was sufficiently documented and acceptable, which means that the main purpose of this part of the supplement to RD&D Programme 2007 has been fulfilled. SSM based its conclusion primarily on the judgement that the waste quantities are too large for an optimal utilization of the existing SFR.

SSM stressed that RD&D Programme 2010 should include an overview of the measures that need to be implemented in the final repository programme.

5.2.1 Operation of the facility

The final repository for short-lived radioactive waste is a hard rock facility in Forsmark with a rock cover of about 60 metres that is reached via access tunnels from the ground surface. The repository is divided into different repository parts, designed with a view to the waste to be emplaced in them, see Figure 5-1. SFR has been in operation since 1988 and has been operated by SKB since 1 July 2009.

The rules for distribution of the waste between the different repository parts in SFR are based on the principles of BAT (Best Available Techniques) and ALARA (As Low As Reasonably Achievable).

The overall performance requirements on waste intended for disposal in SFR are:

- The waste packages shall not give rise to unacceptable dispersion of radionuclides.
- The waste packages shall be able to be handled without unacceptable impact of ionizing radiation on personnel and the general public.

It is above all the origin of the waste that determines in which repository part it is to be deposited. There is an established management system within SKB with procedures for examination of waste data and type descriptions and for handling of waste to be deposited in SFR. Deposition rules and acceptance criteria for waste intended for different parts of the repository are found in SKB's waste manual for low- and intermediate-level waste, and are shown in Table 5-1.

Routing of the waste streams to SFR is based on communication between operating personnel and persons with knowledge of the repository's long-term safety. There is an established forum for experience feedback, the LILW Group, which consists of representatives from the nuclear power plants, Studsvik, SVAFO, SFR, Clab and the department for low- and intermediate-level waste at SKB. The purpose of the forum is to provide information and study questions pertaining to low- and intermediate-level waste, as well as to plan the coming year's waste shipments. The LILW Group meets twice a year.

Programme

Optimization of the emplacement of different wastes in SFR should eventually include both operational and decommissioning waste and be based on the best available knowledge on the activity content of the waste and other properties that can influence the routing of the waste to different repository parts. A plan for how SFR, with its planned extension, can best be utilized will be prepared when the applications for an extension of SFR are submitted.

Table 5-1. Acceptance criteria for waste to SFR. The table is based on the deposition plans of the waste suppliers and SKB.

Waste type	Distribution of the waste between the repository parts			
	BLA	**BTF**	**BMA**	**Silo**
	Overall dose rate requirements for the different repository parts			
	< 2 mSv/h	**< 10 mSv/h**	**< 100 mSv/h (20% > 30 mSv/h)**	**< 500 mSv/h**
Ion exchange resins and filter aids from BWR reactors	After individual examination in exceptional cases	Systems 332, 342 Low-level packages after individual assessment	Systems 332, 342 After individual assessment systems 324, 331	Systems 331, 324, less active systems also permissible
Ion exchange resins and filter aids from PWR reactors and different systems	After individual examination in exceptional cases	Low-level concrete moulds after individual assessment	Systems 417, 330, 342, 334, 324, 336 after individual assessment	Systems 417, 330, 334, 336, 337, 342, 324
Ion exchange resins and filter aids from Clab	–	–	–	Systems 313, 324, 371, 372
Sludge, dried sediment	After assessment dried sediment	Sludge in concrete tanks	Dried sediment in trash and scrap moulds	
Evaporator concentrates from BWR and PWR reactors	–	–	Bottoms deriving from different drainage systems and laundry water	
Ion exchange resins, filter aids and sludge from Studsvik	–	–	After individual consideration, sludge from water purification	After individual consideration, older waste from Ågesta, R2, etc.
Trash and scrap from BWRs	Normally trash and scrap from intermediate building, waste building, turbine and generator building, after assessment waste from reactor building, sorted by dose rate criteria and measured nuclide-specifically	–	Normally trash and scrap from reactor building and turbine, sorted by dose rate criteria and measured nuclide-specifically	Scrap with high dose rates and high nuclide content after assessment
Trash and scrap from PWRs	Normally trash from reactor building, sorted by dose rate criteria and measured nuclide-specifically	–	Normally trash and scrap from reactor building, sorted by dose rate criteria and measured nuclide-specifically	Scrap with high dose rates and high nuclide content after assessment
Trash and scrap from Clab	–	–	Normally trash and scrap	Scrap and filter cartridges
Ashes from Studsvik	–	Ashes from incineration of trash	Ashes from incineration of trash	–
Trash and scrap from Studsvik	Low-level trash R2, HCL, ACL, waste facility, decommissioned facilities, hospitals, institutions, ABB ATOM (Westinghouse Electric Sweden), nuclear power plants	–	Intermediate-level scrap R2, HCL, ACL, waste facility, decommissioned facilities, hospitals, institutions, ABB ATOM (Westinghouse Electric Sweden), nuclear power plants	Smoke detectors containing americium-241

Explanations: 313- Cooling and cleanup system for receiving pools, 324- Cooling and cleanup system for pools, 330- General for chemical and cleanup system, 331- Cleanup system for reactor water, 332- Condensate cleanup system with precoat filter, 334- Chemical and volume control system, 336- Chemical sampling system, 337- Bottom blowdown system, 342- System for liquid radioactive waste, 371- Cleanup of process water, 372- Cleanup of floor drainage water, 417- Bottom blowdown system. R2- Reactor facility R2 in Studsvik, HCL- Hot cell laboratory, ACL- Active central laboratory.

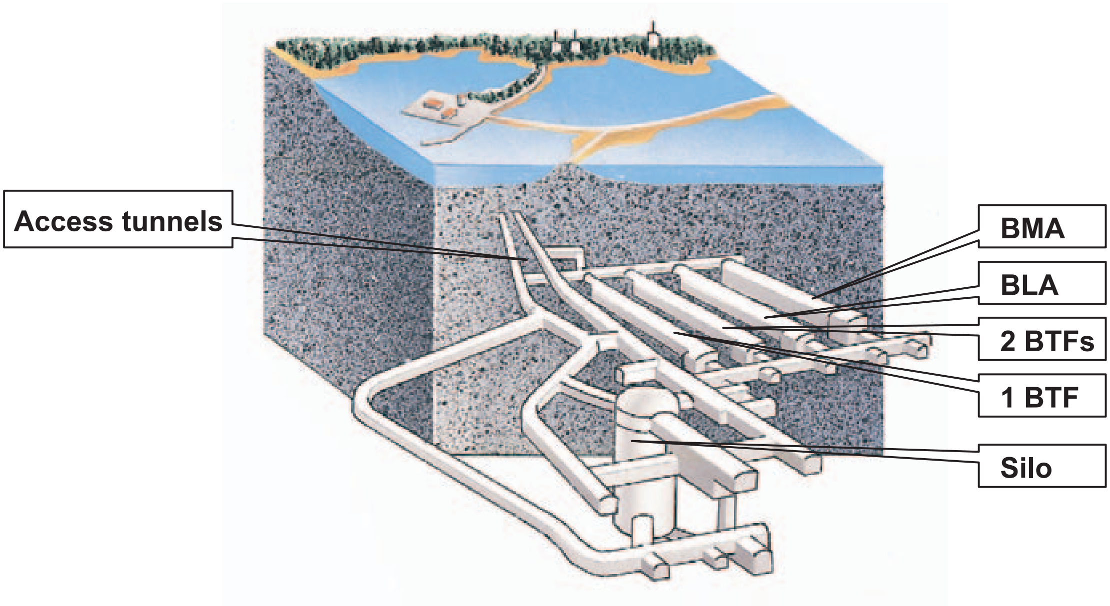


Figure 5-1. Overview of existing SFR facility with access tunnels and the different repository parts marked: Silo and BMA for the intermediate-level waste with the most radioactivity, the BTF repositories for intermediate-level waste with lower activity levels and BLA for low-level waste. (BMA – Rock vault for intermediate-level waste, BTF – Rock vault for concrete tanks, BLA – Rock vault for low-level waste.)

5.2.2 Maintenance

When SFR was built the intention was that the facility would receive short-lived low- and intermediate-level waste up to and including 2010.

Now the nuclear power plants are planned to be operated for a much longer time than was planned when SFR was built. As a result, SFR's operating phase will last longer than originally planned. An increased operating time imposes new demands on maintenance of the facility.

Safety, regardless of the length of the operating phase, is based on the fundamental design of the facility, the use of robust and proven systems and components, and a maintenance programme which, in addition to remedial and preventive maintenance, includes a programme for identification, handling and prevention of age-related deterioration and damage.

Before SKB took over the operation of SFR, maintenance was controlled by procedures developed by FKA. When we assumed responsibility for operation ourselves, we switched to using our own operation and maintenance system called GDU (Swedish acronym for common operation and maintenance system), which was already being used at Clab. GDU is the system OKG uses.

Programme

A study has been initiated to go through the entire SFR facility systematically to determine the status of both systems and buildings. The study will generate suggestions for measures that need to be adopted in the next few years or before the extension is commenced, as well as measures that can be implemented at later stages to enable the facility to be operated for a long time. The study will be concluded during 2011.

5.2.3 Information management regarding waste in connection with interim storage and deposition

The nuclear power plants and Studsvik have systems for registration and reporting of waste packages. These systems are based on SSM's requirements on registration and reporting of radioactive waste.

SKB has its own waste database, Triumf, whose main function is to administer information about content, properties and location of the waste packages in SFR. For waste packages there is information on, for example, the number of waste packages stored at the nuclear power plants and the future annual production of waste packages per waste type. The database also contains information on the waste transport containers and the SFR facility. The information in Triumf is transferred from the waste suppliers' databases by sending an encrypted file containing waste data to Triumf. The data transfer takes place before the waste is shipped to SFR, and all data must be approved by SKB before a shipment can take place. When the shipment is completed and the waste has been deposited, data on where individual packages have been deposited in SFR are entered in the database.

The information in Triumf is also used for reporting to regulatory authorities, planning of shipments and deposition, and issuance of forecasts for future deposition.

Programme

The Triumf waste database is being updated. The report and forecast module is being supplemented with new calculation methods and report production is being further automated. The storage places at SFR are at the same time getting new views, facilitating the deposition work, and a history function is being developed for better traceability.

During the coming year, SKB will consider the possibility of integrating data on future decommissioning waste in the waste database to facilitate the production of forecasts.

Radioactive waste generated at, for example, the nuclear power plants is registered today in local databases at each facility. A common waste database is being developed at SKB for the nuclear power plants, SKB and Studsvik. Studsvik will have access to the database via an external interface.

The common database is intended to manage all information on low- and intermediate-level waste that will be deposited in SFR and in a future SFL, and that arises in connection with the operation of the nuclear facilities in Barsebäck, Forsmark, Oskarshamn and Ringhals, as well as at Clab and SFR. SKB is in charge of administering the database. All stakeholders will be able to enter data on the waste packages in the database when the packages are created at a given facility.

5.3 Construction of final repository for short-lived radioactive waste

SKB's plans for the extension of SFR are related in this section. The planned extension entails an increase of the facility's storage capacity by an estimated 140,000 m^3 from today's capacity of 63,000 m^3. New preliminary studies of the waste volumes that will arise when the nuclear power plants are decommissioned indicate that the extension can be made smaller. When these studies have been completed, they will provide new premises for the extension.

The account covers the main phases up to and including routine operation. The purpose is to provide a complete picture of the current situation in the planning to permit a better overview of the applications of the results from research, development and demonstration in different phases. The focus of the work is the preparation of application documents.

Aside from the planned compulsory consultations pursuant to the Environmental Code, consultation meetings are also being held with SSM. Results, status and plans are presented in detail as they become available within the framework of these consultations. In this way we continuously obtain valuable viewpoints, which can be taken into consideration in the continued work.

Conclusions in RD&D 2007 and the supplement to RD&D 2007 and their review

In the supplement to RD&D Programme 2007, SKB wrote that a more detailed picture of the programme for extension of SFR will be provided in RD&D Programme 2010, with main phases and timetable. We stated that the planned extension will be designed to accommodate all waste from operation and decommissioning of the nuclear power plants, the Ågesta reactor and the research reactors in Studsvik. The applications will include the entire additional extension, but the plans were not final regarding whether the extension will consist of one or two stages.

SSM found that SKB's account in the supplement, regarding plans and programme for SFR, was unclear compared with other accounts submitted to the Authority. SSM stressed that the account must be sufficiently comprehensive for the RD&D Programme to fill its purpose, i.e. it must include an overview of the measures that need to be implemented in the final repository programme. SSM therefore found that the account needs to be elaborated and concretized in RD&D Programme 2010, for example as regards design and capacity of the extension and execution of site investigations, siting and plans for construction.

SKB mentioned the possibility of also using an extended SFR for interim storage of long-lived low- and intermediate-level waste. SSM has in principle no objections to this, provided that the interim storage meets the relevant requirements and that future licences under the Nuclear Activities Act and the Environmental Code permit this.

5.3.1 Main phases and timetable

SKB's overall timetable for execution of the extension of SFR is shown in Figure 5-2.

Preparation of application documents, 2008–2013

The SFR Extension Project commenced in 2008 for the purpose of preparing application documents under the Environmental Code and the Nuclear Activities Act and a construction plan for the extension. The project extends up to and including 2013, when the application documents are planned to be submitted to the environmental court and SSM for preparation for the Government. This phase includes site investigations, design of the facility, assessment of long-term safety, preparation of a preliminary safety analysis report for the operating phase, consultations on and preparation of an environmental impact statement, compilation of application documents, and part of the activity for detailed design and building preparations, including preparation of system documents.

Licensing, 2014–2016

The licensing process starts when the applications under the Environmental Code and the Nuclear Activities Act have been submitted, which is expected to occur in 2013. The applications will then be processed and reviewed by the regulatory authorities, the environmental court, the municipality and the Government. During this period the initiative largely lies with these bodies, and the length of the process depends on how long they take for processing and decision. SKB's main tasks during this time are to participate in the licensing process in various ways and to prepare the work of executing the extension of SFR. If comments of such a nature are received that an updating of the submitted PSAR (preliminary safety analysis report) is warranted before the extension commences, such an updating will be done during this main phase.

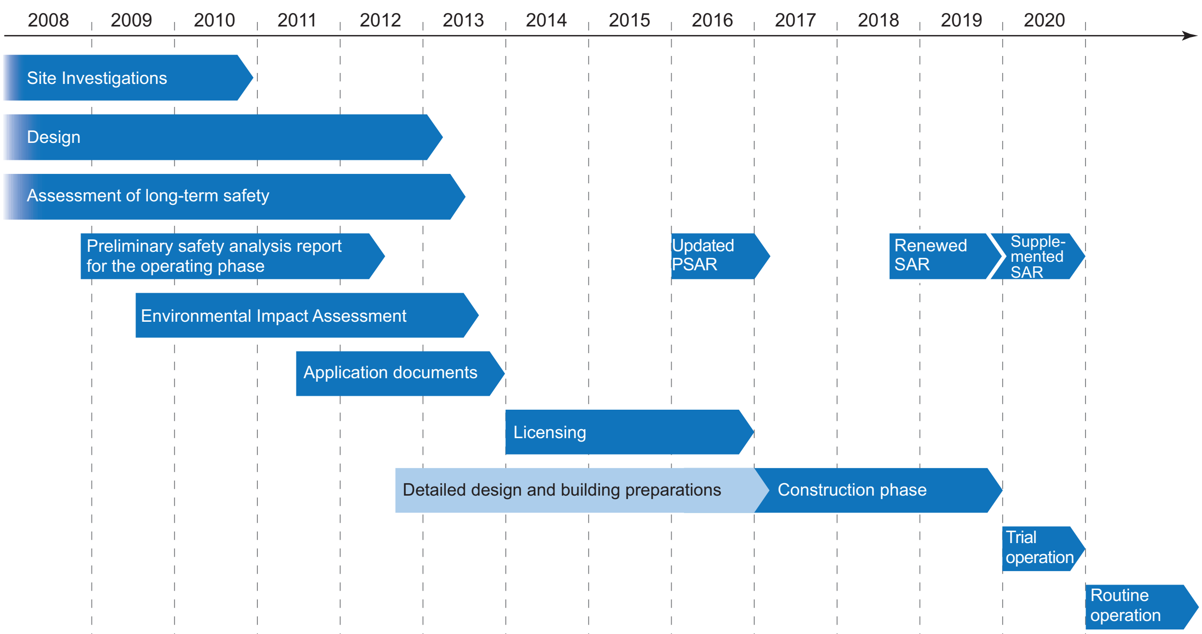


Figure 5-2. Overall timetable for extension of SFR.

Detailed design and building preparations are proceeding in parallel with the licensing process. Provided the licensing process proceeds smoothly, the necessary licences are expected to be obtained so that construction can commence at the beginning of 2017.

Construction and handover to operating organization, 2017–2020

The main phase "construction and handover to operating organization" includes the activities construction, trial operation and handover to routine operation. Before trial operation begins, an updated SAR is produced, which will be supplemented with experience from trial operation before routine operation ensues.

Extension of SFR includes construction, verification of requirements and design premises plus commissioning. The extension is regarded as several successive alterations of the existing SFR. Each alteration that is made of SFR shall be carried out according to the relevant project model for facility alterations, which, among other things, controls commissioning of the facility alteration. Depending on what the work processes look like at the time, the project model for facility alterations in effect at that time will be used.

The extension of SFR may not affect the safety of the existing SFR in an unacceptable manner. To guarantee this, SKB will employ currently applicable systems for safety management with defined operations management levels, which today entails routine monitoring and follow-up of the decisions that are made in safety matters. Each matter for facility alteration will undergo safety review in accordance with the procedures that apply at SKB for primary and independent safety review.

According to the project model for facility alterations, the process is concluded with a handover to the operating organization. The project will then be evaluated by all involved parties, such as purchaser and operations management, by compilation of a final report. When the final report has been approved by the project purchaser and all complaints against the operations management have been resolved, the project is concluded. The decision to conclude the project is then made by the project's purchaser.

5.3.2 Superordinate requirements on the facility

The existing SFR facility has two main safety functions: limitation and retardation. Limitation means that the quantity of permissible radioactivity in the repository is limited. Retardation means that the transport of radionuclides from the waste to the biosphere should be delayed sufficiently long so that they do not have radiological consequences. SFR thus has no absolute isolating function. The same basic safety functions apply for the extension of SFR as for the existing SFR.

As far as is economically reasonable, the facility should be designed so that the risk of radioactivity release and doses to personnel is minimized. Taking into consideration the different types of waste, the disposal chambers should be designed so that they provide the desired barrier function, in interaction with the waste packages. Furthermore, the barrier system and the rest of the facility should be designed in such a manner that it is possible to extend the repository in stages.

The extension of SFR should be designed to accommodate all additional short-lived low- and intermediate-level operational waste and all short-lived decommissioning waste expected to result from the dismantling of all nuclear power plants, including the Ågesta Reactor, the research reactors at Studsvik and Clink. The waste volumes in the facilities at Studsvik – including waste from hospitals, research and industry – are also taken into account in designing the capacity of SFR.

In summary, the extension of SFR should be designed to accommodate:

- all operational waste arising during the currently planned operating time of the nuclear power plants (50 years for Ringhals and Forsmark, 60 years for Oskarshamn),
- all short-lived decommissioning waste from the existing nuclear power plants, including Ågesta,
- all short-lived operational waste from Clab and Clink,
- all short-lived decommissioning waste from Clink,
- the waste volumes and the activity contracted with Studsvik and SVAFO,
- final storage of large components including whole BWR reactor pressure vessels without highly neutron-activated internals.

Furthermore, the SFR Extension Project will investigate the possibility of interim storage of long-lived waste from the nuclear power plants.

Quantity of short-lived operational waste

The quantity of short-lived operational waste that is planned to be disposed of in SFR up to 2069 is calculated today to be about 65,000 m^3. The calculated volume is the sum of the volume deposited thus far (through 31 December 2009) plus the forecast volume based on information reported from suppliers of radioactive waste. The volumes are based on volumes contracted with Studsvik and SVAFO, 50 years' operation of the reactors in Forsmark and Ringhals, and 60 years for Oskarshamn. The operational waste from the encapsulation plant has been estimated to be equivalent to the waste volumes from Clab. Figure 5-3 shows the results of the most recent forecast of operational and decommissioning waste. The forecast for decommissioning waste is based on preliminary versions of decommissioning studies.

Quantity of short-lived decommissioning waste

Short-lived decommissioning waste includes waste from the decommissioning of the Swedish nuclear power plants, the Ågesta reactor, the research reactors in Studsvik, and Clink. According to preliminary estimates, the quantity of short-lived decommissioning waste is about 85,000 m^3. The estimates are based on preliminary decommissioning plans, existing decommissioning studies and indications of volumes from ongoing unit-specific decommissioning studies. When these decommissioning studies have been completed, they will provide new premises for the extension of SFR.

A new estimate of how much decommissioning waste the extension of SFR should be designed for will be compiled in 2011.

Uncertainties in waste volumes

The waste volume that will be disposed of in the future in SFR is dependent on a number of factors. The volume of operational waste is influenced by operating trends at the nuclear power plants and the technology development pursued by the NPPs for waste generation and conditioning, for example compaction of large components, see Section 5.1.

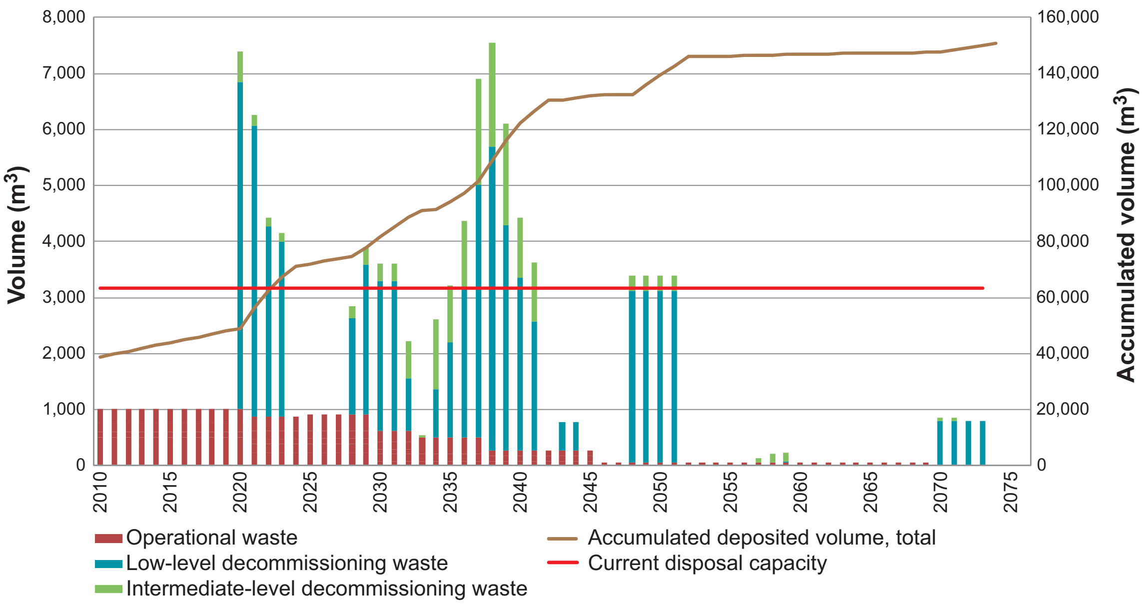


Figure 5-3. *Forecast for volumes of short-lived operational waste and decommissioning waste to SFR. The volume of waste that arises during a year is read on the left-hand axis. The brown curve shows the accumulated volume of waste and is read on the right-hand axis. Accumulated volume can be compared with current disposal capacity (red line).*

A source of great uncertainty is the possibility that some of the decommissioning waste will be released (cleared) from regulatory control or emplaced in a near-surface repository instead of in SFR, see Section 5.5. Based on existing knowledge of release requirements and requirements made on today's surface repositories, it is estimated that roughly one-third of the decommissioning waste can be cleared or emplaced in surface repositories.

Interim storage of long-lived waste

The extension of SFR will be designed for interim storage of long-lived waste from the nuclear power plants. A preliminary estimate of this volume is 2,600 m^3, which includes core components from operation and decommissioning of the Swedish NPPs. The extension will be designed with a view to the needs of the power companies and when SFL is planned to be ready to receive waste.

SKB is studying the technical demands made by interim storage of core components and PWR reactor pressure vessels on the extension of SFR. The studies include the following:

- Generic waste description for interim storage of core components in BFA tanks and PWR reactor pressure vessels.
- Design premises for interim storage of BFA tanks with core components and PWR reactor pressure vessels.
- Consequences for long-term safety.
- Consequences for operation and environs of an abnormal event in connection with underground transport, outloading and interim storage of core components and PWR reactor pressure vessels in SFR.

5.3.3 Siting

Siting a final repository for short-lived decommissioning waste adjacent to SFR appears logical for many reasons, such as the fact that the existing facility works well and the decommissioning waste is of a similar character to the operational waste. SKB has also studied other siting alternatives within the framework of the SFR Extension Project. None of these alternatives has been found to be better than an extension of SFR. In consultations regarding the planned activity, SKB will, in keeping with the requirements of the Environmental Code, take up the siting, scope, design and predicted environmental impact of the activity. The facility's operational safety will be studied and a safety assessment will be conducted to determine whether the selected site is suitable in the long term.

5.3.4 Feedback from site investigations

SKB is conducting site investigations for the purpose of studying the rock in the area in Forsmark being considered for an extension of SFR with regard to properties that are of relevance for the constructability and long-term safety of the facility. The goal was to investigate a rock volume large enough to accommodate the entire extension. The investigations were begun in 2008 and are planned to be concluded during the first half of 2011.

The planning for the extension of SFR started in 2007, when an investigation programme was published /5-1/. The investigation programme was preceded by a feasibility study. The feasibility study examined and systematized the large body of data from previous investigations in the area, both in connection with preliminary investigations and construction of the Forsmark NPP and the existing SFR facility, and in connection with SKB's site investigation for the Spent Fuel Repository. Based on this data review, a first version of a new site descriptive model for geology, hydrogeology and hydrogeochemistry was constructed. This model then served as a point of departure for the continued modelling work.

In conjunction with the site investigations for the Spent Fuel Repository, SKB has presented a series of investigation programmes /5-2, 5-3, 5-4/. These programmes have served as a basis for the investigation programme for the extension of SFR as regards investigation strategy and methodology. Experience feedback from the completed site investigation in Forsmark for the Spent Fuel Repository is also an important basis for the investigation programme, as are lessons gained from the investigations in connection with the construction of SFR in the 1980s. Finally, operating experience the SFR facility was an important support in the formulation of the investigation programme in general, and in particular those parts of the programme that have been conducted from the existing facility.

The choice of investigation strategy has mainly been determined by the following background data and premises:

- The extension of the SFR facility should be directly adjacent to the existing facility, providing geological and other conditions do not prove to be clearly unfavourable.
- Experience from the site investigations in the immediate vicinity of the SFR area will be taken into consideration.
- When executing the investigations, the necessary allowances must be made for the operation of the existing SFR facility.
- The site investigations must be executed within the available time.

Based on the initial version of the site descriptive model, an overall investigative logistics was established where priority was given to an area directly adjacent to the existing SFR facility. The priority area is situated southeast of the existing facility and was chosen due to the good rock quality in the lower construction tunnel in SFR and the fact that ground magnetic surveys do not indicate any major steeply dipping deformation zones in the area.

The investigation programme /5-1/ described detailed investigations in the priority area. A prerequisite for starting the investigations in the priority area was: if the initial site investigations indicate that the priority area does not satisfy the petrological and other requirements that must be met by the facility, the investigations will be moved to another investigation area.

Results from investigations

The site investigations started in April 2008 when the first investigation boreholes were drilled, and the last borehole was drilled from an island southeast of the repository in September 2009. Altogether, four percussion boreholes and eight cored boreholes have been drilled and investigated, see Figure 5-4. Approximately 3,000 metres of drill cores have been analyzed, and preliminary data indicate good rock quality in the area, consisting for the most part of metagranite and pegmatite with an average frequency of 3–4 open fractures per metre.

The investigations in the boreholes have included BIPS (Borehole image processing System), borehole radar, geophysical borehole logging (natural gamma, magnetic susceptibility, temperature, liquid resistivity, density etc.), difference flow logging (fracture transmissivity, electrical conductivity, flow direction, etc.), water chemistry sampling and pressure measurement. The continued work includes data analyses, modelling and hydromonitoring.

The quality work for the investigations is based on experience from the site investigations for the Spent Fuel Repository and has worked well in this project as well.

Experience has been gained from working in a nuclear facility, since activities have been carried out down in SFR. Among other things, a 300 metre long and nearly horizontal borehole has been drilled and investigated. The horizontal borehole investigations have also yielded valuable experience for future projects, and new equipment for moving measurement equipment along the borehole has been developed and tested with good results.

The priority area, which is located southeast of the existing facility, can serve as a suitable site for extension of SFR. This is based on the identification of a sufficiently large rock volume between major deformation zones. Further work remains to be done with evaluation and assessments of long-term safety in particular in order to be able to finally determine the exact location and design of the extension.

Modelling and site description

Modelling will be done in parallel and integrated with the investigations in the various disciplines, mainly geology, hydrogeology and hydrogeochemistry. In general, the modelling project can be said to comprise quality control of data, evaluation and analysis of primary data, three-dimensional modelling and reporting. The final result is a site descriptive model for the investigated area.

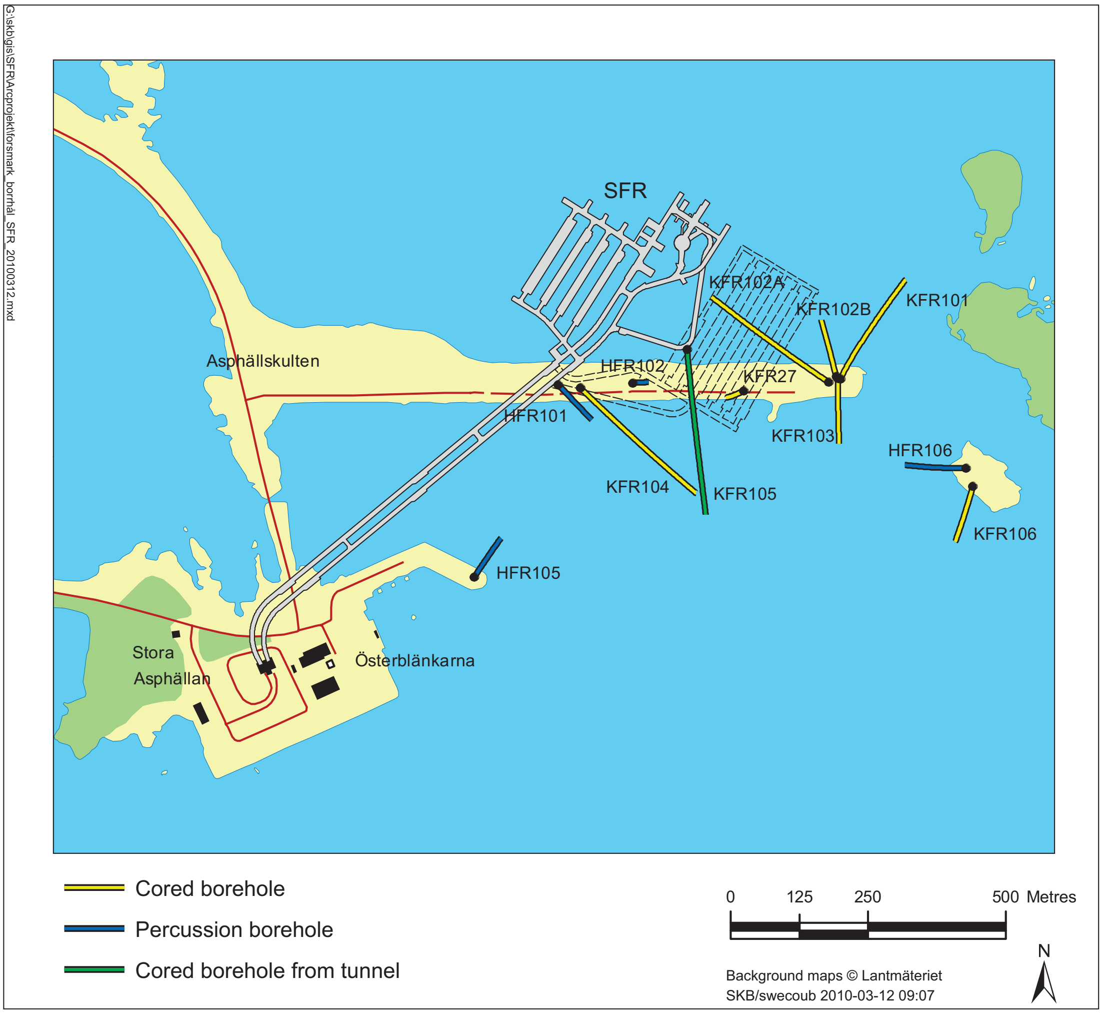


Figure 5-4. *Map showing borehole locations for investigations for extension of SFR. The figure shows an example of a preliminary layout for the extension's deposition area.*

The site description is needed in conjunction with the design work when the final repository is to be positioned in and configured on the investigated site. It also provides a basis for assessment of the long-term safety of the final repository. Furthermore, it can clarify the need for any additional data that needs to be collected. The site description is not only limited to describing the repository site, but also includes its regional environs to the extent this is necessary for the purpose. The geoscientific site descriptive model includes descriptions of the geological, hydrogeological and hydrogeochemical conditions on the site.

5.3.5 Work methodology for design

Facility parts

The facility parts can be divided into rock caverns under ground, buildings above ground and technical installations. The placement of buildings above ground is, for example, dependent on the location of the tunnel portal and possible additional ventilation. Information on the design premises that influence the number of underground rock caverns is provided in Section 5.3.2.

Stepwise execution

Up to the start of construction, design of the SFR extension takes place in two main phases: the system design and the detailed design. The system design is in turn broken down into a number of layout stages of gradually increasing maturity. In the detailed design, detailed design of the facility is carried out with the production of invitation to tender documents and construction documents.

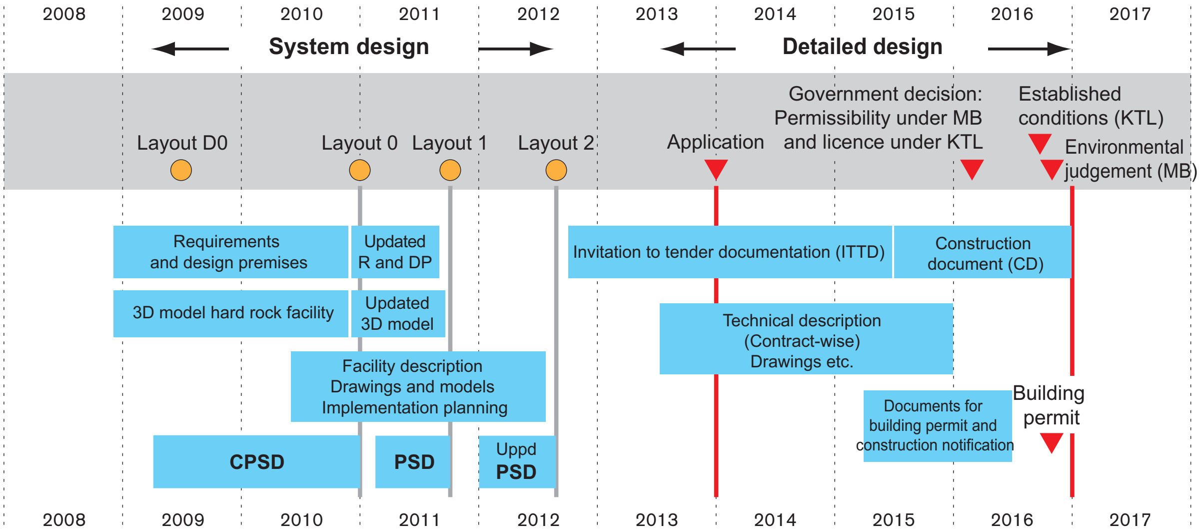


***Figure 5-5.** General structure for production of design basis documents. R and DP – Requirements and design premises. CPSD – Conceptual preliminary system descriptions. PSD – Preliminary system descriptions.*

Layout D0 of SFR was completed during the summer of 2009 and includes extension alternatives where the cross-section of the existing rock caverns comprises a reference design together with the then applicable forecast regarding volume of operational and decommissioning waste, Figure 5-6. The goal of layout D0 was to provide a basis for determining the site-specific configuration of the repository taking into account the geological and hydrogeological conditions on the site.

Layout 0 is a further refined version of layout D0 and is based on studies of preliminary requirements and design premises for installations and equipment, e.g. ventilation, power supply and fire safety and their possible impact on the layout of the rock caverns. During the work with layout 0, data will become available from site investigations and it will be possible to determine constructability and suitable position based on the geological and hydrogeological conditions in the rock mass. The hope is to be able to establish the location of the extension in layout 0, but if this is not possible the area within which the extension can be located must at least be clarified and established.

Based on requirements and design premises for layout 0, which include updated waste forecasts, a 3D model of a possible design of the facility will be constructed. This model will then serve as a basis for the hydrogeological modelling, providing the premises for the long-term safety assessment. Establishment of layout 0 entails that superordinate requirements and design premises are established and that conceptual preliminary system descriptions have been produced.

The purpose of layout 1 is to establish the requirements and design premises that are needed to produce a facility description. In conjunction with layout 1, the 3D model of the facility will also be updated. Preliminary system descriptions will be produced to serve as a basis for a technical description in accordance with the Environmental Code and Chapter 5, “Description of facility and function” in PSAR, General Part 1.

An iterative project involving design, site investigation and assessment long-term safety will be conducted up until mid-2012, before the final location and layout of the extension are confirmed and the system document phase is concluded with the establishment of layout 2. Subsequently, during the construction document phase, detailed design of the facility will be carried out with preparation of invitation to tender documents, detailed technical descriptions for each subcontract (excavation, heating, ventilation, plumbing, electricity, telecommunications, etc.), drawings, construction document and documents for an application for a building permit. Start of construction is planned for the beginning of 2017.

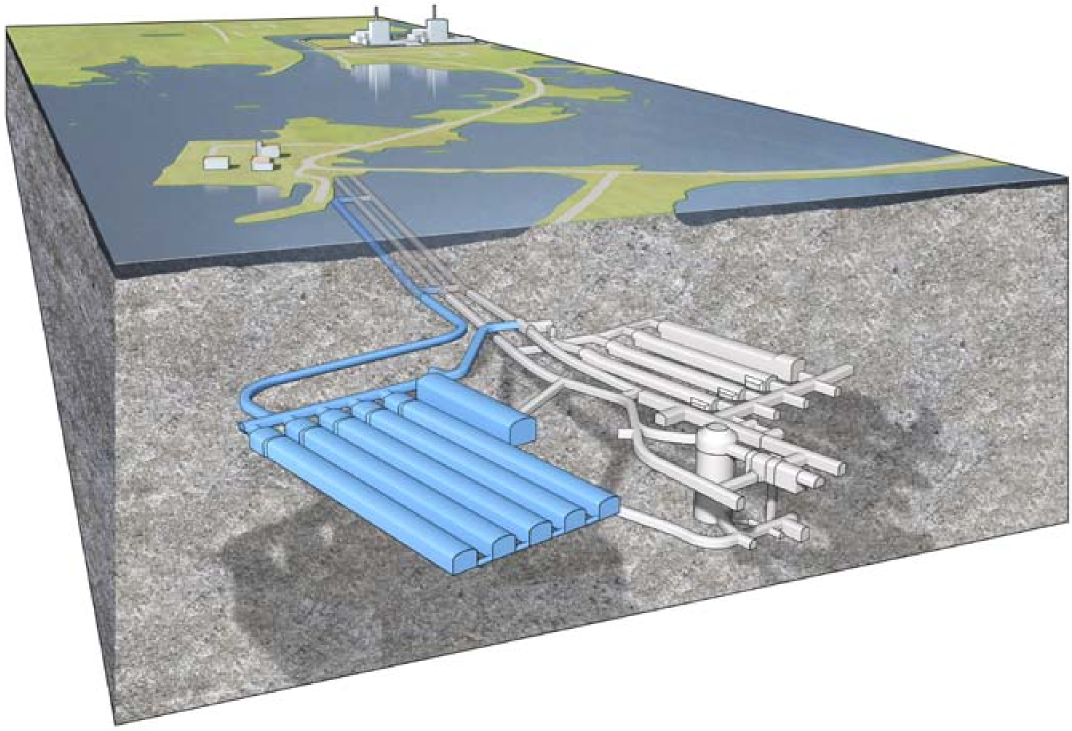


Figure 5-6. *SFR according to layout D0, existing part (light grey) and planned part (blue). Note that the final layout of the extension has not yet been established.*

5.3.6 Safety analysis report

Before a nuclear facility may be constructed and before major rebuilds or alterations of existing facilities are carried out, a preliminary safety analysis report must be compiled in accordance with SSM's regulation /5-5/. The preliminary safety analysis report for the extension of SFR will be based on the facility's existing safety analysis report and will include:

- information on the waste that is planned to be disposed of in the repository,
- information on the layout of the facility after the extension,
- information on planned mode of operation including operating limits,
- descriptions of the safety assessments and other verifying analyses that have been done of new, planned or altered parts or functions of the facility and of parts of the facility that have not been altered but are affected by the alterations,
- references to safety assessments and other verifying analyses.

5.4 Technology development for the final repository for short-lived radioactive waste

5.4.1 Barriers

The final layout of the extension of SFR, for example how many rock vaults are to be built and their placement in the rock, has not yet been established. Most of the waste that will be emplaced in an extended SFR will be low-level and is expected to be able to be disposed of in repository parts with barriers equivalent to today's BLA. A small quantity of waste will be intermediate-level and needs to be emplaced in a repository part with more advanced barriers.

Technology development is thus needed for repository parts for both low-level waste and intermediate-level waste.

Programme

How the barriers in the extended repository parts should be designed will be studied, and the function of the barriers in the existing facility will be evaluated.

The technology development for the low-level waste will be based on disposal chambers of the same type as BLA. Based on operating experience and experience from the most recent safety assessment, for example regarding the possibility of backfilling, changes of the barriers in the repository part and conditioning of the waste will be reconsidered.

For the intermediate-level waste, barriers are needed that limit the water flow through the waste while allowing gas that is formed to escape. Limited water flow through the waste can be achieved by:

- Hydraulic cage consisting of crushed rock and concrete structures, as in BMA.
- Bentonite and concrete structures, as in today's Silo.

The development work will be based on the knowledge and the experience we have of the different configurations of the disposal chambers in the existing SFR. Advantages and disadvantages of different concepts will be studied, and this will determine what development measures are needed.

5.4.2 Closure

The strategy for closure that was evaluated in the safety assessment from 2008 for SFR resembles the one used in previous assessments. It entails that the different disposal chambers are backfilled in different ways and to different extents: plugs of concrete and bentonite are emplaced in all tunnels that connect with the disposal chambers, plugs are emplaced on two levels in the access tunnels and other tunnel systems are backfilled. In its review of the 2008 safety assessment, SSM requested a more well-defined closure strategy.

Programme

SKB intends to develop the strategy for closure of the existing facility and to define a strategy for the extended part of the facility. The closure strategy for the two parts may coincide in many respects, but there is a possibility that different plans may be used for the existing SFR and the extension. The strategy should be sufficiently detailed that it can serve as a basis for the safety assessment that is prepared for the application to extend SFR.

Experience from the most recent safety assessment for SFR will be taken into consideration, as will the assessment for the Spent Fuel Repository.

A reference method for borehole closure has been developed for the Spent Fuel Repository, see Chapter 14. The method will be evaluated in reference to the need for borehole closure for SFR, and may be modified for the depth and the rock conditions that prevail for SFR.

5.5 Surface repositories

Conclusions in RD&D 2007 and the supplement to RD&D 2007 and their review

SSM finds that current conditions that apply to existing surface repositories in Sweden provide a well established regulatory framework for the nuclear power plant owners and SKB. Östhammar Municipality thinks that the need for surface repositories for decommissioning waste needs to be investigated.

Newfound knowledge since RD&D 2007

The issue of final disposal of very low-level and short-lived radioactive waste arises in conjunction with the planning for extension of SFR. The nuclear power companies' decommissioning plans include shallow land burial of very low-level and short-lived decommissioning waste as an alternative to final disposal in SFR. This category of waste represents a large volume, but contains only a relatively small portion of the activity that accompanies the radioactive waste from the nuclear power plants. Normally, very low-level waste is considered to be such that it can be disposed of in simple surface repositories of the type that exist at nuclear power plants and in Studsvik. The nuclear power plants and SVAFO are licensees for these surface repositories, and the conditions for disposal are stipulated in each case by SSM, since generally applicable regulations for shallow land burial are lacking. The regulation for release (clearance) of materials, buildings and land from regulatory control that is expected to enter into force in the near future can serve as a good planning basis when a near-surface repository is established.

For the planning of the extension of SFR, the nuclear power companies need to decide how the very low-level decommissioning waste is to be disposed of. In the current planning, SKB assumes that all material that is not cleared from regulatory control will be disposed of in SKB's facilities.

In view of SSM's work on regulations for clearance and shallow land burial of very low-level waste and emerging industry practice for clearance, estimates can be made of what waste quantities may be considered for shallow land burial. Together with the nuclear power companies, SKB has carried out a feasibility study to shed light on the possibilities and consequences of disposing of some of the decommissioning waste in surface repositories. The goal of the feasibility study is to present possible alternatives for shallow land burial of very low-level waste from decommissioning of nuclear power plants with schematic layout, estimated waste quantities and activity contents, siting and costs.

Programme

Based on ongoing decommissioning studies being conducted by SKB in cooperation with the nuclear power companies and proposed regulations for release (clearance) from regulatory control, an estimate will be made of both waste volumes and activity contents for different institutional monitoring periods.

The feasibility study that was completed during the first part of 2010 will be updated in 2011 when all unit-specific decommissioning studies are finished.

When the study is finished, an evaluation will be made together with the nuclear power companies for a common policy decision on how these volumes are to be managed in the LILW programme. If shallow land burial is the chosen alternative for very low-level and short-lived radioactive waste, the extension of SFR will have to be adapted and concrete plans made for shallow land burial.

6 Management of long-lived low- and intermediate-level waste

Of SKB's three planned repositories for nuclear waste – the final repository for short-lived radioactive waste (SFR), the final repository for long-lived radioactive waste (SFL) and the Spent Fuel Repository – SFL is the repository that is planned to be put into operation last. Before we can put SFL into operation, several important milestones must be passed, such as selection of repository concept and site, investigations, assessment of long-term safety, preparation of applications, construction, etc. SKB plans to apply for a construction licence for SFL in around 2030 and an operating licence in around 2045. SFL will be the smallest of the three final repositories, with an estimated disposal volume of about 10,000 m^3.

In this chapter, SKB relates its plans and programme for disposal of the long-lived low- and intermediate-level waste. In addition to an accounting of waste quantities and when they are expected to arise, the chapter also includes accounts of SKB's plans for choice of repository concept and barriers in SFL, interim storage and conditioning of the waste, and safety assessments. Furthermore, the research programme that will serve as a basis for the safety assessments is presented.

Conclusions in RD&D 2007 and the supplement to RD&D 2007 and their review

The plans for the work with SFL were described in RD&D Programme 2007 with reference to the fact that RD&D Programme 2010 focuses on the LILW Programme.

SKI requested that SKB should describe, in a supplement to RD&D Programme 2007, when the waste intended for SFL will arise, alternative repository designs (with design premises and safety functions), and the content and focus of future safety assessments that are needed to determine and verify acceptance criteria for the waste intended for SFL plus the content of the research and development programme that is needed to support future safety assessments.

SSI called for an integrated account and justification of the strategy for disposal of all long-lived waste from operation and decommissioning and a research and development programme for SFL that takes into account the need of protective capability, strategy for siting, acceptance criteria and guidelines for conditioning, as well as need of interim storage.

In the supplement to RD&D Programme 2007, SKB described the planning for the work with SFL in brief, while more detailed plans were promised in RD&D Programme 2010.

In the review of the supplement to RD&D Programme 2007, SSM mainly singled out four areas to which SKB should give greater consideration in RD&D Programme 2010:

- That SKB's justification of the timetable for the construction of SFL is based on all currently available knowledge of existing waste and of when additional long-lived waste will arise.
- That SKB should concretize its plans for the repository design work.
- That SKB's programme for both repository design and safety assessment of SFL should particularly focus on formulating acceptance criteria for waste.
- That SKB should concretize the research programme it intends to carry out in 2010–2013.

6.1 Long-lived radioactive waste

The long-lived low- and intermediate-level waste consists chiefly of four categories:

- Strongly neutron-irradiated core components. This waste arises in connection with both maintenance and decommissioning of reactors.
- Control rods from BWR-reactors.
- Long-lived waste from activities at Studsvik and from medical care, research and industry. This waste arises continuously and is not associated with the operation or decommissioning of the NPPs.

- Historic waste from research and development within the Swedish nuclear research programmes. This waste is managed and interim-stored by SVAFO.

Conclusions in RD&D 2007 and the supplement to RD&D 2007 and their review

In RD&D Programme 2007, SKB described which types of waste constitute the long-lived low- and intermediate-level waste and how this waste is managed today.

SSI stated in its review that SKB should show what waste quantities can be expected and when the waste will arise. In the review of the supplement to RD&D Programme 2007 as well, SSM says that SKB should present knowledge of existing waste and when additional long-lived waste will arise.

Newfound knowledge since RD&D 2007

The required disposal volume in SFL has been calculated to be 10,000 m^3. The calculation is based on the volume of existing interim-stored waste and a forecast of the future waste from operation and decommissioning of the nuclear power plants, see Figure 6-1. The forecast of the waste quantity and the required disposal volume for future waste is based on the following assumptions:

- All neutron-activated core components in BWR are replaced after 20–25 years of operation. For an operating period of the nuclear power plants of 50 years, one replacement has therefore been assumed, and for 60 years two replacements.
- The control rods in BWRs are replaced after 15–20 years, and after concluded operation of existing nuclear power plants the total number of control rods is calculated to be about 2,500.
- One set of core components from a BWR is estimated to weigh about 70,000 kg on average, which is equivalent to a disposal volume of about 100 m^3.
- No replacements of core components in PWRs.
- Control rods from PWRs are disposed of together with the spent nuclear fuel.
- The core components in a PWR are estimated to weigh 35,000 kg on average, which is equivalent to a disposal volume of about 50 m^3.
- All core components are fragmented and placed in BFA tanks, which have a disposal volume of 9.9 m^3 and a maximum payload capacity of 12 tonnes.
- The BFA tanks are loaded with an average of 6.6 tonnes of waste, which is equivalent to 1.1 tonnes per m^3 of inner volume.

Today's interim-stored waste, which consists mainly of historic waste, needs a disposal volume of about 6,000 m^3, see Table 6-1. This estimate differs from the reference inventory that was used in the safety assessment in 1999, when the historic waste was estimated at 1,800 m^3. The big difference between the estimates is largely explained by the fact that SVAFO is about to place the concrete-filled waste drums in larger drums to achieve safer handling and interim storage. Furthermore, waste that was previously judged to be able to be disposed of in SFR has been transferred to SFL.

The programme for the work with the long-lived low- and intermediate-level waste is described in Section 6.3.

Table 6-1. Weight of existing waste in different interim storage facilities and forecast for need of disposal volume in SFL (year 2009).

Interim storage facility	Weight of waste (kg)	Estimated need of disposal volume (m^3)
Barsebäck in pools	54,616	80
Ringhals in pools	12,330	20
Oskarshamn in pools	1,410	10
Oskarshamn in packages	87,600	420
Forsmark in pools	75,100	100
Clab in pools	135,633	200
SVAFO		4,630
Studsvik		450
Total	366,689	5,910

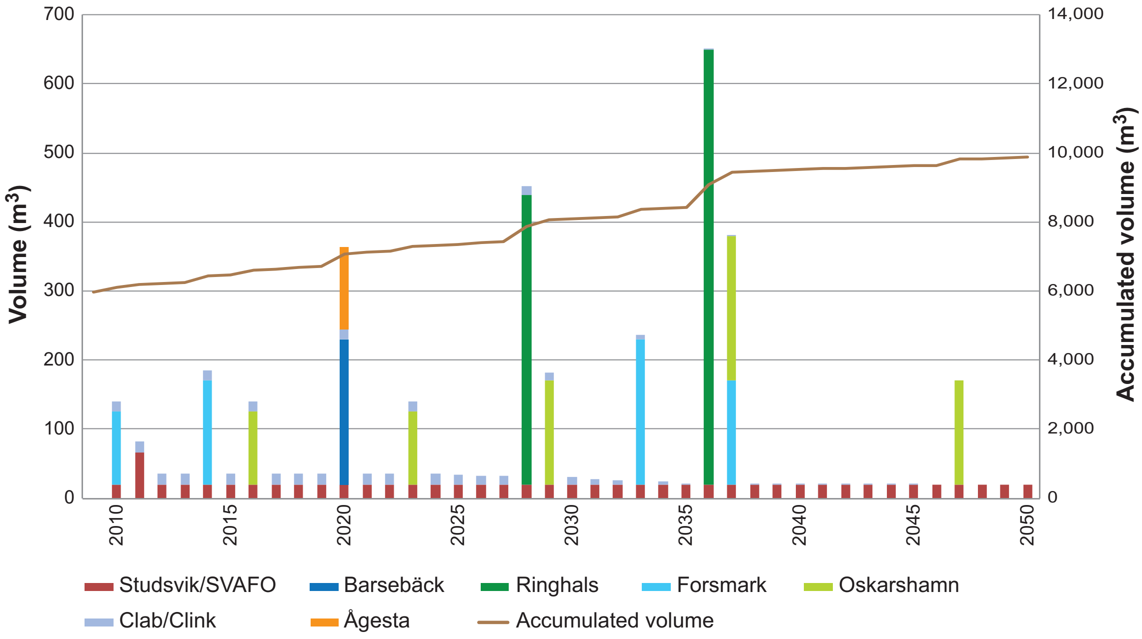


Figure 6-1. *Overall timetable showing when long-lived low- and intermediate-level waste arises. The bars show the volume of waste that arises during one year and are read on the left-hand axis. The curve shows the accumulated volume and is read on the right-hand axis. Information on the existing waste is shown in greater detail in Table 6-1.*

6.2 Interim storage facilities for long-lived waste

Long-lived waste is interim-stored today either in Clab's pools, in pools at the nuclear power plants or dry in containers in different interim storage facilities.

Historic waste is managed by SVAFO and interim-stored, along with waste from Studsvik's activities, in an interim storage facility on the Studsvik site.

Conclusions in RD&D 2007 and the supplement to RD&D 2007 and their review

RD&D Programme 2007 described plans for increased dry interim storage of core components in BFA, as well as Forsmark's plans to build its own interim storage facility.

SSI said in its review that SKB should show how the need for interim storage of the long-lived waste will be met.

The supplement to RD&D Programme 2007 described the plans for interim storage of long-lived waste in BFA, but also the possibilities of such storage in the future extension of SFR. SKB further described the advantages of the freedom of choice that is obtained by storing the waste without conditioning it irreversibly.

In the review of the supplement to RD&D Programme 2007, SSM writes that "the conditioning (irreversible or not) that is done before the acceptance criteria have been finally established should be done in a suitable manner with a view to safety and radiation protection in all future handling steps". SKB interprets this as meaning that SSM accepts SKB's strategy of continuing to store the long-lived low- and intermediate-level waste without final treatment until SFL has been put into operation.

Newfound knowledge since RD&D 2007

The total need for space in interim storage facilities is approximately 10,000 m^3, provided that all long-lived low- and intermediate-level waste is interim-stored before SFL is put into operation. Of this volume, approximately half is managed internally by Studsvik and SVAFO and therefore does not affect the nuclear power plants' planning.

The above data can be compared with existing interim storage facilities (not counting storage pools at the NPPs and in Clab), which have the following capacities:

- BFA – Rock cavern for waste (Oskarshamn) 13,500 m^3
- Interim storage facility at Forsmark 550 m^3
- Interim storage facility at Ringhals 9,200 m^3
- SVAFO and Studsvik interim storage facilities 2,000 m^3
- SVAFO and Studsvik rock caverns 5,000 m^3

The above volumes are not fully available for SFL waste today, since they are also being used for interim storage of other waste. SKB plans to apply for permission to interim-store long-lived waste in SFR. By interim-storing long-lived waste in SFR, we ensure that interim storage space will be available both before and after the NPPs have been decommissioned and the plant sites are being used for other purposes.

Ringhals' PWR reactor pressure vessels constitute a special case when it comes to need for interim storage space. The discussion in Section 6.1 presumes that these vessels are handled in fragmented condition placed in containers, but they may be handled in one piece in the future.

Programme

In order to secure management and interim storage of waste intended for SFL, SKB is conducting and plans to conduct a number of studies, as described below.

Study of interim storage of long-lived waste in the SFR extension

SKB has initiated a study that will serve as a basis for the planning of an interim storage facility for long-lived waste from the nuclear power plants in the extended SFR. The study deals with the premises for and design requirements on such an interim storage facility. The study will define requirements on lift height, radiation protection and in-transport. Furthermore, we need to ensure that we can open the BFA tanks after a period of interim storage. SKB needs to devise a plan of action for how a BFA tank that cannot easily be delidded should be opened.

Development of transport casks for BFA tanks

A type B transport packaging is required to transport long-lived waste placed in BFA tanks. Development of a cask (ATB 1T) for transport of one BFA tank at a time with a maximum surface dose rate of 200 mSv/h has therefore been initiated. The goal is that the tank should be licensed and ready to be put into operation by 2015. In connection with this work, acceptance criteria for waste in BFA tanks have been formulated and entered into SKB's waste manual.

Development of handling equipment for BFA tanks

SKB is pursuing efforts aimed at developing handling equipment for BFA tanks and the ATB 1T transport cask. An example is the equipment needed for packing of BFA tanks and transferring the BFA tank to the transport cask.

The majority of the components for the handling equipment have already been delivered to FKA and tested during January–February 2010. Work with inspection and design documentation is under way and will be finished during 2011.

Studies and tendering processes are under way for transport casks for contaminated handling equipment. After a decision is made on technology and financing, the casks can probably be delivered during 2011. The work in its entirety is expected to be completed in 2011.

6.3 Final repository for long-lived waste

SKB has divided the work with the final repository for long-lived waste (SFL) into two main periods: a short-term period that extends up to and including the first safety assessment in 2016 and a long-term one that extends from 2016 to 2045, when the facility is scheduled to be commissioned. An overall timetable for this work is shown in Figure 6-2. Several important milestones must be passed, such as selection of repository concept and site, investigations, evaluation of long-term safety, preparation of applications, etc. According to the timetable, we plan to apply for a licence to build the repository in 2030, which, taking into account uncertainties etc. is judged to be a realistic point in time, provided we can locate the repository on a site of which SKB previously has good knowledge. The time from applications to commissioning of the repository is more uncertain, since it involves activities in a relatively distant future.

6.3.1 Overall planning and accounts

This section describes the plans for the period 2011–2016 in detail and the long-term plans in general terms. The short-term planning is divided into two periods: 2011–2013 and 2014–2016, which are described separately below.

The period 2011–2013

The focus of the work with SFL during the period 2010–2013 is a concept study which will serve as a basis for the choice of the repository concept. The purpose of the concept study is to carry out an initial screening among possible repository concepts and to identify one or more repository concepts suitable for SFL. They can then be evaluated in the safety assessment that will be presented in 2016.

Furthermore, the groundwork needed for the presentation of a safety assessment in 2016 will be started. Examples of such work are updating of the reference inventory and planning and initiation of various research and development projects. Detailed descriptions of the planned work are provided in Sections 6.3.2, 6.3.3 and 6.3.4.

The period 2014–2016

During the period 2014–2016, the focus of the work with SFL will mainly be on the safety assessment (SA) to be presented in 2016.

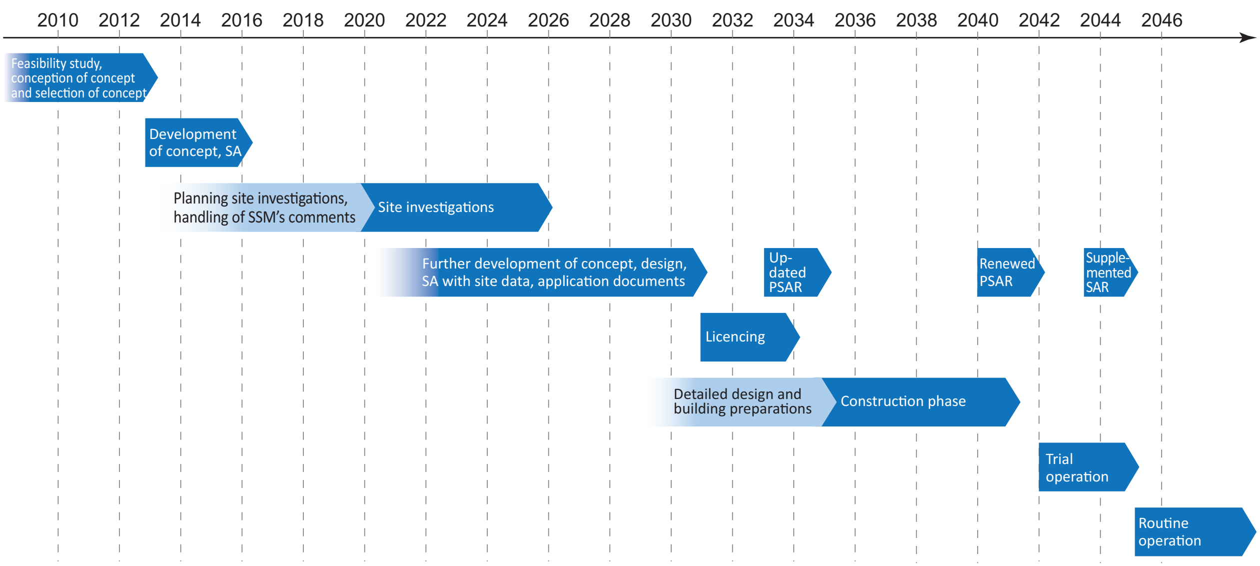


Figure 6-2. *Estimated timetable for the work leading up to the commissioning of SFL. The work is divided into different phases: The first phases comprise conception, selection and development of repository concept, execution of site investigations and preparation of application documents. They are followed by licensing, construction of the extension and handover of the facility to the operating organization.*

Important background material for the safety assessment includes the concept study presented in 2013, the updated reference inventory and the results of research projects carried out for SFL or other repositories. Site data will be obtained from the generic sites used in the previous safety assessment for SFL, or from the site investigations conducted within the Nuclear Fuel Programme.

It will not be possible to formulate acceptance criteria for conditioned waste intended for SFL until the repository concept has been chosen. The nuclear power plants should not commence final conditioning of waste until a verified repository concept exists. Acceptance criteria exist for the waste to be interim-stored, see Section 6.2.

The period 2016–2045

The plans for the SFL work after 2016 are presented schematically in Figure 6-2, where the main activities are marked.

Prior to the site investigations, a siting survey will be conducted. It is estimated that the site investigations can begin in around 2020. After that, design and analysis of the facility and assessment of the repository's long-term safety will be carried out, based on information from the selected site.

6.3.2 Updating of the reference inventory

The waste streams to SFL have changed during the past years, in part due to the decisions on extended operating times for both the nuclear power plants and SFR. SKB therefore plans to compile and present an updated reference inventory by 2013.

Programme

The updating of the reference inventory will be done in close cooperation with the nuclear power companies, SVAFO, Studsvik and other stakeholders. The following waste will be disposed of in SFL:

Historic waste

The historic waste is managed by SVAFO, which in 2009 initiated a supplementary characterization of the concrete-embedded waste that is expected to be finished in the spring of 2011. The purpose is to describe the interim-stored waste and estimate its content of radionuclides. The work also includes identifying those drums that contain large quantities of liquids or other materials that can cause problems in a final repository and to carry out calculations on how reconditioning of the waste will affect the disposal volume.

Waste from Studsvik's activities

Studsvik's activities annually produce waste equivalent to a disposal volume of about 20 m^3. The waste consists mainly of ash and sludge stored in steel drums. Information on this waste is stored in a database. SKB's primary task is to ensure that relevant information is collected and compiled so that it is available to SKB.

Waste from decommissioning of the nuclear power plants

Decommissioning studies are being conducted to characterize the waste from the decommissioning of the nuclear power plants with regard to materials, weight, disposal volume and radionuclide content. An estimate of the needed disposal volume and the points in time when the waste arises is shown in Figure 6-1. The decommissioning study for Barsebäck is finished, and the others are expected to be finished during 2011.

A detailed description of the plans for decommissioning and dismantling of the nuclear power plants is provided in Section 7.2.

Waste from operation and maintenance of the NPPs

An inventory will be made of the waste from operation and maintenance of the NPPs that is in interim storage today. Information on this waste is stored in databases at the NPPs and SKB. The primary task for SKB is to ensure that the information is correct and complete with regard to material composition and radionuclide content.

Furthermore, the forecasts of the waste volumes that will be generated by future maintenance work will be updated by clarifying the NPPs' plans for major maintenance. Forecasts will then be made of the needed disposal volume, and the nuclide inventory will be calculated and compiled.

6.3.3 Repository concept

A BMA-like repository at a depth of about 300 metres was assumed in the safety assessment for SFL presented by SKB in 1999 /6-1/. The core components were placed in long moulds of reinforced concrete with an inner steel cassette. The repository's engineered barriers consisted of concrete and porous concrete. The rock openings were backfilled with crushed rock.

The safety assessment showed that improved engineered barriers needed to be developed.

Programme

The work of identifying possible repository concepts for SFL begun in 2011. The work will for the most part follow the methodology previously used by SKB for the final repository for spent nuclear fuel /6-2/, which includes:

- identification of repository concepts,
- identification and choice of selection method and selection criteria,
- evaluation and choice of repository concepts,
- development and improvement of chosen repository concepts,
- safety assessments and selection of one repository concept.

The work with the repository concept includes choice of methods and technical solutions for:

- conditioning of waste,
- waste packages,
- barriers and engineering materials,
- closure.

The technical solutions, together with the choice of site and repository depth, comprise the complete repository concept. Following is an account of the criteria and properties which SKB must take into consideration in designing the different repository concepts. The repository concept that was evaluated in the safety assessment in 1999 will be included in the concept study and used as a reference to evaluate the long-term safety of the concepts.

The waste which SKB will deposit in SFL consists of materials with different properties. This means that despite its limited volume, the repository may consist of several disposal chambers based on different technical solutions.

Waste package and conditioning waste

Factors to be considered in the choice of waste package and conditioning method for the long-lived waste include the waste's origin, composition, nuclide content and radiation level, and whether it has already undergone some form of treatment or conditioning. In addition, consideration should also be given to the presence of complexing agents and other chemical substances that can affect the long-term safety of the repository.

Information on the waste varies. The historic waste is embedded in concrete today, but its composition and properties are not fully known. The metallic waste from the nuclear power plants will be well characterized. SKB foresees that completely different methods for final conditioning and different types of waste packages will be needed for the different waste types.

There are fewer degrees of freedom in the choice of final conditioning method for the historic waste than for the metallic decommissioning waste. One possible method is to place the containers in an external container to strengthen the properties of the packaging. This container does not necessarily have to be the same as the one described in Section 6.1. There are more degrees of freedom for the metallic waste. This waste will to a great extent be interim-stored in steel containers, BFA tanks, before it is to be deposited. Possible alternatives are therefore to let the waste remain in the tanks and deposit them, or to fill them with concrete before deposition. If the waste is interim-stored for a long time, this allows greater options for conditioning that could provide safety-related advantages during a final disposal phase.

Barriers and engineering materials

Together with conditioning method and waste package, the repository's engineered barriers and the surrounding rock provide protection against the escape of radionuclides. The choice of the repository's engineered barriers is made with consideration given to the waste and how it is conditioned as well as the containers in which the waste is placed.

A basic principle for SFL is that the repository's principal safety function is to retard releases of radionuclides, which is also true for SFR. This is mainly accomplished by the barriers' capacity to limit the transport of groundwater and its capacity to sorb radionuclides.

SFL will contain large quantities of metallic waste, and the release of radionuclides is therefore dependent on whether corrosion can occur. The ability of the barriers to limit this, for example by limiting the availability of water, is therefore of interest.

Other properties to take into consideration in the choice of barriers is that they should be mechanically and chemically stable over very long periods of time and be able to withstand freezing during a permafrost period.

Closure

SKB is about to initiate a study of methods and materials for closure of SFR, see Section 5.4.2. When this study has been completed it may determine how a closure of SFL should be designed. Furthermore, we have a reference design for closure of the Spent Fuel Repository which can also provide guidance.

6.3.4 Natural science research

A detailed body of scientific data will be needed for future safety assessments of SFL. SKB has conducted research and development for SFR and the Spent Fuel Repository over the past few decades. The accumulated body of knowledge will also be useful in the work with safety assessments for SFL.

The plans for the research with the clearest connection to SFL are described in this section. Other research that is being conducted or is planned under SKB's auspices and is relevant for SFL is described in Sections 20.2 and 21.2 and in Chapters 25 and 26.

Programme

A review of the state of knowledge in research areas of importance for SFL has been done during 2009 and 2010, along with a general review of possible technical concepts. The material will be used to make decisions regarding what research projects will be started during the coming three-year period. A number of research projects have been identified and some have already started, mainly in areas linked to the engineered barriers and the properties of the waste.

A regular inventory of the state of research is performed within SKB, leading to the initiation of new research projects aimed directly at SFL and carried out in cooperation with other projects within SKB or by participation in national or international research programmes.

Following is a description of some of the areas identified as important for SFL and within which one or more projects have been initiated during 2009 and 2010 which will last several years.

Age-related changes in cementitious materials

Cement is not a static and homogeneous material, but is altered over time by processes such as carbonatization and leaching-out of pH-enhancing hydroxides such as $NaOH$ and $CaOH_2$. These processes lead to alteration of the structure of the material and of its chemical and mechanical properties by the dissolution of certain phases and formation of others.

It is important that the ability of the cement and concrete structures to limit water flux remains intact as long as possible after closure of the repository in order to limit leaching-out of the structures, metal corrosion and radionuclide transport. The latter phenomena is furthermore dependent on the ability of the cement to sorb radionuclides and thereby contribute to the repository's release-retarding properties.

The chemical and mechanical properties of fresh cement are relatively well known today. The state of knowledge for aged cement, i.e. cement older than 100 years, is poorer, however.

A number of research projects have been initiated to study the processes that are involved in ageing and their effects on the chemical and mechanical properties of the cement. The main purpose of these studies is to gain greater knowledge in areas related to the ageing of cementitious materials and their interactions with surrounding materials, both by real-time investigations in a repository environment and by accelerated experiments in a laboratory environment, and finally by computer modelling.

Corrosion of metals in a repository environment

The rate of release of radionuclides from metallic waste, as well as the life of metallic waste containers and reinforced concrete, are largely dependent on the corrosion rate of the metals in question in a repository environment. The time after which the materials begin to corrode, and the rate, mechanisms and scope of the corrosion process are completely dependent on the metal in question and the properties of the ambient environment. For a more thorough description of the corrosion process, see Section 20.2.12.

Metal in the form of rebar, waste packaging and waste comprises a large portion of the material that will be present in SFL, which means that an understanding of the corrosion processes will be vital in future safety assessments. A project has therefore been initiated where different types of metals have been embedded in cement and placed in boreholes in a repository environment in the Äspö HRL. A first retrieval of this material for investigation of how it has been affected is planned in around 2020, and a last retrieval just before the laboratory is closed.

To supplement these studies, SKB plans to conduct similar investigations in a laboratory environment. There are also plans to conduct studies of the corrosion process in real time by measuring the rate of the ongoing corrosion process by different methods.

Degradation of organic waste in a cement environment

Chemical degradation of organic material in the waste, or in the matrix in which the waste is embedded, can generate products that can affect the repository's long-term safety, for example complexing agents. Organic material is expected to occur both in the historic waste and as an additive in cement and concrete. For a more detailed description of this process and its effects, see Section 20.2.10.

The degradation products can affect the rate at which radionuclides are transported out of the repository, and a good knowledge of what degradation products different organic materials can give rise to is therefore important for future safety assessments. This information is also important for restricting the waste's content materials that give rise to particularly critical degradation products.

A project has been initiated where different types of organic waste have been embedded in cement and placed in boreholes in a repository environment in the Äspö HRL. A first retrieval of this material for investigation of how it has been affected by this exposure is planned in around 2020, and a last retrieval just before the laboratory is closed.

To supplement these studies, SKB plans to conduct similar investigations in a laboratory environment. Before further experimentation is done, a more comprehensive review of the state of knowledge in this area will be done.

Gas permeability of concrete and cement

The repository's ability to release generated gases is important, since excessively gas-tight structures and materials can lead to the buildup of high pressures, which could threaten safety. For a more thorough description of gas transport, see Section 21.2.5.

Gas will be formed in the repository primarily due to anaerobic corrosion of metals, but some contribution is also expected from degradation of organic waste. Since these processes are expected to occur over a long period of time, SKB has initiated a project to study how gas permeability in cement develops in a long-term perspective, and which properties of the cement control this process. A feasibility study is expected to be finished in 2010, after which a decision will be made concerning the future course of the project.

7 Responsibility, planning and technology for decommissioning and dismantling of nuclear facilities

According to Section 12 of the Act (1984:3) on Nuclear Activities, the holder of a licence to own or operate a nuclear power reactor shall, in consultation with other reactor owners, prepare or have prepared a programme for the comprehensive research and development work and other measures specified in Section 10, paragraphs 2 and 3, and Section 11. This programme shall also include decommissioning and dismantling.

This chapter describes how responsibility for decommissioning and dismantling of the nuclear power reactors has been divided between the licensees and SKB. We also describe the plans for how the decommissioning of the reactors and SKB's own nuclear facilities are to be carried out. Furthermore, the manner in which cooperation is pursued nationally and internationally to enhance competence in the field of decommissioning is described.

Decommissioning of a nuclear facility comprises the activities that occur from when the facility has been permanently shut down until it has been released from regulatory control and exempted from conditions under the Nuclear Activities Act. After this there are no radiological or safety-related reasons that prevent the establishment of other activities on the site. The dismantling of a facility is a part of decommissioning and comprises the activities that occur from when dismantling begins until the entire facility has been released for unrestricted use.

Conclusions in RD&D 2007 and the supplement to RD&D 2007 and their review

SSM finds that the compilation of the nuclear power companies' decommissioning plans and SKB's planning needs to be elaborated and supplemented in RD&D Programme 2010. This also applies to the decommissioning plan for the Ågesta combined heat and power plant (CHP). RD&D Programme 2010 should include an account of what mandate has been given to SKB to plan and/or implement measures included in the nuclear power companies' obligations under the Nuclear Activities Act.

In its comments on the supplement to RD&D Programme 2007 and in anticipation of the work with RD&D Programme 2010, SSM noted the lack of a complete picture of the division of responsibilities between SKB and the licensees for the Swedish nuclear power reactors. The mandate that has been given to SKB by the licensees needs to be clarified. An account of responsibility for planning and execution of the component phases in the decommissioning process is needed.

When it comes to rules for release/clearance from regulatory control, SSM and SKB find that the nuclear power companies can refer to the EU's recommendations in the area, which SSM intends to incorporate in future clearance regulations.

At the coordination meetings for RD&D Programme 2010, SSM has requested a general description of the NPPs' plans for decommissioning, final disposal and clearance. Such a description should include what kind of waste arises and when, as well as how it is to be disposed of, along with an account of the flexibility of the plans in allowing for changes in the NPPs' plans. The connections to SKB's plans should also be described.

7.1 Division of responsibilities

According to the Act (1984:3) on Nuclear Activities, the licensees for the nuclear facilities in Sweden are responsible for safely decommission and dismantle the radioactive parts of the facilities. Decommissioning shall be described in plans where the degree of detail in the account increases as the time for decommissioning approaches. Furthermore, the Financing Act (2006:647) stipulates that a licensee shall calculate the estimated cost of decommissioning of NPPs.

The licensees – Barsebäck Kraft AB, Forsmarks Kraftgrupp AB, OKG Aktiebolag and Ringhals AB – are responsible for decommissioning and dismantling of the Swedish nuclear power reactors in

Barsebäck, Forsmark, Oskarshamn and Ringhals. Vattenfall AB is responsible for the Ågesta CHP, while SKB is responsible for its facilities: Clink, SFR, the Spent Fuel Repository and SFL.

The licensee is responsible for the waste until it has been cleared or until SSM has made a decision on closure of the Final repository, and the Government has granted discharge from responsibility under Section 10 of the Nuclear Activities Act.

Figure 7-1 shows an overview of regulatory requirements on decommissioning planning during the life of a nuclear power plant. Decommissioning begins after the facility has been shut down and lasts until the facility can be released (cleared) for unrestricted use. Decommissioning includes defuelling operation, shutdown operation and dismantling operation. Defuelling operation is the activity from final shutdown of the unit until all nuclear fuel has been removed from the unit and hauled away. This is followed by shutdown operation, which lasts until the dismantling process has begun. During dismantling operation, dismantlement and demolishment of the facility proceeds along with other activities required for clearance from regulatory control. After a decision by SSM on clearance, the Authority can release the facility from control under the Nuclear Activities Act.

According to the Environmental Code, an EIS must be submitted both before final shutdown of the facility and as a part of the application for a dismantling licence, see Figure 7-1.

In order to comply with SSMFS 2008:19, radiation protection aspects in connection with a future decommissioning must be taken into account when a facility is built. Sections 4-8 of the regulations regulate decommissioning planning, stating that a preliminary plan must exist. Not later than one year after final shutdown, the licensee must submit a general report describing and explaining goals and measures, and present a timetable for the decommissioning. Not later than four months before dismantling operation is begun, a detailed account shall be send to SSM describing how the work will be conducted. The account shall include how different systems are affected, organization, emergency plans, material quantities, treatment of the waste, management of residual radioactivity in soil or buildings, and quality and monitoring system.

According to Chapter nine of SSMFS 2008:1, the preliminary decommissioning plan shall be supplemented and kept up-to-date as long as the facility is in operation and shall be presented to SSM every ten years. Before dismantling operation may commence, the decommissioning plan must be incorporated in the safety analysis report for the facility. The safety analysis report shall undergo safety review and be reviewed and approved by SSM. The environmental impact statement which is submitted to the environmental court under the Ordinance on Environmental Impact Statements shall be appended to the safety analysis report.

Licensees are responsible for compliance with the regulatory requirements in accordance with Figure 7-1.

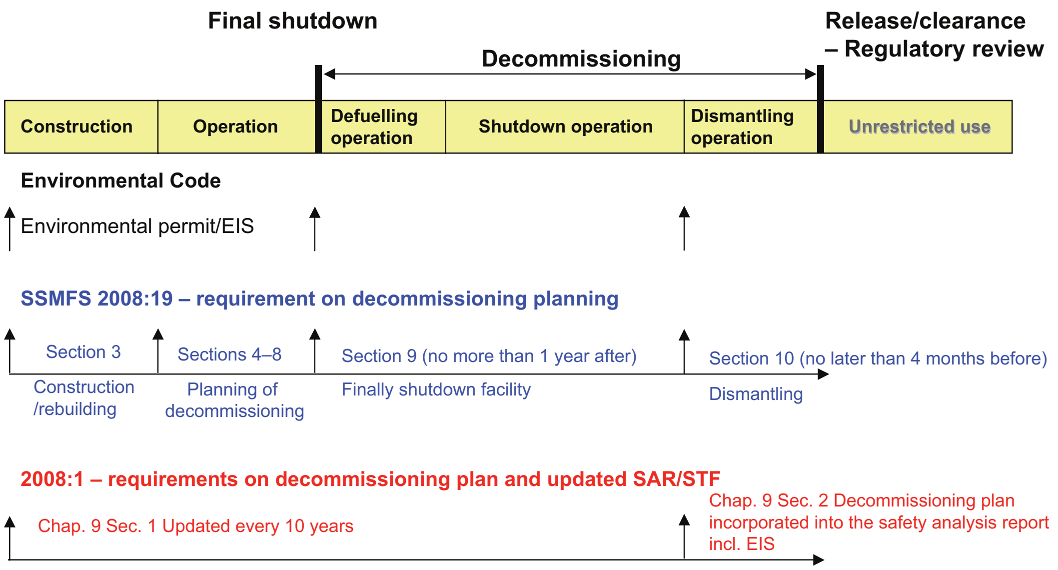


Figure 7-1. *Overview of regulatory requirements on decommissioning planning during the life of the nuclear power plant.*

7.1.1 Division of responsibilities between SKB and licensees of Swedish nuclear power reactors

Responsibility for a nuclear facility rests with the licensee, who is responsible for planning, licensing and dismantling of the facility, as well as disposing of the waste. This includes deciding which strategy and timetable should be employed in decommissioning and dismantling of the facility.

From a national viewpoint, there is a need to coordinate decommissioning and dismantling between the nuclear facilities. Coordination is also needed to ensure that the whole chain from decommissioning to final disposal is optimized. A decommissioning group with representatives from the nuclear power companies, Studsvik, SVAFO and SKB was formed at the start of the 21st century to meet the need for a forum to deal with decommissioning and dismantling issues. The purpose of the group is to focus on technology and logistics issues in connection with the decommissioning of nuclear facilities, for example choice of different technical solutions and treatment and management of the waste. In this work, the nuclear power companies have contracted SKB to participate in the planning and execution of future decommissionings of their NPPs and to operate final repositories for decommissioning waste. SKB's participation mainly involves developing general methods and procedures for the dismantling work, activity and volume estimates, and classification of waste. SKB is also responsible for making estimates of the decommissioning costs of the NPPs in Barsebäck, Oskarshamn, Forsmark and Ringhals.

SKB is responsible for managing the conditioned radioactive waste from decommissioning. Figure 7-2 illustrates the division of responsibilities for management of the decommissioning waste from decommissioning to final disposal via interim storage. SKB is in charge of managing the waste in its facilities and on the ship m/s Sigyn.

In order to dispose of the short-lived decommissioning waste, SKB is planning an extension of SFR, see Section 5.3. Long-lived waste that arises from the decommissioning of nuclear facilities will be interim-stored until SFL has been put into operation, see Chapter 6. SKB shall, in consultation with the nuclear power companies, make sure that the waste is treated and packaged in such a way that it is suitable for interim storage and final disposal.

To gather data as a basis for the design of the transportation and final repository systems and for fee calculations, SKB will continue to carry out decommissioning studies and general cost calculations for decommissioning of the Swedish nuclear power plants. SKB will also keep track of international developments in the decommissioning field and technological advances, for example by participation in the OECD-NEA's cooperative programme and the IAEA's programme.

7.1.2 Division of responsibilities between SKB and licensee/owner of Ågesta combined heat and power reactor

The decommissioning of the Ågesta reactor and the management of waste from decommissioning is financed via fee payments regulated in the Studsvik Act (1988:1597) and by Vattenfall AB.

Vattenfall holds the nuclear licence for Ågesta, but in the winter of 2009/2010 Vattenfall and SVAFO applied for SVAFO to take over the nuclear licence from Vattenfall. The matter has not been decided. A prerequisite if SVAFO is to take over the licence is that the company must be staffed for this task, organizationally and operationally.

Service and maintenance of the facility has been handled for many years by Studsvik Nuclear AB under contract to Vattenfall. This has been approved by SSM. When SVAFO takes over the nuclear licence, SVAFO will also assume responsibility for service and maintenance.

Ågesta's decommissioning work is coordinated with the decommissioning work at other nuclear power plants via the Decommissioning Group. Furthermore, SKB's resources are available if any research is needed.

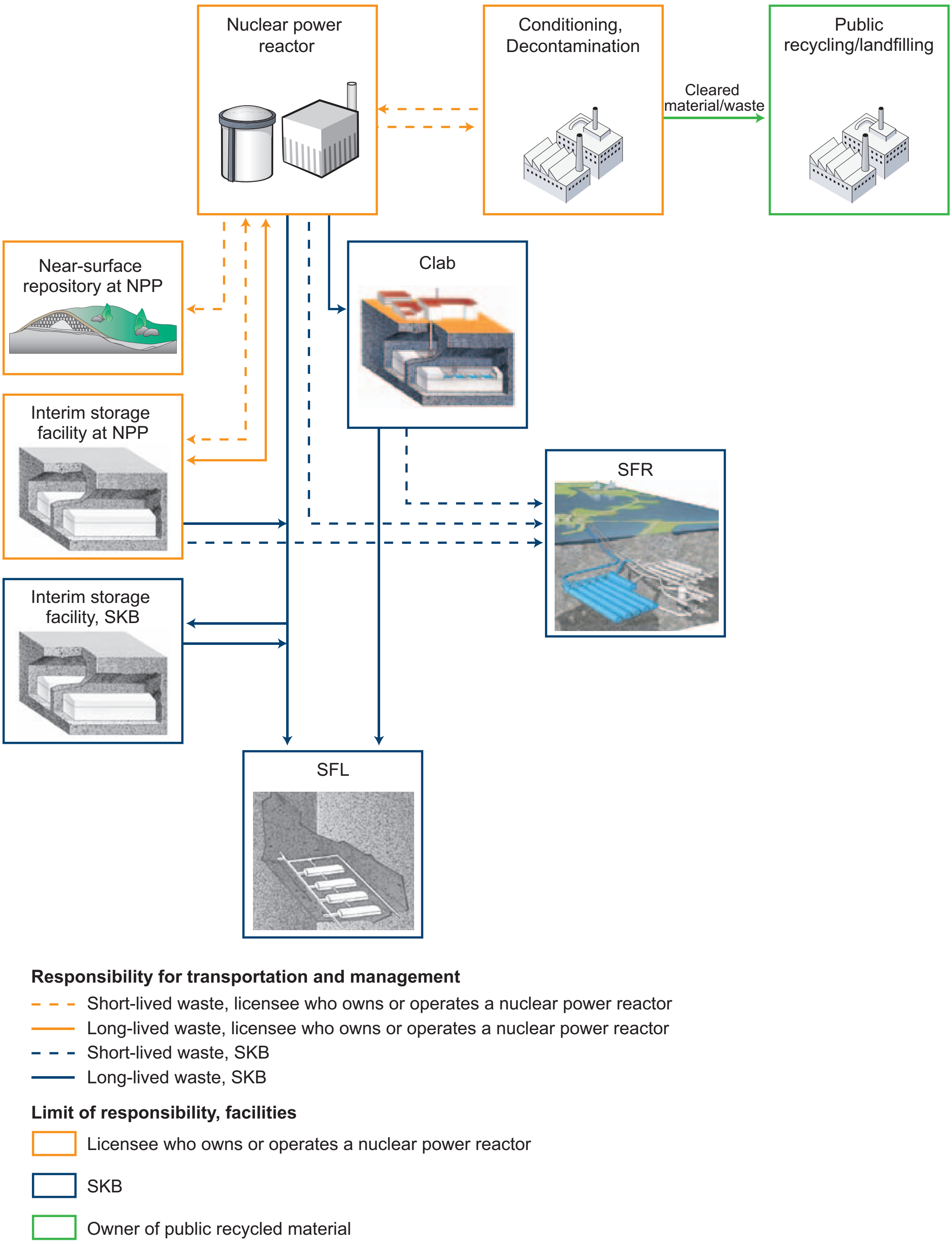


Figure 7-2. *Illustration of division of responsibilities for management of decommissioning waste and its flow. Note that the figure describes responsibility for transport and facility. Responsibility for the waste itself always rests with the nuclear power companies.*

7.2 Planning for decommissioning and dismantling

Since decommissioning often lies far in the future, detailed plans for decommissioning of the nuclear power plants have not yet been prepared by the nuclear power companies. One exception is the plans for the Barsebäck plant, where work is currently under way on in-depth studies for decommissioning.

All licensees for the NPPs, with the exception of Barsebäck, are planning to update their decommissioning plans within the next year. These updatings will be coordinated with the other licensees in the Decommissioning Group, which will be the coordinating forum for this work. Via the Decommissioning Group, SKB intends to update the joint industry documents "Teknik och kostnader för rivning av svenska kärnkraftverk" /7-1/ ("Technology and costs for decommissioning of Swedish nuclear power plants") and "Struktur på avvecklingsplan för kärntekniska anläggningar, guideline" /7-2/ ("Structure of decommissioning plan for nuclear facilities, guideline").

Now that Barsebäck's planning work is being intensified, numerous details and studies will be discussed and coordinated via the Decommissioning Group. Decommissioning issues for the Studsvik reactors, Ågesta and Ranstad, will also serve as a basis for knowledge exchange. These facilities are the first that are planned to be decommissioned, and an exchange of expertise and experience will be important for all members of the Decommissioning Group as well as for the individual decommissioning projects.

The decommissioning of SKB's own facilities lies far ahead in time, but the decommissioning plans are needed as a basis for SKB's programme and to comply with the statutory requirements. The plans for these facilities are summarized in Section 7.2.2.

7.2.1 Decommissioning strategies

Three different strategies for decommissioning are often mentioned in international contexts. The Swedish nuclear power companies mainly plan to carry out decommissioning in accordance with the strategy called "Direct Dismantling". For Barsebäck and Ågesta, which are already shut down, a variant of "Safestore" is being applied.

Direct Dismantling or Early Site Release

Direct Dismantling or Early Site Release entails that all radioactive components and buildings are decontaminated and/or dismantled/demolished shortly after shutdown and the waste is stored on the site awaiting transfer to a final repository or is packaged and transported to an already established interim storage facility or final repository. The facility is then released from regulatory control.

Safestore

Decommissioning to Safestore entails that the facility is kept largely intact and protected for a number of years (20–150). During this time the facility is monitored and "decontamination" occurs by natural decay. When the activity levels have declined, the actual dismantling begins. In many cases it has been possible to dismantle the parts of the facility that generate types of waste for which there are established management and disposal methods, while other parts (mainly reactor and graphite, where applicable) are put in Safestore pending the establishment of final repositories for these waste types (long-lived).

Entomb

The Entomb strategy for decommissioning entails that all radioactive components, systems and building parts are encased on the site in, for example, concrete. Monitoring takes place until the radioactivity has reached such a level that the licensee can be discharged from all responsibilities and obligations for the facility. This strategy is not applied in practice.

7.2.2 Licensees' planning for decommissioning and dismantling

This section contains a compilation of the nuclear power companies' decommissioning plans for Barsebäck, Forsmark, Oskarshamn and Ringhals as well as Vattenfall's plan for the decommissioning of Ågesta. Here the content of the preliminary decommissioning plans sent in by each licensee to SSM /7-3, 7-4, 7-5, 7-6, 7-7/ is described, along with the continued work with the plans. The section also describes the plans for the decommissioning of SKB's nuclear facilities.

Since the number of years until decommissioning and dismantling differs for the licensees, the degree of detail in their planning differs, as is reflected in the text below.

Barsebäck

Barsebäck's two reactors are in shutdown operation, i.e. they have been shut down and the spent nuclear fuel has been removed and Barsebäck Kraft AB (BKAB) has therefore, in keeping with applicable regulations, presented a general decommissioning plan /7-3/ that takes into account the fact that the reactors are in shutdown operation. The final goal of decommissioning of the Barsebäck plant is that the land, plus any remaining buildings including equipment, can be released for unrestricted use.

The facility has been adapted to shutdown operation and a new safety analysis report (SAR) and safety-related technical specifications (STF) have been implemented. During shutdown operation, BKAB has carried out system decontamination on both reactors. Under BKAB's strategy, decommissioning of Barsebäck 1 and 2 and the common facility parts on the site is carried out as a joint project. The dismantling preparations are planned to start in 2018 and dismantling itself in 2020, when SFR is expected to be able to receive the short-lived decommissioning waste. Dismantling, waste management and transportation will be carried out as industrial processes where low radiation doses are prioritized. This entails handling and transporting the radioactive waste as large units, and that clearance measures such as decontamination are only carried out when a great advantage can be identified from an ALARA perspective. The volume of radioactive waste will therefore be slightly greater, but this is compensated for by low radiation doses to personnel, a shorter timetable and a smaller need of resources. A near-surface repository for very low-level waste will not be established at Barsebäck.

The radioactive waste that will be managed by SKB and disposed of in SFR is estimated to amount to about 18,000 tonnes, which corresponds to a disposal volume of about 16,000 m^3. Most of the waste will be produced between 2020 and 2023. The total number of waste packages to be transported to SFR is estimated to be about 800 ISO containers, 62 concrete moulds, 283 steel moulds and 22 BFA tanks. This estimate is based on segmentation of the reactor pressure vessels. Preparations are under way for dismantling and transporting whole reactor pressure vessels, which will lead to changes in the number of packages. The shipments to SFR are expected to be evenly distributed over the years 2020 to 2023. The quantity of radioactive material that can be cleared is estimated at about 2,500 tonnes and consists primarily of parts from the turbine plant that can be sent for melting. Inactive concrete from different buildings is estimated to amount to about 400,000 tonnes, most of which will be able to be used as backfill on the site.

Forsmark

Forsmarks Kraftgrupp AB (FKA) has sent a preliminary decommissioning plan to SSM /7-4/ that applies to all three reactors in Forsmark. The plan is based on operation of the reactors for 40 years. The goal of decommissioning is to remove the radioactive material and release the facility for unrestricted use.

FKA's strategy, according to the decommissioning plan, is to keep the units in shutdown operation for some ten years before dismantling is started. Dismantling will not be started until all reactors have been shut down and system decontamination has been carried out. Dismantling of the reactors will be carried out in sequence. FKA does not rule out the possibility of land fill of very low-level waste.

FKA is updating its decommissioning plan during 2010/2011. The need for an updating stems mainly from the conclusions in the ongoing decommissioning studies concerning planning and strategies, see Section 7.2.3. The decommissioning studies are based on FKA's current plans to operate its reactors for at least 50 years and to keep the shutdown operation period as short as possible and not longer than five years.

Oskarshamn

In its preliminary decommissioning plan /7-5/, OKG AB assumes an operating time of 60 years for its three reactors, which agrees with the current plan. The objective for the decommissioning is to bring the facility to a state where its buildings and land can be released for unrestricted use.

Different decommissioning alternatives have been studied, but the choice of strategy will not be made until closer to the final shutdown of the facilities. The joint waste facility located at O1/O2 must be in operation for a few years after closure of the last reactor. OKG intends to carry out surveys of the activity in the facilities and make new estimates of the waste volume prior to final shutdown of the reactors. A decision will then be made whether system decontamination is to be carried out and whether surface repositories can be utilized.

When ongoing unit-specific decommissioning studies for the Oskarshamn plant are completed, they will serve as a basis for updating the decommissioning plan, which is planned for 2011.

Ringhals

In its preliminary decommissioning plan /7-6/, Ringhals AB (RAB) assumes operating times of 40 years for the R1 and R2 reactors and 50 years for R3 and R4. The goal of decommissioning is to remove the radioactive material and release the facility for unrestricted use.

RAB's strategy entails that most of the dismantling is done five years after final shutdown when system decontamination has been carried out. Furthermore, the nearby unit must be finally shut down before dismantling is begun. According to the plan the reactor pressure vessels will be transported and disposed of as large units, without segmentation. The last facilities on the site to be decommissioned and dismantled are waste and service facilities. According to the plan, this will take place between 2050 and 2053. RAB assumes in its planning that surface repositories can be used for very low-level waste.

RAB's ongoing decommissioning studies will serve as a basis for an updating of the preliminary decommissioning plan during the period 2010–2011. The study assumes that the operating time of all reactors is 50 years. The plan for how the reactor pressure vessels from PWR will be managed needs to be reviewed based on the knowledge gained from the study of handling of whole BWR reactor pressure vessels in the SFR Extension Project.

Ågesta

Vattenfall has contracted SVAFO to decommission and dismantle the CHP reactor in Ågesta. The Ågesta plant is currently in shutdown operation. All fuel and heavy water have been removed from the facility. Otherwise it is largely intact, but two of the four main heat exchangers (the steam generators) have been removed, decontaminated and melted, after which the ingots were cleared. Furthermore, diesel generator sets and batteries have been removed, and now only the rock drainage system is in operation. Follow-up is taking place in accordance with issued radiation protection conditions, and certain equipment (e.g. certain lifting devices) is being maintained to facilitate future dismantling.

Shutdown operation is planned to continue until dismantling starts, no earlier than around 2020 when the current environmental permit expires. A preliminary decommissioning plan has been prepared for the facility. /7-7/, where different objectives for a decommissioning are discussed. The alternative that appears most realistic today is release of the land area and the entire rock cavern for unrestricted use. When the time for decommissioning approaches, one alternative will be selected, explained and presented in detail.

What remains of the radioactive parts is mainly the reactor pressure vessel, two steam generators and 30 controllers with induced activity, plus piping from the primary system. Most of the activity is in the reactor pressure vessel with its internal parts. Two alternatives have been studied for handling them: either transport and disposal of the whole vessel, or segmentation. Regardless of which of the two alternatives is chosen, steps will be taken to minimize the dose to the personnel, for example by remote control handling. The main alternative for managing the reactor pressure vessel for the time being is segmentation, but at present the alternatives cannot be distinguished in terms of dose. More thorough studies will be conducted prior to dismantling.

Earlier dismantling would entail that the decommissioning waste has to be transported and interim-stored at another location, since SFR will not be ready to receive this waste until around 2020. Earlier dismantling is not judged to have any advantages in terms of dose; in fact the dose to personnel would increase since there would be less time for radioactive decay.

Measures will be adopted to avoid radiation doses to the environment and to minimize dose contributions to personnel, in accordance with the ALARA principle. New radioactivity measurements will be performed prior to dismantling, when the degree of detail in the preparations increases.

The decommissioning plan is based on avoiding all unnecessary handling of the waste. The Ågesta plant currently serves as an interim storage facility for radioactive material. Since the plant is being operated with a minimum impact on personnel and the environment, dismantling will not be commenced until final repositories or interim storage facilities are available for the waste.

Unlike other reactors, Ågesta is located far from the nearest harbour. This means that the waste from decommissioning will have to be transported by truck on public roads. The waste will be transported to a facility where it will be treated (volume reduction, conditioning, decontamination) and the radioactivity measured before it is transported to a final repository. Under current laws and requirements, a special permit for transport of hazardous goods by road will be required in order to transport the decommissioning waste from Ågesta. Previous experience exists in the industry of this type of transport, for example the steam generators from Ringhals and today's shipments of low-level waste for treatment in Studsvik.

Environmental reports are published annually, and this will continue during decommissioning according to current requirements. Before an application for decommissioning is submitted, an inventory must be made of hazardous substances and a plan for managing these substances must be presented. Some inventories are already being made of the radioactive and hazardous waste that is planned to be disposed of in SFR, as a basis for an application for a licence for the extension of SFR.

Ågesta's decommissioning study will be updated during the period 2010–2011 in accordance with ongoing decommissioning studies for the nuclear power plants, see Section 7.2.3.

A clearance study has been done for the areas outside the containment in Ågesta and an application for clearance of these areas for unrestricted use was submitted to SSM in 2010.

SKB's nuclear facilities

Clink

SKB is the licensee for Clab and will likewise be the licensee for the integrated facility called Clink when the addition containing the planned encapsulation plant is finished. The decommissioning plan for Clink /7-8/ is preliminary and conforms to the requirements made by the regulatory authorities on SKB for the coming licensing of the addition containing the encapsulation plant.

Clink will be decommissioned when all spent nuclear fuel has been encapsulated and disposed of in the Spent Fuel Repository. The timetable depends on when the last nuclear power reactor is taken out of service. Based on the current reference scenario /7-9/, which is based on 50 years' operation of the nuclear power plants in Forsmark and Ringhals and 60 years for the Oskarshamn Nuclear Power Plant, decommissioning of Clink could be commenced in around 2070 and concluded after 5–7 years. This means that Clink will still be in operation when the last facility in OKG's ownership has been shut down.

During the work of preparing the decommissioning plan for Clink, no reason has emerged why the decommissioning should be more complicated than for the other nuclear facilities whose decommissioning lies closer in time. It should be possible to carry out the decommissioning with a low dose to personnel, and the quantity of short- and long-lived radioactive waste is expected to be limited. According to the current version of the decommissioning plan, the radioactive decommissioning waste will be sent to SFL, but in later versions of SKB's plans it has been redirected to SFR.

The goal of the decommissioning is to remove radioactive material and release Clink for unrestricted use, according to the definition in the Radiation Protection Act. This means that buildings, including all equipment and land, should be released for unrestricted use.

SKB plans to prepare a decommissioning study for Clink during 2011 as a basis for the extension of SFR. This study may demonstrate the need to update the decommissioning plan.

SFR

Despite the lack of radioactive material, SFR's above-ground parts are to be regarded as a nuclear facility and are subject to the requirement on a preliminary decommissioning plan. In view of the fact that the above-ground parts of the facility can be regarded as conventional buildings, such a comprehensive account as that recommended in the published guidelines /7-2/ is not relevant. SKB has therefore sent a simplified account of how the decommissioning of SFR is planned to be carried out /SKBdoc 1049861/ to SSM. The decommissioning plan for SFR will be revised for the application to extend SFR.

The goal of the decommissioning is either to dismantle and demolish the above-ground buildings or to retain them for other use. A choice between these alternatives will be made by SKB prior to the decommissioning. Closure of the tunnels is described in the facility's safety analysis report.

A particular system for final disposal of nuclear waste from decommissioning is not relevant. The system parts that may be slightly contaminated (for example parts of the ventilation system) are dismantled and taken down into the facility's repository parts before the access tunnels are closed. Hauling away of dismantling material from the facility is not likely to be necessary, since dismantling material may be used as backfill in the transport tunnels, provided an environmental permit for such use is issued.

Along with SKB's other nuclear facilities for waste management, SFR will be among the last nuclear facilities to be decommissioned in Sweden. The requisite knowledge and experience will therefore be available from decommissioning and dismantling of other nuclear facilities. Requirements on such experience are very limited, however, since the facility is a conventional industrial plant when decommissioning occurs.

Since no radioactivity is left in the facility once any contaminated parts of the ventilation systems have been removed, there is no risk of radiological accidents during decommissioning. The organization at the facility will be adapted to the chosen strategy for decommissioning and possible dismantling, with observation of the rules and regulations that apply at the time of the decommissioning.

SFL

No decommissioning plan has yet been prepared for SFL, since the facility is so far still in the concept stage. See Section 4.2.2 for decommissioning of SFL.

The Spent Fuel Repository

A preliminary decommissioning plan has been prepared for the Spent Fuel Repository and will be included in the applications under the Nuclear Activities Act for final disposal of spent nuclear fuel and under the Environmental Code for the KBS-3 system.

The final repository for spent nuclear fuel consists of a surface (above-ground) part and an underground part. The surface and underground parts are connected by a ramp and several shafts, for e.g.

ventilation. The underground part consists of a central area and a number of deposition areas. The latter comprise the repository area. The closed underground part constitutes the actual final repository.

Decommissioning begins after operation is concluded, i.e. when all spent nuclear fuel has been deposited and the deposition tunnels have been backfilled and plugged. Decommissioning entails closure of the remaining parts of the underground part and dismantling of the surface part.

When decommissioning starts there will be no contamination in the facility. Dismantling is therefore carried out as for a conventional facility. Decommissioning waste is sorted and recycled where possible, or taken to public landfill. Hazardous waste is managed in compliance with relevant legal provisions. A ground survey is carried out and serves as a basis for site remediation.

The work with the decommissioning plan for the Spent Fuel Repository is handled within the Nuclear Fuel Programme.

7.2.3 SKB's planning for decommissioning and dismantling

By means of its own studies and by actively following what happens internationally, SKB, together with the nuclear power companies, is acquiring the competence needed to plan and execute future decommissioning projects. SKB's objective is to be able to manage the radioactive waste from decommissioning in keeping with the decommissioning plans for the various facilities. The plans for the NPPs and SKB's facilities are updated regularly in dialogue between the concerned parties. The work with the plans is an iterative process involving increasing competence and understanding as the decommissioning studies are updated and refined. The formal forum for this work is the Decommissioning Group.

Studies of decommissioning and dismantling of NPPs have been conducted under SKB's auspices in cooperation with the nuclear power companies for more than 20 years. The studies have been based on designation of a reference plant for the BWR and PWR plants. Detailed decommissioning studies have been conducted for the reference plants, and the results have then been applied to other plants. Strategies and technology have been compiled /7-1/ and are assumed to be the same for all the nuclear power units. The compilation has been used as a guideline in the nuclear power companies' preliminary decommissioning planning.

In order to provide a sound knowledge base for transport, interim storage and final disposal of decommissioning waste, unit-specific decommissioning studies are currently being conducted in cooperation between SKB and the nuclear power companies. Any local deviations from the reference plants are included in these studies. Estimates of waste volumes and estimates of the nuclide-specific content of the waste will be used as a basis for planning the extension of SFR and for assessment of the repository's long-term safety. The studies include surveys of both controlled and uncontrolled areas at the nuclear power plants. Coordination gains are achieved since the entire facility is surveyed at the same time and the licensee gets complete data on both nuclear waste and conventional waste.

In addition to estimates of volumes and costs, a survey is made of the technology and logistics for the decommissioning. The type of dismantling work in a facility varies, so several different techniques are described and a recommendation for the choice of the optimal technique is made for each type of work. Only proven, currently existing techniques are chosen to minimize risks and costs. In some cases the recommended dismantling technique is the same as is used for maintenance work during the operation of the nuclear power plant. The logistics account includes a description of the planned sequence in which everything will be dismantled in order to achieve optimal waste streams at the nuclear power plant and minimize the radiation doses. Dismantling and handling of the reactor pressure vessel and its core components is of great importance for the timetable, since it is one of the major dismantling activities that is begun first.

The logistics for management of the decommissioning waste from all NPPs has been studied in a background report to the compilation of strategies and technology /7-1/. One conclusion is that transport and deposition of decommissioning waste will not be limiting factors for the planning of when the different NPPs can be dismantled.

Joint studies are also examining new technology, such as handling of large components without segmentation.

Ongoing work on regulations at SSM regarding rules for clearance of materials, premises and buildings and rules for shallow land burial of very low-level waste may affect the quantity of waste to SKB's facilities. For the time being, SKB assumes that all material that is not cleared will be disposed of as radioactive waste in SFR or SFL.

In their decommissioning plans, the nuclear power companies present various strategies for management of the very low-level decommissioning waste. In order to reach a consensus, and as a basis for the extension of SFR, a feasibility study was published in 2010 that presents various alternatives for disposal of the very low-level waste. The alternatives described are surface repositories at the NPPs, a single joint near-surface repository, or SFR. An update of the study is planned during 2011, after which a decision is expected to be able to be made on a strategy, see Section 5.5.

7.3 Development of methods and technology for decommissioning

7.3.1 International work and cooperation

SKB is keeping track of and participating in the international development work being pursued within decommissioning and dismantling technology, in part through its chairmanship of the NEA groups CPD/TAG (International Co-operative Programme on Decommissioning/Technical Advisory Group) and DCEG (Decommissioning Cost Estimation Group). The CPD/TAG is treating some 40-odd current international decommissioning projects, while the DCEG is focusing on the costs of decommissioning and dismantling.

Ongoing work within the DCEG involves comparisons and evaluations of costs for decommissioning of facilities in different countries. An important tool in this work is the joint structure "A Proposed standardised list of items for costing purposes, Interim Technical Document (Yellow Book)" /7-10/ which is used to account for the costs. An updating of the tool (Yellow Book) is also being carried out within the DCEG, with the active participation of the NEA, the EC and the IAEA.

Studsvik and BKAB are participating in TAG with the R2 reactor and Barsebäck 1 and 2 as decommissioning projects. BKAB is participating in the WPDD's (Working Party on Decommissioning and Dismantling) survey concerning the management of large components as well as in the EPRI Decommissioning Technology Programme. BKAB is also participating in the IAEA's IDN (International Decommissioning Network).

7.3.2 The nuclear power companies' development work

In the case of Barsebäck, large studies have been conducted for both decommissioning and dismantling of a whole reactor pressure vessel and segmenting of a reactor pressure vessel and core compontents. Furthermore, the activity inventory in the waste that arises from decommissioning of the units has been estimated.

Current and planned studies for Barsebäck are:

- Survey and categorization of facility and environs regarding contamination and environmentally hazardous waste.
- Establishment of radiological criteria for decommissioning.
- Clearance procedures for land and buildings.
- Establishment of requirements for dismantling operation.
- Decommissioning studies dealing with technology, logistics and costs.
- Organization for dismantling operation.

Barsebäck is the NPP that will be decommissioned first, and since, according to the plants' preliminary decommissioning plans, the level of detail in the background data increases as the time of decommissioning approaches, most of the development work is being done for Barsebäck. This work is reported in the Decommissioning Group and is represented in TAG in order to exchange experience and enhance competence in the industry.

In the case of the other NPPs, the development work is taking place mostly via the issues and activities being discussed and reported within the Decommissioning Group.

Important experience is gained when the plants carry out audits and replace components. This provides good knowledge prior to decommissioning with regard to technology and data for estimating the costs of the work and of the management, transport and final disposal of the components.

7.3.3 Clearance

An important parameter for the choice of technology and strategy in the decommissioning work is the applicable release requirements. SSM intends to follow the EU's recommendations in the area, which will serve as a basis for future regulations for clearance.

SSM says that they do not intend at the present time to specify any general clearance levels for land. Instead they wish to establish clearance levels for each individual object based on the properties of the site in question and the future use of the land. The draft edition of SSM's regulation refers to the international recommendations from the ICRP and the IAEA, which stipulate a dose constraint criterion of approximately 300 microsieverts per year for an individual exposed to residual contamination.

The clearance levels for land have a significant bearing on the volume of the radioactive waste that is produced and are thereby a fundamental factor in planning for decommissioning and site remediation. It should be made clear which clearance levels are adequate, both from the public's perspective and for future use of the site.

The Swedish nuclear industry (including SKB) has begun work on a manual to create practical guidelines for clearance and thereby exemptions from the Nuclear Activities Act and the Radiation Protection Act of materials, premises and buildings as well as land. The manual will be completed when SSM's coming regulation on clearance has been published. The procedure described in the manual is based mainly on the Authority's coming regulation. The purpose of the manual is to serve as a tool and a guide in the formulation of company-specific procedures and instructions. It describes the principles, processes and procedures that should be followed during a clearance procedure. The intention is that compliance with the relevant regulations is ensured by following the procedures and principles in the manual. The manual will thereby be of great assistance in the detailed planning of decommissioning.

Part III

The Nuclear Fuel Programme

8 Current situation and points of departure

8.1 Introduction

In RD&D Programme 2007, SKB gave a status report on the planning work for construction and operation of the final repository for spent nuclear fuel (the Spent Fuel Repository). A preliminary execution programme was presented and the work methodology for design, construction and operation that was under development was described. The status report was done for two reasons. One was that the plans for construction and operation comprise planning premises for ongoing and planned technology development. They are thus a necessary background for the report on technology development. The other reason was that SKB considered it valuable to obtain viewpoints, via the review of the RD&D programme, on the execution plan while it was still under development.

The viewpoints on SKB's overall planning of the final repository project offered by SKI in its review of RD&D Programme 2007 mainly consisted of demands for supporting material for future applications. Among other things, they asked for information on planned investigations during construction and operation (detailed characterization), plus more details on the forms for organizational control during construction and operation. Viewpoints on SKB's development work chiefly concerned individual technology areas. This is dealt with in Chapters 10–16.

SKB has continued its planning for construction and operation of the Spent Fuel Repository, with sights set on the account that will be submitted in the applications under the Environmental Code and the Nuclear Activities Act. The execution plan for the Spent Fuel Repository has thereby changed character from being a development matter handled mainly within the RD&D process to being a matter that will be handled within the licensing process. The development of the technology that is needed in different phases of construction and operation will, however, continue to be handled within the RD&D process.

This change of what is reported within the licensing and RD&D processes, respectively – along with the fact that the RD&D programme is being presented shortly before SKB intends to submit the applications – has consequences for what is included in the present RD&D programme. The timing of SKB's RD&D reports is governed by the Nuclear Activities Act, the result being that this RD&D programme is being presented at the same time as the compilation of the supporting material for the applications is in its final phase. An up-to-date account of technology development is then necessarily dependent on background and premises that will not be reported in their complete form until the applications are submitted. This mainly applies to stage breakdown and methodology for construction and operation, plus the consequences of the selection of Forsmark as the site for the Spent Fuel Repository.

In order to provide a background to how the plans for technology development in different areas are linked to the execution of the final disposal of spent nuclear fuel, the execution plans for the encapsulation plant and the final repository are summarized in this chapter. The summary pertains to the period up to the start of operation, i.e. the main phases licensing, construction and commissioning. It is during these phases that the technology developed for the components of the KBS-3 system will be put into use, or readied for use when operation begins. Chapter 9 provides an overview of the programme for technology development and how it is managed to ensure that technology that meets the requirements is delivered as needed during the coming phases. The status and plans for the development work in the different technology areas are then presented in Chapters 10–16 with the necessary background on premises in the form of requirements and chosen reference design.

8.2 Main phases and timetable

SKB's planning for the future management of spent nuclear fuel, from interim storage in Clab via encapsulation to final disposal, takes place within the framework of SKB's Nuclear Fuel Programme. The programme includes licensing, design, construction and commissioning of the encapsulation plant and the final repository for spent nuclear fuel. These two civil engineering projects are beneficiaries of the technology development for the KBS-3 system that is being done with the Nuclear Fuel Programme as the client.

Fundamental premises for the execution plan are presented in the plan of action, see Section 2.3. Other important premises are:

- A total of about 6,000 canisters will be managed and disposed of.
- In routine operation the deposition rate is 150 canisters per year. The system is designed for a maximum deposition capacity of 200 canisters per year.
- The chosen reference design is KBS-3 with vertical deposition in a final repository at a depth of about 500 metres. A switch to horizontal deposition must be possible.
- The encapsulation plant will be built adjacent to Clab, and the two facilities will be operated as one integrated facility called Clink.
- The Spent Fuel Repository will be located at Forsmark and its layout will be adapted to the bedrock and other conditions on the site.
- Operation of the system will start as soon as possible, but with realistic timetables for the licensing process, construction and commissioning.

Figure 8-1 shows SKB's timetable for establishment of the Spent Fuel Repository and Clink, up to the start of operation, when trial operation of the entire system with facilities and transportation

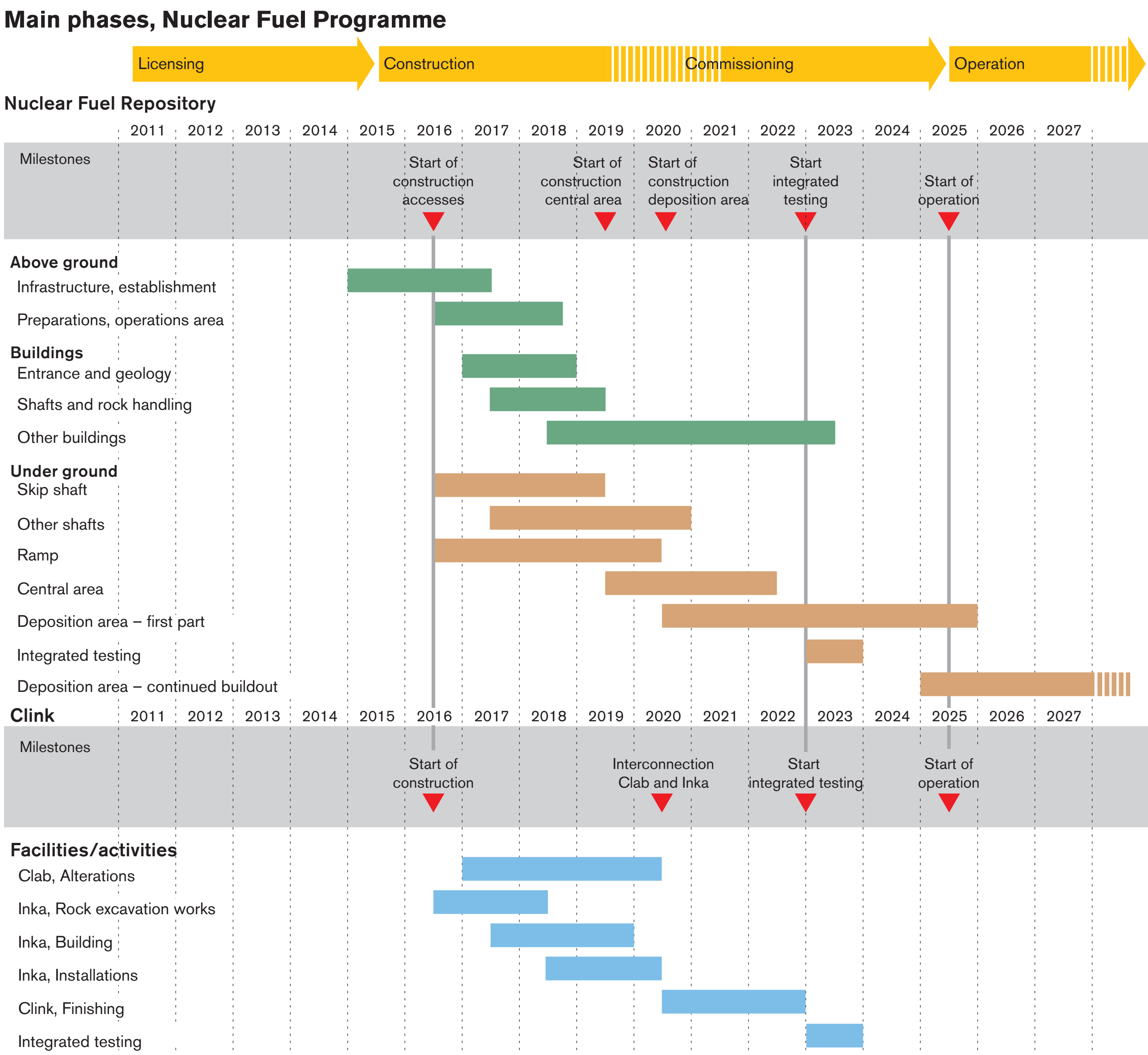


Figure 8-1. *Timetable for establishment of the Spent Fuel Repository and Clink. Designated milestones refer to target times for technology development according to the plan presented in Chapter 9.*

commences. Trial operation subsequently transitions into routine operation. Execution is divided into four main phases: licensing, construction, commissioning and operation. Construction and commissioning overlap in time, since facility parts are commissioned as they are finished and installations are put in place. The activities planned during different phases are summarized for the two facilities in Sections 8.3 and 8.4. The milestones shown in Figure 8-1 refer to distinct "delivery occasions" for results from technology development, i.e. points in time when technology components and solutions should be ready to be put into use or should have reached a certain development phase, see Chapter 9.

8.3 Encapsulation

Figure 8-2 shows a photomontage of the planned encapsulation plant. The building containing encapsulation will be built directly adjacent to Clab. Prior to operation the facilities will be interconnected to a single unit, called Clink. Their operation will then be integrated. Figure 8-3 illustrates the handling sequence for the fuel, from the storage pools in Clab via encapsulation to delivery of the filled and sealed canister.

An application under the Nuclear Activities Act for a licence to build the encapsulation plant and a licence to own and operate it as an integrated facility with Clab was submitted in 2006. Supplements were promised in the application, and in 2007, after SKI's and SSI's initial review, SKB received a number of demands for supplementary information. SKB has responded to these demands, and a supplement was submitted in 2009.

Current and planned technology development mainly concerns the processes around fuel handling, fabrication of components for the canister, plus seal welding and nondestructive testing of components and the seal weld. This is described in Chapters 10 and 11.

Figure 8-2. *Photomontage showing Clink, the integrated facility for interim storage and encapsulation. The buildings outlined in red are the planned encapsulation building and a smaller terminal building.*

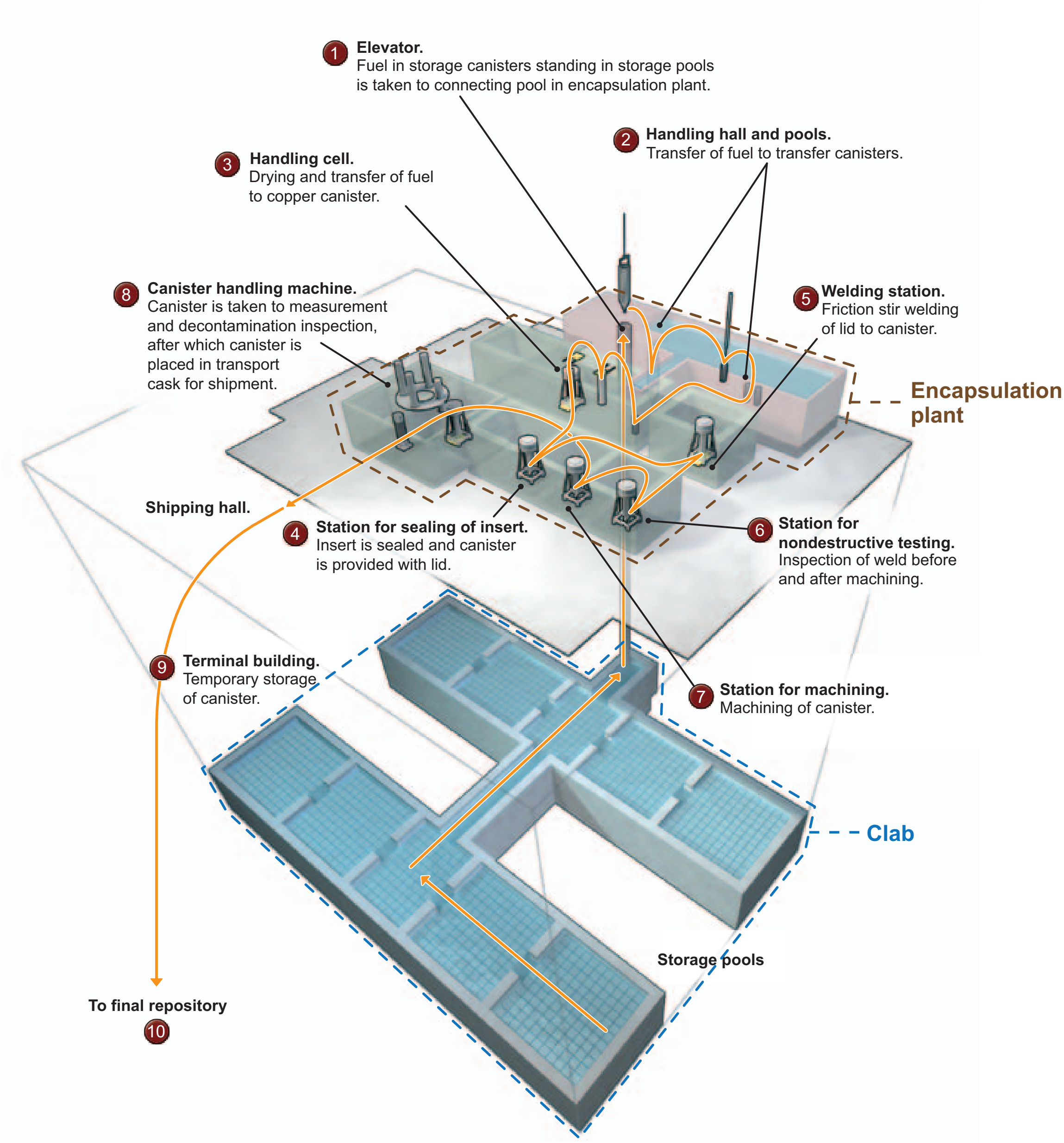


Figure 8-3. The encapsulation process for spent nuclear fuel.

8.3.1 Licensing

Licensing under the Nuclear Activities Act is currently under way for the encapsulation plant. The application was submitted in 2006 and supplemented in 2009. When the application under the Environmental Code for the final repository system has been submitted, the continued licensing process is expected to proceed alongside the licensing of the Spent Fuel Repository, and integrated for the whole system (see Section 8.4.1). During the licensing phase, SKB will continue the design of the encapsulation plant and of all machines and installations. Technology development in accordance with the plans presented in Chapters 9, 10 and 11 will continue in parallel. The work during the licensing phase is supposed to provide a basis for procurement of the facility and system and prepare for construction, organizationally and administratively. Once SKB has been granted a licence, the procurement phase enters its final stage and the construction phase begins.

8.3.2 Construction

SKB presented an account of how the activities during construction and commissioning will be organized, led and managed in an appendix to the application for a licence under the Nuclear Activities Act

/SKBdoc 1056406/. Operation of Clab will continue throughout the construction of the encapsulation plant. A far-reaching adaptation of the construction works will be necessary in order not to jeopardize safety in Clab and to minimize operational disturbances. The extension of Clab (Clab-2) was done in the same way, and experience from that project will be a valuable asset when the encapsulation plant is designed and built.

The construction site for the encapsulation plant will be kept separate from Clab's operations area up until the time the plants are to be connected together. The construction phase will initially be dominated by rock excavation works, which will gradually transition into construction works and finally installation works. During the latter part of the construction phase, organizational and administrative preparations will also begin for the interconnection of the plants and the commissioning of the integrated facility.

8.3.3 Commissioning

The physical interconnection of the plants will take place when the encapsulation plant is finished, alterations have been made in Clab and the subsystems in each facility have been installed and tested as far as possible. Then partition walls will be removed and systems and communication paths will be joined together. This will be followed by integrated testing and running-in of the entire integrated facility, Clink. The concluding integrated testing will include the whole final repository system, with the final repository and canister shipments to it.

8.4 Final repository

After a siting project lasting many years, SKB's Board of Directors decided in June 2009 to select Forsmark as the site for the final repository for spent nuclear fuel. The choice stood between Forsmark in Östhammar Municipality and Laxemar in Oskarshamn Municipality. The reasons for selecting Forsmark will be explained in the applications under the Environmental Code and the Nuclear Activities Act.

With the decision to locate the Spent Fuel Repository in Forsmark, planning of the construction and operation of the repository has entered a new phase. The purely industrial premises for the project have been clarified. Technology development can be concentrated on finding solutions that meet requirements on safety and functionality for the conditions that prevail in Forsmark. The importance of this clarification of premises varies between different technology areas. For adaptation and choice of methods for extraction, sealing and stabilization of the rock, for example, the limitation to the specific rock conditions that we know characterize Forsmark is of great importance. The opposite can be exemplified with the development of machines for canister transport and deposition, which is almost completely independent of site-specific premises.

Figures 8-4 and 8-5 show the design of the Spent Fuel Repository in Forsmark that was conceived during the site investigation phase. The design basis has been that the repository will be located within the identified area with suitable bedrock, and that facilities and activities above ground can be accommodated within the existing industrial area. The result is a repository at a depth of about 470 metres that extends approximately 2.5 kilometres southeast from the NPP's cooling water channel. The facilities above ground are gathered in an operations area at Söderviken, just southeast of the channel.

The repository illustrated in Figures 8-4 and 8-5 is the result of an iterative process. The results from the investigations have served as a basis for a series of progressively more complete site descriptions. Together with general requirements on the repository and other premises, these results have comprised the basis for the site-adapted repository solution. At the same time, safety evaluations have been made and the environmental consequences of a final repository on the site have been examined. The final results of this process have been published in a series of reports, which also constitute supporting documents for the applications which SKB is now preparing. The site description is presented in /8-1/ and the design of the facility's underground parts in /8-2/. A preliminary version of the environmental impact statement for the whole final repository system, including the repository in Forsmark, has been presented within the framework of the EIA consultations. Remaining reports will be presented when an application is submitted.

Figure 8-4. Photomontage with the above-ground parts of the planned final repository in Forsmark. The red outlines show the operations area and the heap for rock spoils. Parts of the nuclear power plant can be seen at the left.

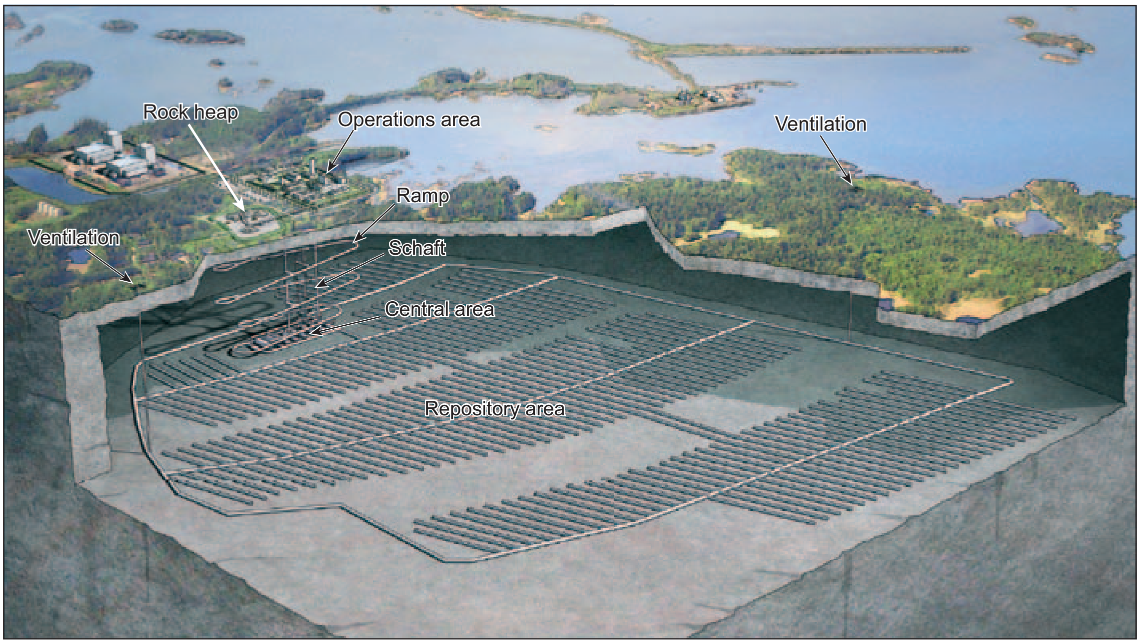


Figure 8-5. The final repository for spent nuclear fuel in Forsmark, fully built-out.

8.4.1 Licensing

Licensing of the Spent Fuel Repository starts when the applications under the Nuclear Activities Act and the Environmental Code have been submitted. How the licensing process is pursued is determined above all by the environmental court, SSM and the Government. SKB foresees the expenditure of considerable resources in providing presentations and any necessary supplements. During the licensing phase, a PSAR (preliminary safety assessment report) will also be written and submitted to SSM.

A principal task for the final repository project during the licensing period is to make all preparations required to begin construction of the Spent Fuel Repository, at a pace geared to the progress of the licensing process. The construction phase will make different demands on SKB's organization and activities than today's. This applies, for example, to management of the project based on the information flow between the construction works and investigations, modelling, design and safety assessment. A central task on the site in Forsmark will be to build up an organization that is suited to this.

In parallel with the licensing process, SKB will proceed with the design of the Spent Fuel Repository. The first step is system design, which will be the last overall design of the whole facility. The next step is detailed design, which is supposed to result in construction documents. Detailed design is not done for the whole facility at once, but at the pace at which documentation is needed for procurements and construction works. Detailed design is therefore carried out during the licensing phase for facility parts that will be built early. This mainly includes the accesses to the repository, i.e. ramp and shafts, and parts of the above-ground facilities. Detailed design for parts that are to be built later, for example deposition areas, is done during the construction phase and partly integrated with the actual construction process.

Construction-related and site engineering investigations are planned as support for the design work prior to the start of construction. Ground surveys will be done as a basis for the placement of buildings and foundation engineering. Near-surface rock will be investigated, for example in planned locations for accesses. The local infrastructure for the facilities will be prepared. This mainly involves working together with Forsmarks Kraftgrupp AB to adapt the infrastructure already in place in Forsmark to meet the needs of the Spent Fuel Repository.

8.4.2 Construction

The construction phase will begin when SKB has obtained all licences and conditions needed to start construction of the final repository. This means, among other things, that a preliminary safety analysis report (PSAR) must have been submitted to and approved by SSM. Construction will be the most labour-intensive phase of the entire final repository project. High demands will be made on effective management and work flows. The mode of working SKB intends to employ is summarized in Section 8.4.4.

Construction of the underground facilities can be divided into three overlapping stages: the first when accesses (shafts and ramp) are driven down to the repository level, the second when the central area's caverns are built, and the third when the first deposition area is established. Figures 8-6, 8-7 and 8-8 illustrate these stages. Construction of the accesses is time-critical for the progress of the entire project. The ramp and the first shaft are excavated in parallel, from the surface downward. Until the shaft has reached the repository level, the rock excavation works are limited to these two faces. When the repository level has been reached, excavation of the central area starts with a rock loading station. When the rock loading station and rock haulage to the surface via the rock hoist (skip) can be put into operation, the capacity of rock handling increases radically and several driving faces can be established. The rock excavation works for accesses and the central area are accompanied by extensive installation works for the equipment that is needed to operate the facility.

Excavation of the accesses and the central area will yield in-depth knowledge of rock conditions and experience that must be capitalized on and translated into e.g. rock support and rock sealing measures in tunnels or changed repository design. A work methodology that permits efficient experience feedback is therefore needed, see Section 8.4.4.

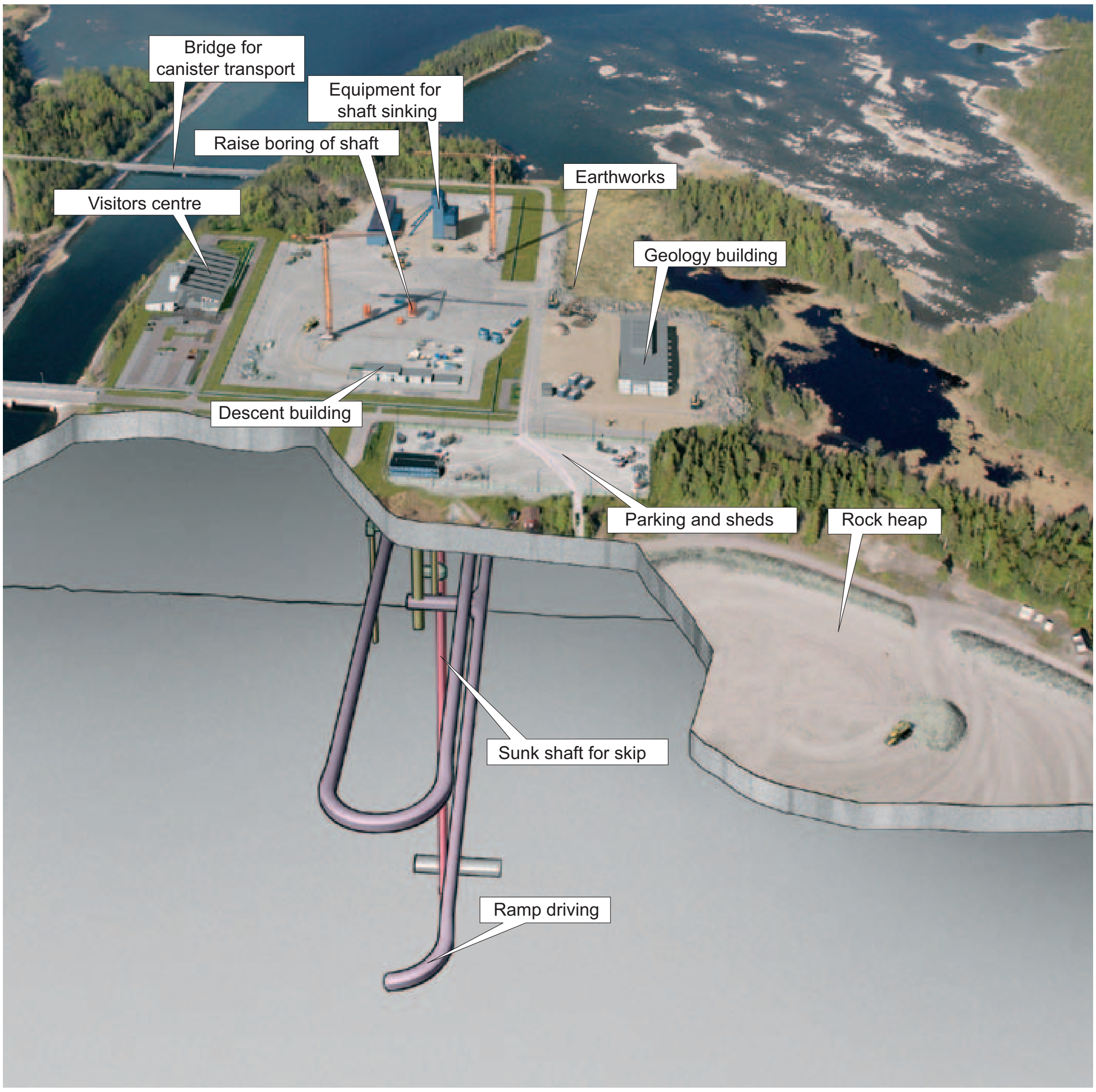


Figure 8-6. *The final repository in the final phase of the first stage of the construction phase, approximately 2 years after start of construction. Driving of the first shaft (the skip shaft) has reached down to the level of the future repository and the ramp is approaching. A few buildings have been erected on the operations area.*

Extensive rock investigations are conducted throughout the construction process (detailed characterization). In the short perspective, the investigations provide the site engineering data which, together with construction experience, permits ongoing adaptation of the construction works to local rock conditions. In the longer term, the investigations also provide a basis for whatever modifications of the repository design may be needed to meet the design premises with respect to e.g. long-term safety. If necessary, site descriptions and safety assessments must be revised in response to the results from the investigations.

As the central area is built, investigations are conducted for the deposition area and a tunnel is driven providing access to this area. From this tunnel a few deposition tunnels are excavated in which deposition holes are bored. One purpose of preparing a deposition area at this early stage is to gather the geoscientific data that are needed as a basis for an updated safety analysis report prior to trial operation, while another is to create room for implementation of the handling technology for deposition and integrated testing of the whole process during the commissioning phase. Then when operation begins, the area will be used for deposition of the first canisters with spent nuclear fuel. Construction of this area is therefore subject to the same technical and administrative requirements that will apply to operation.

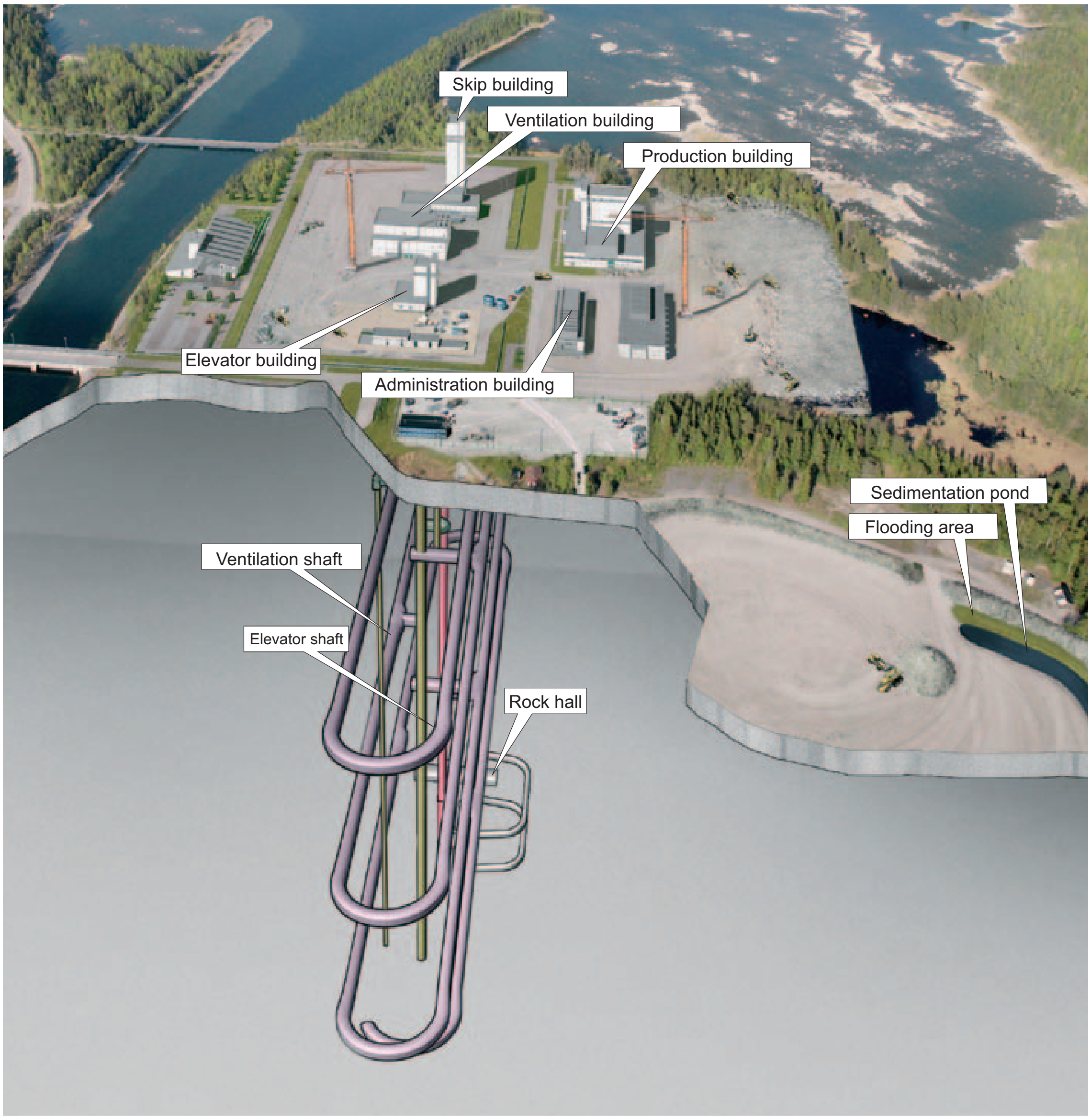


Figure 8-7. *The final repository during the second stage of the construction phase, approximately 4 years after start of construction. Shafts and ramp have been put into operation. Rock haulage takes place via a skip shaft. Construction of the central area's rock cavern begins.*

The facilities on the surface are built at a pace that is geared to the underground works. To start with, parts of the operations area are filled out, handling areas are prepared and temporary construction arrangements are established. The first permanent buildings to be erected are an entrance building for the access ramp and a building that is needed for the investigation activities (geology building). Then come shaft superstructures, facilities for rock handling, production building for buffer and backfill, and in the final phase other buildings for operation and service.

8.4.3 Commissioning

Commissioning of the final repository's subsystems starts and proceeds as the systems are built and installed. The construction and commissioning phases will overlap chronologically. For example, the haulage system for rock spoil (rock loading station, skip shaft etc.) will be put into operation before the first deposition tunnels are even built.

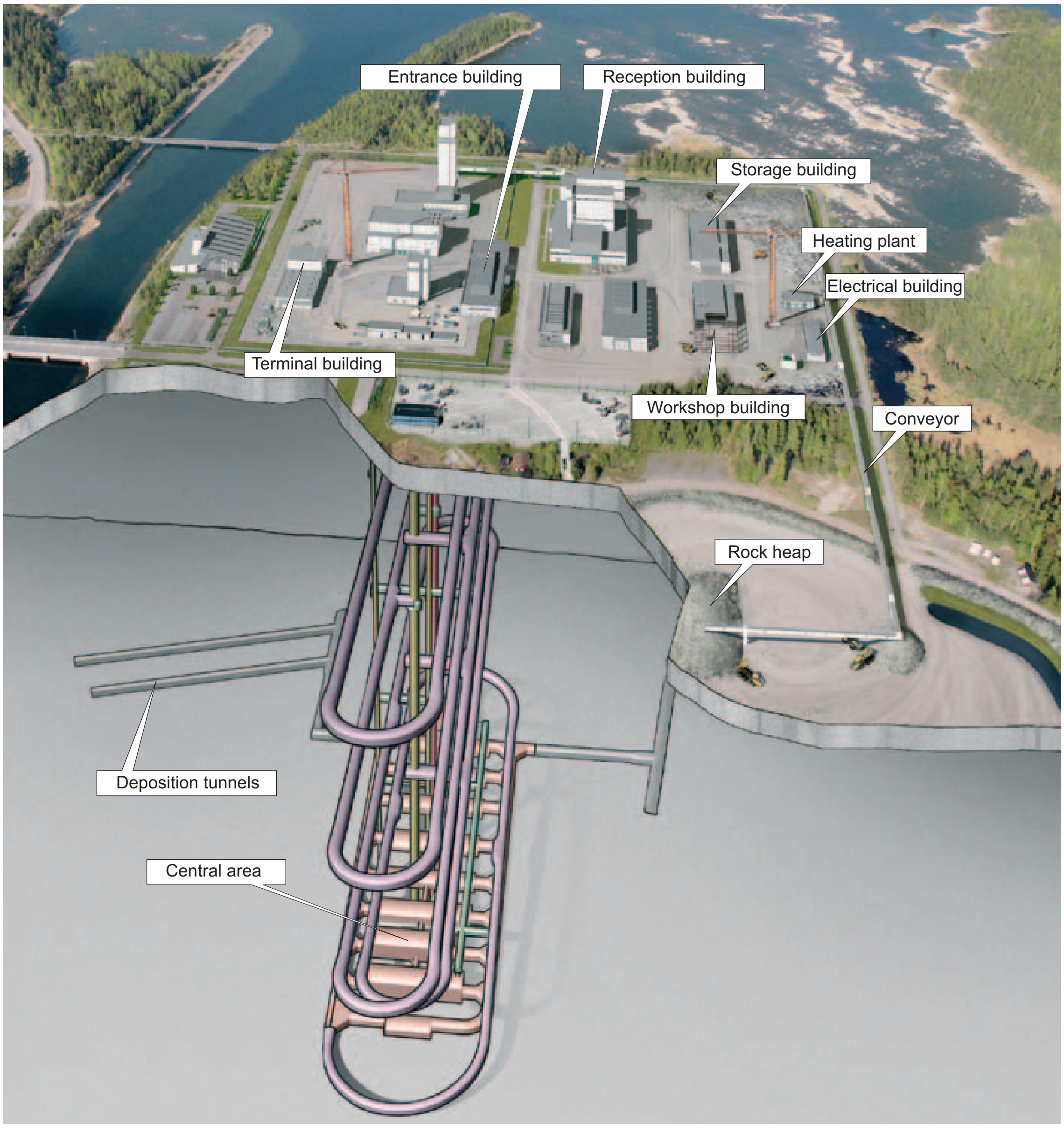


Figure 8-8. *The final repository during the third stage of the construction phase, approximately 6 years after start of construction. Descents and central area are finished. Construction of the first deposition area is in progress. Buildings and facilities on the operations area are being finished.*

In connection with commissioning the systems will be tested, first separately and then gradually more interconnected. As the different parts of the facility are being commissioned, the operating organization will be assembled and personnel will be trained for their duties. Running-in of technology and organization will be concluded with integrated testing of the whole facility under realistic conditions. All operational activities will then be carried out, including deposition of a number of canisters, but without any spent nuclear fuel, in the first deposition area built during the latter part of the construction phase. The integrated testing is concluded with integrated testing of the whole final repository system, with encapsulation plant, transportation system and final repository.

The commissioning phase is concluded when SKB obtains a licence for trial operation of the final repository system. The goal is that all functions and resources, as well as openings for deposition, will then be accessible so that trial operation can be commenced.

8.4.4 Work methodology during construction and commissioning

The planning includes devising a strategy, a methodology and a programme for management and control of the activities, handling of information flows etc. at various stages of design, construction and commissioning of the parts of the final repository. A status report from this planning work was presented in RD&D Programme 2007. Since then the strategy and methodology have been further developed, and an account will be given of the final results in the supporting material for the applications under the Nuclear Activities Act and the Environmental Code. Some principal features are summarized here.

The overall goal of construction and commissioning is that SKB can apply for a licence to take the final repository into operation. In order for this to be possible the following must have been achieved:

- The final repository has been built and commissioned. The facility documentation must also be complete and the operating organization and administrative procedures must be in place and run-in.
- The safety analysis report has been updated in the manner required for an application for trial operation.

Everything that is done during construction and commissioning is aimed at one or both of these goals. In line with this, the activity has been divided into two iterative main processes:

- Safety Assessment
- Construction

The processes with constituent components and interrelationships are illustrated in Figure 8-9. The activities aimed at producing an updated safety analysis report for an application for trial operation are gathered within the main process Safety Assessment. The starting point is the site description that was prepared after completed site investigation and the safety assessment SR-Site. Regular cross-checks and possible updatings will then be made with the guidance of information produced by the main process Construction as a result of, for example, the detailed characterization that is done. Safety evaluations may be needed, e.g. prior to a new construction stage, in order to check that the planned design and execution meet design premises with respect to long-term safety. Conversely, Safety Assessment can provide guidance in the form of requirements and restrictions that must be complied with in order for construction to result in a safe final repository.

All activities needed for the facility to be constructed are gathered within the main process Construction. Solid boxes in Figure 8-9 show the components included during the construction phase. The intention is to apply the same processes during the operating phase, with the addition of components indicated by dashed boxes in the figure. Activities during the construction phase consists of investigations including monitoring, modelling, design with predictions for construction, and production in the form of rock excavation, installation etc. Additional activities during the commissioning phase are facility documentation and organizational preparations for operation.

The methodology for the main process Construction according to Figure 8-9 mainly applies to the underground facilities and consists of rock construction in accordance with the Observational Method. This method is suitable because the exact rock conditions where the facility parts are to be built cannot be fully determined in advance. A tool is therefore needed to gather information from both investigations and actual inspections as well as experience from the construction works. This information must then be interpreted so that it can be translated, via design and construction predictions, into adaptation of the construction technology or the design of the facilities. The purpose of the Observational Method is to systematize this iterative mode of working. This makes heavy demands on smoothly functioning information and work flows, but also on an ability to interpret and understand the information so that the right measures can be adopted. In daily application, this may mean for example that rock support and sealing measures can be planned with the support of the latest information obtained from the rock excavation works. In the longer term, the same principle is used for stepwise buildout of the facility, where the detailed planning of each stage is based on the latest information from investigations and experience from previous stages.

The mutual control between the main processes can also be handled to some extent within the application of the Observational Method. If the information warrants far-reaching changes that entail revisions of site descriptions and safety assessments, however, this must be handled at higher decision-making levels.

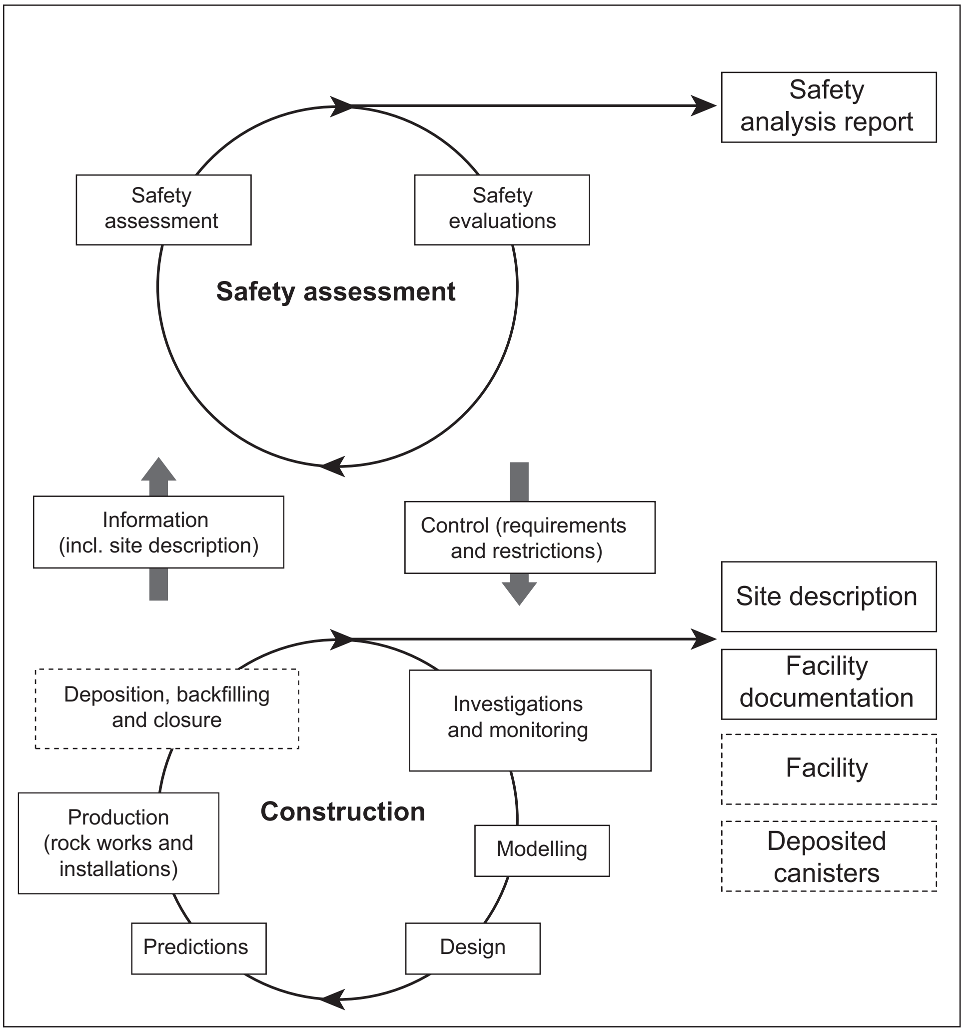


Figure 8-9. *The main processes Safety Assessment and Construction, with constituent components, work flows and important relationships. Dashed components are added when the final repository is put into operation.*

9 Overview – technology development

SKB develop technology for building and operating the final repository system so that the requirements on long-term safety, low occupational radiation dose and good external environment are met. This chapter describes how technology development is managed to ensure that technology that satisfies these requirements is delivered as needed to the Nuclear Fuel Programme during construction and commissioning. The chapter also provides an overview of the goals and scope of the technology development with regard to the different barriers. The concrete development work is then described in subsequent chapters.

The development work requires extensive technical resources. SKB's own laboratories – the Äspö HRL, the Canister Laboratory and the Bentonite Laboratory – are built and equipped for full-scale tests, demonstrations and dress rehearsals. Most of these types of development activities will therefore be conducted at these facilities. There are other possibilities as well, such as Posiva's demonstration facility Onkalo, located in Olkiluoto in Finland, which is under construction. There are also underground laboratories and laboratories for metallurgical research available in Europe and other parts of the world. In addition, there are industrial facilities in many countries with knowledge and resources in areas where SKB needs to conduct development work.

The bedrock in Forsmark differs in essential respects from that in the Äspö HRL. The rock stresses are higher in Forsmark, and the permeability of the rock to water at repository depth is much lower than at Äspö. These and other differences need to be taken into account in the development work, but the Äspö HRL nonetheless remains SKB's most important resource for research, development and demonstration requiring an underground environment at the relevant depth. Furthermore, cooperation with Posiva is being intensified, and certain experiments may be conducted in Posiva's Onkalo facility, where the rock conditions in many respects resemble those in Forsmark. The development work in many areas – such as canister fabrication and boring of deposition holes, as well as transport vehicles and handling machines – is more or less independent of differences in rock conditions.

9.1 Points of departure

The structure with division of technology development into a number of production lines which SKB presented in RD&D Programme 2007 has been further developed. According to this structure, the development work is conducted within production lines for fuel, canister, buffer, backfilling, closure and underground openings. Furthermore, overall systems are developed for e.g. logistics and machines that are unique for the final repository and that are therefore not available on the market.

Technology development has now reached the point where a reference design for the KBS-3 system is finalized and has been shown to meet the design premises that have been formulated. At the same time, a feasible way towards production and an inspection programme has been found. Continued technology development is needed as we proceed from schematic solutions to solutions that are tailored to an industrialized process with established requirements on quality, cost and time.

Technology development has come the farthest for components of crucial importance to long-term safety, such as the design of the canister. The design of certain components, including the canister, may need to be modified, at least in details. Moreover, possibilities are being explored for further improving the design to simplify the technical execution with undiminished or improved safety.

9.1.1 Design premises

Technology development is supposed to deliver solutions for the design and site adaptation of Clink and the Spent Fuel Repository. The solutions should meet design premises both with regard to long-term safety and with a view to the fact that the different barriers make demands on each another.

SKB's current design premises with regard to long-term safety, "Design premises for a KBS-3V repository based on results from the safety assessment SR-Can and some subsequent analyses" /9-1/, are primarily based on conclusions from the SR-Can safety assessment plus a number of supplementary analyses. Updated safety assessments can occasion clarifications or revisions of the design premises. A first revision, based on the conclusions and recommendations that emerge from SR-Site, is performed when the applications have been submitted.

An example of design premises conditional upon the fact that the different barriers make demands on each other is the tolerance requirement for the dimensions of the deposition hole. Given the dimensions of the precompacted buffer blocks, the dimensions of the deposition hole must lie within certain tolerances so that the buffer density will remain within the permissible interval. These types of design premises are presented for the different production lines and may need to be revised if the reference design is changed.

9.1.2 Site-adapted solutions

The selection of Forsmark as the site for the Spent Fuel Repository means that it is now possible to focus continued technology development efforts on the conditions that exist there. We no longer need to devote our efforts to solutions that have to work under a variety of rock conditions. The low frequency of conductive fractures means that it should be possible to simplify or skip certain technology development projects. An example is arrangements to deal with water in the deposition holes. Rock engineering development can be focused on excavation in an environment with fracture-poor and relatively heavily stressed rock.

9.2 Control and reporting

Responsibility for control of the technology development within the different production lines rests with the Nuclear Fuel Programme's client function, whose task is to:

- Develop and administer the requirements and design premises that control technology development.
- Coordinate the work between the different lines so that the KBS-3 system as a whole satisfies the requirements regarding long-term safety and operation.
- Control the development work to meet the needs of the two civil engineering projects for Clink and the Spent Fuel Repository, as far as the level of development that must have been reached for different parts by different times to serve as a basis for design, construction, installations and functional tests.

To achieve this control SKB applies systematic requirements management, see Section 1.5. This means that decisions to modify solutions in relation to the current reference design or reference activity are made in a systematic and controlled fashion. The purpose is to ensure that everyone works from the same premises. The reference design and reference activity are defined as the solutions that apply on a given occasion and that everyone who works with development of facilities and technology are supposed to use as premises, with the exception of studies of alternative designs. All tasks for the different production lines are defined in accordance with a delivery control model for technology development (see Section 9.2.1).

All results from the technology development within the production lines serve as a basis for the safety analysis report for the encapsulation plant and the final repository. Results and execution must therefore meet requirements on quality assurance, traceability and documented evaluation of compliance with requirements. SKB's management system shows how this is accomplished.

9.2.1 Delivery control model

SKB has developed a methodology for control of technology development within the Nuclear Fuel Programme, a delivery control model. The methodology entails specifying what degree of maturity the development of different components should have reached by the following milestones: start of construction, start of buildout of first deposition area (Spent Fuel Repository) or interconnection of Clab and Inka (Clink), start of integrated testing and start of operation. The delivery control model is thus aimed at clarifying how the different components of technology development are linked to the execution of the final repository system. Furthermore, the delivery control model should ensure that the client give the contractors clearly defined assignments so that the right things are delivered in relation to the requirements and the needs of the two construction projects.

The basic idea behind the model is that development is divided into a number of phases, and that for each phase there is a specification of what should have been achieved and thus what should exist as a basis for a decision to proceed to the next phase. Compared with the delivery plans for the technology development presented in RD&D Programme 2007, this entails a stricter systematics and subdivision of the work that applies to all development areas.

The delivery control model divides technology development into the following phases, see also Figure 9-1:

- **Concept phase:** The purpose of the concept phase is to specify the requirements on the subsystem or the component, make a broad evaluation of conceivable solutions and propose one or more technical solutions to proceed with in the next phase. This entails that a reference design (or several alternative reference designs) is established for the subsystem, that it has been shown how this (these) reference design(s) can be verified against the design premises defined for the concept phase, and that a feasible way to production and an inspection programme has been found.
- **Design phase:** The purpose of the design phase is to produce a design of the subsystem or component, to verify that it satisfies the requirements, and to formulate proposals for production, inspection and maintenance of the subsystem or component. The design phase may be iterative since it may turn out that the proposed solution does not satisfy the requirements, cannot be produced, or cannot be inspected in an efficient manner. As a rule, the design phase consists of two stages: initial system design and final detailed design.
- **Implementation phase:** The purpose of the implementation phase is to build up production and inspection systems. This phase also includes the documentation, including any licensing, that is needed for operation of the subsystem or component. The goal of the implementation phase is that the system or component is run-in and ready to be handed over to operation.
- **Administration phase:** The administration phase begins when the system or component has been put into operation. The goal of this phase is to make use of operating experience in a structured way as a basis for possible modifications of both the production apparatus and the product. If and when it is warranted, a change case or project is initiated.

When SKB submits applications for licences to build and operate the final repository system, technology development has in principle passed the concept phase. In some respects it has come considerably farther. Prior to the start of integrated testing, development of all systems and components for operation should be at the end of the implementation phase.

9.2.2 Reporting

The results of technology development are documented by the contractor and reported to the client as a basis for the safety analysis report. There the contractor specifies the reference design, shows that the reference design fulfils the design premises, explains how the reference design can be accomplished with methods for inspection/verification of achieved results, and describes the initial state of the different barriers. In the licence applications for construction and operation of the final repository, this reporting is done in line reports.

9.3 Technology development needs

When SKB submits the applications for the Spent Fuel Repository and Clink, technology development, with the delivery control model's definition, must have at least achieved the goals for the concept phase. Reference designs that fulfil the design premises must have been reported and feasible ways to production and inspection programmes must have been found. For many subsystems, considerable design work remains before they are ready for implementation. The development needs are discussed in general terms in the following sections and are summarized in Table 9-1.

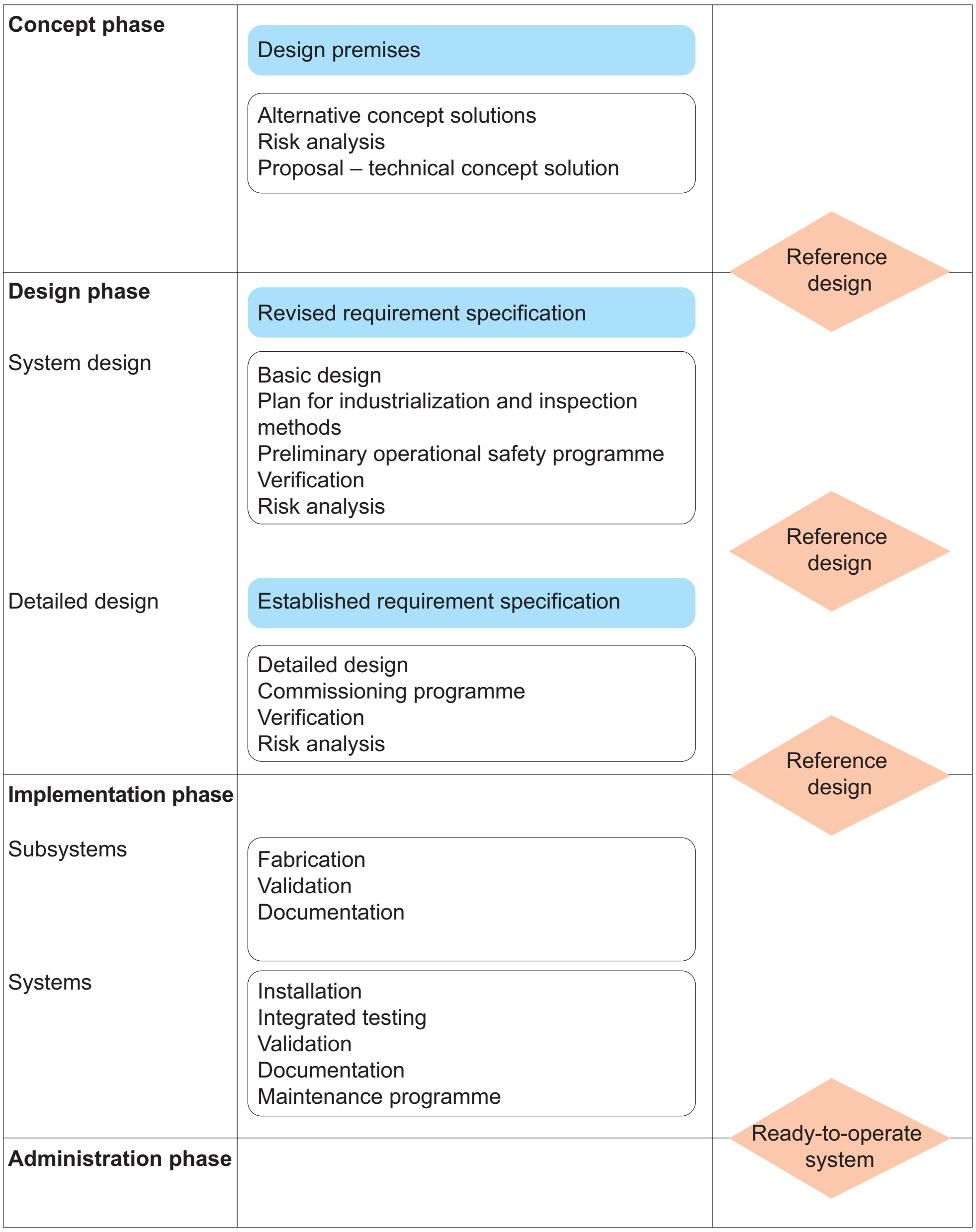


Figure 9-1. *Delivery control model for SKB's technology development.*

9.3.1 Overall needs

In the line reports, which serve as a basis for the licence applications for construction and operation of the final repository, SKB intends to show that the chosen reference design fulfils the design premises. In the continued development work it is important to focus the efforts on solutions that are efficient for the integrated system. This brings up a number of overarching development questions. Moreover, the methodology that has been chosen to formulate design premises with respect to safety entails that these premises are updated when a safety assessment has been conducted. The following general measures have been deemed necessary:

- After each major updating of the safety analysis report, the design premises are updated based on the analyses and conclusions of the safety analysis report. This first revision will pertain to the design premises published in /9-1/. It will be done based on the conclusions and recommendations that will emerge from SR-Site and the assessment of operational safety for the final repository system. Further revisions will be done if information is subsequently forthcoming that is deemed to be of importance for safety.

- An integrated set of requirements is drawn up for fuel and canister, concerning for example decay heat and water in the canister. This set of requirements is needed for system design of the encapsulation plant.
- An integrated set of requirements is formulated for accesses, as well as for deposition holes and deposition tunnels. The purpose is to allow solutions that take into account the conditions at Forsmark while also being more production-optimal. This includes requirements for accesses, inflows, the geometry of the floor of the deposition tunnel, the bevelling of deposition holes, tolerances for deposition holes and buffer, and closure.
- Integrated production adaptation needs to be done with regard to rock excavation, inspection programme, deposition and backfilling. The work involves making sure that the construction and production processes that have been developed within the different production-lines function together in a rational and efficient manner, based on the further developed integrated set of requirements for the different parts of the final repository.
- Database structures and procedures for the information flow in the execution process are developed and maintained. They are important tools for presentation and integration of inspection programmes within the different lines and for verifying that the repository has been designed in accordance with established design premises.

9.3.2 The fuel line

The characteristics of the spent nuclear fuel that influence the design of the Spent Fuel Repository and how its properties are to be determined are described in the fuel line. The design of the repository in turn imposes requirements on the choice of fuel assemblies for encapsulation and on the handling of the fuel. Parameters that must be determined by measurement or calculation or a combination thereof include radionuclide inventory, decay heat, radiation, criticality and quantity of water in the canister. The inventory of radionuclides is dependent on the fuel's enrichment, burnup and decay time. The quantity of uranium/HM (heavy metal) is also of importance. There may therefore be a great difference between fuel assemblies with different operating histories, but this variation is limited by the fact that the fuel assemblies in a canister may not have a combined decay heat in excess of 1,700 W. This makes it possible to calculate the consequences for long-term safety in a simplified manner without underestimating them. The fuel parameters that influence the handling of the fuel and are needed for safety assessment are also of importance for the safe and efficient operation of the nuclear power plants. Programs for calculating these parameters are therefore available today. SKB intends to use the same, or same type of, program.

Further development efforts mainly involve gathering knowledge as a basis for the design of the encapsulation process. In addition, work is needed to obtain more detailed knowledge about the fuel. The planned development work covers the following areas:

- Continued measurements of decay heat on individual fuel assemblies in Clab. The purpose is to obtain a better body of statistics. The goal is that a measurement method for determination of decay heat should be developed by the time the encapsulation plant begins to be built.
- Study different techniques for drying the fuel in Clink, including PWR fuel with control rods. The work includes inventorying possible techniques, choosing a method and planning the evaluation of the chosen method. The goal is to be able to decide on a method for drying prior to detailed design of Clink.
- Analyze the probability of criticality in canisters with odd fuel types, such as PWR fuel with high enrichment and low burnup, and study how such PWR fuel should be handled.

The plans to meet these needs are described in Chapter 10.

9.3.3 The canister line

The canister line should show that the chosen reference design for the canister is technically possible to achieve and that it fulfils the stipulated design premises. Development of the canister has passed the concept phase and parts are well along in the design phase.

Development of the different processes for fabrication of canister components, welding to seal the canister and inspection of components and welds has reached different stages. The copper components can be fabricated by means of several hot forming methods. SKB has fabricated components that meet requirements on dimensions, material structure and material composition. Welding of copper components by FSW (friction stir welding) has been carried out with high reproducibility and low occurrence of defects. Canister inserts for BWRs assemblies can be fabricated today with a quality that meets requirements on dimensions and material properties with ample margin. Development is under way to reach an equivalent quality level in the fabrication technology for PWR inserts. The methods for inspection of canisters are being further developed in various respects.

Development is also needed as a basis for more detailed acceptance criteria, industrialization of testing and inspection methods in connection with fabrication and handling in Clink, and development and studies of how the canister is handled and deposited. This includes adapting the processes for welding and nondestructive testing (NDT) of the seal weld to the radiation environment in Clink, plus establishing a testing procedure and verifying the reliability of ultrasonic NDT of canisters, as well as finding technology for detection and size determination of surface defects and deformations.

Further development work within the canister line therefore mainly includes:

- Canister design – analyses of the canister.
- Fabrication and testing of inserts.
- Fabrication and testing of copper components.
- Sealing and testing of the weld.
- Handling and deposition of canisters in the Spent Fuel Repository including development of ramp vehicle for underground transport of the canister to the transloading station in the central area and further development of the deposition machine.

The plans for this work are described in Chapter 11.

9.3.4 The buffer line

The buffer line should show that the chosen reference design for the buffer fulfils the design premises and is technically feasible. Development of the buffer has passed the concept phase. A reference design has thereby been established which fulfils stipulated design premises.

Studies show that it is possible to achieve density and material composition of compacted blocks and pellets within the intervals required according to the design premises. For extreme combinations of geometry of the deposition holes and density of buffer blocks and pellets, however, the water-saturated density of installed buffer may locally lie outside the limits of the acceptable density interval.

With conventional technology it should be possible to further reduce the variation range of density and material composition, which is deemed to be advantageous from a safety viewpoint, even if no formal requirements have been established. Further development work within the buffer line has the following primary goals:

- Within the framework of the chosen reference design, develop the design and the technology for installation of the buffer so that system design can be finalized before the start of construction and so that implementation can be initiated before integrated testing begins.
- Further develop the method and design the equipment for fabricating (pressing), machining and handling buffer blocks so that this is ready when system design of the production building commences.

In addition, there are a number of development questions that span the whole buffer-backfill-closure system. For example, buffer installation could be simplified by eliminating the bottom plate and buffer protection, provided the deposition sequence with backfilling is also modified. The following work is planned:

- Mine the outer section of the Prototype Repository at the Äspö HRL. The purpose of this full-scale demonstration is to gain further knowledge on how buffer, plug and backfill work.
- Study and evaluate alternative methods for deposition and backfilling sequences.
- Develop system and equipment for transport of buffer and backfill material between the production building and the deposition area in the repository.

The plans to meet these needs are described in Chapter 12. In addition there is a need for further model studies and experiments to gain a sufficient understanding of certain important processes in the buffer-backfill-plug system, see Chapter 24. These processes include: erosion of buffer before full water saturation has been achieved, plus water saturation, self-healing and homogenization in buffer and backfill. The goals are to determine the evolution of density and swelling pressure for the interacting buffer-backfill-plug system during the period between installation and repository closure, and to formulate more practically useful design premises for these system parts.

9.3.5 The backfilling line

The backfilling line should show that the chosen reference design for backfilling is technically feasible to achieve and that it fulfils the stipulated design premises. Development of backfilling has passed the concept phase. The expected very low inflows of groundwater in Forsmark should facilitate installation of the backfill.

Development work remains to be done for the design of plugs as well. This includes what tightness criteria are needed in order to ensure controlled swelling of the backfill and to prevent "piping erosion" in buffer and backfill.

In addition to the overall goals for the whole buffer-backfill-closure system presented in Section 9.3.4, the continued development work within the backfilling line has the following primary goals:

- Finish the design of the backfilling process within the chosen reference design so that system design can be finalized before the start of construction and so that implementation can be initiated before integrated testing begins.
- Finish design of the plug for deposition tunnels so that system design can be finalized before the start of construction and so that the implementation phase has been reached before integrated testing begins.

The plans to meet these needs are described in Chapter 13.

9.3.6 The closure line

With the exception of certain boreholes, repository closure will not begin until all spent fuel has been deposited. Detailed design and implementation of the closure technology will therefore not become urgent for another 50 years at least. The current reference design, which is based for the most part on the reference design for backfilling, will therefore probably be modified and simplified so that it meets the less stringent requirements that should apply to closure compared with backfilling. This includes both material composition and geometric configuration. The requirements that need to be made on rock excavation in parts of the accesses constitute an important exception. These need to be established before the start of construction to ensure that the design and production of the accesses permit an expedient closure.

In addition to the overall goals for the whole system of buffer, backfilling and closure presented in Section 9.3.4, the continued work within the closure line has the following primary goals:

- Further develop the reference design for repository closure, primarily to establish requirements on rock excavation in accesses before the start of construction. A new concept phase is completed. After revised reference design, development measures as a basis for system design will follow.

The plans to meet these needs are described in Chapter 14.

9.3.7 The rock line

The rock line should show that the chosen reference design of the Spent Fuel Repository's underground openings is technically feasible to achieve and that it fulfils the stipulated design premises. For a few design premises, uncertainties remain that require further technology development. Further development is also needed to translate the results of full-scale experiments and demonstrations to industrial processes.

Design and construction of the hard rock facilities will be done using the Observational Method. Application of this method needs to be further developed to ensure that the repository is site-adapted with respect to requirements on long-term safety and that the methodology is expedient for the Spent Fuel Repository. In order for this method to be used, a number of rules must be stipulated in advance regarding how geoscientific information should be interpreted and decisions made, for example regarding acceptance of intended deposition positions.

The most important input data for configuration of underground openings and site adaptation of the layout are the results of detailed characterization with associated modelling. Available technology for investigations and modelling needs to be further developed in certain respects. Detailed characterization will be carried out in steps prior to and during ongoing production of excavations, and coordination between these activities needs to be clarified. It must also be ensured that the needs of the safety analysis report are met.

The industrialization of the rock excavation process includes production methods that are adapted to the requirements made for rock excavation, stability and tightness. The engineering materials that are used must meet requirements with respect to long-term safety. The main requirement is on low pH for cementitious materials such as injection grout, shotcrete, bolt grout and structural concrete.

In the area of equipment, further development is needed of e.g. tunnel drill rigs with high drilling precision, grouting equipment, machines for rock support with wire mesh and the boring machine for deposition holes. Experience from experiments at the Äspö HRL show that present-day machines for rock excavation and grouting in accesses, above all tunnel drilling rigs and grouting equipment, have the potential to meet the requirements made for the final repository's accesses.

Development of plans for delivery inspections of building materials and execution and result inspections of the rock excavation works is important for verifying that stipulated requirements have been met. This is an integrated part of the industrialization of the rock construction process, which should be finalized when the excavation of the repository's accesses begin. Documentation of where and how the tunnels have been constructed will be important for e.g. future safety analysis reporting.

Further development work within the rock line has the following primary goals:

- Further develop the methodology for underground design and the application of the Observational Method, above all with regard to strategies for detailed adaptation of the deposition areas and coordination of detailed characterization and excavation work.
- Further develop methods and equipment for detailed characterization with associated modelling so that the detailed characterization programme can be executed in a rational and efficient manner from the start of construction.
- Further develop production methods adapted to the requirements made for rock excavation, stability and tightness. The primary goal is to be able to stipulate performance requirements in the construction documents for the accesses.
- Ensure that approved and duty-proven engineering materials are available in time for construction of the accesses.
- Ensure that the special machines that are needed – such as a tunnel drilling rigs with high drilling precision, grouting equipment and machines for rock support with wire mesh – are available in time.
- Develop the boring machine for deposition holes so that it is available when the first deposition area is built.
- Develop inspection plans, including format and procedures for documentation and for as-built plans, so that they are evaluated and finalized when the first rock excavation works start.

The plans to meet these needs are described in Chapter 15.

9.3.8 Production system and logistics

In the Spent Fuel Repository, many activities and material flows will take place simultaneously. An overall production system is needed in order to be able to plan, coordinate and control these activities in an efficient manner. Advanced production systems exist and are used in many industrial sectors. The similarities between the final repository and a mine are such that systems used in the mining industry are of particular interest to SKB.

The production system that SKB plans to use can be divided into five subareas, see Figure 9-2. The names and scope of the system's subareas have not yet been established, but will be developed during the continued work of designing facilities, systems and components.

Large quantities of materials and products will be handled in the final repository. This requires well functioning logistics. Tools for logistics simulation are therefore being studied and developed in parallel with the design of the facility. The purpose is to construct a model that can simulate the logistics of all activities at the final repository during the operating phase, i.e. for both buildout and deposition activities.

The logistics model is divided into submodels in the manner illustrated in Figure 9-3. Further breakdown will, however, be required to make the model manageable. The interfaces between the different submodels are important. The submodels can be developed at different rates as input data become available.

In order for the logistics model to be able to convey information to those who work with different parts of the activity in the facility, the software for logistics simulation must be able to use the 3D models of the facility and different vehicles and equipment that are constructed in the design process.

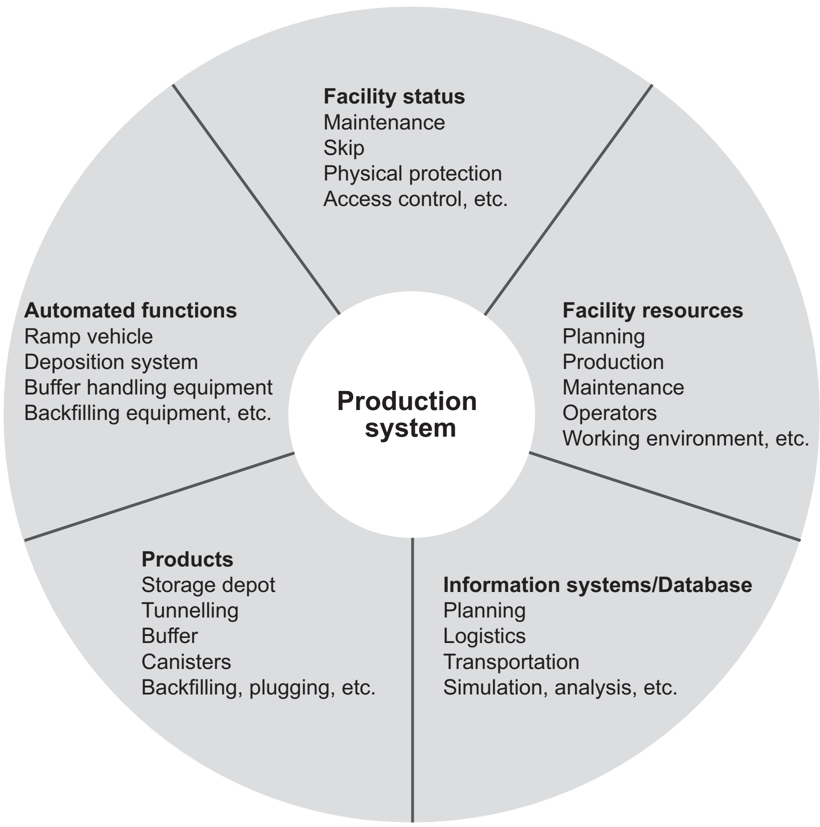


Figure 9-2. *Preliminary structure for the production system for the final repository.*

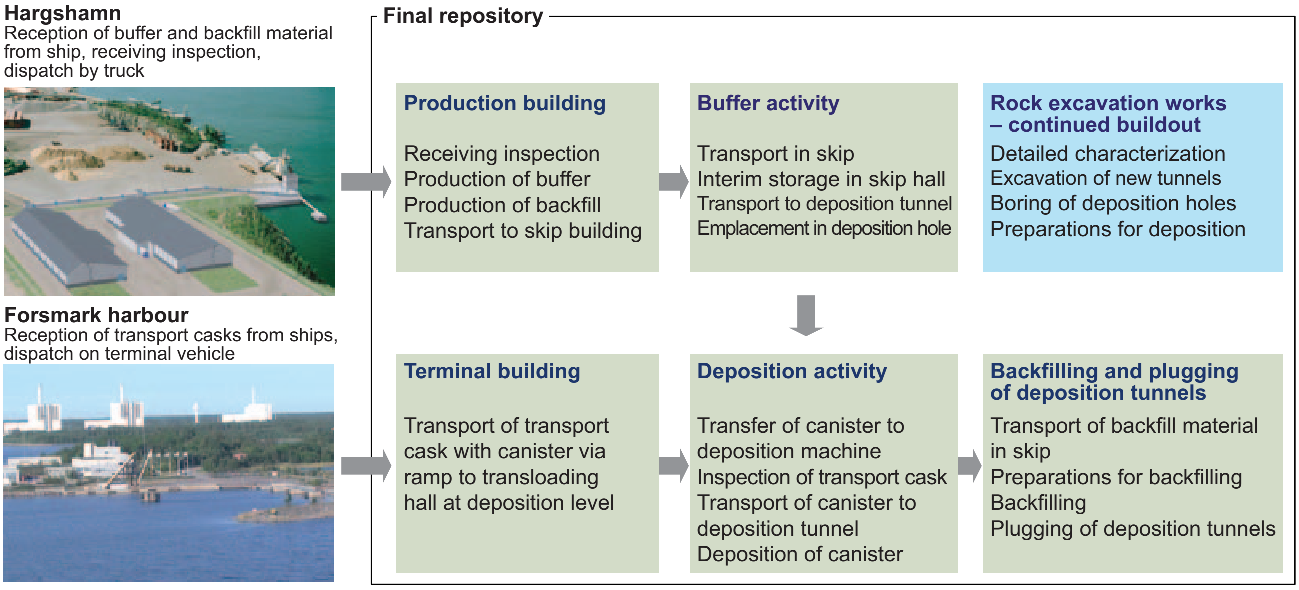


Figure 9-3. *Schematic illustration of the logistics model for the final repository.*

Software that could meet SKB's requirements on being able to both simulate and plan the activity was inventoried during 2009. Based on the inventory, one program was selected for further studies. Since then SKB has carried out a demonstration project to show that the selected programming tool has the potential to produce the desired results. After evaluation of the demonstration project, further logistics studies are planned in the steps and at the rate input data become available. The work will focus on analyzing flows of materials, machines, vehicles and personnel within the facility, structuring this information in a database, developing the basic model from the demonstration project, and clarifying interfaces between constituent submodels. The submodels can then be developed separately, but in the end it should be possible to simulate all activity at the final repository.

9.3.9 KBS-3H

Together with Posiva, SKB is studying whether horizontal deposition can constitute an alternative to vertical deposition. The work carried out during the period 2004–2007 was reported at the end of 2007 /9-2/. Based on the results achieved, SKB and Posiva decided to continue the development work with the main goal of developing the technology for KBS-3H to the point where it is possible at a later stage to demonstrate the technology on a full scale. The work was initiated in 2008 and the programme for the next few years includes the following main activities:

- Design of a KBS-3H repository.
- Demonstration at the Äspö HRL.
- Studies of key issues relating to long-term safety.

Plans for these activities are presented in Chapter 16.

9.3.10 Summary

Table 9-1 summarizes how far the technology development in different areas should have reached in relation to defined important milestones for the execution programme in accordance with Figure 8-1.

Table 9-1. Planned status of technology development prior to different stages of construction and commissioning.

Development need	Prior to start of construction	Prior to buildout of deposition areas – interconnection with Clink	Prior to integrated testing	Prior to start of operation
Overarching				
Design premises with regard to operational (pre-closure) safety and long-term (post-closure) safety.	Revised with respect to results from safety assessments.	Possible revision if new knowledge of importance for safety has come to light.		
Integrated set of requirements for fuel and canister concerning for example decay heat and water in the canister.	Ready in time for system design of the encapsulation plant.			
Further developed, integrated set of requirements for accesses, as well as for deposition holes and deposition tunnels.	Set of requirements revised in time for system design of the Spent Fuel Repository.	Set of requirements revised in time for detailed design of the deposition area.	Administration and improvement.	Administration and improvement.
System for information management in the execution process.	Detailed design finished.	Implementation finished.	Administration and improvement.	Administration and improvement.
Tool for production control and logistics.	System design finished.	Detailed design finished.	Implementation in progress.	Implementation finished.
The fuel line				
Measurement method for determination of decay heat and other fuel parameters.	Implementation finished at start of construction of the encapsulation plant.	Administration and improvement.	Administration and improvement.	Administration and improvement.
Method for drying fuel.	Method chosen prior to detailed design of Clink.	Design in progress.	Implementation in progress.	Administration and improvement.
Criticality analyses for PWR fuel with high enrichment and low burnup.	Chosen alternative for handling of such fuel is described in the application for burnup credit for Clab.	Implementation in progress.	Administration and improvement.	Administration and improvement.
The canister line				
Design and analysis of the canister.	Detailed design finished.	Implementation in progress.	Administration and improvement.	Administration and improvement.
Methods and equipment for fabrication and testing of inserts.	Detailed design finished.	Implementation in progress.	Implementation in progress.	Administration and improvement.
Methods and equipment for fabrication and testing of copper components.	Detailed design finished.	Implementation in progress.	Implementation in progress.	Administration and improvement.
Methods and equipment for welding and NDT of welds.	System design finished.	Detailed design finished.	Implementation in progress.	Implementation finished.
Encapsulation process with nuclearization, including weld and NDT.	System design finished.	Detailed design finished.	Implementation in progress.	Implementation finished.
Handling and deposition of canisters in the Spent Fuel Repository.	System design finished.	Detailed design finished.	Implementation in progress.	Implementation finished.
Special machines: Ramp vehicle and deposition machine.	System design finished.	Detailed design finished.	Implementation in progress.	Implementation finished.
The buffer line				
Further development of the buffer within selected reference design.	System design finished.	Detailed design finished.	Implementation in progress.	Implementation finished.
Method and equipment for fabricating bentonite blocks and pellets.	System design finished.	Detailed design finished.	Implementation in progress.	Implementation finished.
Full-scale demonstration of function of buffer and backfilled tunnel.	Mining of Prototype Repository finished.			
Alternative methods for deposition and backfilling sequences.	Concept phase finished. Possible decision on changed reference design.			
Quantitative description of evolution of density and swelling pressure for the interacting buffer-backfill-plug system for the time between installation and repository closure.	Input data for possible revision of design premises for backfill and plug ready.			
Transportation system for buffer and backfill material.	System design finished.	Detailed design finished.	Implementation in progress.	Implementation finished.

Development need	Prior to start of construction	Prior to buildout of deposition areas – interconnection with Clink	Prior to integrated testing	Prior to start of operation
The backfilling line				
Further development of backfilling within the chosen reference design.	System design finished.	Detailed design finished.	Implementation in progress.	Implementation finished.
Plug for deposition tunnel.	System design finished.	Detailed design finished.	Implementation in progress.	Implementation finished.
The closure line				
Reference design and installation method for closure.	Requirements on rock excavation in accesses established.	System design started.	System design in progress.	System design finished.
The rock line				
Methodology for underground design.	Methodology for accesses ready for implementation.	Methodology for deposition area ready for implementation.	Administration and improvement.	Administration and improvement.
Tools for detailed characterization.	Instruments and methods for investigations in accesses ready to be implemented.	Instruments and methods for investigations in deposition area ready to be implemented.	Administration and improvement.	Administration and improvement.
Execution methods, building materials and special machines.	Methods etc. for accesses ready to be implemented.	Methods etc. for deposition area ready to be implemented.	Administration and improvement.	Administration and improvement.
Boring machine for deposition holes.	System design finished.	Detailed design finished.	Implementation in progress.	Implementation finished.
Tools for data management and visualization.	Detailed design finished.	Implementation finished.	Administration and improvement.	Administration and improvement.

10 Technology development, fuel handling

This chapter deals with the technology development that is planned for handling of the spent nuclear fuel in accordance with the requirements imposed by final disposal. Issues of importance for long-term safety that concern the fuel after closure of the repository, for example fuel dissolution, are discussed in Chapter 22.

10.1 Requirements and premises

The spent nuclear fuel consists of fuel assemblies from the twelve Swedish nuclear reactors. Figure 10-1 shows fuel assemblies from a BWR reactor (BWR assemblies) and a PWR reactor (PWR assemblies). In addition to the fuel from the twelve reactors, a small quantity of odd fuels from the early part of the Swedish nuclear power programme, as well as from research, must be disposed of.

The properties of the spent nuclear fuel influence the design of the Spent Fuel Repository. The design of the repository in turn gives rise to requirements on the choice of fuel assemblies for encapsulation as well as on the handling of the fuel. Properties of the fuel that give rise to requirements on handling are:

- enrichment,
- burnup,
- decay time (time since the assembly was removed from the reactor).

Enrichment and burnup affect the fuel's reactivity and the probability of criticality. The fuel's burnup and decay time, along with the quantity of uranium, determine its radionuclide inventory and radioactivity. Radioactivity is the source of radiation and heat output in the fuel. The fuel's reactivity, heat output and radiation impose requirements on handling. In addition to the fuel parameters that impose requirements on handling, certain properties of the fuel need to be known for the analyses included in the safety analysis report. These properties include quantities and composition of engineering materials and radiation history, which affects the radionuclide inventory, see also Sections 22.1.2–22.1.4.

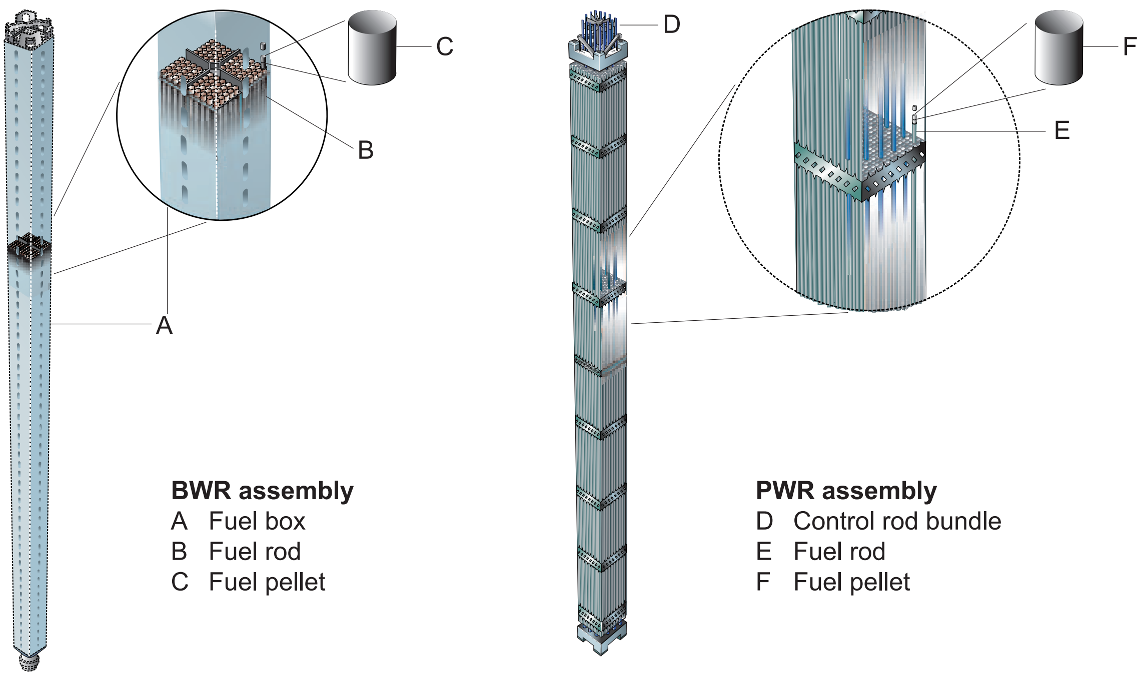


Figure 10-1. Fuel assembly from a BWR reactor (left) and a PWR reactor (right).

All fuel parameters that affect the handling of the spent nuclear fuel also need to be known for the operation of the nuclear power reactors. The fuel suppliers and the reactor owners have therefore developed calculation and verification programmes for the parameters. The programmes are quality-assured and approved for their purposes by the authorities.

In addition to the fuel parameters mentioned above, the quantity of water and water vapour remaining in the canister when it is sealed imposes requirements on handling. Since water and water vapour can form nitric acid via radiolysis and cause corrosion, the amounts in the canister must be limited.

Requirements related to the design and long-term safety of the Spent Fuel Repository

Decay heat

The fuel's decay heat is dependent on its burnup and decay time. The decay heat in a fuel assembly is also dependent on the quantity of uranium. The decay heat affects the temperature in the final repository. The current requirement is that the temperature in the buffer may not exceed 100 degrees. The fuel assemblies enclosed in a canister must therefore be chosen with a view to their burnup and decay time so that the total decay heat in the whole canister is below the maximum acceptable level. This level is affected by the thermal conductivity of the canister, the buffer and the rock and is 1,700 W according to the reference design, see also Section 22.2.3.

Radiation

Radiation on the outside of the canister can lead to corrosion of the copper canister. Analyses show that the corrosion process can be neglected if the surface dose rate on the canister does not exceed 1 Gy/h /10-1/. The radiation, like the decay heat, is dependent on the fuel's burnup and decay time, see also Section 22.1.2. The radiation that reaches the outside of the canister is also affected by the canister's radiation-attenuating capacity. For combinations of fuel assemblies that can be accepted for encapsulation with respect to the decay heat requirement, the dose rate on the surface of a canister according to the reference design lies well below the acceptable level. During handling it should be verified that the canister's surface dose rate does not exceed the acceptable level.

Water and water vapour

In order to prevent corrosion of the insert in the canister, the quantity of water and water vapour in the canister must be limited and most of the air present in the canister must be replaced with an inert gas. The atmosphere in the canister must therefore be replaced with argon (> 90 percent) and the fuel assemblies dried so that the quantity of water in the insert does not exceed 600 grams, see Sections 11.1.3, 22.1.5 and 23.2.2.

Criticality

Criticality may not occur in the canister under any circumstances. The fuel's reactivity and the probability of criticality are dependent on its enrichment and burnup, as well as on the fuel geometry and the materials surrounding the fuel. For a given enrichment, geometry and environment, the probability of criticality declines with the burnup. In order to prevent criticality, fuel assemblies for encapsulation must be chosen with a view to their enrichment and burnup, as well as the canister's geometry and materials, so that criticality cannot occur during handling or after deposition, even if the canister is water-filled. The requirement that is made is that the effective multiplication constant (k_{eff}) for the fuel placed in the canister, including uncertainties, must be less than 0.95, see also Section 22.2.4.

Requirements related to the operation of the KBS-3 system

Encapsulation

It is an advantage if the canisters can be filled to their maximum capacity. Fuel assemblies for encapsulation should therefore to minimize the number of canisters and so that all fuel positions in the canisters are filled, at the same time as the requirements related to criticality and decay heat are complied with. Furthermore, fuel assemblies should be chosen so that the number of lifts and moves is minimized.

Operational safety and radiation protection

With a view to radiation protection and operational safety, the activity content of the canister and the radiation on its surface need to be known. Both activity content and radiation are limited by the maximum permissible decay heat. In conjunction with encapsulation, it must be verified that activity content and radiation do not exceed the levels used as premises in the report on operational safety.

Safeguards

SKB must comply with safeguards requirements from both Swedish authorities and international inspection bodies. After encapsulation it is no longer possible to inspect individual fuel assemblies; instead, each canister will comprise a unit for safeguards inspection. This means that each canister must be assigned a unique identity.

10.2 Current situation and programme

SKI pointed out in the review of RD&D Programme 2007 that SKB should explain how it is ensured that data on the spent nuclear fuel is correct before it is encapsulated. If the documentation is incomplete or has deficiencies, SKB should present a plan of action to deal with this. It should also be indicated when SKB needs to know that sufficient information exists on the fuel.

The fuel parameters that affect the handling of the fuel in the facilities and are needed for assessment of operational and long-term safety are also of importance for the safe and efficient operation of the nuclear power plants. The parameters are reported by the NPPs prior to transport to Clab. SKB intends to calculate radionuclide inventory, radiation and decay heat with the same, or the same type of, program as that used by the NPPs. If necessary, SKB can carry out verifying measurements in conjunction with delivery to Clab. SKB is developing a special database for documentation of the fuel's properties.

The following sections present SKB's development plans within the areas relevant to the fuel line. Radionuclide inventory is further discussed in Section 22.1.2 in Part IV.

10.3 Decay heat and radiation

Decay heat must be calculated for each fuel assembly. If necessary, for example if there are uncertainties in the data reported by the NPPs, the calculations are supplemented by verifying measurements.

SKB has followed the development of calculation programs for decay heat since the mid-1990s and has developed methods for supplementary measurements. Calorimetric measurements have been used for accurate determination of decay heat. These measurements take a long time, and in view of the large number of fuel assemblies a faster method has also been developed – gamma scanning. In gamma scanning, the gamma radiation from the fission product cesium-137 is measured. The radiation intensity from cesium-137 exhibits a nearly linear relationship with the decay heat. The method can be used for fuel assemblies with both short and long decay time.

The methods for measurement of decay heat have been tested on fuel assemblies in Clab. The results are summarized in /10-2/ and show small differences between the two measurement methods. The work has been conducted in part together with Oak Ridge National Laboratory, which develops calculation software. The results have also been used to validate and further develop the calculation programs. Since decay heat and radiation are dependent on the same parameters, the same calculation programs and methods can be used to determine and verify them.

Programme

For the purpose of obtaining a better body of statistics and to include new fuel types, the measurements of decay heat on individual fuel assemblies in Clab will continue for a number of years to come. SKB's goal is that a measurement method for determination of decay heat and other relevant fuel parameters should be available when construction of the encapsulation plant begins.

10.4 Water and water vapour

During the design of the encapsulation plant, two methods for drying of fuel have been studied: vacuum drying and drying with hot air. Vacuum drying is the reference method. Both methods have been found to satisfy the requirements on drying of undamaged fuel. Regarding damaged fuel, uncertainties exist as to whether the ability of the methods to expel the water through pinholes in the fuel cladding is sufficient. Furthermore, uncertainties exist regarding their ability to expel water efficiently from the PWR fuel's control rods, see Figure 10-1.

Programme

SKB has started a project for the purpose of studying different drying techniques. The project involves a market survey, choice of method and a plan for verification and validation of the drying method, plus a plan for qualification of the equipment. SKB intends to make a decision on a method for drying before the detailed design of Clink.

10.5 Criticality

In the criticality safety analyses for transportation of spent nuclear fuel to Clab, no credit has been taken for the decrease in reactivity that occurs due to burnup of the fuel when it is irradiated in the reactor (burnup credit). For storage in Clab, no burnup credit has been used for enrichments up to 4.2 percent. However, for interim storage of PWR fuel with an enrichment of 5 percent, and for the encapsulated spent nuclear fuel, burnup credit must be used to show that the criticality requirements have been met. The method used by SKB for burnup credit follows a method developed by Oak Ridge National Laboratory. All currently known fuel types in the Swedish nuclear power programme have been analyzed together with information on the canister in accordance with the reference design /SKBdoc 1193244/.

Analyses for the reference canister and all currently known fuel types show that fuel that has been used normally in the nuclear reactors has a burnup that permits it to be placed in the canister from a criticality safety viewpoint with ample margin. The analyses include material defects in the canister insert. PWRs fuel with high enrichment and low burnup does not meet the requirement on criticality safety, however. At sufficiently low burnup, this applies even if only one fuel assembly is placed in the canister. Such low burnup can only occur due to severe fuel damage or unplanned premature final shutdown of a nuclear power plant.

Programme

SKB intends to supplement completed criticality analyses by determining which geometric configuration in the canister is most reactive. The results can be used to estimate the probability of criticality in connection with deviations in the geometries of the canister and the fuel.

SKB will also study how PWR fuel that does not satisfy the criticality conditions should be handled. There are a number of different ways to store and handle such fuel in Clab, and to encapsulate it. The alternative chosen by SKB will be presented in conjunction with the application for burnup credit for Clab.

10.6 Safeguards

Safeguards enable regulatory authorities and inspection bodies to ensure that nuclear material is not diverted. SKB's facilities must comply with the requirements that are made on safeguards by both Swedish regulatory authorities and international inspection bodies. This means that there must be an administrative system for accounting of nuclear material and where it is located, plus technical systems for inspection and supervision that it is not diverted.

In the case of encapsulated fuel, the safeguards system will contain information on the individual canisters' content of nuclear material, which fuel assemblies the canisters contain, when the fuel was encapsulated, transported and arrived at the Spent Fuel Repository, where the canisters are deposited, and the total quantity of nuclear material in the repository. In other words, it must be possible to identify individual canisters and their content.

Safeguards that meet the requirements can employ conventional technology and do not require any technology development. In the case of new nuclear facilities, safeguards must be taken into account in the design stage so that inspection and supervision are facilitated. An important component of the nuclear safeguards system in the Spent Fuel Repository is being able to verify that the facility has been built in accordance with approved drawings. This is done so that the inspection bodies can ensure that there are no routes out of the facility that have not been indicated and that there are no areas where activities, other than those indicated, are carried out.

SKI pointed out in its review of RD&D Programme 2007 that SKB states that visual verification of the fuel assemblies will take place before the steel lid is lifted onto the canister, but without explaining how this verification will be done or documented. SKI regarded this as a critical point in the handling, since SKB switches here from handling individual fuel assemblies to regarding the canister as the smallest unit.

SKB considers that verification can be done by using several inspectors and/or by photographing. Inspection records and photographs can be kept in accordance with stipulated requirements. SKB described the planned safeguards within the KBS-3 system in the supplement to the application for Clink that was submitted to SSM in October 2009. The spent fuel's content of nuclear (fissionable) material is dependent on its enrichment and burnup and can be calculated with the same program as other fuel parameters that must be known. The content of nuclear material can thus be determined and verified at the same time as the decay heat, see Section 10.3.

11 Technology development, canister

The canister's purpose and function is to contain the spent nuclear fuel and prevent the escape of radionuclides to the environment. The canister should also attenuate radiation and prevent criticality.

This chapter deals with the technology development being conducted by SKB to fabricate, seal, transport and deposit the canister that will be used. The chapter also deals with the analyses that are performed to verify the function of the canister and determine what requirements should be made on fabrication and inspection of the canister and on systems for transport and handling of the sealed canister.

SKB's reference canister is a cylindrical container with an impervious shell of copper and a loadbearing insert of nodular iron in which spent nuclear fuel is placed, see Figure 11-1. The insert is available in two variants: one that holds twelve fuel assemblies from BWR reactors and one that holds four fuel assemblies from PWR reactors. SKB has also chosen reference methods for fabrication of the canister's components and for welding and sealing. The copper tube is fabricated by extrusion, copper lids and bottoms are forged and the insert is cast. The copper bottom is welded on and the canister is sealed by friction stir welding (FSW).

Development of the canister has passed the concept phase and parts are already well along in the design phase. Section 11.2 provides an overview of the current situation and programme, followed by a table that also summarizes the comments of the regulatory authorities on the account given by SKB in RD&D Programme 2007. The work with detailed design of the canister is then described in Section 11.3, fabrication of canister components in Sections 11.4–11.5 and welding in Section 11.6. Furthermore, the development work prior to nuclearization of testing and inspection methods in connection with sealing and handling in Clink is described in Section 11.6, and the development of methods for handling and deposition of the canister in the Spent Fuel Repository in Section 11.7.

11.1 Requirements and premises

SKB stated in RD&D Programme 2007 that the design premises that were presented were not complete, and in its review the authority concluded that "SKB needs to continue to develop the design premises so that they can provide better input data for the choice of materials, design and inspection of the canister".

SKB has continued the work of specifying design premises for the canister. An important basis for this is the list of design loads and detailed specifications of what requirements the canister must meet in the Spent Fuel Repository which SKB has compiled as a result of the work with the most recent assessment of the repository's long-term safety, SR-Can /11-1/, and the analysis of the canister's strength and damage tolerance (design analysis) which SKB has now completed /11-2/. In addition to the analysis of loads in the repository, an analysis has been made of the loads to which the canister is subjected in the facilities and during transport, which has also involved calculating loads on the canister during lifts by the lid.

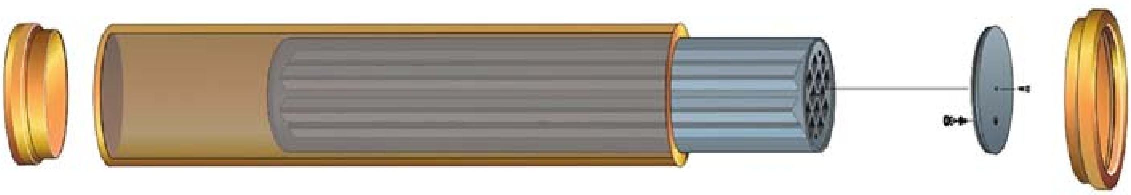


Figure 11-1. Exploded drawing showing the components of the reference canister. From left: copper bottom, copper tube, insert and copper lid.

The design premises for the canister relate to:

- its barrier function in the Spent Fuel Repository,
- the spent nuclear fuel to be encapsulated,
- production of canisters and encapsulation,
- operation of the KBS-3 system.

Provided that operation of the facilities and the transportation system is normal, it should be possible to deposit the canisters in a safe way without affecting the properties that are important for the barrier function in the Spent Fuel Repository. This entails that acceptance criteria will be needed for repository operation, for example for the highest permissible temperature on the canister surface, indentations or scratches on the copper shell and presence of chemicals. The work of specifying these criteria is under way.

11.1.1 Barrier function in the Spent Fuel Repository

In the Spent Fuel Repository, the canister is supposed to enclose the spent nuclear fuel and prevent the escape of radioactive substances. This means that:

- The canister shell must resist the corrosion to which it is exposed in the Spent Fuel Repository. The copper in the shell should therefore have a nominal thickness of five centimetres and it should be of high quality to prevent intergranular corrosion. Oxygen concentrations of a few tens of ppm can be permitted.
- The canister must withstand an isostatic load of 45 MPa, which is equal to the sum of the maximum swelling pressure that is expected in the buffer and the maximum groundwater pressure at repository depth. As a result, properties with a bearing on the strength of the insert are important, as is the creep ductility of the copper shell.
- The canister shell must be leaktight and the loadbearing capacity of the insert must remain intact after a shear movement in the rock of five centimetres with a velocity of up to one metre per second. This means that the insert's fracture mechanical properties and dimensions are important, as is the copper shell's elongation at break and fracture toughness.
- The canister must also prevent criticality, which means that the insert must be designed so that criticality cannot occur even if water has entered a defective canister. Verification of safety against criticality is based on a material composition of the nodular iron of $Fe > 90$ percent, $C < 4.5$ percent and $Si < 6$ percent, plus a fuel configuration that is determined by the geometry of the reference canisters.
- The canister must contribute to keeping the radiation on the surface below 1 Gy/h.

11.1.2 The spent nuclear fuel

The canister must be able to hold the different types of spent nuclear fuel included in the Swedish nuclear fuel programme. To permit safeguards inspection, each canister will have a unique marking that can be read when the canister has been placed in the deposition hole.

11.1.3 Production of canisters and encapsulation

It must be possible to fabricate and seal canisters with high reliability. It must also be possible to inspect them to ensure that they fulfil established criteria.

At encapsulation the following must be observed:

- Fuel assemblies must be chosen so that criticality in the canister is prevented.
- The atmosphere in the insert must be replaced with > 90 percent argon and the fuel assemblies dried so that the quantity of water in the insert does not exceed 600 grams. This limits the amount of nitric acid that can form in the canister and affect the insert's long-term function.
- The radiation level on the outside of the canister should be less than 1 Gy/h. If this radiation level is not exceeded, corrosion of the copper shell due to nitric acid formation outside the canister will be negligible.

The current situation and the programme that concerns these three points is presented in Chapter 10.

11.1.4 Operation of the KBS-3 system

It must be possible to transport, handle and deposit the canister in a safe manner without the properties of importance for its barrier functions in the Spent Fuel Repository being significantly affected.

The properties of the canister materials have been investigated at different temperatures, and the creep properties of copper in particular have been investigated at up to 175°C without any sign of material degradation. Both copper and nodular iron retain important properties such as failure strength and yield strength at temperatures of up to 125°C. During handling, the outside temperature of the canister should therefore be limited to 100°C, which allows ample margin to avoid adverse material impact.

11.2 Current situation and programme

Barrier function in the Spent Fuel Repository

SKB has performed deterministic analyses of the strength of the insert and thereby verified that the insert for BWR fuel meets the strength requirements. The analyses have been performed assuming the material properties that have been achieved in fabrication under production-like conditions. Analyses of damage tolerance show that the insert possesses high damage tolerance under isostatic loading. An updated probabilistic analysis of an isostatic load of 45 MPa shows a negligible risk that a canister with a BWR insert will collapse. Damage tolerance is much lower under shear load, and calculations show that fairly small surface defects can initiate fracturing. Preliminary analyses of the PWR insert show that it also meets the strength requirements, but representative material data from serial production are needed to complete the analyses.

The analysis of the strength and damage tolerance of the PWR canister will be supplemented with deterministic analyses for the different loading cases strength and a probabilistic analysis for an isostatic load of 45 MPa. In addition, probabilistic analyses are planned of the shear load case for both variants of the canister.

It has been shown by creep modelling that the integrity of the copper shell is maintained under the loads in the repository. Tests performed on cold-deformed material show that the creep ductility of the copper material decreases with increasing cold deformation. Modelling and tests show that handling damage can give rise to relatively extensive cold deformation.

One area where continued work is planned is the technical specifications for the insert. The extensive strength analyses provide input data for evaluation of the specifications, which are either verified or, if reason exists, are modified or supplemented.

Production of canisters and encapsulation

A development programme for fabrication of BWR inserts has been carried out. To verify that the goal of the programme had been achieved, a demonstration series of five inserts was fabricated under production-like conditions and evaluated. Material data from this evaluation were used to verify the strength of the BWR insert by calculation. Further development of the fabrication technology for PWR inserts has been given priority and is taking place in parallel at two foundries.

The fabrication process for copper tubes has been further developed and the geometric accuracy of extruded tubes has been improved. What remains is to reduce the variation in grain size. This microstructural variation does not constitute a limitation from the viewpoint of strength and long-term integrity, but could affect the reliability of quality inspections by ultrasonic testing. To address this problem, both simulation of the extrusion process and laboratory-scale extrusion tests are being carried out. The results of these activities will provide guidance on what changes can be made in the fabrication process to mitigate the problem.

SKB has shown that the weld metal has virtually equivalent chemical and mechanical properties to the parent metal. In order to reduce or eliminate the defects that sometimes occur in the weld metal in the form of oxide inclusions or so-called joint line hooking, SKB has further developed process control, the welding system and the welding tool. The occurrence of oxide can be eliminated by welding under shielding gas, and welding trials have been performed with good results.

The development work for the welding process involves several organizations: The Center for Friction Stir Processing in the USA is working with tool development, Lund University with process control and ESAB with development of the welding system. Development of welding is being led by SKB, and the results are being implemented in the Canister Laboratory's welding system, where evaluation of the developed technology is also carried out.

Work is under way to formulate acceptance criteria for the canisters. Regarding the inserts, these criteria are largely based on damage tolerance analyses. Certain questions need to be further analyzed in order for SKB to determine the detailed set of requirements regarding defects and thereby determine the underlying requirements on nondestructive testing (NDT). The situation for the copper shell is different, since the design requirements are linked to the material's creep ductility and the thickness of the intact copper barrier. In terms of production, the limits for the copper material stipulated in the technical specifications have been achieved.

Development of the technology for NDT continues at the Canister Laboratory. The goal is to develop and test the technology for testing of the canister's components and welds in order to be able to ensure that the acceptance criteria regarding defects are fulfilled.

SKB uses ultrasonic testing that is adapted to detect the kinds of defects that might occur. This means that the testing is adapted with respect to the range of sizes, positions and orientations the defects can be expected to have. Only a few defects have been detected in the fabricated copper components. Better data on possible material defects are available for the FSW welds, since it has been possible to push the welding process outside the intended process window, whereby various types of defects have been created.

The fact that SKB regularly tests canister components and welds by NDT not only contributes to more efficient process development, but also provides valuable experience of testing systems, design of testing methods and testing procedures, and methods for interpreting and evaluating the test results. The benefits of feedback between testing and fabrication can be illustrated by an example involving the casting technology for inserts. By developing a method for ultrasonic measurement of the position of the outer channel tubes in the BWR insert, it has been possible to determine the position of the tubes along the length of the entire insert. Testing showed that the tubes were bent in some cases, a phenomenon that could be traced to a certain procedure in the casting process. By modifying the casting process, an important parameter for the strength of the inserts could be assured.

In the case of the nodular iron inserts, testing of the outer parts has been prioritized, since they are of great importance for strength, while testing methods for checking the homogeneity of the inner parts of the insert will be developed later.

Handling and deposition of canisters

SKB has conducted a feasibility study of solutions for shipments of canisters from the Spent Fuel Repository's operations area on the surface to repository level. The feasibility study shows that a concept where a transport cask (KTB) with canister is transported on a separate loadbearing transport frame provides efficient handling and facilitates salvage in the event of an accident on the ramp.

A new deposition machine has been developed in a prototype version. The machine has been tested in manual operation at the Äspö HRL. In parallel with these tests, the machine has been equipped with an advanced navigation and positioning system.

Summary of current situation and programme

Table 11-1 summarizes the regulatory authorities' comments on the account of the development programme for the canister provided by SKB in RD&D Programme 2007, as well as the current situation and programme for the development work.

Table 11-1. Summary of the regulatory authorities' comments on the account of technology development for the canister in RD&D Programme 2007, SKB's current situation, and SKB's programme.

Regulatory authorities' comments on RD&D Programme 2007	SKB's current situation	SKB's programme
Canister design – analysis of the canister		
Lack of complete design analysis.	Completed design analysis for the copper shell.	Analyze effects of local surface deformations.
	Completed strength and damage tolerance analysis for BWR inserts and preliminary analyses for PWR inserts.	Carry out final strength and damage tolerance analysis for PWR inserts with relevant material data.
	Completed probabilistic analysis of isostatic load case for BWRs.	Carry out probabilistic analysis of isostatic load case for PWRs, and of shear load case for both BWRs and PWRs.
Lack of requirements on materials and geometric tolerances and defects of importance for fabrication.	Identified requirements for position of channel tubes.	Revise specifications for the inserts based on the completed and supplemented (see above) design analysis.
	Investigated cold working effects that can occur in connection with fabrication and handling damage.	Establish acceptance criteria for cold working and other defects in connection with fabrication and handling.
Fabrication of inserts		
SKB has fabrication methods that need to be further developed.	Further developed fabrication process for BWR inserts and verified this with a demonstration series.	
	Carried out factor tests for different parameters in casting of PWR inserts and developed the method at two foundries.	Optimize the process and verify that it satisfies the requirements with a demonstration series for PWR inserts at a foundry.
Nondestructive testing (NDT) of inserts		
Development and qualification of several supplementary testing methods is important.	Developed ultrasonic technology for testing of the insert's most loaded outer part (< 200 millimetres) and inspection of the distance of the channel tubes to the surface of the insert.	Further develop method for NDT of the insert's outer areas by ultrasonic technique and supplementary technique for detecting surface defects.
	Ultrasonic technique has been used for testing of about 20 inserts.	Develop ultrasonic technique for NDT of the insert's inner part.
	Reliability has been studied.	Develop acceptance criteria for testing.
Fabrication of copper components		
SKB has fabrication methods that need to be further developed.	Improved shape accuracy of extruded tubes by modification of the process system.	Carry out a demonstration series for verification of achieved results.
	Causes of structural irregularity in extruded copper tubes have been analyzed by simulation and laboratory tests.	Finish laboratory studies and implement the results in the fabrication process. Verify that the process meets the requirements by means of a demonstration series.
	Cold working effects in forging of copper lids have been studied. Method for stress-relief annealing has been tested.	Implement stress-relief annealing in the fabrication process for lids/bottoms.
	Developed the forging technology for reduced grain size variation.	Demonstrate developed forging technology and method for stress-relief annealing of lids and bottoms.
	Improved the forging process and determined process parameters based on factor tests for forging of copper lids.	Carry out and evaluate a demonstration series to verify technology development for forging of lids and bottoms.
	Further developed the fabrication of tubes by pierce and draw processing and forging.	Further develop the fabrication of tubes by pierce and draw processing and forging.
Nondestructive testing (NDT) of copper components		
	Implemented preliminary ultrasonic testing methods. Evaluated detection capability and reliability.	Further develop technology for ultrasonic testing. Study and characterize possible defect types and formulate requirements for their detection and size determination. Develop technology for surface inspection.

Regulatory authorities' comments on RD&D Programme 2007	SKB's current situation	SKB's programme
Seal welding		
SKB must show that defects such as oxide particles can be eliminated and joint line hooking can be overcome.	Improved exactness in clamping of the canister and process control so that joint line hooking can be controlled. Optimized the design of the welding tool.	Optimize the control system and automate testing. Verify that the process meets operational requirements and that the weld metal meets requirements on properties.
	Demonstrated by trial welding that the welding process and the welding system are robust and stable.	Gather data for system in encapsulation plant.
	Started testing of welding in shielding gas.	Finish process for welding in shielding gas to minimize oxide formation.
	Microstructural investigations of the weld metal are being conducted.	Perform verifying microstructural investigations of the weld metal with respect to e.g. oxides and joint line hooking.
Nondestructive testing (NDT) of seal welds		
	Implemented ultrasonic testing for inspection of welds (tested approximately 70 welds). Evaluated detection capability and reliability in testing.	Formulate detailed acceptance criteria for defects and further develop ultrasonic testing, if necessary.
		Develop technology for surface inspection of welds.
Handling and deposition of canisters		
An account is desired of how loading, unloading and deposition of the canister will be done.	Completed feasibility study of transport solutions. Developed deposition machine with navigation and positioning system.	Conduct full-scale tests with the deposition machine at the Äspö HRL.
		Clarify the requirements on ramp vehicles and prepare documentation as a basis for procurement.

11.3 Canister design – analyses of the canister

In its review of RD&D Programme 2007, SKI emphasized that "... there is still lacking a complete design analysis of the canister including canister dimensions and a compilation of the strength analyses performed with reference to current design premises. SKB should include such an integrated design analysis of the canister including safety margins when applying for a licence to build the final repository." Furthermore, the authority pointed out that "certain remaining requirements on the different materials of the canister are lacking as well as certain final geometric tolerances which are important for the fabrication of the canister. Information is also lacking on the maximum permissible defects in different components of the canister, which are important for fabrication inspection."

SKB has carried out strength analyses and damage tolerance analyses (design analysis) for the canister in order to verify that the reference canister fulfils requirements and design premises /11-2/.

The strength analysis of the BWR insert with steel lid shows that it has large safety margins at an isostatic load of 45 MPa. For this load case relatively large defects can be permitted in the material in the cylindrical part of the insert and some geometric displacement of the insert's steel cassette can be tolerated. The plastic deformations and creep deformations in the copper shell are generally small (less than 1 percent). The calculations show that large strains (up to 30 percent) can occur locally in the end of the joint line between lid and cylinder. The global strain in the area is much less (12 percent), which prevents propagation.

The design analysis shows that shear load in combination with glaciation would not be more serious for the canister insert than shear alone. The analyses show that the reference canister can withstand a shear movement of 5 centimetres. The insert is the canister component which in this case is the most sensitive to surface defects. For the "least favourable" case with respect to buffer density, angle of attack and point of attack, semi-elliptical surface cracks (ratio length:depth = 6:1) with a depth of just over 4 millimetres can be permitted. A shear movement of 5 centimetres perpendicular to the canister results in a maximum plastic shear in a small local area of the copper shell of just over 20 percent. This is the same order of magnitude as for slow creep in the final repository. Further creep after a shear is expected to result in only a small addition (in the order of a couple of percent) to the total deformation. This creep relaxes the stresses that have arisen in the copper shell during the shear.

Programme

Verification that the reference canister meets the stipulated requirements will continue, with further analyses of the canister in BWR and PWR versions. The results of these analyses will serve as a basis for modification of e.g. acceptance criteria for nondestructive testing of inserts.

The deterministic and probabilistic analyses that have been done to demonstrate the strength of the BWR insert under isostatic load will be supplemented with equivalent analyses for the PWR insert as soon as representative material data are available from the demonstration series which SKB plans to carry out for PWR inserts.

The deterministic analysis that has been done for BWR inserts includes damage tolerance in connection with a shear movement in the rock and will be supplemented with a probabilistic analysis. Equivalent deterministic and probabilistic analyses will be carried out for PWR inserts. The extensive strength analyses provide data for evaluation of the established fabrication specifications, which are either verified or, if reason exists, are modified or supplemented.

The analyses of the creep properties of the copper shell will continue as a basis for further development of the creep model (see also Section 23.2.3). This work includes further studies of the influence of cold working on the creep properties of copper and implementation of this in the creep analyses. Cold working effects include the effects of local damage that can occur during handling of canister components or the finished canister.

11.4 Fabrication and testing of inserts

11.4.1 Fabrication

The insert is the canister's loadbearing component and is fabricated of nodular iron. The insert contains a steel cassette that creates the channels where the fuel assemblies are placed and is fitted with a steel lid. The inserts are cast and then machined to their final dimensions.

SKB has been developing the fabrication of inserts in cooperation with suppliers for a long time. In its review of RD&D Programme 2007, SKI conceded that methods exist for fabricating canister components and whole canisters, but concluded that the methods need to be further developed. Further, the authority pointed out that further work is needed with regard to serial production.

During the period 2007–2009, 24 nodular iron inserts were fabricated, of which 13 are PWR inserts and 11 BWR inserts /SKBdoc 1175208/. A development programme for fabrication of BWR inserts has been carried out. The programme was concluded with the casting of five inserts under production-like conditions in a demonstration series. Reliability in the fabrication of BWR inserts was analyzed by investigating the material properties of test pieces taken from the bottom, middle and top of the inserts. All inserts met the fabrication requirements.

On evaluation of the demonstration series, SKB was able to conclude that considerable progress had been made in the fabrication technology. The range of variation in the process had decreased considerably. It is particularly gratifying that the ductility of the material is much more even than before, as shown by fracture toughness testing. Based on data from the demonstration series, a probabilistic analysis has been done showing that the probability of local ductile deformation of the insert at maximum isostatic load is negligible /11-2/.

Development of the technology for casting of BWR inserts is considered finished, inasmuch as that the requirements that have served as a basis for the development work have been met. Further work may be needed to meet additional requirements in connection with the planned revision of the technical fabrication specifications for the insert. As a part of its cooperation with Posiva, SKB is also participating in the development of BWR inserts at a Finnish foundry.

The lessons learned from the development of the BWR inserts have been a point of departure for the equivalent development of PWR inserts. The decisive differences between BWR and PWR inserts from a fabrication viewpoint are that both the quantity of cast metal and the dimensions of the channel tubes are greater in PWR inserts. This has led to new questions regarding variations in material structure and deformations of the channel tubes. The latter has been dealt with by improved technique for the compaction of the mould sand that holds the channel tubes in place during casting. Further optimization of this compaction technique is planned. Progress has also been made in keeping the variations in material structure within acceptable limits.

A systematic method has been used to investigate the most important process parameters during casting. During 2008 and 2009, two rounds of factor testing, with four fabricated PWR inserts and three varied process parameters in each round, were carried out at one of the foundries to which the development work had been concentrated. The second factor test is currently being evaluated. The state of knowledge in the summer of 2010 is that at least one of the two casting processes is judged to give satisfactory results in terms of material properties and form accuracy.

As mentioned above, work based on the completed design analysis is under way to compile the properties that are important for the canister's mechanical integrity. This work will serve as a basis for the final evaluation of the casting processes. When technology development for PWR inserts has achieved the established goals, a demonstration series will be carried out at one of the foundries. The range of variation in the process will be examined in the evaluation of this serial production. Statistical measures for the important insert parameters are used in probabilistic analyses for the insert.

Some occurrence of material defects is considered to be unavoidable in such large castings as these inserts. The nondestructive ultrasonic testing that is regularly performed on fabricated inserts provides knowledge on the occurrence of defects. Detected defects are analyzed in certain cases by radiographic and metallographic examinations. The information is used both in the development of the fabrication process and to optimize the ultrasonic testing.

Figure 11-2 shows a steel lid for an insert. The lids will be delivered to the canister factory complete with finished dimensions and holes for valves for atmosphere exchange plus a centre screw for fitting to the insert. The current choice of material is steel grade S355J2, which has been used in a strength analysis. The analysis showed that the chosen dimensions and steel grade can withstand the design load of 45 MPa.

Figure 11-2. *A part of a machined insert and a steel lid with hole for valve for atmosphere exchange plus a centre screw for fitting to the insert.*

Programme

The continued development work includes:

- evaluating both casting processes for PWRs and carrying out a demonstration series at a foundry,
- investigating the range of variation for important parameters of the insert,
- determining different types of defects in the inserts,
- optimizing the packing technique to ensure the form accuracy of the channel tubes,
- revising the technical specifications for the insert,
- carrying out fitting tests for the steel lid.

11.4.2 Testing of inserts

In its review of RD&D Programme 2007, the authority pointed out that development and qualification of testing methods is just as important for the nodular iron insert as for the shell and the seal weld, and that the nondestructive testing that is needed should be based on design premises and strength analyses in accordance with current practice at the nuclear power plants. SKI further considered it necessary and important that SKB develop several different testing methods that complement each other.

SKB intends to develop testing methods that will ensure that the requirements stipulated in the design analysis presented in Section 11.3 are met. For certain parts such as steel lids and bolts, the alternative of purchasing standard products with quality certificates will be explored, which SKI in its review of RD&D Programme 2007 thought should be possible as long as requirements on the subcontractor are studied and reported.

In recent years, the focus has been on implementing the preliminary testing methods for quality inspection of the nodular iron inserts that are included in the programme for trial fabrication presented in RD&D Programme 2007, and evaluating the reliability of the methods. The work within nondestructive testing is compiled in /SKBdoc 1179633/ and summarized below.

As a part of the implementation of the testing methods, instructions have been prepared for quality assurance of ultrasonic testing. They are based on two different types of methods. One is TRL (Transmitter-Receiver-Longitudinal), where a transmitter generates longitudinal waves with a 70 degree angle that are received by a receiver. This method is suitable for examining material with a coarse structure and is used to scan the outer area of the insert. The other type, called phased array where the sound impinges on the surface at a right angle, is used in areas down to about 200 millimetres below the insert's mantle surface. Phased array testing permits electronic focusing for increased capacity to detect defects at greater depths. Some twenty or so inserts, both BWR and PWR, have been tested with these methods, whereby only single discrete defects have been indicated. The TRL method has given certain indications which on evaluation have turned out to be porosity variations. Phased array testing has also been used to measure the distance between the mantle surface and the outer corners of the channel tubes. This distance is an important parameter for the strength of the insert.

Preparatory studies have been made of how the central parts of the insert should be tested. One possibility is to use through-transmission testing, where sound is sent from a transmitter on one side of the insert to a diametrically positioned receiver. Preliminary tests show that this is feasible, but also makes high demands on the stability of the equipment (fixture) that fixes and positions the sensors. Development within this area is being pursued in cooperation with Posiva.

The reliability of the preliminary testing methods has been studied in cooperation with BAM (Bundesanstalt für Materialforschung und -prüfung) in Berlin. The TRL method has been studied on spark-machined grooves (for simulation of crack-like defects). The results indicate a detection capacity ($a_{90/95}$) of 2–3 millimetres for near-surface defects and 4–9 millimetres for defects at a depth of 50 millimetres. The phased array method has been studied on side-drilled holes with results indicating a detection capacity ($a_{90/95}$) of 2–8 millimetres, which decreases with increasing distance from the mantle surface. These tests are mainly aimed at studying the sensitivity of the testing method in different areas and will be supplemented with studies on real defects so that the reliability of the testing can be described.

Programme

The plans to devise acceptance criteria for nondestructive testing are based on the results of a recently completed design analysis. Supplementary deterministic calculations will be performed as a part of this effort.

Owing to the fact that the shear load case according to the design analysis imposes high demands on detection of relatively small defects in the outer area of the insert, further development of NDT will include methods for testing of surface and near-surface defects. Several methods will be evaluated such as inductive testing, magnetic powder testing and different variants of ultrasonic testing, for example TOFD (Time of Flight Diffraction).

SKB also plans to optimize the testing methods and develop and implement technology for testing of the area between the channel tubes in both the BWR and PWR inserts, as well as to determine the detection capacity of the method for real defects. Modelling will be carried out in support of the experimental work.

11.5 Fabrication and testing of copper components

11.5.1 Fabrication of copper components

SKI stated in its review that while methods do exist for fabricating canister components, they need to be further developed, and that there is also a need for development in those cases where SKB has chosen a reference method. SKI found SKB's development programme to be appropriate and headed in the right direction.

During the period 2007–2009, SKB made seven large copper ingots for tube fabrication and 14 smaller copper ingots for fabrication of lids and bottoms. During the same period, seven copper tubes were fabricated by extrusion, eight by pierce and draw processing and three by forging.

The geometric accuracy of extruded tubes has been improved. The tendency of bent tubes from the extrusion process has been eliminated by improvements in the process system; a guide tube has been fabricated and used to improve the straightness of the copper tube. After the introduction of the guide tube, all extruded tubes have met the fabrication requirement on straightness.

Areas with increased sound attenuation (bands) have been found in ultrasonic testing of extruded tubes. Detailed studies of these bands show that grain size differs compared with other parts of the tubes. The average grain size in the bands is 170–250 µm, compared with 70–150 µm in other parts. The bands have normal mechanical properties and they meet the fabrication requirement for grain size, which is ≤ 360 µm. The effect of the bands is poorer testability, since ultrasound is sensitive to variations in the grain size of copper. To address this problem, the extrusion process is simulated and extrusion tests are performed on a laboratory scale. In parallel, methods for reducing the disturbance sensitivity of ultrasonic testing for the material structure are being investigated.

A trial series of 10 copper lids was fabricated for the purpose of finding the best process parameters with existing tools in the forging press used for trial fabrication. A demonstration series of 10 lids was then carried out with parameters based on the outcome of the trial series. The results showed that the material structure was satisfactory and that the strength properties meet the fabrication requirements. The lid blanks also had a good geometric shape and could be machined to lids in a satisfactory manner. However, the current forging method subjects the lids to cold working. Process optimization and experiments with stress-relief annealing to eliminate cold working are being conducted.

Programme

Further development of the fabrication technology for the canister's copper components includes:

- Studying the extrusion process system based on new knowledge of the causes of grain size variations and, if possible, implementing the necessary modifications in the system.
- Carrying out and evaluating a demonstration series to determine what material structure can be achieved in extruded tubes.

- Continuing the development of fabrication by pierce and draw processing and forging.
- Continuing studies of cold working effects in copper and process optimization of the current forging method for local fabrication. Other forging methods may also be evaluated.

11.5.2 Testing of copper components

In recent years, SKB has focused on implementing the preliminary methods for quality inspection of copper components that were described in RD&D Programme 2007 and evaluating the reliability of the methods. The work that has been done within nondestructive testing is compiled in /SKBdoc 1179633/.

Quality assurance instructions have been compiled for ultrasonic testing of the copper components by means of the phased array method. Some twenty or so copper tubes and an equal number of copper lids have been tested by means of this method, whereby only small defects have been indicated. The testing has shown that sound attenuation varies in the components, which in some cases leads to poorer detection capacity.

The reliability of the preliminary testing methods has been studied in cooperation with BAM (Bundesanstalt für Materialforschung und -prüfung). The phased array method has been studied on artificial reference defects (flat bottom holes), and the results indicate a detection capacity ($a_{90/95}$) within a range of 2–5 millimetres.

The work with acceptance criteria for the testing of copper components does not conform to the same model as for the insert. Copper is not sensitive to stress concentrators in connection with either plastic strain or creep strain. Moreover, the copper shell is only loadbearing during lifts. The acceptable defects that have emerged in the damage tolerance analysis for lifts by the lid are based on limit load calculations and are very large. This means that detection requirements that are based on the effect of the defects on the thickness of the corrosion barrier are comparatively much stricter. Acceptance criteria for NDT must be supported by a description of possible defects. So far, defects have occurred very sparsely during trial fabrication. Only one or two forging defects in the lid have been found, despite extensive testing of both tubes and lids/bottoms. Accumulation of knowledge concerning different defects and how they can be detected will continue, as will the work of clarifying the detection requirements for these defects. This is planned to be done both by flow simulations of the pierce and draw and extrusion processes, which provides information on the orientations and positions of natural defects, and by the hot forming of material on a laboratory scale where the process is disturbed so that defects occur.

Current testing methods will be supplemented with surface-scanning methods to increase detectability for surface defects as well as impact deformations, which can give rise to cold working effects.

Programme

The development work for testing of copper components includes:

- studying possible defect types in hot-formed copper and characterizing them, as well as formulating detection and size determination requirements for them,
- further developing technology for ultrasonic testing, including surface inspection of the copper components,
- studying what effect varying grain size has on the detection capacity of ultrasonic testing and optimizing the method on this basis.

11.6 Sealing and testing of the weld

11.6.1 Sealing

In RD&D Programme 2007, SKB reported that the weld metal in friction stir welding has, in most cases, equivalent chemical and mechanical properties to the parent metal. However, investigations showed that certain defects can occur, such as oxide inclusions and joint line hooking. In its review of RD&D Programme 2007, the authority said that it remained for SKB to show by further tests that oxide particles can be eliminated and joint line hooking can be overcome. The authority also pointed out that it is of great interest that SKB complete its programme to automate the FSW process in order to achieve as high repeatability as possible in the future.

With the choice of FSW as a reference method in 2005, SKB focused research, development and demonstration on this method /SKBdoc 1175162, 1175236/. After 75 lid and bottom welds (the lid weld is done with insert and the bottom weld without insert) consisting of nearly 350 separate welding cycles, both the welding process and the welding system can be considered robust and stable.

Before the work of automatically regulating the welding parameters within the process window was begun, the welding process was optimized with respect to stability and repeatability within as wide a process window as possible. This resulted in new parameter settings and a new design of the tool shoulder, see Figure 11-3. Furthermore, the design of the tool probe has been changed to optimize its life, i.e. the safety factor against failure, since a probe is only intended to be used for one weld. First it was found that a surface coating, chromium nitride (CrN), permits full-turn welding without detectable cracking of the surface of the probe. In order to further reduce the risk of failure, the length of the MX notches was reduced. In cases where cracks occurred during welding they were localized to MX notches 22–25 millimetres down from the shoulder, where tool failure has occurred in tests at maximum tool temperature. The tool therefore now has MX notches only 17 millimetres down from the shoulder, see Figure 11-4.

In several full-turn welds done in conjunction with optimization of the welding process with respect to the stability and repeatability of the process (parameter study), it was found that the heat input needed to keep the tool temperature at about 850°C was repeatable. Due to the fact that the thermal conditions during a welding cycle vary, for example during the downward sequence and the upward sequence, the heat input must be varied during the cycle, see Figure 11-5.

The only discontinuity that occurs during welding within the relatively wide process window (790–910°C for the tool temperature) is joint line hooking, which occurs when the tip of the probe penetrates too deeply and the flow of material pulls the vertical joint line towards the surface. It is above all during the overlap sequence that the tip of the probe goes too deep. Joint line hooking can be greatly reduced by using a shorter tool, as has been demonstrated in a small welding series carried out after the demonstration series. A reasonable assessment is that joint line hooking can be limited in production to a radial extension of two millimetres.

Extensive tests have been done on lid welds made at the Canister Laboratory to judge the long-term properties of the welds and how well the weld metal conforms to the specification for the reference canister. The weld metal has been found to meet the requirements regarding material composition, material properties and dimensions.

In RD&D Programme 2007, SKB presented tests of the mechanical and chemical properties of the weld metal. Since then SKB has performed chemical analyses of the weld metal in a number of measurement points in several lid welds /SKBdoc 1175162/. Three measurement points along the radial centreline – top, middle and root – from both the single weld area and the overlap area have been analyzed along with a reference point in the lid. The results show that when weld metal from welding in air is examined, the oxygen concentration varies from about 3 to 44 ppm. This shows that the weld metal can contain regions or bands with small quantities of oxide inclusions, primarily concentrated in the overlap area. When weld metal from a weld performed in argon shielding gas was examined, the oxygen concentration varied between 1 and 2 ppm, which indicates that oxide inclusions in the weld metal can be minimized or prevented if welding is done in shielding gas. A major survey of the occurrence of oxide inclusions and their influence on the properties of the weld metal is being carried out.

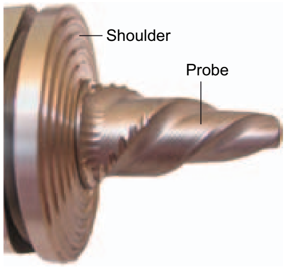


Figure 11-3. *Welding tool with convex shoulder.*

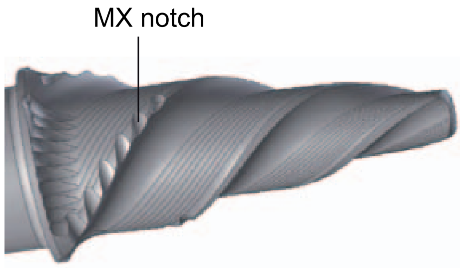


Figure 11-4. *Probe with MX notches only 17 millimetres down from the shoulder.*

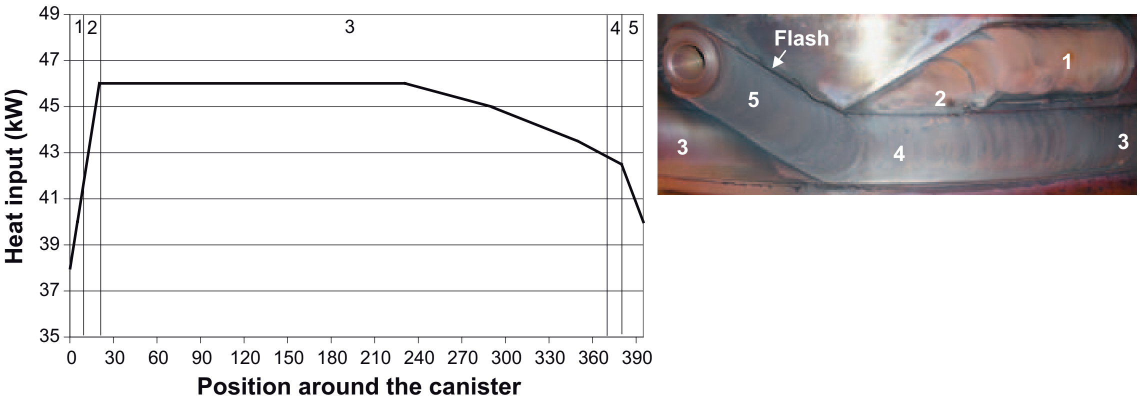


Figure 11-5. *Required heat input for a tool temperature of 850°C as a function of position in the welding cycle. Sequences in the welding cycle: 1. acceleration, 2. downward sequence, 3. joint line sequence, 4. overlap sequence and 5. parking sequence.*

Analyzed concentrations of phosphorus, sulphur and hydrogen in the weld metal (see Table 11-2) are acceptable and comparable to the levels in the parent metal. The occurrence of metallic elements has also been analyzed, since small particles can be dislodged from the tool probe and get mixed into the weld metal. The tool probe – which is made of Nimonic 105, an alloy of nickel, chromium, cobalt, aluminium and titanium – is now surface-treated with chromium nitride (CrN). This surface treatment has reduced the quantity of nickel particles in the weld metal.

Programme

The welding tests have shown that repeatability and reliability in the welding process are very high, but further optimization of both the process and the system is needed. The steps that remain are to:

- commission the regulator and then optimize the regulator's settings and function so that the human factor is eliminated as far as possible,
- weld more lids in argon in order to get a better picture of the weld metal's properties, since the provisional shielding gas chamber has only been used for one full-turn weld. Figure 11-6 shows the shielding gas chamber that is planned to be used for most lid welds,
- adapt the development work with regard to the microstructure and oxide content of the weld to the results of ongoing material investigations of the weld metal at e.g. KIMAB and Helsinki University of Technology,
- gather material on the systems in the canister factory and the encapsulation plant.

Figure 11-6. Illustration of the new shielding gas chamber.

Table 11-2. Concentration of phosphorus, sulphur and hydrogen as well as metallic elements in weld metal.

	Concentration in the weld metal (ppm)		
	Welds (FSWL27, 35, 36) – tool without surface treatment	Weld (FSWL69) – tool with surface treatment	Weld (FSWL51) – tool without surface treatment, with shielding gas
Ni	21	< 1	9
Co	9	< 1	3
Cr	5	< 1	3
P	52.7	49.5	61.5
S	4.4	< 5	4.3
H	0.31	0.3	0.55
O	17	32	1.6

11.6.2 Testing of welds

Experience accumulation from weld testing has mainly occurred in the form of feedback to development of the welding process. When it comes to possible defects, the state of knowledge for welds is better than for hot-formed copper, since the welding system is at the Canister Laboratory and is small-scale in comparison with the industrial processes of forging, extrusion and pierce and draw processing. The welding process is being optimized to meet the detection requirements for various defects, including changes in tool design and parameters. This includes establishing a reliable process window. Although the probability is low, there is a possibility that new types of defects could be generated in the new process. This will be investigated by means of different process tests. If reason exists, the NDT methods will be optimized to detect new types of defects. A special question that will be further studied is the possible occurrence of volumetric defects in the welding process, which determines whether the welds need to be X-rayed. As for the copper shell in general, methods will be developed to detect surface defects.

Programme

The development work for testing of welds includes:

- studying possible defect types in welds and characterizing them; formulating requirements for detection and size determination of these defects,
- further developing technology for ultrasonic testing,
- studying the reliability of different NDT methods for weld testing; in particular, determining whether radiographic testing is needed,
- develop technology for surface inspection of welds,
- gather background material for the NDT systems in the encapsulation plant.

11.7 Handling and deposition of canisters in the Spent Fuel Repository

Handling and deposition of the canister in the final repository includes the whole handling chain, from the time the canister in its transport cask (KTB) arrives at the terminal building above ground until it has been inspected and deposited, i.e.:

- Transport down the ramp of the canister in KTB.
- Transloading of the canister to the deposition machine at repository level.
- Deposition of the canister.

In the review of RD&D Programme 2007, the authority requested information on how far SKB had come in its development of the handling chain for the canister at the final repository. The authority has also asked for an account of how loading, unloading and deposition of the canister will take place and what needs to be automated due to radiation.

11.7.1 Transport on the ramp

Transport of the canister transport cask (KTB) down to repository level will take place via the descent ramp on a vehicle specially intended for this purpose. A KTB with canister weighs about 85 tonnes.

A feasibility study of the set of requirements for underground transport and suitable transport solutions was carried out during the period 2007–2009. The study shows that a concept where the KTB is transported on a separate loadbearing transport frame has significant advantages. It permits efficient handling of the KTB and facilitates salvage of the vehicle in the event of an accident on the ramp. A vehicle solution called “Self Propelled Modular Transport (SPMT)” has proved to be suitable, see Figure 11-7. It has a modular construction, which simplifies repairs and replacement of components. SPMT is a serial-manufactured product and is judged to be capable of meeting the requirements for ramp transport after certain modifications.

Programme

During the period 2011–2013, SKB plans to clarify the requirements and prepare the documentation for a future procurement. The possibilities of coordinating the design of transport frames with terminal transport duties will be explored. The purpose is to purchase a vehicle with properties similar to those of the future ramp vehicle. This vehicle will permit testing of the ramp vehicle concept and detailed testing of technical solutions.

11.7.2 Deposition

SKB has developed a new deposition machine, which was delivered to the Äspö HRL in the autumn of 2008. The machine has been tested in manual operation for nearly 2 years. In parallel with the manual tests, the machine has been equipped with an advanced navigation and positioning system.

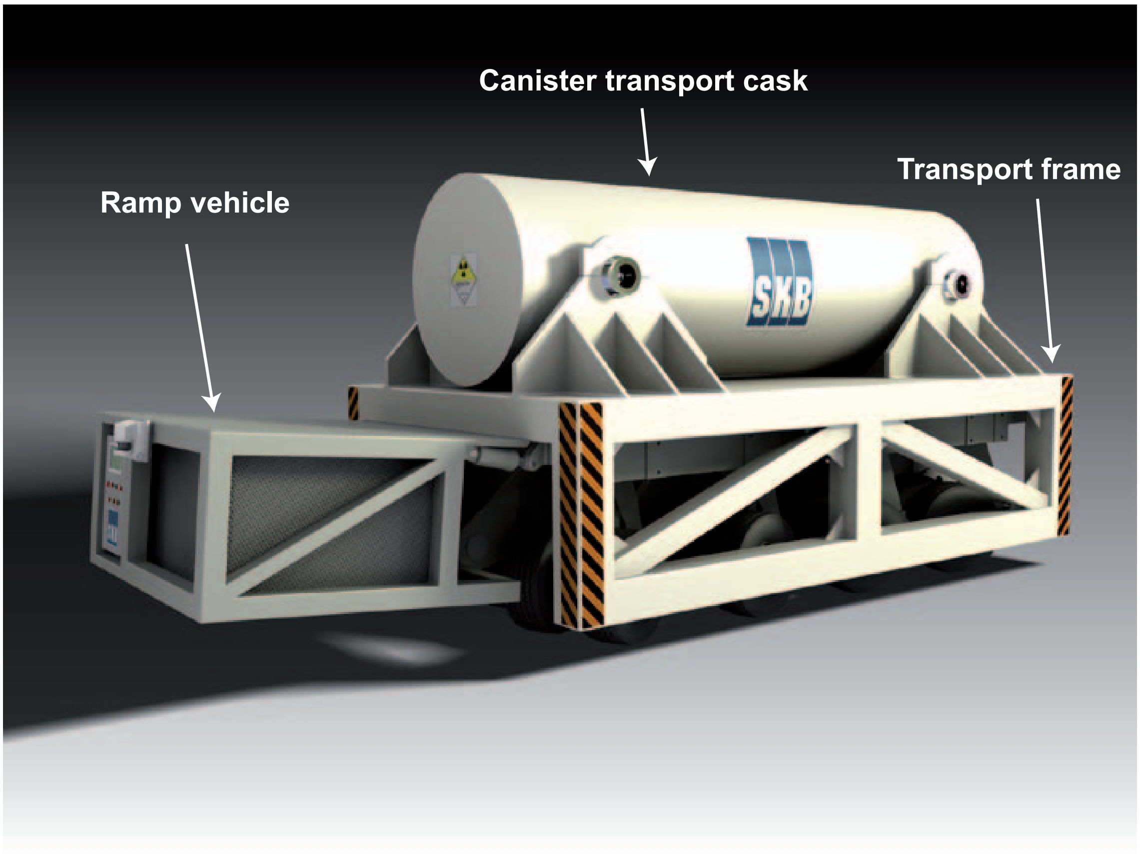


Figure 11-7. *Concept for vehicle for underground transport of canisters via the ramp.*

Programme

In planned full-scale tests at the Äspö HRL, the new deposition machine will carry out around 1,000 depositions during a two-year period. Tests of the navigation system are included. In the final repository, the deposition machine will travel from the central area to the deposition tunnel and the selected deposition hole. The most critical part of the navigation sequence, when the machine straddles the radiation shielding hatch, is included in the tests being conducted at the Äspö HRL. The deposition machine is also equipped with a positioning system that enables the canister to be positioned centrally in the buffer rings in the deposition hole.

The full-scale tests are expected to be concluded in 2011. The main purposes are to:

- collect and evaluate data on the reliability and availability of the machines and the subsystems in fully automated operation,
- verify that deposition with fully automated navigation and positioning can be carried out with satisfactory accuracy and safety,
- compile a list of service work that is required in continuous operation.

12 Technology development, buffer

The buffer line includes fabrication, handling and installation of the buffer that surrounds the canister in the deposition hole. This chapter briefly describes the buffer's reference design, the requirements made on the buffer to achieve the initial state and the technology development SKB plans to carry out during the coming years. In parallel with, and as a basis for, this technology development, research and development on the clay barriers continues. This is described in Chapter 24 in Part IV.

12.1 Requirements and premises

The function of the buffer is to prevent flowing water from coming into contact with the canister and the spent fuel. Any transport of corrosion products and radionuclides through the buffer will be dominated by diffusion. Furthermore, the buffer should limit the stresses on the canister caused by any displacements in the rock. In order for the buffer to perform these functions, hydraulic conductivity must be low. Moreover, the buffer must retain its dimensions and be able to self-heal, which means that any cracks and inhomogeneities should close when the buffer swells so that the buffer retains its function. The buffer must also be physically and chemically stable in a long time perspective.

The requirements on the buffer are thus mainly related to its barrier function in the Spent Fuel Repository. In order to secure this barrier function, the following requirements have been formulated on installed buffer /12-1/:

- the content of the clay mineral montmorillonite in the buffer material shall be 75–90 percent of the total dry weight,
- the content of organic carbon in the buffer material shall be less than one percent,
- the total sulphur content may not exceed one percent, and the sulphide content may not exceed 0.5 percent,
- the water-saturated density of the installed buffer shall be between 1,950 and 2,050 kg/m^3,
- the buffer's dimensions shall be in accordance with SR-Can /12-2/,
- the temperature in the buffer may not exceed 100°C,
- minimum swelling pressure, maximum hydraulic conductivity, stiffness and strength shall be retained despite changes in temperature and pressure and movements in the buffer.

In addition there are requirements related to production and operation. Generally formulated, the buffer shall be based on proven or tested technology. Buffer with the specified properties shall be possible to produce and install with high reliability.

12.2 Current situation and programme

Based on the requirements set forth in the preceding section, SKB has devised a reference design for the buffer. According to this reference design, the buffer consists of uniaxially compacted blocks and rings of MX-80 bentonite clay. The gap between the buffer blocks and the wall of the deposition hole is filled with pellets, see Figure 12-1. In addition to the composition of the buffer, the reference design also includes the fabrication process and the chosen installation technology. It must be possible to install buffer blocks and canister up to three months before the outer gap is filled with pellets and the deposition tunnel is backfilled. To this end the reference design includes a buffer protection and a bottom pad, designed to connect with the buffer protection.

In its review of RD&D Programme 2007, SKI pointed out that SKB should provide a more detailed description of what technique is to be employed during installation of the buffer to prevent unduly rapid saturation of the buffer. As regards fabrication of the buffer blocks, SKI noted that SKB had abandoned isostatic pressing as a reference method, without giving any reasons for this. Furthermore, SKI pointed out that SKB had not specified what should be included in a programme for quality

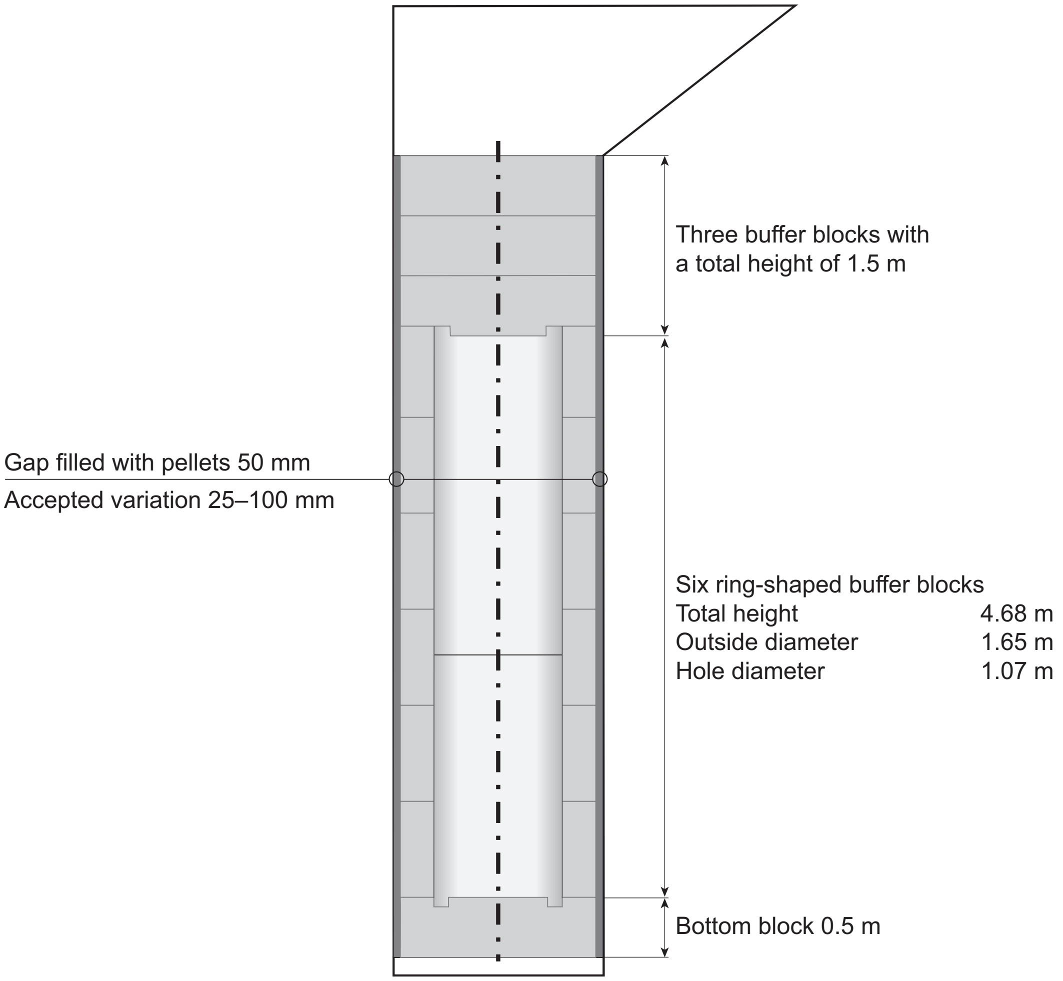


Figure 12-1. *Reference design for the buffer.*

assurance of the buffer. SKI and the Swedish National Council for Nuclear Waste consider it essential that SKB should develop a quality programme for buffer fabrication. SKI also said that further trial fabrication may be needed to show that sufficiently high quality can be achieved for the chosen material under conditions that more closely resemble serial production. Regarding the choice of material for the buffer, SKI maintained that bentonite has a complex and varying chemical and mineralogical composition and that this is important for the long-term evolution of the buffer. Besides specifying a minimum concentration of montmorillonite, SKB therefore also needs to specify the concentration interval for other minerals, for example certain trace minerals that can have an impact on long-term safety. Furthermore, in order to make optimal use of available resources, SKB should draw up a plan for which material is to be used in various new tests.

Since 2007, SKB has developed and tested both the buffer protection and the bottom pad. The components have been tested on a large scale at the Äspö HRL. Furthermore, a system for drainage in the deposition holes during installation has been tested /12-3/.

In the Bentonite Laboratory, SKB has examined the function of the outer gap, which is filled with bentonite pellets. The purpose of these tests has been to study how the buffer is affected by inflowing water. The phenomena that have been studied are erosion of the bentonite, heaving of buffer blocks and buildup of water pressure in the buffer /12-4/.

Since RD&D Programme 2007 was published, SKB has produced some fifteen or so full-scale buffer blocks for the tests at the Äspö HRL. In connection with the fabrication of these blocks, the bentonite has been conditioned by in situ addition of water to the clay, see Figure 12-2.

There is an ample supply of bentonite on the world market. Several suppliers today can deliver a product that meets SKB's requirements on the buffer material /12-5/.

Figure 12-2. At left: Eirich mixer used for conditioning of bentonite at the Äspö HRL. At right: A bentonite block is placed on a shipping pallet.

According to SKB's delivery control model (see Section 9.2), technology development for the buffer has passed the concept phase and is in the design phase. The work to be done before the start of repository construction has been divided into three parts. In the first part, system design is carried out within the chosen reference design. Here the reference design is described, along with how far development has advanced, and suggestions for further development are presented. In parallel with the development of the reference design, alternative designs are developed and evaluated. These two parts of the development work will be pursued in separate projects (see Sections 12.2.1 and 12.2.2). The third part involves the development of the fabrication process for buffer blocks (see Section 12.3). In addition, some technology development for the buffer will take place within the project "Mining of the outer section of the Prototype Repository" (see Section 12.4).

12.2.1 System design of buffer

In conjunction with the development of the reference design for the buffer, SKB saw a need, within the framework of the chosen reference design, to further develop the technology for installation of the buffer to obtain an efficient work process. To a great extent, the various components in the reference design have been tested previously, both in the laboratory and in field tests. The areas where there is still a need for testing and further development are described in the following section. All components included in the reference design for the buffer will be tested together, as a part of the system design (see "Full-scale tests of the reference design", later in this section). The results of the work will serve as a basis for the detailed design of these components.

Components in the reference design

The components in the reference design that need to be further developed are the bottom pad, the buffer protection, the radiation shield and the cover plate. The bottom pad is supposed to provide a flat and horizontal surface on which the bentonite blocks can be stacked vertically and to serve as an attachment for the buffer protection. According to the reference design, it consists of a copper plate grouted onto a cement slab; other solutions will be considered. The buffer protection is supposed to prevent water from being absorbed by the buffer during the installation phase. The radiation shield is supposed to provide protection during emplacement of the canister and buffer. A prototype of the radiation shield is being tested. After buffer and canister have been emplaced in the deposition hole, the hole is covered with a cover plate. It is supposed to prevent material from getting into the deposition hole and to facilitate the work of installing the backfill. Development of these components has begun. In the continued development work it is important that all components function together and that they fulfil the requirements and premises presented in Section 12.1.

Equipment for installation

Before the canister is placed in the deposition hole, the buffer of highly compacted bentonite, in the form of blocks and rings, must be in place. Installation of the buffer requires equipment that is capable of both lowering bentonite rings and bentonite blocks into the deposition hole and, if necessary, lifting them up again. Such equipment has been developed in a prototype version. It consists of a gantry crane and a vacuum lifting tool, see Figure 12-3. The equipment is equipped with a positioning system so that the blocks will end up in exactly the right positions in the deposition hole. Tests of the functions of the vacuum lifting tool began in 2009. The tests include showing that the tolerance requirements for the buffer can be met.

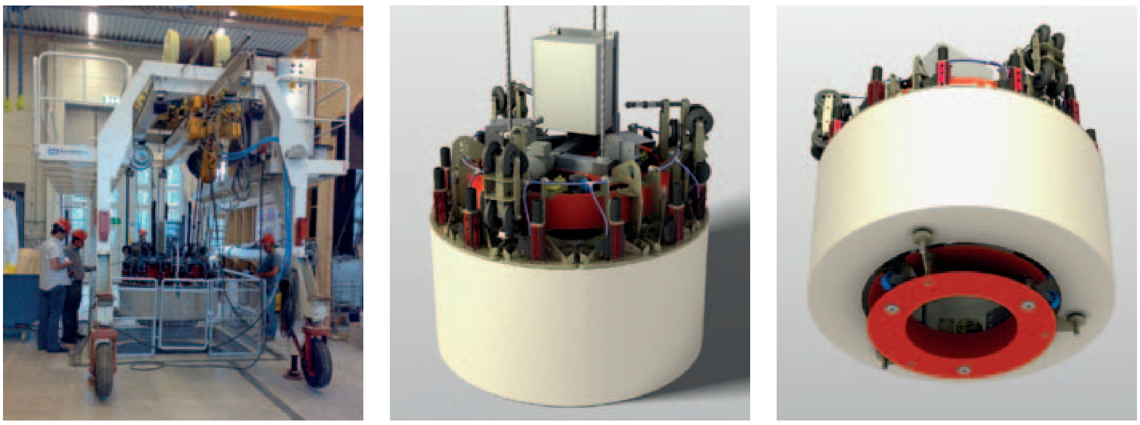
During the detailed design phase, the prototype of the lifting tool will be tested together with a prototype of a buffer lowering machine equipped with navigation system. Via the prototype tests, SKB will show that all steps in the buffer lowering procedure can be carried out under the conditions that will prevail in routine operation.

The development programme for installation of the buffer includes testing and demonstration of the method at both the Bentonite Laboratory and the Äspö HRL. The purpose of the planned tests and demonstrations is to confirm that the chosen method and the lifting tool work as planned. Testing of the lifting tool has begun at the Bentonite Laboratory. True-to-scale blocks and rings of concrete are being used for the tests. Later on, tests will be performed in tunnels at a depth of 420 metres at the Äspö HRL.

Furthermore, new equipment for filling the gap between the buffer blocks and the wall of the deposition hole with pellets will be designed, manufactured and tested. The requirements on the equipment have been specified. The equipment must be capable of filling the gap relatively quickly and without creating too much dust.

Inspection and measurement methods

In order to ensure conformance with the reference design, different inspections and measurements will need to be performed. For example, information is needed on the dimensions of the deposition holes in order to determine the precision and accuracy of the density of the installed buffer. Measurement methods and data on measurement accuracy will be developed within the rock line. Another example is measurement methods for determining the chemical composition of the bentonite. Methods exist and have been used /12-5/, but further studies and tests are needed to determine the accuracy of the methods. This work should lead to a better idea of the measurement accuracy of the methods.



a) b) c)

Figure 12-3. *Equipment for emplacement of buffer blocks: a) Vacuum lifting tool together with gantry crane. b) Vacuum lifting tool viewed from above and to the side. c) Vacuum lifting tool viewed from below and to the side.*

Test of function of subcomponents and subsystems

According to the reference design, it should be possible for the buffer together with the canister to remain in the buffer protection up to three months before the buffer protection is removed and the outer gap is filled with pellets. During this time, changes can occur in the buffer blocks. For example, a redistribution of water can cause cracks in the blocks. This has previously been investigated by laboratory tests. Supplementary tests, both on a laboratory scale and on a larger scale with full-scale bentonite blocks, will be conducted. In the laboratory-scale tests it is possible to simulate either the expected temperature difference over the buffer in the repository (from canister to rock wall) or the temperature gradient over the buffer. When the temperature difference is simulated in laboratory-scale tests, the gradient is greater compared with full-scale conditions. Laboratory-scale tests can therefore cause changes in the buffer that do not occur in reality. Larger-scale tests are therefore necessary to be able to simulate both expected temperature difference and the temperature gradient over the buffer simultaneously.

Tests have been performed to investigate how the system with bentonite blocks and pellets functions and reacts when water is added. The tests have studied how water is transported and distributed in the pellet-filled gap and how the stack of bentonite blocks is deformed at different inflows. These tests have been conducted on a laboratory scale as well as with full-scale blocks /12-4/. Supplementary tests are planned.

Full-scale tests of the reference design

Tests to study how the different subcomponents function together are important. SKB intends to conduct tests where all constituent components in a deposition hole are tested together. These tests are planned to be conducted at the Äspö HRL during 2012, provided that the proposed new components and equipment have been fully tested (see “Components in the reference design” and “Equipment for installation” in this section).

12.2.2 Alternative reference design

In addition to design and components, the reference design also includes the deposition sequence, i.e. the duration and timing of the installation of buffer, canister and backfill. The rock conditions in Forsmark, with very low hydraulic conductivity of the rock at repository level, means that it should be possible to reduce the permissible inflow of water in a deposition hole. It is then probable that the installation of the buffer can be simplified, which would entail a change in the reference design. Studies of alternatives to the chosen reference design are therefore planned. These studies will also concern the backfill. Examples of parts that may need to be changed are:

- *The buffer protection and the inflow of water into the deposition hole*. If installation can be done fast enough, possibly in combination with tougher requirements on permitted inflow into the deposition hole, it is possible that the buffer protection can be omitted.
- *Water ratio, dimensions and density of installed buffer blocks and pellets*. If the buffer protection is omitted, the buffer may have to be changed so that it alone can resist the effects of water inflow into the deposition holes. For example, blocks with a higher water ratio, other dimensions and/or other density may be needed.
- *Simplification of the bottom pad.* If the requirements on the evenness and inclination of the bottom surface can be relaxed, the pad can be simplified or possibly omitted.
- *Variations in the geometry of the deposition holes*. If it can be shown that it is possible to produce deposition holes with narrower tolerances, regarding both out-of-plumbline and diameter variations, the chances of remaining within the interval stipulated for buffer density (1,950 to 2,050 kg^3) are improved, even in unfavourable cases.
- *Accuracy in installation of canister and buffer.* If the tolerance requirements on the positions of the buffer blocks and the canister can be relaxed, deposition can be simplified.

12.3 Compaction technology

Two methods can be used for fabrication of buffer blocks: uniaxial and isostatic compaction. In uniaxial compaction, bentonite is pressed into blocks in a stiff mould. In isostatic compaction, pressing is done by the even-sided application of pressure on a flexible mould filled with bentonite. The methods have been tested by SKB with good results, and the assessment is that both produce blocks of equivalent and acceptable quality.

Uniaxial compaction has been chosen as the reference method for fabrication of buffer blocks. The reason for the choice is that SKB has shown that this technology can be used to fabricate buffer blocks of good quality and with dimensions in accordance with the reference design. Isostatic pressing of buffer blocks with the same dimensions requires a larger press than SKB currently has at its disposal.

Further development of uniaxial technology will continue and will be focused on testing of new solutions to minimize the friction between mould and bentonite. The technology that is used today to fabricate large buffer blocks requires a conical mould. New technology to minimize or completely eliminate the conicity will be tested. Trials will also be performed in an attempt to fabricate blocks of greater height than today.

Posiva has carried out and plans to carry out additional tests with isostatic compaction of bentonite blocks on a larger scale than the tests performed by SKB. SKB intends to participate in these tests via cooperation with Posiva. The purpose is to improve the basis for comparisons between the compaction methods with regard to quality, technology and cost for production of buffer blocks. The outcome of the comparisons will serve as a basis for the final choice of fabrication method. This choice needs to be made in time so that it can serve as a basis for the design of the production plant.

12.4 Prototype Repository

Several large-scale tests where the properties of the buffer have been studied have been carried out at the Äspö HRL. Examples are the *Canister Retrieval Test* (CRT) /12-6/ and the *Temperature Buffer Test* (TBT) /12-7/. In both of these tests, the buffer had a similar configuration and dimensions as in the reference design. Water was supplied artificially to the buffer via filters on the walls of the deposition holes. The deposition holes were sealed with plugs in the upper part of the holes.

The biggest large-scale experiment in the Äspö HRL, the *Prototype Repository* /12-8, 12-9/, consists of a total of six full-scale deposition holes in a deposition tunnel that has been backfilled with a mixture of bentonite (30 percent) and crushed rock (70 percent). The overall goal of the Prototype Repository is to test and demonstrate the integrated function of subcomponents in a final repository, under realistic conditions and on a full scale. Data from the test is compared with model-calculated predictions.

The Prototype Repository, which was installed during 2001 and 2003, is divided into two sections. According to SKB's plans, the plug that seals the Prototype Repository plus the outer section with backfill and two deposition holes will be mined in 2011 (see Figure 12-4). The buffer in these two deposition holes has been supplied with water the natural way via the rock surface in the deposition holes and from the backfill in the deposition tunnel. The buffer is not water-saturated, and density and water ratio are expected to vary depending on to what extent water has been supplied to the buffer.

Planning of the project is currently under way. A detailed plan will be drawn up for the mining method, duration, resources in the form of machines and personnel, and cost.

The goals of the mining of the Prototype Repository are to:

- by extensive sampling, obtain a picture of density and water saturation of buffer and backfill in the outer section of the Prototype Repository, and to study any changes in the buffer. During the course of the test, the buffer has absorbed water from the surrounding rock. Furthermore, it has been exposed to high temperatures for a long time. The highest temperature measured in the buffer since heating of the outer section started in 2003 is about 85°C. This may have affected the properties of the buffer material. Such changes have been studied and will continue to be studied in experiments that are specially designed for this,

- characterize the contact surfaces between buffer and backfill, and between backfill and tunnel wall, after eight years' wetting. These parts of the buffer and backfill can be studied in detail when the Prototype Repository is mined,
- confirm or reject indications of small changes in the rock mass around the two deposition holes. Measurements during the wetting and heating phase indicate that certain changes have occurred in the rock, for example small movements along fracture planes. When backfill, buffer and canister have been removed, the rock in and around the deposition holes can be studied,
- determine the positions and form of the canisters after mining. The canisters have been subjected to swelling pressure from the buffer. This may have changed their positions, and possibly their shape,
- identify any corrosion of the canister. Equipment for studying corrosion of copper in the buffer has been installed. These measurements will be accompanied by sampling of the buffer around the canister. Since the buffer is not water-saturated, the environment of the canister has not been oxygen-free during the course of the test. It is therefore doubtful whether sampling can contribute any knowledge relevant to the question of copper corrosion under oxygen-free conditions. Furthermore, impurities, mainly lubricant from fabrication of the buffer blocks, can interfere with the analysis of chemical processes on the canister surface,
- register any changes in or damage to the plug. During a short period of the test, the outer plug has been exposed to high pressures, which may have caused permanent effects,
- study biological and chemical activities in buffer and backfill when the test is mined. During the course of the test, water and gas have been sampled and analyzed chemically and biologically. Sampling during mining can verify these analyses.

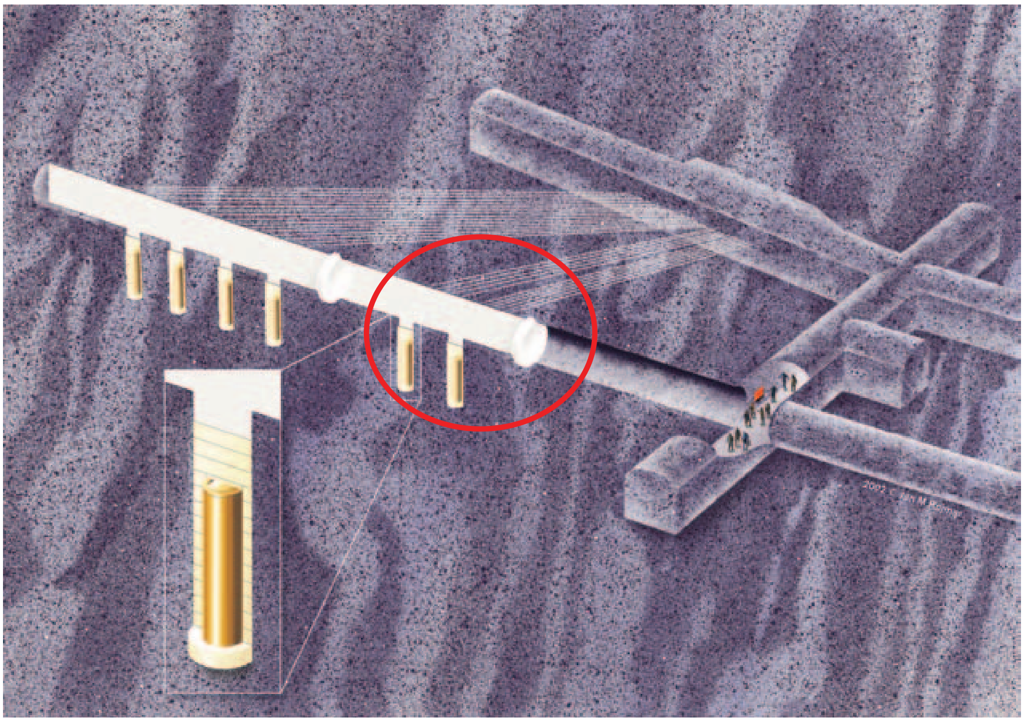


Figure 12-4. *Prototype Repository at Äspö HRL. According to SKB's plans, the circled part of the test will be decommissioned in 2011.*

13 Technology development, backfilling

The backfilling line includes manufacture, handling and installation of backfill in deposition tunnels and in the uppermost parts of the deposition holes. When a deposition tunnel is backfilled, a plug is built near the mouth of the tunnel opening into the main tunnel. The plug is a part of the backfilling line.

This chapter describes the overall reference design for backfill and plug, the requirements that are made on the backfilling line in order to achieve the initial state /13-1/, and the further development of the chosen reference design that is planned during the coming years. The backfill is presented in Sections 13.1–13.2 and the plug in Sections 13.3–13.4. Parts of the development work are being pursued in cooperation with Posiva.

13.1 Requirements and premises – backfilling

The requirements on the backfill can primarily be related to its barrier function in the Spent Fuel Repository, to requirements on other barriers and to production and operation of the final repository.

The backfill's barrier function is to limit the water flow into the deposition tunnels. This leads to the following requirements:

- Hydraulic conductivity < 10^{-10} m/s.
- Swelling pressure > 0.1 MPa.

The barrier function also includes limiting the upward swelling of the bentonite in the deposition hole. This means that the density and deformation properties of the backfill, in its initial state and after complete water saturation, must be such that swelling of the buffer results in a buffer density at water saturation that is not less than 1,950 kg/m^3, with sufficient margin for material losses and uncertainties.

Moreover, the backfill may not give rise to appreciable deterioration of the barrier function of other barriers and should retain its barrier function for a long time in the environment that is expected to prevail in the final repository. In order for the backfill to maintain its barrier function, its density must be sufficiently high over the cross-section and along the length of the deposition tunnel. In order to achieve this density at installation of the backfill, requirements are made on the geometry of the deposition tunnels and on the maximum water inflow that can be accepted during installation. These requirements are presented in Chapter 15.

In addition there are requirements related to production and operation. Generally formulated, the backfill must be based on proven or tested technology. Backfill with the specified properties must be possible to produce and install with high reliability.

13.2 Current situation and programme – backfilling

Development of a backfilling concept with naturally swelling clay has been in progress for a long time. Since RD&D Programme 2007, the work has been focused on developing methods and equipment for the reference design chosen by SKB. The reference design is based on backfilling with precompacted bentonite blocks and pellets. The tunnel floor and the bevel to the deposition holes are filled with pellets, which are compacted and levelled to an even and stable surface. Bentonite blocks are then stacked on the bed, and the remaining space between the stack and the tunnel wall is filled with bentonite pellets.

In its review of RD&D Programme 2007, SKI pointed out that SKB should prepare a quality programme for fabrication and installation of backfill and that SKB should demonstrate that they can handle the backfill under the conditions that can be expected to prevail on the selected site. Furthermore, SKI wanted SKB to present plans for full-scale tests at the Äspö HRL. These activities have been initiated and are planned to be executed in the system design phase for backfilling, which is described in Sections 13.2.1–13.2.5.

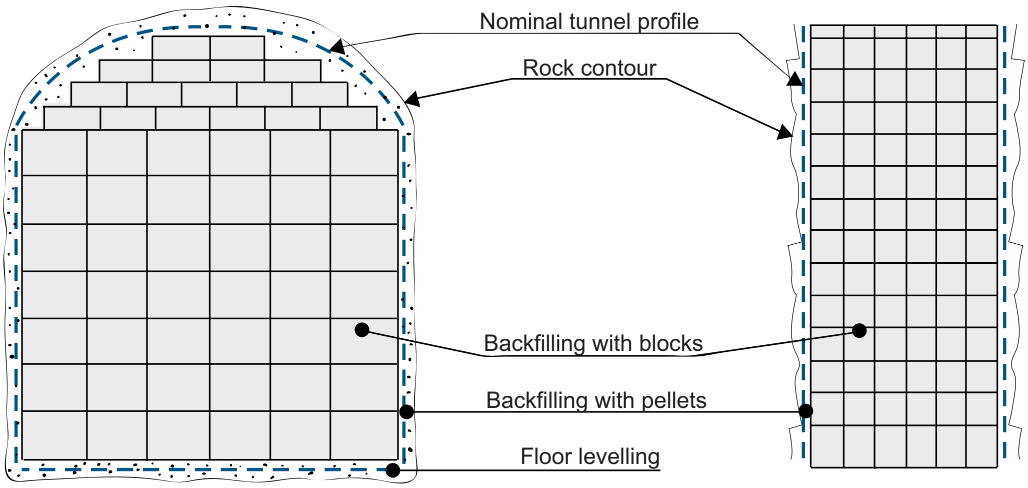


Figure 13-1. *Cross-section and longitudinal section of a tunnel with backfill blocks and pellets.*

Technology and methods for preparing the clay prior to pressing of blocks are known and proven, as is the method of fabricating blocks by uniaxial pressing. Blocks of the size specified in the reference design have not yet been pressed, but there are no indications that this would not be possible with the same good results as for smaller blocks.

Different methods for installation of blocks in deposition tunnels have been studied. A concept has been chosen where precompacted bentonite blocks are emplaced individually. This concept is now undergoing further development, where prototype equipment for handling and installation will be developed and tested both at the Bentonite Laboratory and under more realistic conditions at the Äspö HRL. The alternative of installing modules of blocks may also be interesting but is more difficult to test on a large scale /13-2/.

Since the presentation of RD&D Programme 2007, the concept for installation of pellets between bentonite blocks and tunnel wall has been developed and tested /13-2/. For bevelling of the upper part of the deposition hole, an alternative involving backfilling with pellets or granules has been tested at the Bentonite Laboratory. Preliminary results indicate that it is possible to backfill the bevel with clay material in this form. Method and equipment for this need to be developed, however.

Tests have been conducted to understand and control the processes that take place in pellet filling during the installation phase. The purpose has been to determine what inflow of water to the deposition tunnel can be permitted during the backfill installation phase. This work will continue in order to ensure that backfilling meets the stipulated requirements.

SKB's programme for technology development of the backfill has passed the concept phase and is now in the design phase, which is divided into system design and detailed design (see Section 9.3.5). The design work will be pursued with the goal of finalizing the system design within the chosen reference design prior to the start of construction of the Spent Fuel Repository. This will be followed by detailed design and subsequent implementation.

System design of backfilling according to the delivery control model entails the following goals for the development work:

- Further develop the conceptual design of the barrier.
- Further develop the conceptual design of the production system for backfilling.
- Verify that design premises and requirement specifications are fulfilled.
- Prepare a costed plan for industrialization/implementation with inspection methods that show how the subsystem can be implemented and inspected so that stipulated requirements are fulfilled.
- Perform technical risk analysis, which includes risks related to both the design and the plans for industrialization/implementation with inspection methods.

The work covers the following areas:

- Choice and specification of material.
- Installed density and geometric configuration of the tunnel.
- Methods for fabricating and handling backfill components.
- Methods for installing backfill in deposition tunnels.
- Methods for inspecting material, components and installed backfill.

During detailed design, the equipment needed for production and installation of backfill will be finally designed and tested. The backfilling process as a whole with its inspection methods will be tested on a full scale and in a realistic underground environment.

13.2.1 Choice and specification of material

The content of montmorillonite in the backfill material must be sufficient for the backfill to satisfy the requirements on hydraulic conductivity and swelling pressure. The montmorillonite content also affects the material's compression properties and thereby its ability to limit the upward expansion of the buffer. The reference design entails a bentonite clay with a montmorillonite content within the range 45–90 percent.

The work during the system design phase is aimed at gathering more data on backfill material and devising a strategy for how different materials with varying properties can be handled during production and installation of the backfill.

In the reference design, the same material is used for the pellet fill as for the blocks. However, the properties of the pellet fill need to be optimized with respect to its function in the deposition tunnel. The pellets are blown in as soon as possible after a section with blocks has been installed in the deposition tunnel. The idea is that the pellet fill should absorb water seepage and in this way protect the blocks during the installation phase. Dripping water on the blocks otherwise causes erosion relatively quickly, which can lead to instability in the stack of blocks. Related to this optimization, work will be pursued to determine the limit for acceptable water inflow in a deposition tunnel during the installation phase. The point of departure for this work is the conditions that prevail in Forsmark and the results of previous tests performed at the Äspö HRL and the Bentonite Laboratory /13-3/.

13.2.2 Installed density and geometric configuration of the tunnel

The geometry of the tunnel influences the density that can be achieved in backfilling. A smooth tunnel profile with small deviations permits a larger portion of the profile to be filled with blocks, resulting in higher density /13-2/. The reference design stipulates that at least 60 percent of the tunnel volume should be filled with blocks and the rest with pellets.

Further modelling of the buffer/backfill interface is planned to study more closely how the tunnel geometry influences the ability of the backfill to limit the upward swelling of the buffer.

13.2.3 Fabrication and handling of components and material

Backfill blocks are fabricated by uniaxial pressing using commercially available equipment and technology, see Figure 13-2. Pressing of a large number of small blocks has shown that blocks can be fabricated with very small variations in size and density /13-4/.

During the next few years, backfill blocks will be pressed for use in various tests in the Äspö HRL. Experience from these pressings will further improve the body of data available on the pressing of backfill blocks.

An overall concept for handling, storing and transporting backfill material and backfill components has been developed. Buffer material will be handled with the same system, but backfill is the dominant portion in terms of volume. The components in the handling chain are shown schematically in Figure 13-3. The principle in designing machines and equipment is to use available and proven technology wherever possible. In cases where special equipment needs to be developed, prototypes will be devised and tested during the system design phase.

Figure 13-2. Commercial press for block fabrication.

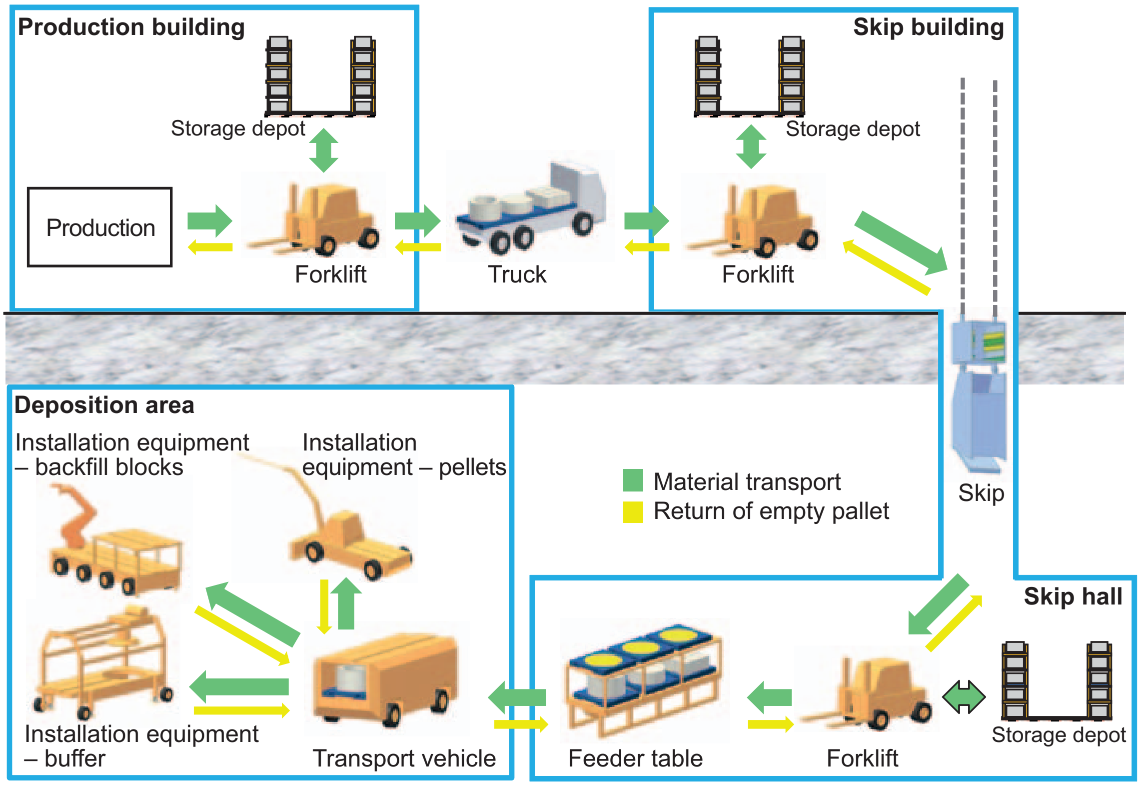


Figure 13-3. Overall concept for the transportation system for buffer and backfill.

13.2.4 Installation of backfill in deposition tunnels

Installation of backfill blocks is an operation that requires equipment with high capacity. At the same time the work must be done with precision to minimize the gaps between the backfill blocks and thereby ensure high quality (correct density) of the backfill. The equipment must also be able to be relocated quickly, since backfilling is done in sections of 6–10 metres at a time, alternating with other activities in the deposition tunnel. This means that the equipment must be removed from the tunnel each time a canister is to be deposited.

During the period 2007–2010, studies have been done to clarify whether the material can be handled with manually controlled machines or whether robot technology should be used. Robot technology is judged to be of interest for further study, since robot solutions are available on the market that can meet both environmental requirements and requirements on capacity and precision. Studies have also been made of the handling technology for the blocks. As in the case of the buffer blocks, the plan is to use vacuum technology to lift the blocks, since this method is gentle on the blocks and enables them to be positioned close enough to each other.

Computer simulations of the robot technology will continue during the system design phase, to be followed by full-scale tests at the Äspö HRL. At the same time, a concept will be developed for the equipment that will carry the backfilling equipment.

Figure 13-4. *Robot for installation of backfill blocks.*

Figure 13-5. *Stacking tests performed at the Bentonite Laboratory on Äspö.*

13.2.5 Inspection of material, components and installed backfill

It has been clarified which parameters are important to inspect during production and installation of backfill to verify that the backfilling process meets stipulated requirements and that the initial state is achieved. The accuracy needed in inspections of important parameters will be determined in the system design phase. Inspection methods will be further developed and tested to ensure the reliability of the inspection methods so that sufficient accuracy is achieved.

13.3 Requirements and premises – plugs

The deposition tunnel is closed by building a concrete plug just inside the tunnel opening. The premises and requirements for the plug have been changed compared with the plug design used in the Prototype Repository. Experience from the plug design in the Prototype Repository is described in /13-5/. The primary requirement on the plug is that it should resist the groundwater pressure and the swelling pressure exerted by the backfill in the deposition tunnel. The plug design should also prevent water transport out of the deposition tunnel.

The concrete plug in itself cannot meet the watertightness requirement. The design therefore consists of a plug section, a seal section and a filter section, see Figure 13-6. Prefabricated concrete beams separate the sections from each other. The concrete plug needs to remain leaktight until the bentonite seal is fully water-saturated. The bentonite seal should then seal against leakage and the concrete plug only act as a mechanical restraint.

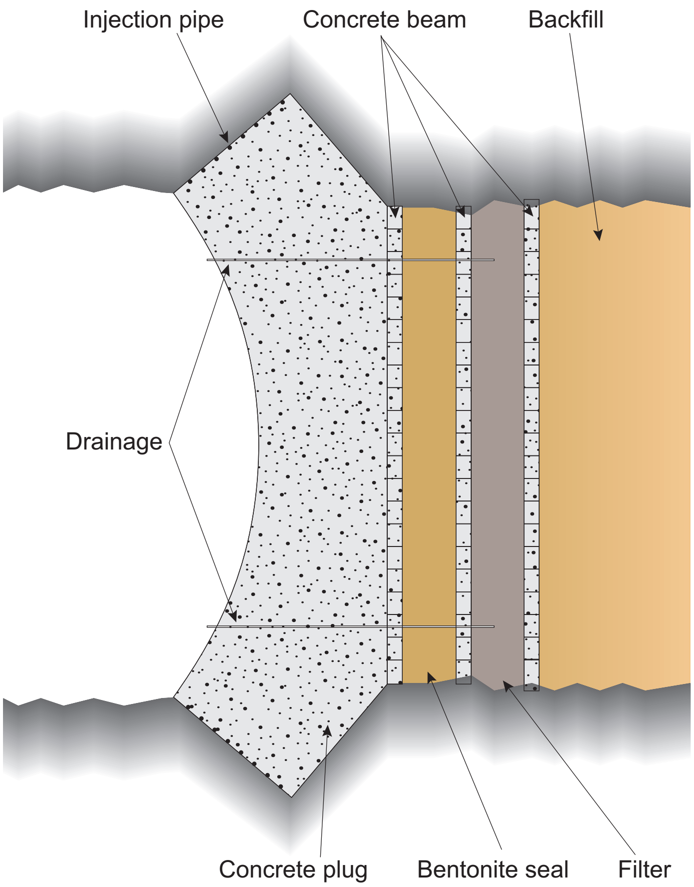


Figure 13-6. *Section through a plug in a deposition tunnel.*

The three most important general requirements made on the plug design with respect to function are:

1. The concrete plug, and in particular the adjacent bentonite seal, shall ensure that any water that leaks into the deposition tunnel after backfilling is kept in. Since the water pressure that is built up at the depth in question is higher than the bentonite's swelling pressure, the bentonite seal inside the concrete plug cannot absorb inflowing water to begin with. The result may then be so-called *piping* and subsequent erosion. This is the reason the plug must be watertight. The bentonite does not generally seal until the voids in the buffer and backfill have been filled with water and water pressure is absorbed by the plug.
2. In order for the concrete plug to be watertight it must be able to withstand the load generated by the water pressure on the inside of the plug.
3. The concrete plug must function until the water pressure on the outside of the plug (towards the transport tunnel) is equal to the water pressure on the inside, i.e. until there is no pressure gradient over the plug.

13.4 Current situation and programme – plugs

Development of the plug is in the design phase. Work is under way on system design, and the objective is to conduct a full-scale test at the Äspö HRL at the end of 2011.

The chosen reference design for backfilling with precompacted blocks of swelling clay entails that the backfill nearest the plug may need to be adapted so that the swelling pressure against the plug is not too high. Furthermore, the concrete must be made with low-pH cement. This, together with the need for a very tight plug design, entails that some parameters need to be verified by testing.

Two design alternatives were evaluated in the concept phase: arched plug and friction plug. The conclusion following calculations /13-5, 13-6/ was that the arched plug, together with bentonite seal and filter (Figure 13-6), is a more watertight design than the friction plug. The calculations for the arched plug are based on data for a concrete recipe with low-pH cement /13-7/.

SKB sees great advantages in making the concrete plug without reinforcement. The advantages of an unreinforced plug of low-pH concrete over a reinforced one are:

- No risk of rebar corrosion.
- Crack risks related to the rebar due to shrinkage of the low-pH concrete are eliminated.
- Time gains in installation.
- Lower costs.

The goal of the continued development work is a finished system design for the plug. This entails the following:

- Further develop the conceptual design.
- Prepare a costed plan for industrialization/implementation with inspection methods that show how the subsystem can be implemented and inspected so that stipulated requirements are fulfilled.
- Verify that design premises and requirement specifications are fulfilled.
- Perform technical risk analysis, which includes risks related to both the design and the plans for industrialization/implementation with inspection methods.

In the ongoing project for system design of the plug, construction documents are being prepared for an unreinforced plug of low-pH concrete. The project also includes cooling and contact grouting, as well as execution descriptions with inspection programme. Cooling and concrete grouting are used to ensure that the concrete plug is firmly confined into the rock, and to seal the potential leakage pathway between rock and concrete. The planned technology for cooling and contact grouting does not differ much from that used for the plugs in the Prototype Repository /13-8/.

Before a full-scale test can be realized, the following work must be done:

- Modelling and tests of bentonite seal and filter to ensure function.
- The requirements on the tightness of the plug need to be concretized in terms of flow and geometry.
- The concrete recipe must if possible be adapted to the requirements and premises made by the arched plug. Certain parameters – such as shrinkage, tensile strength and creep – need to be studied more closely.
- An inspection method for tightness must be developed.
- Production adaptation of rock excavation method for slots.

In addition, extensive design work is required for the test setup with regard to pressure buildup and instrumentation as well as execution. In the longer perspective, the design will be optimized and production-adapted. This includes studying alternative solutions that meet stipulated requirements but are simpler to achieve.

***Figure 13-7.** Cast plug for the inner section in the Prototype Disposal Tunnel at the Äspö HRL.*

14 Technology development, closure

Closure of the repository includes backfilling and sealing of all openings except the already backfilled deposition tunnels, plus closure of investigation boreholes. With the exception of certain investigation boreholes, closure will not begin until all spent fuel has been deposited. Design and execution therefore still lie more than 50 years in the future.

Research and development specifically related to technology for repository closure has not yet been carried out. On the other hand, SKB has for many years studied and conducted considerable research, including full-scale tests on backfilling and closure of deposition tunnels. SKB has also developed and tested technology for closure of investigation boreholes. Experience and results from these efforts will be important for the future development of the technology for repository closure.

In its review of RD&D Programme 2007, SKI called for an account of how backfilling of other parts of the repository than the deposition tunnels will be carried out. SKI and SSI also commented on the need for long-term tests at the Äspö HRL and technology development for plugging of investigation boreholes. With reference to newfound knowledge about piping/erosion, buffer erosion and reaction between cement and bentonite, SKI considered that SKB needs to investigate whether the methods for plugging of investigation boreholes with bentonite need to be updated.

14.1 Requirements and premises

Underground openings

The closure must not adversely affect the function of the other barriers to any appreciable degree. It must also retain its barrier function for a long time in the environment that will prevail in the Spent Fuel Repository. The closure in the main tunnels is supposed to prevent the backfill in connecting deposition tunnels from losing its barrier function by expanding or being transported out of the tunnels. The closure in one underground opening must not affect the closure in an adjacent underground opening so that its function is jeopardized. In the uppermost parts of the ramp and the shafts, the closure (the top seal) should be designed so that it greatly hinders intrusion into the repository.

According to given design premises /14-1/, the closure shall prevent the formation of water flow paths between the repository area and the ground surface that could jeopardize the rock's barrier function. This requirement is judged to be met if the hydraulic conductivity in all openings except deposition tunnels, central area and the uppermost parts of the accesses is less than 10^{-8} metres per second. The value is an average for long distances. Higher hydraulic conductivity can be accepted in sections where a tunnel or the ramp passes zones with elevated transmissivity.

In the underground openings where there are limiting requirements on hydraulic conductivity, the density of the chosen material must be high enough for the closure to maintain its barrier function. It should also be possible to install the closure in these openings with sufficient density. To achieve this, demands must be made on the geometry of the underground openings and on acceptable water inflow during installation.

Like other systems and components in the final repository, the closure must be based on proven or tested technology. Furthermore, it must be possible to fabricate, install and inspect the closure with high reliability.

Boreholes

Boreholes, drilled both from the surface and from openings in the repository, must be sealed. According to given design premises /14-1/, the boreholes must be sealed so that they do not unduly impair the ability of the repository to prevent the escape of radionuclides. A preliminary assessment is that this can be achieved if the borehole seal has a hydraulic conductivity that is less than 10^{-8} metres per second. Higher hydraulic conductivity is accepted in sections where the borehole passes zones with elevated transmissivity.

14.2 Current situation and programme

14.2.1 Reference design

As a basis for the safety assessment SR-Site and based on stipulated requirements and design premises, SKB has established the following reference design for closure of the different parts of the final repository. The reference design applies until results from research and development with a focus on repository closure warrant changing the design.

Main tunnels, transport tunnels and central area

Figure 14-1 is a schematic illustration of the chosen reference design for closure of main and transport tunnels and the central area. The reference design for main and transport tunnels is backfilling with compacted blocks and pellets of swelling clay, according to the same concept as for backfilling of deposition tunnels, see Section 13.2.4. The reference design for closure of the central area is backfilling with crushed rock that is compacted.

On passage of water-conducting zones, it may be necessary to install plugs. Plugs may also be needed where different closure materials border on each other, for example where the transport tunnels connect to the central area. In the current phase of technology development, the same conceptual reference design applies to plugs for closure of main and transport tunnels as to plugs in deposition tunnels.

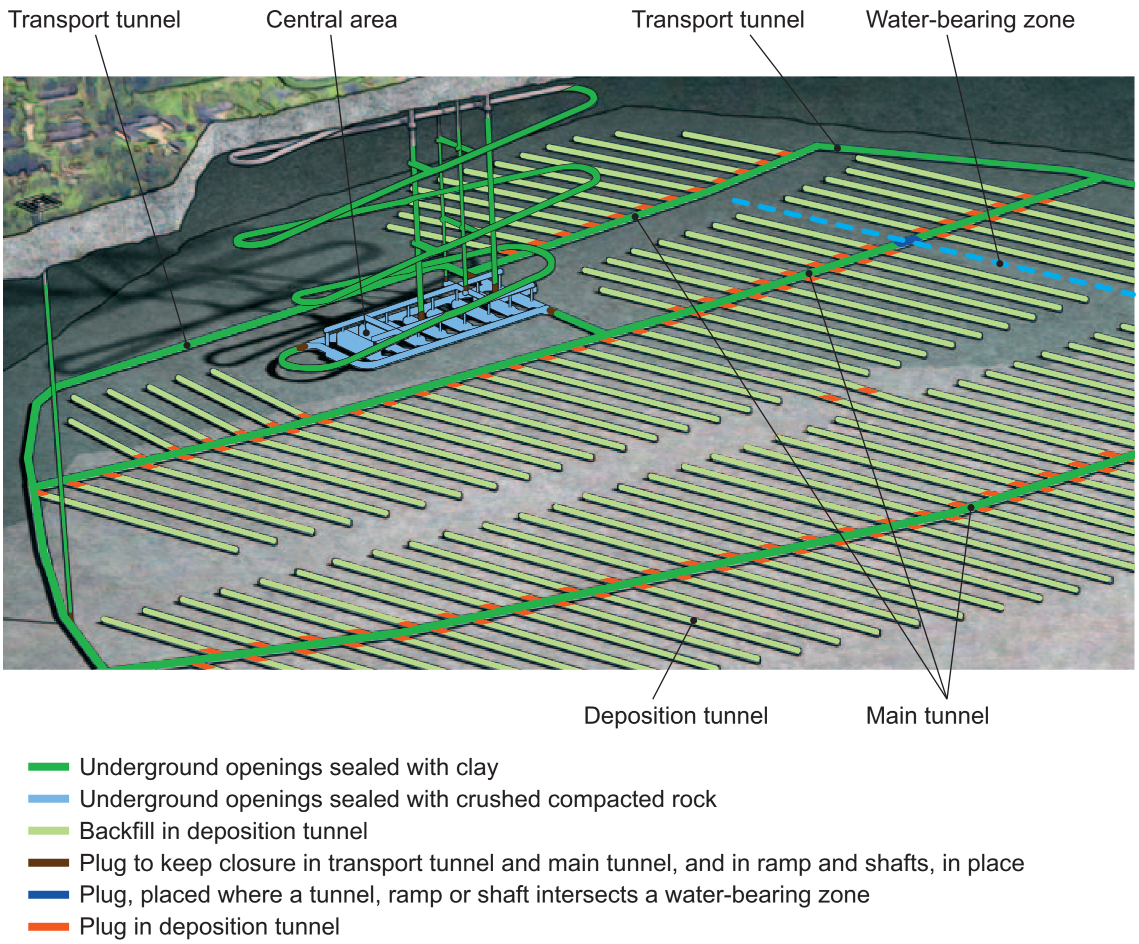


Figure 14-1. *Schematic illustration showing reference design for closure of main and transport tunnels as well as the central area.*

Ramp and shafts

The reference design for closure of ramp and shafts, up to the top seal, is the same as for main and transport tunnels. The chosen reference design for closure of ramp and shafts, including the top seal, is illustrated in Figure 14-2.

Top seal

The reference design for closure of the upper part of ramp and shafts (from the –200 metre level up to –50 metres) is backfilling with crushed rock that is compacted. To hinder intrusion in the repository, the uppermost parts (from about –50 metres up to the ground surface) are sealed with coarser rock material.

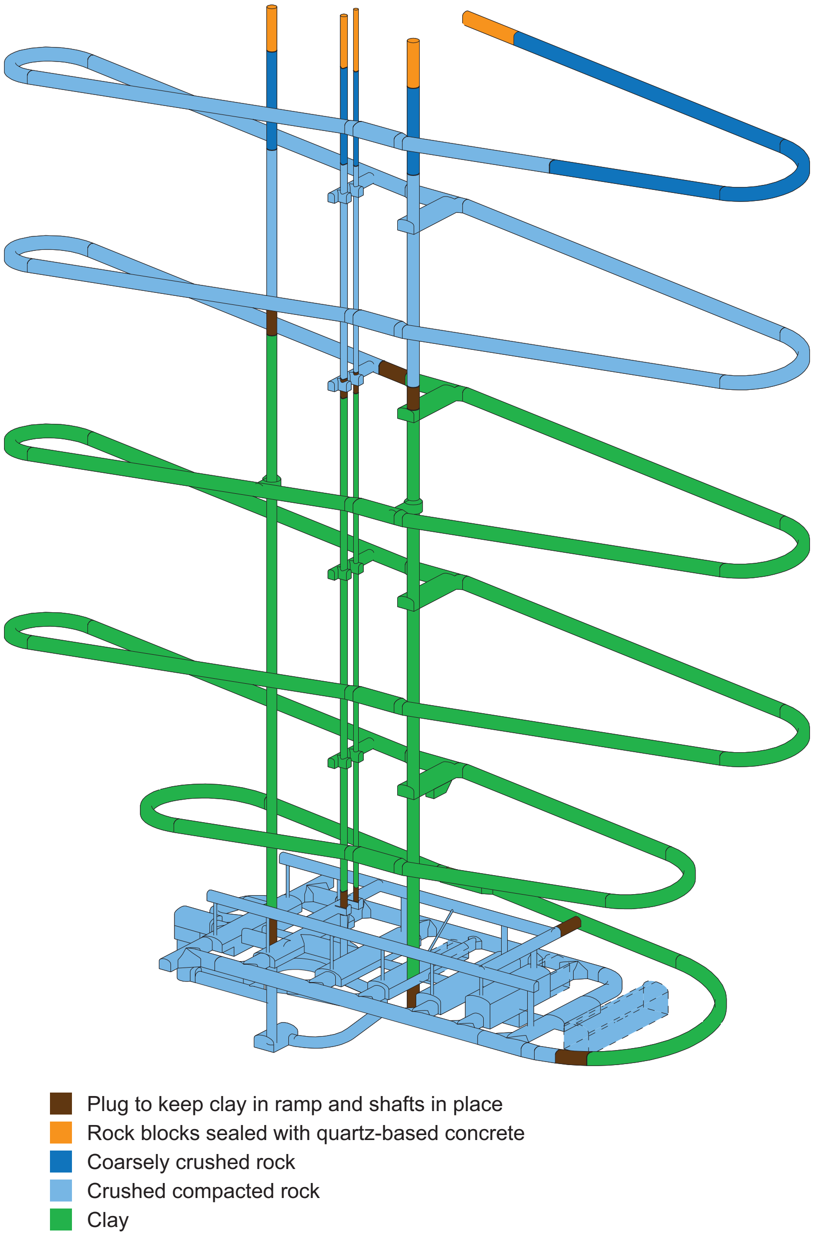


***Figure 14-2.** Schematic illustration showing reference design for closure of ramp and shafts.*

Investigation boreholes

The reference design for sealing of investigation boreholes is perforated copper tubes filled with highly compacted bentonite. In sections where the borehole passes water-conducting zones, the clay may erode. These passages are therefore filled with quartz-based concrete, which is permeable and resistant to erosion. The uppermost part of the borehole is sealed with a material that resists the clay's swelling pressure and withstands mechanical stresses. Cylindrical rock plugs in combination with in situ-cast concrete plugs and well compacted till have been chosen for the reference design. The design is illustrated in Figure 14-3.

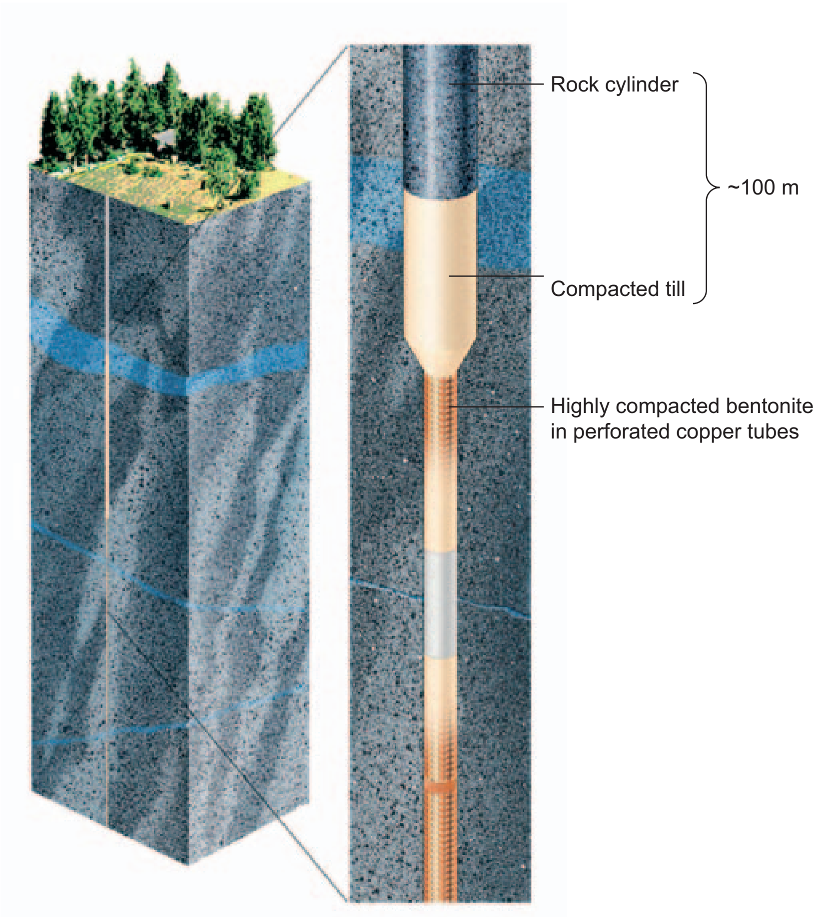


Figure 14-3. *Reference design for closure of investigation boreholes.*

14.2.2 Programme

SKB plans to study alternative design concepts for repository closure during the next three years. The intention is that the work should provide a basis for system design of the possibly revised reference design resulting from the concept studies.

The planned efforts will mainly consist of desk studies; no extensive tests are planned. The results of the work with system design for the backfill and of the work with development of plugs for the deposition tunnels (see Sections 13.2.1 and 13.4) will comprise an important basis for the project. Parts of the work may be done in cooperation with Posiva.

The following activities are planned:

- Preliminary proposals will be prepared for site-adapted solutions for closure of the repository in Forsmark. The studied concepts will be based on revised requirements on the closure, including plugs, established with the guidance of results from SR-Site. The revised requirements will probably warrant the study of closure concepts that are simpler than the reference design presented in the current version of the closure production report. Simpler closure concepts may entail the choice of another material composition and geometric configuration. In order to ensure that the alternative closure concepts do not lead to poorer safety than in SR-Site, sensitivity analyses, mainly hydrogeological simulations, will be performed of the long-term safety of a repository in Forsmark.
- Design requirements for the rock excavation in ramp and shafts under the top seal need to be determined, with reference to e.g. the possible presence of an excavation-damaged zone with elevated hydraulic conductivity around the tunnel periphery. This will take place in a special study. The purpose is to ensure that the design and production of the accesses makes it possible later to achieve a suitable closure. There are water-bearing structures in the rock in the upper part of ramp and shafts. For these parts of the accesses, it has therefore not been meaningful to require that the closure should have low conductivity.
- Studies of plugs. In conjunction with repository closure, both temporary and permanent plugs will be needed for different situations and different requirements. The studies will include specification of design and performance requirements as well as proposals for the design of plugs that meet these requirements. Completed and ongoing work with the plug in the deposition tunnel comprises an important basis for the studies.
- The current stage of the borehole closure project will be concluded and the results from the work to date will be summarized. Furthermore, proposals for efforts in areas where the project identifies a need for in-depth efforts will be presented. A question that was noted in the review of RD&D Programme 2007 is how the occurrence of piping/erosion and undesirable reactions between cement and bentonite can affect plugging of investigation boreholes. The reference design for sealing of boreholes may possibly need to be modified, for example by applying one of the other concepts for borehole sealing that SKB has studied /14-2/.

15 Technology development, rock

The rock line includes detailed characterization, design, construction and maintenance of the Spent Fuel Repository's underground openings. The development work spans a wide field and is concerned with methods for investigation, characterization and rock construction, including rock sealing and support measures, as well as development of special equipment. The selection of Forsmark as the site of the Spent Fuel Repository has great significance for the technology development programme.

The activity in the rock line is based on the requirements on the repository's underground openings with respect to long-term safety, occupational safety and efficiency. The requirements must be translated into specifications for the execution of investigations, design and construction. Technology development should ensure that solutions that meet the requirements are available. A parallel and equally important activity is the documentation of completed design and construction. This includes documentation of on-site conditions, how the facility has been adapted to the site (as-built plans) and how the different parts of the facility have been built (quality documentation).

15.1 Requirements and premises

The rock surrounding the repository's underground openings is one of the repository's barriers. It must provide a stable chemical and mechanical environment for the canisters, the buffer and the backfill. Furthermore, the rock must retard the transport of radionuclides, above all by slow groundwater movements. The underground openings must be located and designed in an optimal way in relation to the properties of the rock on the selected site /15-1/.

The underground openings can themselves cause changes in the surrounding rock, for example altered groundwater conditions, stress redistributions and excavation-induced damage. This can affect both the premises for the engineered barriers and the rock's own barrier properties.

The design premises for long-term safety stipulate requirements on the geotechnical design of the facility, how the facility should be adapted to the properties of the bedrock, and how construction is allowed to affect the surrounding rock. The design premises also contain requirements imposed by other barriers and requirements related to the function of the facility during its operating time.

The following requirements are particularly significant for ongoing and planned technology development within the rock line:

Requirements related to the rock's barrier function

- The deposition holes must be located more than 100 m from deformation zones with a trace length at the ground surface of more than 3 km.
- As far as reasonably possible, the deposition holes must be located so that shear movements cannot arise that cause greater loads than what the canister is designed for.
- Within the deposition area, the groundwater must fulfil the criteria stipulated in SR-Can /15-1/.
- The spacing between the deposition holes must be chosen so that – for specified properties of fuel, canister and buffer – the temperature of the buffer does not exceed 100°C.
- The total water volume that flows into a deposition hole from the time the buffer is exposed to inflowing water until it is saturated should be limited.
- Before canister emplacement, the connected effective transmissivity, integrated along the full length of the deposition hole and averaged around the hole, must be less than 10^{-10} m^2/s.
- Excavation damage should be limited and may not result in a connected effective transmissivity, along a significant portion (at least 20–30 m) of the deposition tunnel and averaged over the tunnel floor, higher than 10^{-8} m^2/s.
- Under the top seal, the integrated effective connected hydraulic conductivity in the backfill in tunnels, ramps and shafts and in the excavation-damaged zone surrounding them must be less than 10^{-8} m/s.

Other requirements

In order that buffer and backfill can fulfil their barrier functions, requirements are made on the design of deposition holes and deposition, as well as the maximum permitted inflow of water. Furthermore, there are requirements on maximum water inflows for main and transport tunnels as well as ramp and shafts.

15.2 Current situation and programme

In RD&D Programme 2007, SKB gave an account of development plans regarding investigation and characterization, grouting, drilling and blasting of underground openings, rock support and boring of deposition holes. In its review of the programme, SKI stated that:

- SKB should report detailed plans for designing and implementing a large-scale measurement experiment of the excavation-disturbed zone around a blasted tunnel under realistic rock mechanical and hydrogeological conditions.
- The choice of reference method for excavation of deposition tunnels should be made in conjunction with the submission of the application for construction of the repository.
- There is a need for further initiatives for investigations and measurements in deposition tunnels and deposition holes, but measurements in pilot holes for the deposition holes should also be taken into account.
- Planned research initiatives concerning grouting at great depth are urgent, and SKB should also consider the possibilities of relocation or new formation of leakage.

In addition, SKI offered viewpoints on SKB's programme for specifying the composition of low-pH cement, methods for rock excavation and boring machine for deposition holes.

Since RD&D Programme 2007 was presented, SKB has carried out an extensive full-scale test to demonstrate technology for sealing by grouting under realistic conditions, see Section 15.5.1, as well as extensive investigations regarding the extent and properties of the excavation-damaged zone around a drill-and-blast tunnel, see Section 15.5.2. SKB has also devised a framework programme for detailed characterization. The programme will be presented in conjunction with the application process, and planned development initiatives are summarized in Section 15.4.

SKB will describe and justify reference solutions for design and execution of the final repository's underground facilities, including choice of excavation method for deposition tunnels, in conjunction with coming applications. In SKB's opinion, the reference solutions fulfil the stipulated design premises in all essential respects and are technically feasible to realize. Certain aspects require further technology development, however. Furthermore, technology that has been demonstrated in tests and experiments needs to be further developed for industrial application.

15.2.1 Programme overview

The overall needs and goals that guide the development programme for the rock line are presented in Section 9.3.7. The programme's main tasks and goals can be summarized as follows:

- Further develop the methodology for underground design and the application of the Observational Method, above all with regard to strategies for detailed adaptation of the deposition areas and coordination of detailed characterization of the rock and production of excavations.
- Further develop methods and equipment for detailed characterization with associated modelling, in a first step prior to construction of the accesses, and in a second step prior to build out of the deposition areas.
- Further develop production methods adapted to the requirements made for rock excavation, stability and tightness. The primary goal is to be able to stipulate performance requirements in the building documents for the accesses.
- Ensure that approved and duty-proven engineering materials are available in time for construction of the accesses.
- Ensure that the special machines that are needed – such as a tunnel boring machine with high boring precision, grouting equipment and machines for rock support with wire mesh – are available in time.

- Develop the boring machine for deposition holes, the goal being that the technology will be available when the first deposition area is built.
- Develop inspection plans, including format and procedures for documentation and for as-built plans, so that they are evaluated and finalized when the rock excavation works start. The programme for this will be presented in conjunction with coming applications.

The timetables for the rock line's technology development differ somewhat from those for other production lines, since much of the technology must be ready to be put into operation at the start of construction. This applies to technology for investigations, design and construction of the accesses to the final repository, including shafts and ramp. The remaining technology must be ready to be put into operation as the buildout of the central area and subsequent deposition areas at repository level progresses.

The continued planning of the final repository in Forsmark is based on the site-adapted layout that was produced during the site investigation phase, called Layout D2 /15-2/. The uncertainties that remain regarding this layout must be addressed in the coming design and construction work. The methodology for this is based on the Observational Method, see Section 15.3. The methodology and the actual application of the Observational Method in Forsmark needs to be developed, however, in particular as regards strategies for detailed adaptation of deposition areas and coordination between detailed characterization and production.

Methods for investigations as well as interpretation of results and modelling have been developed over a long period of time. The application of these methods in the study area investigations, at the Äspö HRL and in the site investigations has given SKB a broad knowledge base for future detailed characterization. Advanced and effective methods for borehole investigations are available, as are procedures for all steps from planning to reporting of investigations. The additional requirements that arise during the construction phase nevertheless entail a need for further development of investigation methods and modelling strategies. The plans for this are described in Section 15.4.

The site investigations have also yielded knowledge and experience concerning quality assurance and documentation of investigation data and results from modelling. However, the construction phase involves larger quantities of information, and up-to-date information is required to meet the needs of design and construction. Information needs to be available for safety evaluations, as well as for regulatory authorities and other stakeholders. This requires further development of e.g. tools for data management, modelling and visualization. The planned development work is described in Section 15.6.

The plans for remaining development of execution methods, construction materials and special machines (see Section 15.5) are influenced to a high degree by the selection of Forsmark. The properties of the site are described in the site description /15-3/ and the remaining uncertainties have been summed up in a separate report /15-4/. The consequences of these uncertainties for the proposed repository layout have been analyzed in /15-2/. The following points are of particular significance for further technology development:

- The upper part of the bedrock contains locally highly conductive fractures that must be passed by the accesses to the repository. Strategies and methods for detecting and characterizing these structures are important in order to be able to adopt the right sealing measures in time. At greater depth, the frequency of conductive fractures is very low, but grouting may nevertheless be necessary in places. Current status and planned development work for grouting are described in Section 15.5.1.
- The rock stresses are relatively high. Knowledge of the magnitude and orientation of the rock stresses at repository depth is therefore particularly important in order to minimize the risk of overloading of the rock around underground openings. Measurements of the rock stresses and ongoing observations of signs of overloading, especially stress-induced spalling, will be important for the detailed design of the orientation and cross-section of the tunnels. The plans for obtaining reliable knowledge of the rock stress situation are described in Section 15.4.

Development needs also exist within the area of execution methods, materials and machines that are essentially independent of the site specific rock properties. These needs include rock excavation (Section 15.5.2), low-pH materials (Section 15.5.3) and equipment for boring and finishing deposition holes (Section 15.5.4).

15.3 Methodology for underground design

Current situation

The report on underground design in phase D2 /15-2/ describes the main features of the methodology that will be employed in design and construction so that the requirements on long-term safety and efficiency are met. An overall prerequisite is being able to site-adapt the facility to take maximum advantage of the rock conditions on the site. Good knowledge of the rock properties and their spatial distribution is therefore important. Even though we already have good knowledge of the range of variation of the rock properties in Forsmark, we don't know the exact locations of, for example, fractures that can necessitate the rejection of deposition holes (see Section 15.4) or of areas where the rock has lower thermal conductivity. Only when these data are available, i.e. when underground investigations have been carried out, can the final layout of the repository, including exact canister positions, be determined. In order to nevertheless get a controlled design process, principles based on the Observational Method will be used.

The Observational Method is a design method that is applied in underground construction when it can be difficult to predict in advance all parameters that may affect the project. The approach was described for the first time in the late 1960s /15-5/. Possible and acceptable behaviour must be established for the design in question. By "behaviour" was originally meant the strength and stability of the design. In the case of the final repository the concept has been broadened to include adaptation to the design premises, including requirements with regard to long-term safety plus performance requirements related to other barriers, for example maximum permitted in-leakage for installation of the backfill.

In the case of the Spent Fuel Repository, today's knowledge level is represented by the results of the site investigation, plus analyses of the prospects for building the facility /15-2/, including assessment of consequences for long-term safety and environmental impact. In order to ensure the acceptable behaviour of the design, the Observational Method prescribes a monitoring plan, plus a plan of contingency actions in the event the design does not behave within the framework of what is acceptable. Regarding the repository's strength, stability and watertightness, the Observational Method can probably be applied as intended according to /15-6/. Further development is needed, however, to meet SKB's special needs to adapt the details of the layout to ensure an efficient and rational utilization of the good rock conditions in Forsmark. To achieve this, it is necessary to specify how detailed characterization is to be executed and interpreted.

Programme

SKB intends to further develop the design method that was employed to obtain the current preliminary design of the final repository /15-7/. Furthermore, the requirements must be adapted to the feedback from SR-Site, and the scope of the continued design work is specified so that resources to prepare construction documents for the final repository's accesses can be procured. During this design work, the requirement specifications that will apply to the rock excavation works are gradually developed, first for the accesses and the central area. Prior to the licence application for trial operation, all procedures for construction and investigations must be developed to a level such that integration of progressive investigations and modelling provides the up-to-date input data needed to guide the rock excavation works in the deposition area.

15.4 Tools for detailed characterization

Current situation

In RD&D Programme 2007, SKB described the development of methods and instruments considered necessary to be able to carry out geoscientific investigations of the rock during construction and operation of the final repository. SKB wrote then that the extensive activities during the site investigations in Forsmark and Oskarshamn and the other research and development activities conducted since the mid-1970s had provided access to methods, instruments and experience that will be useful for the investigations during construction of the final repository. At the same time, SKB pointed out that the mode of working will have to be developed to meet the higher demands on speed and efficiency in construction and operation. Certain instruments and methods need to be improved and adapted for underground investigations, and completely new methods may have to be used for certain purposes.

SKB has devised a framework programme for the geoscientific and surface ecological investigations that are required for construction and operation of the final repository. This detailed characterization will provide a basis for the final adaptation of the repository to prevailing rock conditions. The framework programme (detailed characterization programme) will be presented when the applications are submitted. During the years up to the start of construction the programme will be progressively developed and defined, based on results from SR-Site and its review. Figure 15-1 provides an overall picture of the process for the development of the detailed characterization programme.

The detailed characterization programme also includes plans for continued development of methods and tools for investigations, modelling, data management and quality assurance. These are priority tasks for SKB, since well functioning investigation methods and efficient modelling tools are crucial success factors in being able to carry out design and construction in the intended manner.

Programme

Here the areas are described where greater development efforts are required or where development is particularly important in view of the need for information to be able to assess long-term safety.

Figure 15-2 shows an overall timetable for the development of instruments and methods for investigations deemed to be particularly important. Development work and tests for certain methods and instruments should be completed before the rock excavation works begin. Others do not have to be completely developed before tunnelling is commenced in the repository area. It is important that the methods are fully proven and performance-tested before they are used, especially the methods that will be used in deposition tunnels and deposition holes.

Knowledge and experience gained from investigations in conjunction with other underground projects will be studied as a part of the preparations for start of construction. Posiva's ongoing tunnelling for a final repository in Olkiluoto, Finland, is of particular interest, since the issues related to long-term safety are the same there as in SKB's final repository project. SKB and Posiva also plan to collaborate on the development of instruments and methods in a number of areas. Ongoing tunnelling works in the Stockholm area, as well as underground projects in other parts of Sweden and other countries, can also provide important experience.

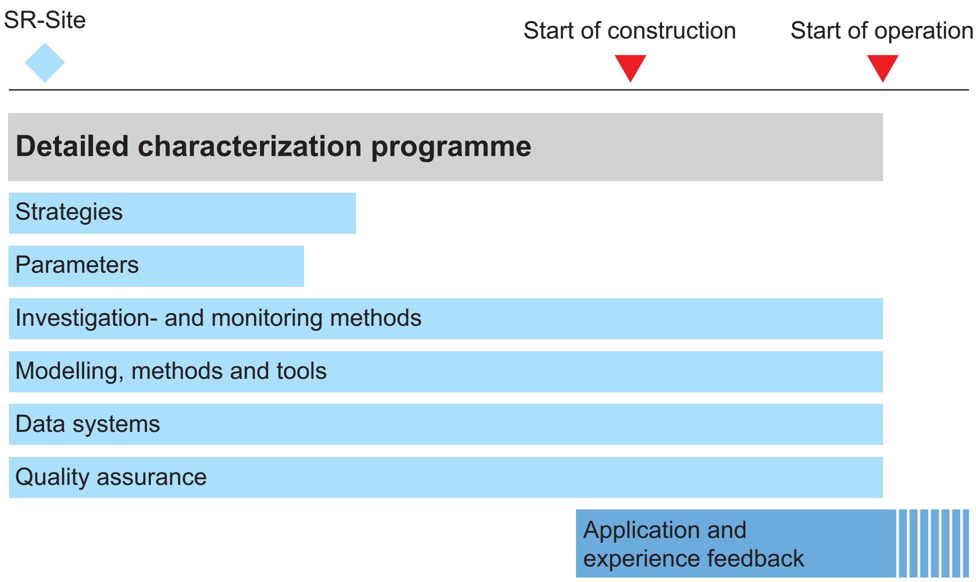


Figure 15-1. *Overall timetable for development of the detailed characterization programme up to construction and operation.*

Figure 15-2. *Overall timetable for development of methods and instruments for investigations deemed to be particularly important by SKB.*

Geodesy

Available methods for measurement of borehole deviation are still associated with considerable uncertainties. Industrial development in this field will be followed, and SKB's own technology development of equipment for measurement of borehole deviation cannot be ruled out.

Installation of buffer in the deposition hole imposes requirements on the straightness of the hole as well as deviations from the plumb line and intended geometry in other respects. SKB plans to develop instruments and methods to verify these requirements. The technology must be fully developed and tested in good time before the start of construction of the deposition area.

Sweden is currently changing its national coordinate system. The national reference system for position indications, RT 90, will be replaced by SWEREF 99, while the system for height indications, RH 70, will be replaced by RH 2000. SKB plans to switch to the new systems before the start of construction of the final repository. Transforming all data from the site investigations that refer to older coordinate systems will entail a great deal of work.

Geology and geophysics

The geological documentation of the final repository's underground openings will require a tunnel mapping system that meets high standards with regard to quality of results, robustness and user-friendliness. Adaptation of a system based on photogrammetry with the use of digital photography for rock mapping of tunnels has begun. The development work, including tests at the Äspö HRL, is planned to be finished in good time before the start of construction.

Identification and characterization of "long fractures" (individual fractures or minor deformation zones) is a priority task for detailed characterization, since such fractures can disqualify canister positions. According to the design premises (linked to the EFPC criterion /15-1, 15-8/), a deposition hole intersected by a "long fracture" that can be followed around the full perimeter of the tunnel should be rejected. Furthermore, if a fracture intersects five or more deposition holes it should be rejected.

In addition to the geological mapping methods for tunnels and boreholes, a combination of geophysical and hydraulic methods probably has to be applied to identify and characterize long fractures. The basic methods are established, but some upgrading is required and methodology for co-interpretation of measurement data and modelling needs to be devised. This work has been initiated and is being pursued partly in cooperation with Posiva. The final results will not be applied until the first deposition area is built, but since the question is of great importance the technology may need to be tested and fine-tuned for several years before it is put into routine use.

A total of 6,000 deposition holes will be bored, investigated and approved in the final repository. This will require well developed technology for mapping of the surface of the borehole wall. Adaptation and further development of the technology for rock mapping in tunnels is planned for application in deposition holes. Scaling down from tunnel to deposition hole requires considerable development in order for photogrammetry to be used. The mapping technique will be put into use when the first deposition area is built. Testing and fine tuning at both Onkalo and the Äspö HRL is foreseen as part of the implementation phase.

Rock mechanics

One of the most important remaining uncertainties in the site model for Forsmark concerns the mechanical state of stress, primarily the magnitudes of the principal stresses. The planning work prior to further investigations includes identifying the method combinations that have the greatest potential for reducing these uncertainties.

Borehole measurements with overcoring and hydraulic methods will be less resource-consuming to do during the construction phase than during the site investigations, since they can be done in short holes drilled from the tunnels. When shafts and tunnels begin to be driven, convergence measurements will be employed as a proven alternative and complement to borehole measurements. In preparation for the convergence measurements, the measurements and analyses conducted by Posiva in Onkalo are being studied.

Two other methods, SLITS (SLim borehole Thermal Spalling) and LSG-LVDT (Long Strain Gauges-Linear Variable Differential Transformer), are under development. The development work is being pursued in cooperation between SKB and Posiva. In the event of a successful outcome, both methods will be used. The SLITS method is based on the fact that heating of a rock volume around a loaded borehole can give rise to an additional load, leading to spalling. The method has potential to become a robust way to keep track of the orientation of the biggest primary stress. The LSG-LVDT concept is an overcoring method where measurement is performed on a larger scale (larger diameter boreholes) than in the case of conventional overcoring methods. SKB plans to calibrate the method against known conditions in the Äspö HRL and also intends, in cooperation with Posiva, to integrate LVDT measurement with the SLITS method in holes drilled from tunnels.

Thermal properties

There is a need to further develop field methods for characterization of the bedrock's thermal conductivity that are more rational and less costly than today's. SKB has considered some alternative development variants for determining the thermal conductivity of the rocks, in situ and on a scale that is relevant for the canister. The development need is difficult to judge at present. SKB is participating in a working group under the ISRM (International Society of Rock Mechanics) that is supposed to propose a methodology for determination of thermal properties.

Hydrogeology

SKB has a number of hydrogeological methods at its disposal for use both on the surface and under ground. A priority task in the planning of the detailed characterization is to select the investigations methods that will be used under ground. The bedrock in Forsmark is characterized by high water flow near the surface, while very low conductivity is prevalent at repository depth. These extremes can make special demands on the measurement methods. The measurements that will be performed concern characterization and monitoring in boreholes, plus documentation of in-leakage to tunnels and deposition holes. Two criteria are crucial. Firstly, the hydraulic tests must meet stringent requirements on measurement accuracy, and secondly, rational logistics and flexible adaptation to other activities in conjunction with construction and operation must be achieved.

SKB plans to use a combination of outflow tests, injection tests, flow logging and interference tests. Feasibility studies will be initiated to determine the testing methods and combinations of methods that appear optimal in terms of quality and investigation logistics. Requirement specifications will be prepared for equipment and evaluation methodology will be established. This will be followed by technical development, where deemed necessary.

Special development initiatives are planned for methods for measuring the small inflows that are expected to occur to pilot holes for deposition holes and finished deposition holes. Development of methods for measuring small in-leakage in tunnels is also planned. This work will be pursued partly in cooperation with Posiva.

Hydrogeochemistry

Most hydrogeochemical field and laboratory methods that SKB has at its disposal meet the requirements on quality and functionality. However, substantial development needs have been identified on two points. One is complete chemical characterization with a mobile measurement cell, under ground and with on-line measurement of Eh, pH, EC, O_2 and water temperature. The other is determination of the colloid concentration by means of single particle counting.

An initial trial with a portable instrument, which could be used for pH and Eh measurements at the Äspö HRL, was conducted in the Prototype Repository. All instruments have so far been equipped with specially adapted electronics, since commercial instruments have been found to have problems in the tunnel environment, even though they work well in a laboratory environment.

Transport properties

The discipline includes both field and laboratory methods. Only minor development needs have been identified. Further development is planned of a simplified application of long-term experiments in individual boreholes for determination of diffusion properties (LTDE-SD), see also Section 25.2.13. Development is also planned of a new laboratory method for determination of sorption coefficients, matrix diffusivity and matrix porosity based on electron migration, as a complement to through-diffusion measurement. Furthermore, a new variant of laboratory determination of inner surface areas in the rock (the BET surface area) for whole drill core pieces is under development, which could become a new standard method, see also Section 25.2.15.

Monitoring

When underground openings become accessible, the monitoring will be broadened to include the inflow of groundwater to different tunnel sections. Traditional technology is used for this, but certain improvements are foreseen. In preparation, experience from the fine sealing project at the Äspö HRL /15-9/ and results from ongoing BeFo projects will be studied and evaluated.

SKB has initiated a study of the premises for installing a local seismic network in Forsmark. The purpose is to continuously record, during the construction of the final repository, natural or induced seismic activity of considerably lower magnitudes than can be recorded in the national seismic network. It will be possible to record both natural earthquakes and blasting. Not only can a database of earthquakes be built up, but measurement data can be analyzed with respect to parameters that contribute to a better understanding of the rock's response to tunnelling. If the study shows that it is warranted to install a local seismic network, the objective is that it should be in operation before the tunnel works begin.

Investigation drilling

Technology for investigation drilling that meets the needs of detailed characterization is judged to be available for the most part. Limited studies and development work will be initiated, however. SKB's high demands on the straightness of the pilot boreholes require method development. A first step will be to conduct a thorough review of existing methods for guidance of boreholes as a basis for possible method development.

A number of drilling-related parameters will be recorded while drilling is being done, a process called MWD (Measurement While Drilling). An ongoing project is investigating whether the tools that exist for analysis of MWD data can provide better predictions of the rock properties than those that have been possible so far. Recent developments on the hardware side are also being studied in the project.

Modelling

Planned development activities for modelling concern methodology for fulfilling essential design premises, or are linked to the utilization of new types of underground data. An essential component is modelling of long fractures in conjunction with application of the EFPC criterion and criteria related to the hydraulic properties of fractures for evaluation of deposition positions. A review is also needed of the methodology for imaging of objects from mapped tunnel surfaces to three-dimensional representation in relevant models. Furthermore, a review will be conducted of the methodology for interpretation of hydraulic properties of flowing structures based on measurements in tunnels, pilot holes and probe holes, with due consideration given to observation scale, two-phase flow and skin effects. Other development of methodology and tools for modelling that are also important for detailed characterization is dealt with in Section 25.3.

15.5 Execution methods, construction materials and special machines

15.5.1 Grouting

Current situation

The rock line's biggest project since RD&D Programme 2007 is the fine sealing project. In this project, an 80 metre long tunnel, called the TASS tunnel, was built in the Äspö HRL at a depth of 450 metres /15-9/. The main purpose was to show that the rock at this depth can be sealed so that the requirement on limited inflow to the backfill can be met.

Before grouting was done and the tunnel was excavated, the rock along the tunnel route was characterized by means of three cored boreholes. Based on the rate of leakage into the boreholes, the rate of leakage into the tunnel (without sealing) was predicted to be 30–120 litres per minute and 60 metres of tunnel. The level of sealing that was achieved, mainly by pre-grouting, corresponded to an in-leakage rate of 1 litre per minute and 60 metres of tunnel. The post-grouting that was done confirmed previous experience that post-grouting under high water pressures is complicated.

As a basis for selecting fan geometries and pumping pressures, predictions were made of the fracture distribution and the smallest fracture that had to be sealed, and penetration lengths were calculated /15-10/. Sealing was done with both conventional grouting fans outside the tunnel contour and with a new geometry where the grouting holes were kept inside the tunnel contour. The alternative borehole geometry also yielded satisfactory sealing results. Two different grouting materials were used: a cement-based grout devised by SKB and Posiva, and silica sol. The project has also yielded valuable knowledge on how the properties of the grouting materials affect the grouting process and its result. This provides a means for controlling material properties and execution to achieve the requisite watertightness with controlled spreading of grout.

Construction of the fine sealing tunnel has generated a large quantity of data which, together with data from the previously built TASQ tunnel /15-11/, provides good opportunities for testing different models for inflow and sealing for the purpose of understanding and predicting these processes.

Programme

SKB's development work to ensure limited inflow to the final repository is aimed at first gaining a fundamental understanding of the mechanisms allow sufficiently watertight underground openings to be achieved with acceptable materials. This understanding will then be translated into methods for investigation and construction, as well as development of equipment and material. In order to have practical control over the work, achieve the requisite results in the field and do this with reasonable efficiency, it is necessary to have smoothly functioning communications, management and handling.

The grouting technology SKB needs must 1) meet the requirements related to long-term safety, 2) cope with the specific sealing scenarios that follow from the site specific hydrogeological conditions, and 3) meet the requirements on quality-assured and efficient execution.

The KBS-3 method does not impose any particular requirements on the long-term durability of the sealing. Sealing by grouting is done for the purpose of temporarily limiting the inflow of water so that buffer and backfill can be installed without eroding. As is evident from Section 15.1, very low inflows must be achieved in deposition holes and deposition tunnels. In addition there are restrictions regarding the impact which the rock surrounding tunnels may be subjected to, for example the creation of new channels for water flow and the introduction of foreign materials into the rock mass.

Inspection of watertightness and in-leakage is essential for result follow-up of grouting, from an environmental monitoring viewpoint and for a better understanding of the hydraulic response of the rock to the construction activity. Inspection of watertightness and in-leakage will comprise an integral part of construction inspection, but technology and procedures need to be further developed.

The site investigation in Forsmark has shown that the rock mass at repository depth has few conductive fractures. The inflows predicted beneath the more water-bearing shallow rock (without grouting) are consequently low, which means that systematic pre-grouting is not judged to be necessary /15-2/. The primary task in the design work at repository level is then to identify the sections along planned deposition tunnels that are intersected by water-bearing fractures where selective grouting may be needed. The development work for grouting within the deposition area is therefore aimed primarily at solving a forecasting problem – being able to predict when grouting is needed based on investigations in pilot and probe holes – and is mainly conducted as a part of the detailed characterization programme.

Use of silica sol is new within rock grouting. There are therefore reasons to actively gather new knowledge and experience from studies and applications in other projects. The results obtained thus far need to be translated into technology for routine industrial production. The waiting time after grouting, before drilling and blasting for the next round can begin, is generally an important question. Reduced waiting time permits great gains in tunnelling productivity and economy. In the case of the Spent Fuel Repository, this can be important mainly for driving of the accesses through the more conductive, superficial part of the bedrock. SKB's intention is to pursue continued development in this area in collaboration with the rest of the underground construction industry.

15.5.2 Rock excavation

Current situation

SKB has chosen conventional drill-and-blast as the rock excavation method for the final repository. The biggest advantages of drill-and-blast are high flexibility, mature technology and comparatively low cost, not just for rock excavation itself but also for subsequent work /15-12/. The method can easily be adjusted to different rock conditions by adaptation of tunnel shape and blasting design to prevailing requirements and site conditions. More detailed reasons for the choice of method and a comparative study of different rock excavation methods will be presented in the application.

In conjunction with rock excavation for the fine sealing tunnel on Äspö (see Section 15.5.1), an evaluation was made of how smooth a contour can be achieved with modern drilling equipment and of what factors affect the extent of blast damage in the tunnel contour /15-13/. Since the primary goal of the project was to obtain a smooth and watertight tunnel, there was plenty of time to evaluate results from one excavation stage to the next (about 20 metres of tunnelling) and take corrective action. The blasting technology that was applied was based on research results regarding how blast damage can be calculated /15-13/ and the results of previous blasting on Äspö /15-14, 15-15/. The results showed that the tolerance requirements made by the backfill on the tunnel contour could be met with good margin, see Figure 15-3 /15-16/.

To study the blast damage in the fine sealing tunnel, radar measurements were made on the tunnel wall and blocks excavated from the wall were subjected to extensive investigations.

The radar measurements were carried out with high-frequency ground-penetrating radar (GPR) /15-17/ in a grid with a point-to-point distance of 0.5×0.5 metre on the tunnel wall. The results are illustrated in Figure 15-4. The Figure shows the distance into the rock from the tunnel contour where elevated dispersivity has been measured (the dispersion zone). Dispersivity represents the frequency dependence of the velocity and amplitude of the radar waves and is affected by the mechanical properties of the rock. An increased occurrence of fractures results in increased dispersivity. The variation in dispersivity may thus reflect the relative fracture frequency in the tunnel wall, and the depth of the dispersion zone gives an idea of the depth to which the fracture frequency is elevated.

Figure 15-4 indicates that areas with greater depth of the dispersion zone as a rule coincide well with the locations of the inner parts of the blasting rounds, where a higher charge concentration is required locally to initiate detonation in the boreholes. Documented natural fractures contribute to the deeper dispersion zone visible in the middle of the figure.

In order to study more closely the effect of blast damage, eight blocks were sawn out from an eight metre long section in the tunnel wall. The blocks were one metre long, 1.5 metres high and about 0.8 metre deep. Each block was sawn into slabs about ten centimetres thick. The slabs were studied in detail with respect to natural and blast-induced fractures.

A three-dimensional model of the extent of the fractures was created /15-18/, see Figure 15-5. Most blast-induced fractures were too short to be modelled between two slabs. The model showed that the blast damage fractures were limited to the inner part of the blasting round, see also Figure 15-4. In most of the model the blast damage is limited to short fractures, sub-parallel to the contour.

Samples over different fractures were extracted and their transmissivity was investigated in the laboratory. The investigation was performed with the aid of a permeameter, which permitted creation of an isostatic pressure to simulate the in situ stress state. The equipment had a detection limit for transmissivity of about 10^{-9} m^2/s. There were not many fractures induced by blasting to test, but the transmissivity of a number of them was below the measurement limit. Supplementary injection tests in short boreholes within the expected excavation-damaged zone were performed at the places where the blocks had been extracted. In some holes the local transmissivity was less than about 10^{-9} m^2/s, while in others it was up to about 10^{-6} m^2/s. Of greater importance, however, is the fact that the induced fractures did not form a continuous network, which meant that the requirement on a maximum transmissivity over a long stretch of tunnel (20–30 metres) could be met /15-19/.

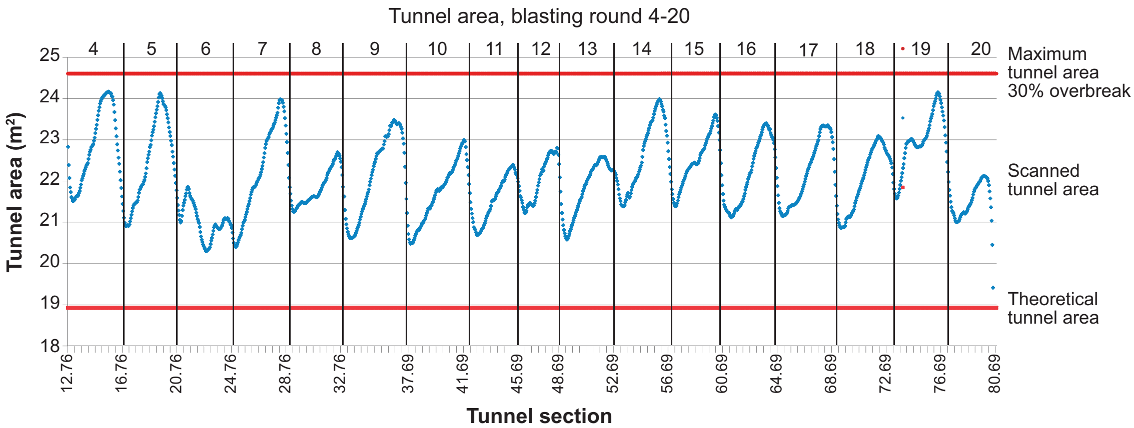


Figure 15-3. *Measured cross-sectional area in the fine sealing tunnel, starting with the fourth blasting round, which was done with full area. The red lines indicate the desired cross-sectional area, a deposition tunnel and the largest permitted cross-sectional area.*

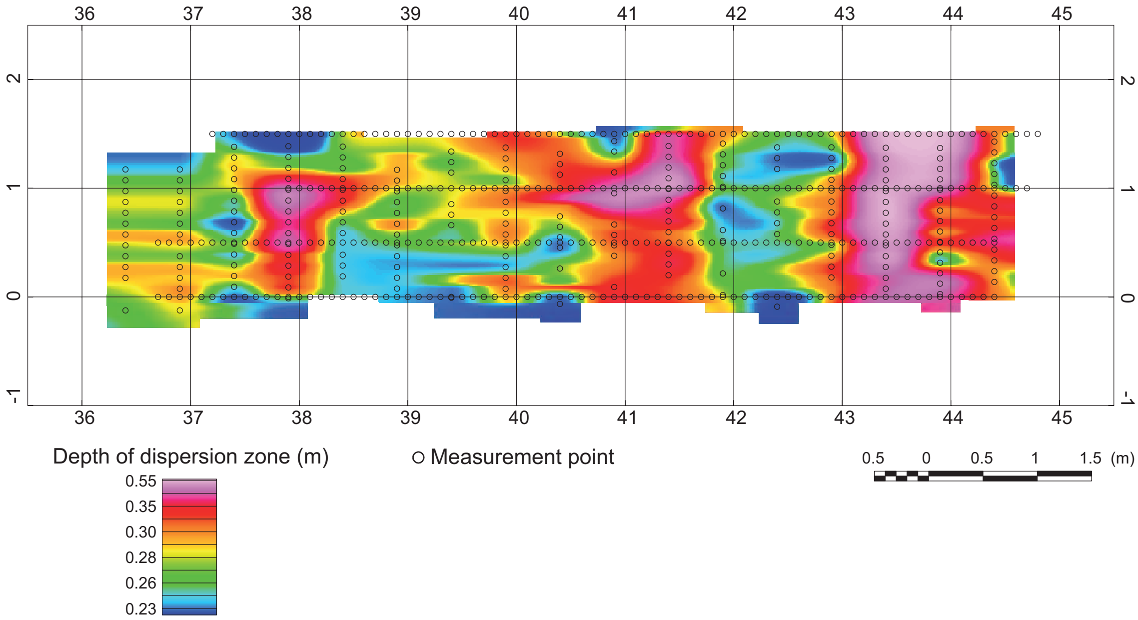


Figure 15-4. *Depth of the dispersion zone (zone of elevated dispersivity) according to radar measurements over the tunnel wall in the fine sealing tunnel at the Äspö HRL. The dispersion zone is interpreted as a measure of the depth to which the fracture frequency is elevated /15-17/.*

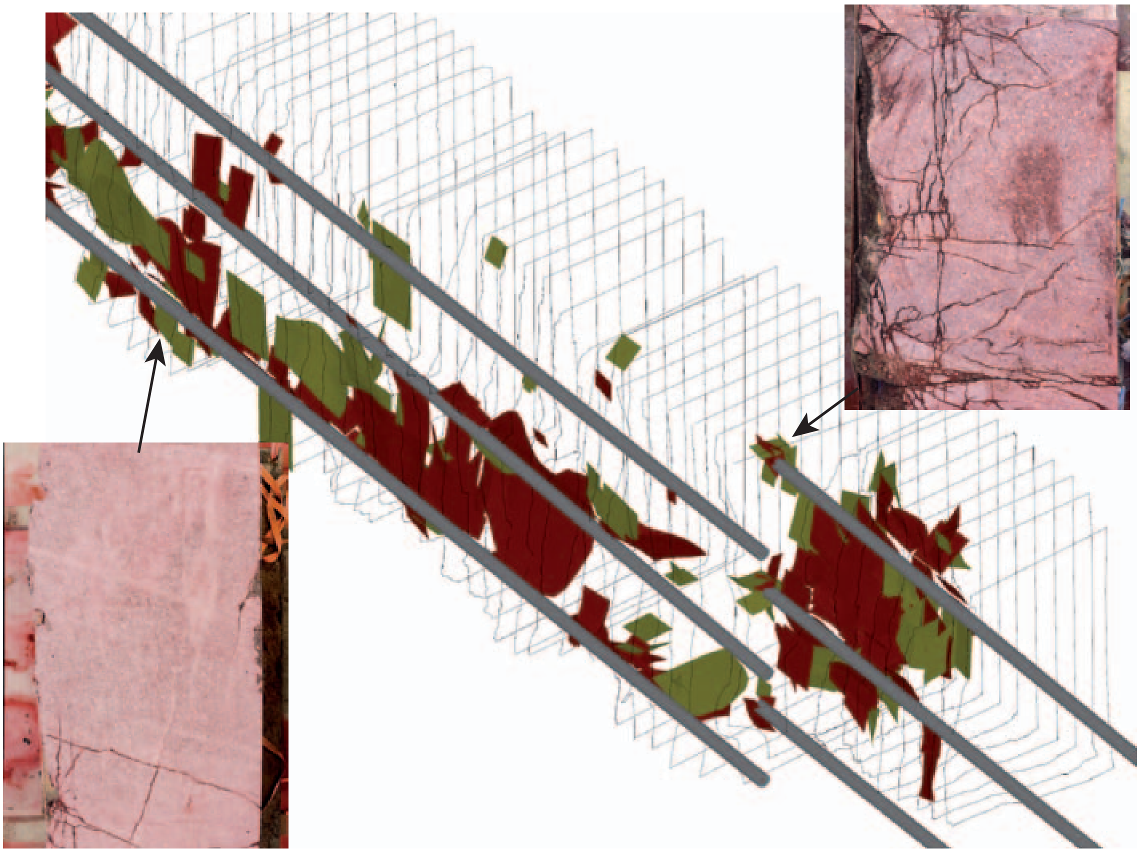


Figure 15-5. *Section from fracture model from fine sealing tunnel. The model covers three boreholes in the contour and extends over three blasting rounds. The photo at the upper right is from the end of a round where the charge concentration is relatively high. The green and reddish-brown fracture planes show the extent of the blast damage. The photo at the lower left comes from a section in the middle of a round and shows – besides natural fractures – some blast damage fractures sub-parallel to the tunnel wall.*

In summary, the investigations of the fine sealing tunnel show that with the employed blasting technique and the applied quality control, it was possible to meet the requirements on the tunnel contour – at least in the wall. This applies with respect to both geometry and not creating a continuous flow path along the tunnel periphery. However, only cartridged explosives were used in order to optimize charge control. Attempts to charge with string emulsion indicated problems in having adequate control over the charge distribution /15-20/.

Programme

Experience from the rock excavation experiments that have been conducted are positive with respect to the requirements the backfill makes on the tunnel contour, and the requirement of not creating a continuous excavation-damaged zone that causes flow paths along the tunnel. Further development is needed to fulfil these requirements in routine production, however. This applies to application of the requirements in design, with regard to both site adaptation of the layout and performance requirements for the different unit operations. The methods for determining the scope of blast damage also require further development.

A number of development questions require collaboration with manufacturers and suppliers. Development of existing technologies to meet SKB's project-specific requirements may for example include:

- Improvement of drilling equipment to ensure precision in the drilling towards the intended endpoint.
- Adaptation of equipment to obtain more information via on-line recording of drilling parameters (MWD).
- In cooperation with suppliers, determine competence requirements for drilling equipment operators.
- Improvement of charging technique. Charging with cartridged explosive can be done with good control of the charge concentration. But emulsion explosive, which is mixed at the tunnel face when it is to be used, has many advantages with respect to safety and working environment. However, the equipment does not have enough precision to control the charge towards the narrow tolerances that apply in cautious blasting. Development of better control equipment for automatic charging is important so that charging can be done with sufficient precision.
- Electronic detonators are necessary to obtain the instantaneous detonation of a number of parallel holes that is required according to /15-13/ in order to minimize the blast damage.

Collaboration with other tunnel construction clients, primarily infrastructure-building agencies and Posiva, can drive technology development and permit costs to be shared with other stakeholders. When it comes to requirements on execution and control of contour accuracy and excavation-damaged zone, SKB seeks collaboration with other actors.

15.5.3 Materials with low pH

Below a certain depth in the Spent Fuel Repository, only low-pH materials may be used. This requirement influences the choice of grout and the materials that can be used for rock support, primarily shotcrete. Grouts are dealt with in Section 15.5.1. The current situation and programme for rock support materials are summarized below.

Current situation

Within the framework of the EU project ESDRED, tests were conducted in 2006 with low-pH shotcrete for rock support at the Äspö HRL. The shotcrete that was used lacked reinforcement, however.

In 2009, SKB conducted new tests with low-pH shotcrete at the Äspö HRL, in an experimental niche (NASA 0408A) at a depth of 56 metres and in the fine sealing tunnel at a depth of 450 metres. These field tests were preceded by laboratory tests at CBI in Stockholm to develop the recipe for the concrete. Material and shotcreting tests were then conducted on a pilot scale at Vattenfall's concrete laboratory in Älvkarleby. This time they included admixture of steel fibres in the shotcrete for reinforcement in the way that is usually done for rock support with shotcrete.

SKB has also installed 20 rock bolts with low-pH grout in the aforementioned experimental niche. The bolts will be monitored over a ten-year period. Of particular interest is the durability of the grout and possible increased corrosion of the bolts due to the fact that they are anchored with a grout with a pH of about 11, compared with ordinary grout with a pH of about 12.5.

Limited tests to determine corrosion of steel in low-pH concrete were also initiated in 2009, and these tests will be monitored for about 10 years. SKB is also the initiator and coordinator of an international project started in 2008 for the purpose of devising a uniform method for measuring the pH of cement products. Besides SKB, other participants in the project are Posiva in Finland, Nagra in Switzerland, ENRESA in Spain and NUMO and JAEA in Japan. pH measurements using the same method are also being performed by CEA in France and RAWRA in the Czech Republic.

Programme

The programme mainly entails monitoring the Äspö tests with rock bolts and shotcrete as well as the corrosion tests on rebar. The international pH project will preliminarily be concluded in 2011, but the network that is established may lead to new cooperation projects.

In conjunction with the planned extension of the Äspö HRL with new rock caverns, grouting may be done with low-pH grout, just as in the fine sealing project. Low-pH materials will also be used for grouting of rock bolts and for rock support to the extent these types of support are needed. Design and construction of an arched plug (see Section 13.4) may also require supplementary tests with low-pH concrete made according to different recipes.

15.5.4 Deposition holes

Current situation

Boring machine

At the Äspö HRL, SKB has bored some fifteen deposition holes for different experiments and demonstration projects. They have been done using TBM technology, adapted for boring vertically downward. Furthermore, SKB has experience of boring of deposition holes using reverse raise boring, both from trial boring of three holes in Posiva's underground tunnel for the VLJ repository for low- and intermediate-level waste in Olkiluoto /15-21/ and from boring of long horizontal deposition drifts for the development of KBS-3H at the Äspö HRL, see Chapter 16.

In the autumn of 2006, a feasibility study was conducted of possible methods for excavation of deposition holes. The following methods were then considered of interest to study:

- Reverse raise boring
- Shaft boring machine (SBM) or tunnel boring machine (TBM)
- Water cluster
- Air cluster
- Water jet
- Core drilling

Based on the feasibility study, reverse raise boring was chosen as a reference method. The reasons were that reverse raise boring has high efficiency, high durability, high safety, meets the requirements on the geometry of the deposition holes and causes little environmental impact. Furthermore, the method is well established technology, which reduces the time and costs required for adaptation to SKB's needs and contributes to efficient operation and maintenance.

During 2007 and 2008, SKB conducted a preliminary design study of a machine that fulfils the relevant requirement specification for the deposition holes. The design is based on a machine for conventional raise boring of shafts, but has been modified to push the reaming head downward instead of pulling it upward. The machine is automated and can position itself correctly in the tunnel. It also carries the drill bit for the pilot hole and the drill pipes required to drill the pilot hole to full depth. The drill pipes are handled by a robot arm mounted on the machine. When the pilot hole is finished, the whole drill string is raised and the pilot bit is replaced with a reaming head which enlarges the deposition hole to full diameter. But first the pilot hole must be characterized and approved. Removal of the drill cuttings is planned to be done with water instead of vacuum/air, since using water requires much less energy.

Finishing of deposition hole

The bottom of the deposition hole has to be levelled so that a flat surface is obtained for buffer blocks and buffer rings to be installed in accordance with the relevant requirements. The current reference solution for this is described in Section 12.2.1. Experience from bottom levelling of the deposition holes in the Prototype Repository shows that the technology needs to be further developed. One practical problem is that the work has to be done on the bottom of a roughly eight metre deep shaft with a diameter of 1.75 metres.

Finishing the deposition hole also requires bevelling of the shoulder between the deposition tunnel and the hole to provide the space needed for lowering a canister. Bevelling by wire sawing with diamond wire has been done for two deposition holes in the demonstration tunnel at the 420-metre level at the Äspö HRL. Both drilling and wire sawing could be done with standard equipment. Schematic drawings of equipment for drilling of the holes for the diamond wire and subsequent sawing are shown in Figure 15-6.

Programme

Boring machine

SKB intends to proceed with the design of a boring machine for deposition holes, according to the method reverse raise boring. The geometric requirements presume that deposition holes can be bored with small tolerances regarding diameter and plumb line. This makes special demands on positioning of the machine, stiffness in the system, and monitoring of deviations during boring. The goal is to be able to complete the design work so that the finished equipment is ready to be put into use when the first deposition area is built. Tests of e.g. handling methods for the drill cuttings and details in the design of the drill bit. Further development of technology for investigation of deposition holes (pilot holes after reaming to full diameter) is planned to be integrated with the development of the boring equipment.

Finishing of deposition hole

Different solutions will be evaluated for bottom levelling. The programme for the next few years includes in-depth studies of how to level the bottoms of the deposition holes in a rational manner by material removal or casting of a level bed and a suitable diffusion barrier. Practical tests of the chosen method are foreseen, but the timetable for these tests is dependent on when new deposition holes are bored in Onkalo or the Äspö HRL.

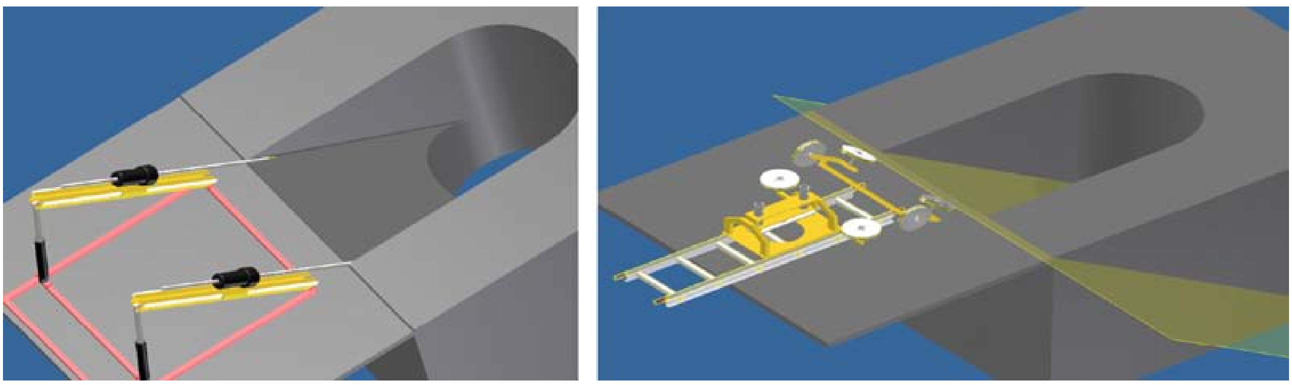


Figure 15-6. *Bevelling of deposition hole by wire sawing. The left-hand figure shows drilling of the holes for the diamond wire, done with standard equipment mounted on a fixture. A diamond-studded wire can then be pulled through the holes to saw the bevel. The right-hand figure illustrates this. The equipment will be designed so that it can saw both side walls and the bevel without changing setup.*

15.6 Tools for data management and visualization

In order for the final repository project to be carried out within the given framework, data management must be efficient and traceable. Modelling, design and construction will require short lead times from ordering of investigations to delivery of quality-assured data. This is the starting point for further development of databases and tools for modelling and visualization. The work will begin with an evaluation of existing systems and tools, and a plan of action to guide the development work will be drawn up. The systems that are needed must be developed and tested before the start of construction.

15.6.1 Databases

The site investigations have entailed an extensive use of databases for gathering and storing of data which has subsequently been retrieved and utilized for modelling, design, etc. Based on this experience, SKB intends to further develop and adapt the database system to meet the needs that will exist during construction and operation. The system must then be able to link and present geometric information on boreholes and tunnels, along with results from investigations in them. This is necessary in order to be able to check that the design premises are fulfilled. The system must also be able to be used for presentation of data as a basis for both day-to-day decisions and long-term planning.

15.6.2 Visualization system

The deterministic geological modelling during the site investigations was done with the aid of the SKB-developed system RVS (Rock Visualization System), which is based on the commercial CAD program MicroStation. The need for fast, daily geological modelling for construction makes demands on robustness and user-friendliness which the system does not meet today. A requirement specification that meets the needs of the new work processes during construction and operation is being prepared as a basis for further development. Commercially available systems, mainly from the mining industry, and their potential for adaptation to SKB's applications will be evaluated.

16 KBS-3H – horizontal deposition

The KBS-3 method permits canisters to be emplaced either vertically (KBS-3V) or horizontally (KBS-3H). Vertical deposition is the reference design, but SKB is also exploring the possibility of switching to horizontal deposition at a later stage.

In both KBS-3V and KBS-3H, the canister is surrounded by a buffer of bentonite, see Figure 16-1. In KBS-3H, no deposition tunnels are needed, since the long horizontal deposition drifts are bored directly from the main tunnel, see Figure 16-2. A row of so-called supercontainers is deposited in the deposition drifts. A supercontainer consists of a canister surrounded by bentonite buffer, held together by a perforated outer metal shell. A distance block of bentonite clay is placed between the supercontainers, partly to seal the drift so that water flow along the drift is prevented, and partly so that the temperature in the buffer will not get too high. A drift end plug will be installed at the mouth of the deposition drift. The plug will hold the supercontainers and distance blocks in place until the main tunnel has been backfilled. The deposition drifts will be spaced at a distance of between 25 and 40 metres, depending on the properties of the rock, mainly its thermal conductivity. The rock volume that needs to be excavated for a KBS-3H repository is considerably less than for vertical deposition. The facilities on the surface are only marginally affected by which of the designs is chosen.

In the early 1990s, SKB explored the possibilities of emplacing the canisters horizontally /16-1, 16-2/, and in 2001 a research, development and demonstration programme for KBS-3H was presented /16-3/. The purpose was to determine whether horizontal deposition can offer an alternative to vertical deposition. During 2002–2007, SKB carried out studies in accordance with this programme in cooperation with Posiva. This work was reported at the end of 2007 /16-4/. Based on the results achieved, SKB and Posiva decided to continue with the next stage of the research programme. This work was initiated in 2008.

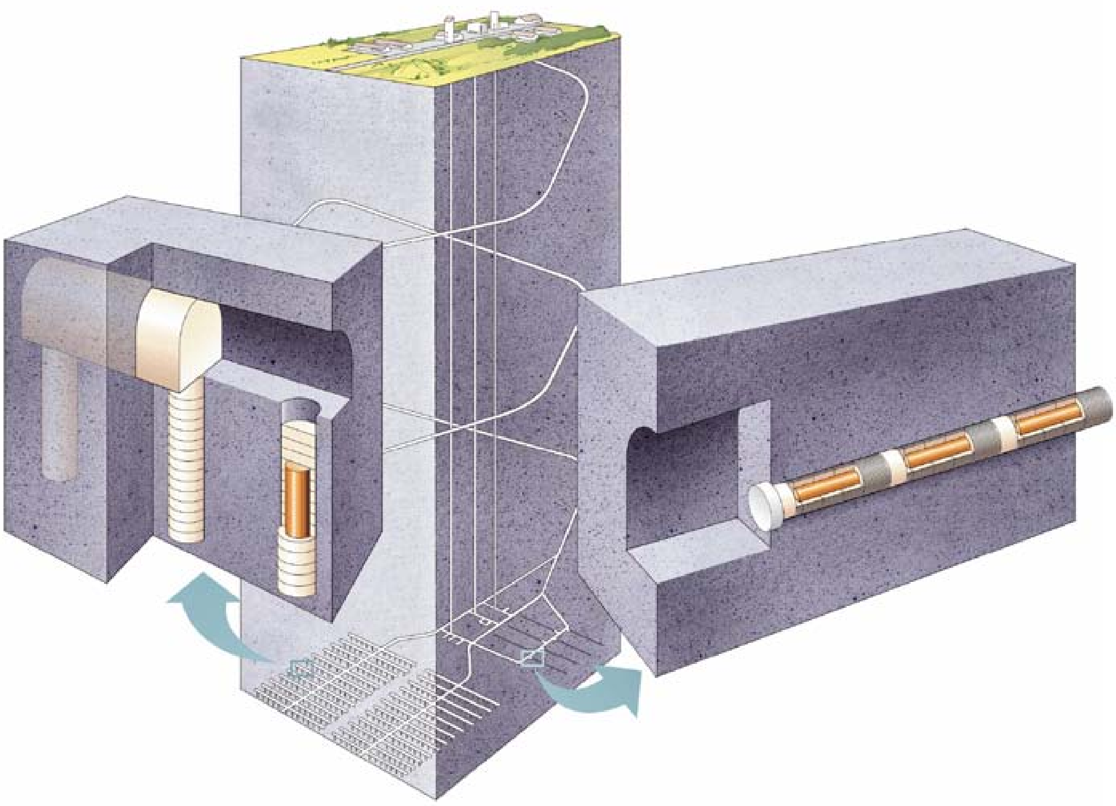


Figure 16-1. *Schematic illustration of the KBS-3 repository with vertical (KBS-3V) versus horizontal (KBS-3H) deposition.*

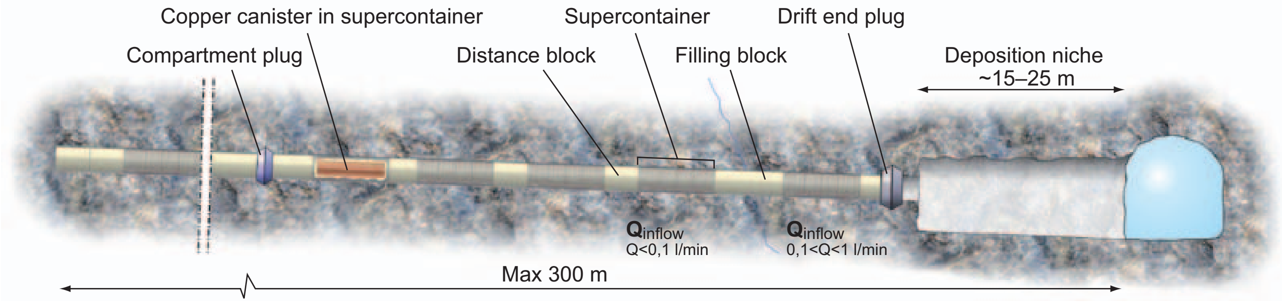


Figure 16-2. *Schematic illustration of a KBS-3H repository.*

16.1 Current situation and programme

In its review statement on RD&D Programme 2007, SKI raised a number of issues which, in their opinion, need to be addressed regarding horizontal deposition. Among other things, SKI called for information on the function of the distance blocks in the event of uneven wetting of the bentonite. Wetting is of importance for the buildup of swelling pressure in the contact between deposition hole and rock. In particular, wetting may be uneven in positions with rock breakout and/or water inflow in the deposition hole. SKB now makes the assessment that with a Dawe-type design, the risk of uneven wetting can be eliminated, see Section 16.1.1.

SKI further said that SKB should study whether there are differences between KBS-3H and KBS-3V with regard to sensitivity for mechanical impact on the rock, particularly at high rock stresses. If the loads lead to spalling or rock breakout in a deposition hole, this may lead to practical difficulties in getting the container in place. According to SKI, the same may apply if, due to the damp atmosphere in the deposition hole, the bentonite begins to swell before the container has reached its final position.

Since RD&D Programme 2007 was presented, equipment for deposition of supercontainers and distance blocks has been developed and tested in one of the two full-diameter horizontal deposition drifts at the Äspö HRL. Boring of the drifts, 95 and 15 metres in length, has in itself comprised an important test. After some modifications, the technique for transporting supercontainers in the drifts has been shown to work, and the equipment has been driven nearly 50 km, forward and back, in the 95-metre drift. Now work is under way to improve and optimize the technique. The goal is to obtain a functional methodology and equipment with high reliability. The two horizontal deposition drifts are also being used for testing different components. For example, post-grouting is being studied with the aid of a Mega Packer, as well as plugging with a compartment plug.

An assessment has been made of the long-term safety of a hypothetical KBS-3H repository in Olkiluoto and reported by Posiva /16-5/. The assessment is based on a preliminary design and layout of the repository.

The main goal of ongoing activities is to further develop the design of KBS-3H so far that it is possible in the next stage to demonstrate the method on a full scale. To achieve this, evidence is required showing that:

- the design is feasible with respect to construction, fabrication and installation,
- buffer in supercontainers and distance blocks, as well as supercontainers and plugs, function as intended after deposition, so the initial state is achieved,
- the materials that are used for supercontainers and plugs do not impair the buffer's barrier function, i.e. that the buffer meets the requirements with regard to long-term safety.

Provided that SKB and Posiva decide to proceed with KBS-3H, the following activities are planned during the relevant RD&D period:

- prepare production line reports, specific for KBS-3H, and update other design-specific documentation,
- verify components on full scale by demonstration,
- perform an assessment of the long-term safety of a hypothetical KBS-3H repository in Forsmark.

16.1.1 Design of a KBS-3H repository

The design of KBS-3H has been further developed since RD&D Programme 2007. The principle is still the same (Figure 16-2), but a technique has been developed for drainage, artificial watering and air evacuation. The technique is called Dawe (Drainage, Artificial Watering and air Evacuation) and now comprises a reference design for KBS-3H. Dawe is based on artificial watering to achieve faster and more even contact between buffer and rock wall. Before the supercontainers and the distance blocks are deposited, a pipe is installed for air evacuation along the tunnel wall. During installation, seepage water runs out along the tunnel floor, which is permitted by the fact that supercontainers and distance blocks stand on spacers and water can then run beneath them. Due to the fact that the voids in the deposition drift are then artificially filled with water through the compartment plug (so that the buffer will swell rapidly), uneven wetting and large pressure gradients in the system can be avoided. The air evacuation pipe is pulled out of the deposition drift before the buffer builds up too much swelling pressure. In order that the blocks will not be harmed by humidity during installation, the buffer in the Dawe design has a higher water content /16-6/.

In RD&D Programme 2007, SKB noted which development issues are important in order for KBS-3H to achieve a technical level equivalent to that of KBS-3V. These included methods to handle water inflows and layout studies to provide a basis for judging which rock stresses need to be taken into account and how they affect the risk of spalling, see Section 16.1.3.

In order to assure the quality of the buffer, the water inflow along the deposition drifts must be managed. By post-grouting with a Mega Packer, it is possible to seal a deposition drift at points where the drift is intersected by a conductive fracture without drilling any grouting holes outside the drift contour. The Mega Packer consists of a steel cylinder that leaves a 1.5 centimetre wide gap to the walls of the deposition drift, see Figure 16-3. The gap is closed in both ends of the cylinder using two hydraulic packers, creating a 1.5 metre long isolated section. The fracture can then be characterized hydraulically, whereupon the rock in the packered-off section can be injected with silica sol or low-pH cement. The method has been tested in the Äspö HRL with very positive results /16-6/.

In order to carry out the installation of the supercontainer, and to be able to use a larger portion of the deposition drift even where large inflows occur in individual fracture zones, it must be possible to screen off very transmissive fracture zones. This can be done by placing compartment plugs on both sides of the water-bearing fracture zones in the deposition drift. SKB has developed technology for this. A slot is sawn out and a fixing ring over which the deposition equipment can pass is grouted in. Later, when the deposition work outside the plug's position has been completed, a steel plug is fitted in this ring. The plug has a number of holes, permitting pellet filling, artificial watering and air evaluation of the section inside it. Testing of a compartment plug has been under way since 2009 at the Äspö HRL, see Section 16.1.2.

Compartment plugs are used to facilitate wetting of the buffer in the deposition drifts in the Dawe design. The 300 metre long deposition drifts are divided into approximately 150 metre long compartments. Supercontainers and distance blocks are placed in compartments where the water inflow is very low (< 0.1 litre per minute), see Figure 16-2. The main function of the distance blocks is to separate the supercontainers hydraulically and thermally. Filling blocks of bentonite are placed in compartments where the inflow is greater than 0.1 litre per minute but less than one litre per minute. They must primarily be able to resist erosion and support the function of the distance blocks. A drift end plug is placed in the mouth of the deposition drift. It is supposed to seal tightly until the deposition niche and the main tunnels have been backfilled.

Alternative materials for the metal shell around the supercontainer have been studied. The choice of reference material – steel, copper or titanium – will be made during the current project phase. The most important factor for long-term safety is the effect of the material on the bentonite, which has been investigated in comprehensive studies, see Section 16.1.3. Corrosion products and wall thickness influence the choice of material. This in turn affects the size of the bentonite blocks, and ultimately the density achieved at full water saturation. The studies have therefore included how different materials would affect both the properties of the buffer and the size of the deposition drifts. Copper has poorer strength properties than steel and titanium. If copper is chosen, the bottom part of the supercontainer must therefore be made thicker, and the width of the spacers that create space for the deposition machine's sled must be increased.

A preliminary layout of a hypothetical KBS-3H repository in Forsmark has been devised. The main tunnels have roughly the same locations as in the layout for the KBS-3V repository. The deposition tunnels have been replaced by the KBS-3H repository's deposition drifts. Two alternative layouts have been studied: one with 30 metres and one with 40 metres distance between the deposition drifts. The deposition drifts are divided into two compartments by a compartment plug. A KBS-3H repository in Forsmark is estimated to hold about 7,000 canisters if the deposition drifts are spaced 30 metres apart and just under 6,000 canisters if the spacing is 40 metres. Despite the fact that the repository area per canister is greater for a KBS-3H repository than for a KBS-3V repository, the total rock volume that needs to be excavated is considerably less.

Programme

The Dawe design with its various components will be evaluated and verified in greater detail. Among other things, a comprehensive testing programme will be carried out in the Äspö HRL, see Section 16.1.2.

Ongoing work on reports for production lines and facility description will continue, as will ongoing work with geometric criteria for the deposition drift and boring technology. The latter work is aimed at studying the need to update the requirements on the deposition drift and finding equipment to measure and verify that the drift satisfies these requirements.

Additional issues that will be studied are:

- installation of the air evacuation pipe along the wall before supercontainers and buffer components are lifted in, and removal of this pipe through the compartment plug,
- effects of materials in supercontainers and plugs.

16.1.2 Demonstration in the Äspö HRL

The deposition equipment was delivered and installed in the Äspö HRL in 2006. A series of tests and trials have been performed with the equipment. The deposition machine has been driven nearly 50 kilometres in the 95-metre drift, with supercontainer and distance blocks made of concrete, but also without load. The tests, which were conducted on a full scale, showed that the equipment works properly /16-7/. Continued technical development work has been started to ensure long-term functionality.

The Mega Packer was tested in the 95 metre long deposition drift at a depth of 220 metres, see Figure 16-3. A total of five positions were grouted, with an in-leakage rate of between 0.15 and 2.40 litres per minute. Silica sol was used as grout, and the total outflow from the deposition drift was reduced from about four litres per minute to about 0.4 litre per minute /16-8/.

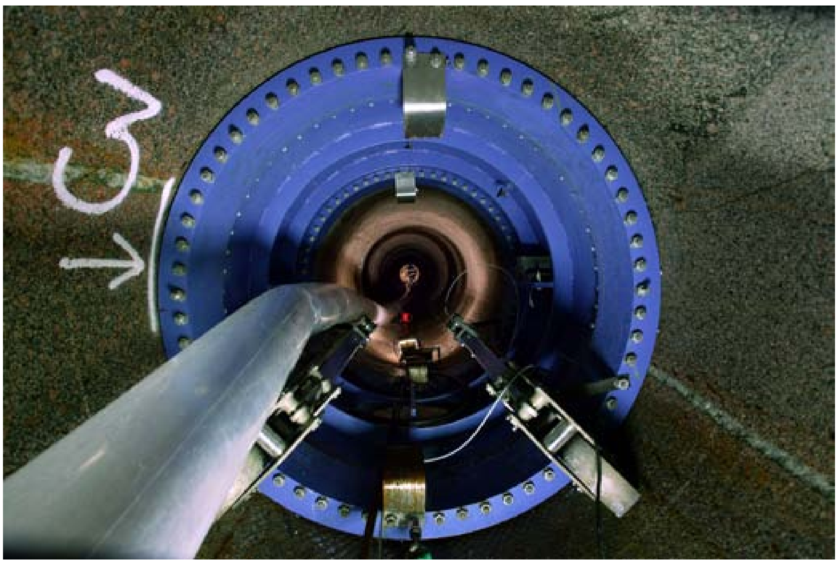


Figure 16-3. Mega Packer for post-grouting in a horizontal deposition drift placed in the 95-metre drift in the Äspö HRL.

A compartment plug was installed in a slot sawn out in the 15 metre long deposition drift. The plug was pressurized incrementally to 50 bar. Initially the leakage around the plug exceeded the set limits, but after grouting with silica sol in preinstalled grouting hoses in the casting, a tightness well on a par with stipulated values was achieved. The test shows that the concept with compartment plugs is feasible and that sufficient watertightness can be achieved. The drift end plugs have a design similar to that of the compartment plugs, and the positive results with the compartment plugs indicate that the drift end plugs should also perform satisfactorily. Additional tests of the compartment plug are planned within the current project phase. Results and experience will be reported when the current project phase is concluded.

Laboratory tests of removal of water pipes in the Dawe design show that the pipes can be removed as planned. But the tests have also indicated that removal must take place at a relatively early stage of the swelling process, since the force required to pull out the pipes increases rapidly as the buffer begins to swell. Erosion tests indicate that short pipes can be used when the compartment is filled, which should reduce the risks during removal when only one pipe remains, the air evacuation pipe.

Programme

At present, work is under way to develop a programme for further testing of KBS-3H at the Äspö HRL. Preliminarily, the programme will consist of subsystem tests where several components are tested together.

In order for deposition to be possible in the nearly 300 metre long deposition drifts, the drifts must meet strict geometric requirements. This is a key issue for horizontal deposition. To this end, drilling must achieve straight pilot holes which can then be reamed up to full diameter. Some pilot holes will therefore be drilled and inspected for straightness. The drilling tests will be done in the Äspö HRL, at a depth equivalent to repository depth. Parts of a pilot hole will be reamed up to full diameter. In parallel with the drilling tests, the geological and hydrogeological conditions will be studied and comparisons will be made between evaluated data from pilot holes and data from the full-scale deposition drift. The deposition drift will also be used for trials with the Mega Packer at relevant groundwater pressure, and possibly also tests of spalling processes in the rock wall. A deposition drift at repository depth will also enable system tests to be carried out under realistic conditions in the future.

In addition to the tests at repository depth, integrated tests are also planned to demonstrate and test several of the KBS-3H repository's components together. These tests will preliminarily be conducted in the existing 95 metre long deposition drift at a depth of 220 metres. These subsystem tests are aimed at:

- demonstrating full-scale fabrication, handling and transport of a supercontainer with bentonite buffer,
- demonstrate the technique of depositing a supercontainer and buffer components of bentonite,
- test the Dawe design with full-scale artificial watering of the buffer in the supercontainer and distance blocks,
- demonstrate the function of the compartment plugs in connection with artificial watering.

16.1.3 Long-term safety

A KBS-3H repository bears great resemblances to a KBS-3V repository. The similarities include the properties of spent fuel, the copper canister and the bentonite buffer as well as conditions in the geosphere and the biosphere. A safety evaluation for a KBS-3H repository can therefore be based on knowledge obtained by the KBS-3V programmes in Sweden and Finland, and then focus on issues that are specific for KBS-3H. The most important issues are:

- The deposition drifts in KBS-3H are long and there is a risk that piping and erosion will occur in the buffer and distance blocks before the bentonite has become water-saturated.
- How processes that can affect several canisters in the tunnel can be avoided, such as spalling of the rock and formation of transport pathways along the deposition tunnel.
- In its present reference design, KBS-3H has several components of steel. The iron will corrode, leading to hydrogen gas formation. This can in turn sometimes lead to increased microbial activity and the formation of pressure gradients in the system. The iron can also affect the physical and chemical properties of the bentonite.

A preliminary assessment of long-term safety has been carried out for a KBS-3H repository with data from Olkiluoto, the site selected for a final repository in Finland. The work has been described in a number of interim reports plus a summary report /16-5/. The safety assessment was based on the original design of KBS-3H, called the Basic Design, which, in a later stage of the project, was not considered sufficiently robust and feasible. Instead, the Dawe design has been chosen as the reference design. The conclusion of the safety assessment is that KBS-3H is a promising design for a repository on this site, in terms of long-term safety and for the conditions prevailing at Olkiluoto.

The safety assessment has been reviewed by the Finnish Radiation and Nuclear Safety Authority (STUK), which offered comments at the end of 2009. In its comments, STUK takes up effects of iron components, the functionality of the buffer and spalling. Via its external expert group BRITE (the Barrier Review, Integration, Tracking and Evaluation group), SSM has also conducted a thorough evaluation of the work with KBS-3H. In its report, the expert group raises a number of issues that require further research in order for a complete evaluation of feasibility and long-term safety to be carried out for KBS-3H /16-9/.

Most of the issues raised by STUK and SSM have been studied within the KBS-3H project during the ongoing project phase. The issue of the impact of iron on bentonite has been further studied since 2007, and preliminary results show that iron reacts with the bentonite clay in several ways in conjunction with the corrosion processes. The preliminary geochemical modelling with data from Olkiluoto showed that the impact of iron on bentonite was limited to a couple of centimetres for a period of up to several hundred thousand years /16-10, 16-11/. In response to the problems that can arise due to the fact that iron affects the buffer, the KBS-3H project has studied the alternative materials copper and titanium, see also Section 24.2.18.

The evolution of the buffer has been studied under conditions that are specific for KBS-3H. For example, erosion during water saturation (Dawe) has been tested on a laboratory scale. Theoretical studies have been made of spalling of the deposition drifts. The studies indicate that the rock stresses will probably not lead to spalling. Additional thermally induced loads that could initiate spalling will not occur earlier than 1–10 years after deposition.

Programme

The effect and extent of the affected zone in interactions between iron and buffer will be evaluated with the aid of results from long-term tests on a laboratory scale. Data will be compiled and used to model reaction and transport mechanisms in the buffer. Corrosion of iron gives rise to the formation of hydrogen gas, so the effects of hydrogen and corrosion products on the pore water in the buffer will be modelled. The effect of the supercontainer in the boundary zone between buffer and rock will be clarified, and the acceptable size of the groundwater flow disturbances and the impact on mass transport will be calculated. The goal is that, together with other material studies (titanium and copper), the studies should provide a basis for choosing a reference material for all metal components in KBS-3H repository.

Ongoing work to design the KBS-3H repository's components is directly linked to long-term safety; for example, the buffer's ability to develop a counterpressure is important for minimizing the risk of thermally induced spalling. Design premises for the KBS-3H repository's components with respect to long-term safety will be compiled.

Providing that SKB and Posiva decide to continue the work with KBS-3H, a site-specific assessment of the long-term safety of a KBS-3 repository in Forsmark will be conducted, after SR-Site has been completed. The assessment will be based on the results of SR-Site, with supplementary analyses of the parts that are specific for KBS-3H. The purpose is to be able to compare the safety of the two repository designs and, at a later stage, make a well-founded choice between KBS-3V and KBS-3H.

Part IV

Research for assessment of long-term safety

17 Overview – research for assessment of long-term safety

An overview is provided here of SKB's natural science research programme within the framework of our safety assessments, i.e. the assessments of the long-term safety of the Spent Fuel Repository and the extension of SFR (the final repository for short-lived radioactive waste). The plans for the final repository for long-lived low- and intermediate-level waste, SFL, are described in Part II, the LILW programme. Much of the research that relates to the Spent Fuel Repository and SFR is also relevant to the research that will be conducted for the safety assessment for SFL.

Research concerning safety assessment of a final repository for radioactive waste includes general areas not specifically related to any given repository system and research related more directly to the different repository systems.

The research that is conducted for the purpose of learning more about the long-term safety of the final repository for spent nuclear fuel has mainly been conducted within the framework of the SR-Site project. The basis for the focus of the research on processes in the engineered and natural barriers incorporated in the repository concept can be found there. Table 17-1 provides an overview of the processes and the scope of the efforts that are planned during the coming three-year period.

Research specifically related to the safety assessment for the extension of SFR is dealt with in Chapters 20 and 21. Research on buffer also includes research on clay barriers in SFR and is dealt with in Chapter 24. The processes that are studied for the safety assessment for the SFR Extension Project (PSE) are compiled in Table 17-2.

A general introduction to the different research areas is given below. In addition, a summary is provided of the link between the research programme and the various projects being conducted at the Äspö HRL, along with an overview of research at Nova FoU (Nova R&D) in Oskarshamn.

The research programme also includes keeping track of other methods for disposal of the spent nuclear fuel. SKB is following research on partitioning and transmutation (P&T) as well as disposal in deep boreholes.

17.1 General research areas

Certain research areas are of a general nature and not linked to any specific repository system. These areas are: Safety Assessment (Chapter 18), Climate Evolution (Chapter 19), Geosphere (Chapter 25) and Surface Ecosystems (Chapter 26).

Future climate change may lead to glaciation and permafrost. These two phenomena have great impact on the natural environment around a final repository. The climate can therefore indirectly affect the barriers in a repository and thereby the outcome of a safety assessment.

A large number of processes in the rock (the natural barrier) influence the outcome of a safety assessment. Important processes in the geosphere include fracturing, groundwater flow, water chemistry and earthquakes. Radionuclide transport and retention in the rock are included in the modelling of these processes.

Site data and models of the ecosystem in Forsmark serve as a basis for the research within Surface Ecosystems. The research includes work with numerical models for dose calculations.

17.2 Research linked to the repository system

The research that specifically relates to the final repository for spent nuclear fuel and the SR-Site safety assessment takes place within the research areas Fuel (Chapter 22), Canister (Chapter 23), and Buffer and Backfill (Chapter 24).

Table 17-1. Processes in fuel, canister, buffer and backfill, and geosphere.

Code: Major initiatives | Moderate initiatives | Minor initiatives/monitoring during coming three-year period

Type of process	Fuel	Canister	Buffer and backfill	Geosphere
R (radiation-related)	Radioactive decay 22.2.2			
	Radiation attenuation/heat generation 22.2.3	Radiation attenuation/heat generation 23.2.1	Radiation attenuation/heat generation 24.2.2	
	Induced fission – criticality 22.2.4			
T (thermal)	Heat transport 22.2.1	Heat transport 23.2.1	Heat transport 24.2.3	Heat transport 25.2.2
			Freezing 24.2.4	
H (hydraulic)	Water and gas transport in canister cavities, boiling/condensation 22.2.1		Water transport under unsaturated conditions 24.2.5	Groundwater flow 24.2.3
			Water transport under saturated conditions 24.2.6	Gas flow/dissolution 25.2.4
			Gas transport/dissolution 24.2.7	
			Piping/erosion 24.2.8	
M (mechanical)	Thermal expansion/cladding failure 22.2.1	Deformation of cast iron insert 23.2.2	Mechanical processes (including swelling) 24.2.9	Movements in intact rock 25.2.5
		Deformation of copper canister under external pressure 23.2.3		Reactivation – movement along existing fractures 24.2.7
		Deformation from internal corrosion products 23.2.4		Fracturing 25.2.8
				Time-dependent deformations 25.2.9
		Thermal expansion 23.2.1	Thermal expansion 24.2.10	Thermal movement 25.2.6
		Radiation effects 23.2.5		
C (chemical)	Advection and diffusion 22.2.1	Corrosion of cast iron insert 23.2.6	Advection 24.2.12	Advection/mixing – groundwater chemistry 25.2.10
	Residual gas radiolysis/oxygen formation 22.2.1	Galvanic corrosion 23.2.1	Diffusion 24.2.13	Diffusion – groundwater chemistry 25.2.12
	Water radiolysis 22.2.5	Stress corrosion cracking of cast iron insert 23.2.1	Osmosis 24.2.14	Reactions with the rock – groundwater chemistry 25.2.14
	Metal corrosion 22.2.1		Ion exchange/sorption 24.2.15	
	Fuel dissolution 22.2.6	Corrosion of copper canister 23.2.7	Montmorillonite transformation 24.2.16	Microbial processes 25.2.16
	Dissolution gap inventory 22.2.1	Stress corrosion cracking of copper canister 23.2.8	Iron-bentonite interactions 24.2.17	Decomposition of inorganic engineering material 25.2.17
	Speciation of radionuclides, colloid formation 22.2.7	Earth currents – stray current corrosion 23.2.1	Dissolution/precipitation of impurities 24.2.18	Colloid formation – colloids in groundwater 25.2.18
	Helium production 22.2.8		Cementation 24.2.19	Gas formation/dissolution 25.2.20
			Colloid release/erosion 24.2.20	Methane ice formation 25.2.21
			Radiation-induced montmorillonite transformation 24.2.21	Salt exclusion 25.2.22
			Radiolysis of pore water 24.2.22	
			Microbial processes 24.2.23	
Integration/modelling				DFN 25.3.1
			THM evolution unsaturated 24.2.11	THM evolution 25.3.2
				HC evolution 25.3.3
Radionuclide transport			Advection/mixing 24.2.24	Advection/mixing 25.2.11
			Diffusion 24.2.25	Diffusion 25.2.13
			Sorption 24.2.26	Sorption 25.2.15
			Speciation 24.2.27	
			Colloid transport 24.2.28	Colloid transport 25.2.19
	RN transport, near-field	RN transport, near-field	RN transport, near-field 24.3.15	RN transport, geosphere 25.3.4

Table 17-2. Processes in short-lived low- and intermediate-level waste and barriers for this.

Code:	Major initiatives	Moderate initiatives	Minor initiatives/monitoring during coming three-year period

Type of process	Waste	Engineered barriers and backfill
T (thermal)		Heat transport 21.2.2
	Freezing 20.2.2	Freezing 21.2.3
H (hydraulic)	Water uptake in ion exchange resins and bitumen 20.2.3	
	Water transport 20.2.4	Water transport 21.2.4
	Two-phase flow/gas transport 20.2.5	Two-phase flow/gas transport 21.2.5
M (mechanical)	Expansion/contraction of the waste 20.2.6	Expansion/contraction 21.2.6
	Fracturing 20.2.7	Fracturing 21.2.7
	Breakout 20.2.8	Breakout 21.2.8
C (chemical)	Dissolution/precipitation 20.2.9	Dissolution/precipitation 21.2.9
	Degradation of organic compounds 20.2.10	Chemical cement and concrete degradation 21.2.10
	Corrosion 20.2.12	Corrosion 21.2.11
	Speciation 20.2.11	Sorption 21.2.12
	Diffusion 20.2.13	Diffusion 20.2.13
	Advection and mixing 20.2.14	Advection and mixing 20.2.14
	Colloid formation and transport 20.2.15	Colloid formation and transport 21.2.15
	Microbial activity 20.2.16	Microbial activity 21.2.16
	Radiolytic degradation 20.2.17	
Modelling – radionuclide transport		Development of calculation codes 21.3.3

Research relevant to SFR and the safety assessment for the Extension Project (SR-PSE) is being conducted in the areas of Short-Lived Low- and Intermediate-Level Waste (Chapter 20) and Engineered Barriers in SFR (Chapter 21).

Spent Fuel Repository

The level of knowledge for the SR-Site safety assessment is now deemed to be high enough to bound the importance of identified uncertainties. The research programme is continuing so that we can gather further knowledge and quantify remaining uncertainties. In this way, it should be possible to make more realistic assessments of the Spent Fuel Repository's safety margin in future safety assessments.

The properties of the spent fuel and the processes that occur if the fuel comes into contact with water is an important part of the research for the SR-Site safety assessment. Some of these processes are strongly bound to the initial state (type of fuel, burnup etc.), and information on this can be found in Chapter 10 (Part III) as well as in Chapter 22.

The ability of the canister to isolate the fuel is vital, and research within the framework of the safety assessment is focused on the processes that can be expected to occur after deposition. Important processes are corrosion and mechanical load. Knowledge concerning the initial state of the canister is presented in Chapter 11 (Part III) and in Chapter 23.

All processes in the buffer after deposition – for example water uptake and swelling, or freezing and erosion – are important for the outcome of the SR-Site safety assessment. Many processes in the backfill are virtually identical to those that occur in the buffer, and research on the backfill is therefore presented jointly with the buffer in Chapter 24.

SFR

Short-lived low- and intermediate-level waste is deposited in SFR in Forsmark today. Work is under way at SKB on the safety assessment for the Extension Project. The processes that occur in this type of waste are specific for the waste type, and the research programme focuses on corrosion and degradation of organic compounds in the waste, see Chapter 20.

The engineered barriers that are used in SFR and are planned for the extension are greatly affected by processes that occur in cement and concrete, which is reflected in the research programme, see Chapter 21. Research on the processes that occur in the clay barriers that are used in SFR (the silo buffer) is presented together with research on buffer and backfill in Chapter 24.

17.3 Research at the Äspö HRL and Nova R&D

17.3.1 Research at the Äspö HRL

The Äspö HRL is a cornerstone of SKB's programme for research and technology development. The purpose of many of the projects being pursued at the Äspö HRL is to gain better knowledge of long-term safety. These projects mainly have to do with processes in the canister, the buffer and the bedrock. A number of examples of projects in the Äspö HRL that are completely or partially focused on long-term safety are given in Table 17-3, along with reference to where in this RD&D programme the project is described.

Interest in research at the Äspö HRL is great not only in Sweden, but also internationally. Numerous organizations from different countries are participating in the joint international work being conducted in the Äspö HRL. The members of the Äspö International Joint Committee are Andra (France), BMWi (Germany), CRIEPI & JAEA (Japan), NWMO (Canada) and Posiva (Finland). The foreign organizations are participating both in the experimental work and in the modelling work being done by the Task Forces.

Table 17-3. Overview of Äspö projects with a link to research on long-term safety. For the location of the projects in the Äspö HRL, see Figure 1-10.

Project	Process	Section
Minican	Deformation due to internal corrosion products	23.2.4
LOT	Montmorillonite transformation	24.2.16
	Corrosion of copper canister	23.2.7
Alternative buffer materials	Water transport under saturated conditions	24.2.6
	Iron-bentonite interactions	24.2.17
Lasgit	Gas transport/dissolution	24.2.7
	THM evolution, buffer	24.2.11
TBT	Water transport under saturated conditions	24.2.6
	THM evolution, buffer	24.2.11
Canister Retrieval Test (CRT)	Water transport under unsaturated conditions	24.2.5
Apse (Pillar Stability Experiment)	Movements in intact rock	24.2.5
Prototype Repository	Heat transport (buffer)	24.2.3
	THM evolution, buffer	24.2.11
	Water transport under unsaturated conditions	24.2.5
	Water transport under saturated conditions	24.2.6
	Piping/erosion	24.2.8
	Heat transport (geosphere)	25.2.2
	Thermal movement	25.2.6
	Fracturing	25.2.8
	Time-dependent deformation	25.2.9
	Integrated modelling THM (geosphere)	25.3.2
True	Advection/mixture – radionuclide transport	25.2.11
	Integrated modelling – radionuclide transport	25.3.4
LTDE	Diffusion – radionuclide transport	25.2.13
	Reactions with the rock – sorption	25.2.15
Colloid Transport Project	Colloid formation	25.2.18
Microbial Projects	Microbial processes	25.2.16
Matrix Fluid Chemistry Experiment	Diffusion – groundwater chemistry	25.2.12
PADAMOT	Diffusion – groundwater chemistry	25.2.12
SWIW tests	Integrated modelling – radionuclide transport	25.3.4

17.3.2 Broadening of research at Nova R&D

Nova FoU (Nova R&D) (www.novafou.se) is a platform for research and development under the auspices of Nova Högskolecentrum in cooperation between SKB and Oskarshamn Municipality. As such, Nova FoU is a centre of excellence where several universities, including Linnaeus University, and other educational institutions and companies both from Sweden and other countries carry out research and development projects. To help Nova FoU achieve its ambition of being a world-class centre of excellence, information on the new research opportunities offered via Nova FoU, linked to SKB's laboratories, will be intensified nationally and internationally.

By agreement, Nova FoU has access to SKB's laboratories (the Äspö HRL, the Bentonite Laboratory and the Canister Laboratory), data and expertise. The research and development projects are conducted in these facilities with different universities and companies as owners. The projects take advantage of the unique opportunities offered by SKB's laboratories, databases and competence infrastructure. The research can be of a varying nature within multiple scientific fields, but the focus is on geosphere research and development with links to hydrogeochemistry, hydrogeology, geology, and soil and rock engineering. The principal categories are basic research, environmental research and instrument development. Examples of current projects at Nova FoU are shown in Table 17-4.

Results from the geochemistry, microbial and hydromodelling projects will contribute to a better understanding of the environment in which the Spent Fuel Repository will be located.

Geochemical processes are being studied by Linnaeus University's geochemical research group, which is based on the Nova FoU platform. Studies at Äspö, Laxemar and Forsmark have mainly focused on surface water and groundwater /17-1, 17-2, 17-3, 17-4, 17-5/, but have now branched out to include geology (fracture-filling minerals) and biogeochemistry (biological and organic materials) as well. At present the group consists of a professor, a post-doc, a research assistant and three doctoral candidates.

Table 17-4. Examples of projects within Nova FoU linked to the Äspö HRL.

Project	Organization	Description
Postgraduate studies	Linnaeus University	Linnaeus University's geochemical research group at the Äspö HRL.
Microbes	University of Gothenburg	Study of geogas-driven biosphere in the Äspö HRL.
Microbes II	University of Göttingen	Biomineralization, biogeochemistry and biodiversity involving studies of microorganisms at Äspö.
Near-coastal modelling	KTH	Hydrological transport pathways and coastal dynamics with integration of transport and transformation processes in water from land to sea.
Alfagate	NeoSys AB	3D location of individuals. Development and adaptation of RFID technology, creation of a software-independent solution and integration with other operating systems on Äspö.
SAFESITE	NeoSys AB	Integrated fire protection. Fire alarm and security for the final repository, development and adaptation of RFID technology.
EoS	Oskarshamn municipality	Low-grade waste heat. Research and technology development to recover and utilize waste heat from industry.

17.4 Other methods

SKB is following the development of other methods besides KBS-3 for management and disposal of spent nuclear fuel. These other methods are Partitioning and Transmutation (P&T, Section 27.1) and disposal in deep boreholes (Section 27.2).

18 Safety assessment

18.1 Introduction

The safety assessment is the instrument that is used to determine whether a repository for radioactive waste satisfies the regulatory requirements on long-term safety that are made on such a facility. In Sweden, the primary requirement is formulated as a risk limit, and a central part of the safety assessment consists of quantitatively estimating the radiological risk associated with possible releases from the repository in question. The safety assessment also plays an important role in providing feedback to the RD&D programme by indicating i) areas where greater knowledge could lead to more realistic and thereby often more favourable outcomes of the assessments, and ii) possible improvements of the design of a repository.

At present, SKB is conducting safety assessments as support for the licence applications to build a final repository for spent nuclear fuel at Forsmark and for the licence applications for an extension of the existing final repository for short-lived low- and intermediate-level waste at Forsmark (SFR).

All essential development of methodology for safety assessments is taking place within the framework of the two projects where the aforementioned assessments are being conducted. Complete accounts of methods and applications are given in the reports from the projects, and only a summarizing overview is therefore provided in this RD&D report. Matters relating to the safety assessment for the Spent Fuel Repository and for SFR are discussed in Sections 18.2 and 18.3, respectively.

The work concerned with the long-term safety of the planned repository for long-lived low- and intermediate-level waste (SFL) is treated in Part II (the LILW programme).

18.2 Methodology for assessment of the long-term safety of the Spent Fuel Repository

18.2.1 Methodology in SR-Can and its review

A ten-step methodology was used in the SR-Can safety assessment /18-1/. The methodology was reviewed by SKI and SSI within the framework of the consultations with SKB during the site investigation phase /18-2/. In summary, the then-regulatory authorities found the following, based on their own work and reports from external consultants:

- “SKB’s methodology for safety assessment largely complies with the authorities’ regulatory requirements, but parts of the methodology need to be further developed prior to a licence application.
- “SKB’s quality assurance of the safety assessment SR-Can is insufficient.
- 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 verify that the assumed initial state of the repository is realistic and achievable.
- The account of the risk of early releases should be strengthened.”

These main conclusions agreed well with SKB’s own view of important points in the work remaining to be done up to the applications; most are expressed already in the SR-Can report.

SKI’s viewpoints on safety assessment methodology in the review of RD&D Programme 2007 were largely based on the results of the regulatory authorities’ joint review of SR-Can. SKI summarized its view of the safety assessment as follows:

- “SKI notes that SKB has developed a methodology relating to safety assessment with an appropriate design in relation to SKI’s and SSI’s regulatory requirements. This conclusion is based on SKI’s and SSI’s joint review of the safety assessment in SR-Can.

- Like SSI, SKI considers that it is crucial for the future development of the programme that SKB provides appropriate feedback to the need of research and development from the critical issues identified in the review of SR-Can.
- It is very important that SKB prior to SR-Site raises the level of ambition for the quality work in conjunction with safety assessment. SKI finds that the consultations between the authorities and SKB could be used for a continued dialogue to avoid unnecessary lack of clarity on methodological issues, interpretations of regulatory requirements and reporting forms."

Programme

All method development for the Spent Fuel Repository is taking place within the framework of the SR-Site safety assessment project, which will be reported as a part of the documentation in support of the licence applications to build the final repository for spent nuclear fuel. A detailed account of method development is therefore not given here. The need for method development after submission of the applications will also be presented in SR-Site.

The methodology used in SR-Site is based in large part on the methodology used in SR-Can, and it is important to note that the regulatory authorities found in their review of SR-Can that the methodology has an appropriate design in relation to relevant regulatory requirements.

In order to ensure that the viewpoints expressed in the review of SR-Can are heeded, the review report has been systematically examined and more than 200 measures for development and improvement have been identified. These will be documented internally by SKB within the framework of the SR-Site project. As far as the viewpoint of the regulatory authorities in the review of RD&D Programme 2007, that SKB should provide appropriate feedback to the need of research and development from the critical issues identified in the review of SR-Can, reference is made with regard to process understanding to relevant other parts of this RD&D programme (fuel, canister, buffer, rock, etc.).

A number of themes of relevance to SR-Site have been taken up in the consultations with SSM, in accordance with SKI's review viewpoint. Here accounts have been given of the quality work within SR-Site, of the methodology for SR-Site and of new findings surrounding buffer erosion.

18.3 Assessment of long-term safety of SFR

Since RD&D Programme 2007, SKB has published two relevant documents related to long-term safety that have been reviewed by SSM:

- Safety Analysis Report for SFR, SAR-08, delivered April 2008 (in Swedish only) /18-3/,
- Supplement to RD&D Programme 2007, delivered in April 2009 (in Swedish only) /18-4/ .

The parts of the review of the two documents that concern the assessment of the long-term safety of SFR are presented in the following sections, while the parts of the review that concern research are mainly presented in Chapters 20 and 21.

Supplement to RD&D 2007 and results of its review

In the supplement to RD&D Programme 2007, SKB presented a plan for continued methodology work within the SFR Extension Project, PSE. Since the project extends up to 2013, the plan included the time interval encompassed by this RD&D programme.

The plan was dealt with very cursorily in regulatory review /18-5/.

SAR-08 and results of its review

In April 2008, SKB submitted the SAR-08 safety assessment to SSM. The Authority reviewed the safety assessment during 2008 and 2009 and issued a decision in the matter in December 2009 /18-6/. SSM is of the opinion that:

"The Authority judges that the submitted report analyzes and evaluates long-term safety in a better way than previous reports. The calculation cases and scenarios that are derived and analyzed are structured in a manner that satisfies the Authority's requirements and complies in essence with the directions issued by the Authority. This facilitates assessment against the prescribed risk criterion that applies to the long-term consequences of the final disposal of nuclear waste."

However, SSM considers the plan presented by SKB for closure of the facility to be inadequate and ordered SKB to:

"Present a concrete and coherent plan for the measures which, in accordance with applicable radiation safety conditions, need to be taken upon closure of the facility by not later than 30 April 2010."

SKB has responded to the Authority's order by the designated deadline.

SSM also ordered SKB to supplement the safety analysis report by 31 December 2013 with:

- "an account of the anticipated barrier degradation in the repository part BMA based on all reasonable and probable degradation processes,
- a sensitivity analysis of the risk and importance of changed redox conditions at repository depth,
- a detailed justification of the assumed parameter distributions associated with the groundwater flow in the repository and its vicinity, plus a well-founded discussion of which model variants have been rejected."

Methodology in SAR-08

SAR-08 employed a methodology that largely corresponds to the ten-step methodology used in SR-Can. Among other things, SKB introduced in SAR-08 safety functions that are used to identify the scenarios that are analyzed in the safety assessment. Concerning methodology, SSM writes in the review report /18-6/ that:

"The new safety assessment methodology for SAR-08, which greatly resembles that developed by SKB in SR-Can, is based on well-defined steps such as: FEP processing, initial state, safety functions, reference evolution, selection of scenarios, selection of calculation cases, dose and radionuclide transport calculations, weighing together of risk, and safety evaluation. SSM finds this methodology to be expedient and concrete and considers that it offers good potential for conveying a clear picture of features, events and processes that affect the repository's long-term radiation safety. The use of graphical aids in SAR-08 such as interaction matrices and information flow diagrams contributes to the clarity of the presentation."

In its review, SSM pointed to a number of areas where the safety assessment should be improved. The Authority notes, for example, that the effect of degradation of the plugs that are installed to reduce the water flow in the repository parts, as well as the effect of other redox conditions than those analyzed, should be further studied or that the argumentation should be better substantiated. It may therefore be advisable to take a look at the safety functions that were used in SAR-08 and possibly supplement them.

The Authority is also of the opinion that:

"One area that can be improved is, however, the account of how identified features, events and processes have been taken into account in the development of the conceptual calculation models."

We plan to improve the presentation by using more structured background reports, such as the process reports that were used in the most recent safety assessments for the Spent Fuel Repository. An initial indication of how this may look is shown in Chapters 20 and 21, where research and study needs for an SFR repository are presented in a similar process-based way as for the Spent Fuel Repository.

Regarding calculations and uncertainties in data, SSM says:

"On the whole, the calculation cases provide a good picture of the events and processes that can affect the long-term radiation safety of the repository. In some cases, however, the calculations are not optimally structured to shed light on the safety-related importance of remaining uncertainties.

An example is the calculation cases that are supposed to illustrate the importance of degraded barriers (early degradation of barriers and extreme permafrost)."

Further work is thereby required to substantiate the argumentation that is offered both in the description and in the identification of scenarios. In the most recent safety assessments conducted for the Spent Fuel Repository, for example SR-Can, uncertainties in parameters have been identified and analyzed in a special data report.

Programme

Methodology development for assessment of long-term safety for SFR is taking place within the framework of the safety assessment project. The purpose of the project is to conduct an assessment of the long-term safety of both the existing facility and an extended SFR.

The supplement to RD&D Programme 2007 contains a description of the methodology that is intended to be used in the Extension Project's safety assessment. A report structure is also presented there. The plan is relatively unchanged compared with the one presented in the supplement, but for the sake of completeness the report structure is summarized below:

FEP analysis

An FEP (Features, Events and Processes) report summarizes all features, events and processes of importance for the repository system. As far as possible, the FEP report is based on work already done in SR-Can and SR-Site. Based on identified FEPs, process reports are produced for different system parts. The structure of the process reports is presented below.

Process reports

A necessary part of a safety assessment is presenting our understanding of the processes that can take place in the repository system. This process understanding is based on, among other things, the research that is being conducted within and outside SKB and on the FEPs that have been identified (see above). Process reports were compiled in SR-Can (and are being compiled in SR-Site) for different parts of the system in accordance with a predefined template. A similar template is being prepared for the safety assessment for an extended SFR. Process reports will be written for the applications concerning an extended SFR and will deal with:

- geosphere,
- waste,
- barrier system,
- climate,
- biosphere and surface systems.

Data report

The data report will present and qualify data used in the safety assessment. The data report for the safety assessment of an extended SFR will conform to a similar structure as the data report for SR-Can /18-7/ and SR-Site.

Radionuclide transport calculations

All models used in SAR-08 have been implemented in a new code for radionuclide transport.

The work has continued with a deeper sensitivity analysis than the one performed within SAR-08. Extensive development is being done on biosphere modelling within the framework of the SR-Site safety assessment. This includes collection of site-specific data and development of models and modelling tools. The models used for the safety assessment of an extended SFR will, where applicable, be based on the results obtained within SR-Site.

Repository design

The work of designing the repository is being pursued in cooperation between design, safety assessment and technology development. The long-term properties of the repository components are taken into consideration in connection with geotechnical design and choice of technology. The work includes developing engineered barriers (bentonite and concrete structures) as well as tunnel backfill and plugging. Tunnel backfill is the material used to backfill tunnels and other disposal chambers.

Initial state

The work of preparing descriptions of the initial state will be done in cooperation with other sub-projects within the SFR Extension Project (PSE).

Reference evolution

The work of preparing descriptions of the reference evolution of SFR will be carried out in cooperation with the site investigation project. This work is based to different degrees on similar work that has been done within SR-Site. It can be assumed that biosphere evolution will be nearly identical. However, the evolution of the barrier systems may differ in the safety assessment of an extended SFR, compared with the KBS-3 repository dealt with in SR-Site.

Safety functions

The Authority's view of the safety functions used in SAR-08 is presented in the preceding section, and SKB interprets the Authority's reply as indicating that SSM is positive to the use of safety functions. SKB intends to develop the concept further.

Scenario analysis

Safety functions, in combination with the stipulated requirements on future human actions (FHA), were used in order to devise the scenarios that were subsequently analyzed in SAR-08. How scenario analysis is to be conducted within the safety assessment for an extended SFR will be decided during the course of the work. As is mentioned in the preceding section, the Authority considers that SKB has handled scenario selection in an acceptable manner, but calls for additional scenarios in the safety assessment.

19 Climate evolution

In the time perspective in which the safety of the final repositories for spent nuclear fuel and for short-lived radioactive waste (SFR) are being studied, i.e. hundreds of thousands of years and longer, the climate in Scandinavia has varied enormously. The climate has alternated between warm interglacial conditions similar to what we have today to periods with full ice age conditions. Even though the climate *as such* does not have a great direct impact on repository performance, other processes related to future climate variations will probably have a great impact. An example is the growth of ice sheets and permafrost and changes in sea level. These processes in turn affect, for example, groundwater flow, groundwater chemistry and stresses in the Earth's crust, which are also of importance for repository performance. The climate at the ground surface also affects the evolution of the biosphere. The biosphere in turn has a great influence on human activities, which should be able to proceed in the landscape without man being affected by nearness to the repositories.

Knowledge of climate and climate variations is therefore of great importance in SKB's safety assessments for the final repositories for radioactive waste. The following descriptions of the climate evolutions in SKB's safety assessments mainly apply to the final repository for spent nuclear fuel. However, large parts of these descriptions are also relevant for other types of repositories. Where issues specifically concern SFR, this is stipulated.

19.1 Climate evolutions in SKB's safety assessments

SKB's climate programme has two main purposes today: i) to contribute information for selecting, substantiating and justifying the climate evolutions used in the safety assessments, and ii) to provide detailed information for the descriptions of different climate domains, which can then be used in other parts of SKB's research programme.

As was noted in RD&D Programme 2007, it is not possible today to predict future climates in the long time perspectives that are analyzed in SKB's safety assessments. It is nevertheless possible to estimate the magnitude of possible future climate changes, based for example on knowledge of natural historical climate variations and simulations of future climates. In other words, it is possible to describe relatively well the bounds within which the Scandinavian climate may vary, even in very long time perspectives. Within these bounds we can identify certain characteristic climate domains (temperate, periglacial and glacial climate domains) that are of importance for repository performance. As before, we are therefore focusing our research efforts in the climate field on identifying and understanding conditions and processes within these climate domains. If it is possible to show that the repository meets the safety requirements during the different conceivable climate domains, and during transitions between them, it is not necessary in the assessment of repository safety to take into account the actual future evolution of the climate in time and space to as great a degree. In descriptions of possible future climate changes, it is important also to consider cases where human impact is included.

SKB's approach is to handle the complex questions surrounding future climates by first constructing a reference evolution. This is an example of how different climate domains might conceivably succeed one another during a future glacial cycle, i.e. during the coming 100,000 years or so. The reference evolution consists of a repetition of conditions reconstructed for the last glacial cycle, including the Weichselian glaciation. The reference evolution provides process knowledge on how the different climate-related processes(ice sheet, permafrost, shoreline displacement) function and interact.

Besides being used in the assessment of long-term safety, the reference evolution is also a suitable scientific point of departure for an extended analysis of the impact of the climate on the repository. Based on the reference evolution, our knowledge of possible climate variations and our knowledge of the repositories' performance and safety, a number of other climate evolutions are selected in a structured manner. These cases cover all conceivable situations where climate-related processes that could potentially have a major impact on the performance of the repository are not covered by the reference evolution, for example thicker ice sheets or deeper permafrost. For these cases as well, the way in which the different climate domains succeed one another over time is described, along with how parameters such as ice sheet thickness, permafrost depth and shoreline displacement vary.

Since it is not possible to describe with sufficient certainty an *expected* climate evolution during the next 100,000 years, the climate evolutions that are constructed should not be regarded as attempts to predict future climates. They are rather *relevant examples* of future evolutions that describe climate-related processes in a realistic and integrated way in a 100,000-year perspective.

The climate evolutions that include a clear human impact via increased greenhouse effect include a global rise in sea levels caused by increased melting of the Greenland ice sheet in particular and a thermal expansion of the surface water in the world's oceans. Including such climate evolutions is necessary in view of the fact that Forsmark is located directly adjacent to the present-day coastline.

In the reference evolution, the reconstruction of the ice sheet during the Weichselian is important for how the climate domains succeed one another. The reconstruction of the Weichselian glaciation, whose results were described in RD&D Programme 2007, is used in turn to study shoreline displacement (see Section 19.3), permafrost (see Section 19.4) and the occurrence of glacial earthquakes (see Section 25.2.7).

In order to gain a better understanding of the change and variability we can expect of the climate in the future, SKB has, in addition to earlier studies reported in RD&D Programme 2007, also carried out studies on climate variations during the past three-year period. The purpose of one project was to study extreme climate situations in Sweden in a 100,000-year perspective, while the purpose of other projects has been to study climate variations from geological climate archives, and specifically to compile paleoclimatic information for selected periods during the Weichselian, see Sections 19.2 and 19.5.

With the current approach and safety assessment methodology, the identified climate domains – and their evolution in time and space in the selected climate cases – serve as a basis for SKB's safety assessments. The climate programme is being carried out in close cooperation with our programmes for hydrogeology, geochemistry, biosphere, thermo-hydro-mechanical processes, buffer and canister, see Figure 19-1.

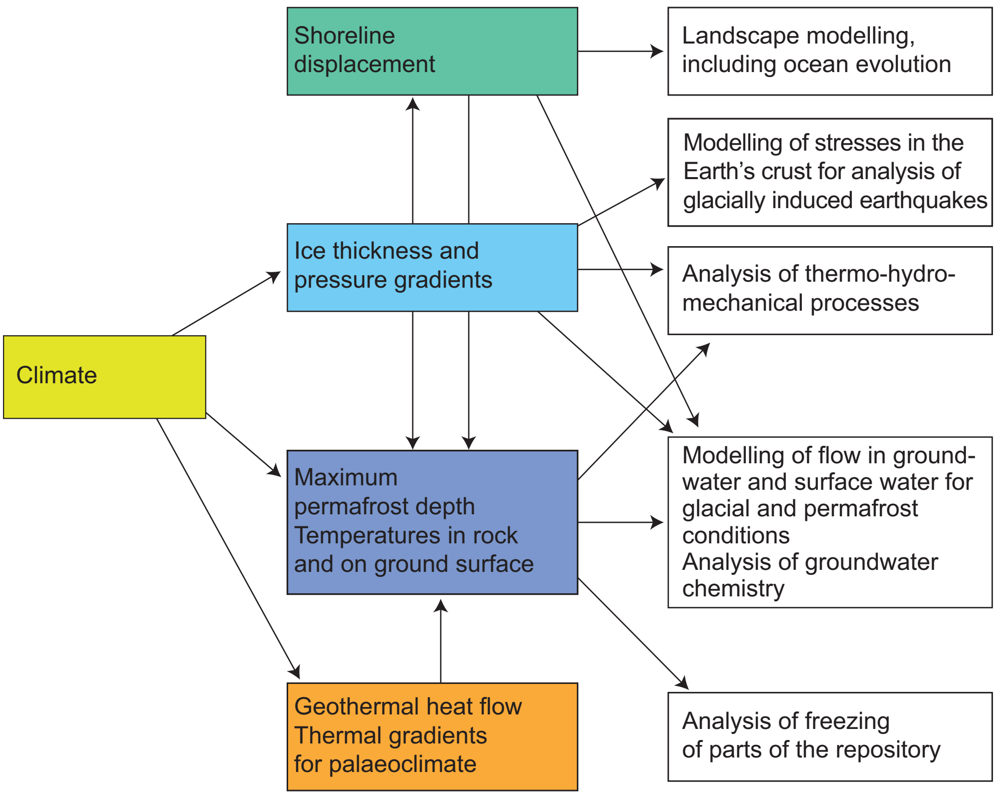


Figure 19-1. *Diagram showing how different parts in the climate programme are linked to other parts in the SR-Site safety assessment. Coloured boxes show phenomena studied in the climate programme. Certain boxes show other activities in SR-Site where data from the climate programme are used. The arrows indicate principal data flows in SR-Site.*

The research area concerned with future and past climate evolution is very active today. As a complement to its own research, SKB is therefore following current research in international scientific journals and at scientific meetings, as well as the work of bodies or organizations that deal with climate issues.

19.2 Ice sheet dynamics and glacial hydrology

The glacial climate domain is defined as areas covered by glacial ice, in other words glaciers or continental ice sheets. The most important research areas for this climate domain are ice sheet dynamics, glacial hydrology and the glacial history of the Weichselian.

A typical Quaternary glacial cycle spans a period of around 100,000 years. During the most recent glacial periods, all of Sweden was dominated by the glacial climate domain during the maximum stages. The average extent of the continental ice sheets during the Quaternary Period was, however, much less than that; on average, the areas around Forsmark and Oskarshamn were in all probability ice-free.

The thermal and hydraulic properties of the ice sheet determine how the ice affects its bed (including rock and groundwater) and thereby also how it affects a final repository. The glacial meltwater is ion-poor and oxygen-rich. Groundwater recharge during periods of glacial climate domain therefore causes water with such properties to be transported downward in the rock. Some of SKB's efforts are therefore spent on studying how groundwater is recharged and transported down through the rock under glacial conditions, see Section 25.2.3, and how a groundwater of glacial origin affects, for example, the properties of the buffer clay, see Chapter 24. When a continental ice sheet advances and retreats, the rock stresses in the concerned area are affected, which could lead to a reactivation of existing fracture zones in the form of glacial earthquakes, see Section 25.2.7.

Conclusions in RD&D 2007 and its review

SKI appreciated the way in which climate issues had been integrated in the safety assessment and thought that SKB had made great progress in understanding how climate changes affect a repository. SKI pointed out that the issue of glacial hydrology needed to be better integrated with other climate-related issues and that differences in assumptions between the calculations for glacial hydrology and the evolution of the inland ice sheet needed to be clarified. SKI also pointed out that SKB should deal with the uncertainties that arise due to the fact that most process studies have been done on relatively small glaciers, while most modelling is concerned with large ice sheets. SKI further pointed out that calculated maximum ice thicknesses in Laxemar and Forsmark are reasonable, but that SKB should describe the model better and the simplifications on which it is based.

SKI further commented that SKB should link the efforts to gain a better understanding of the hydrological conditions in and around an ice sheet to the efforts related to reactivation of existing fractures in the rock (Section 25.2.7). SKI pointed out that SKB should give a more detailed account of the risk and consequences of glacial erosion at both candidate sites. SKI appreciated SKB's plan to utilize the Greenland ice sheet as an analogue of future glacial conditions at Forsmark. However, SKI complained of the lack of a reference to the feasibility studies that were being conducted on Greenland.

The Swedish National Council for Nuclear Waste writes that the hydrogeology of the site must be taken into account, and that the different climate scenarios should be taken into account when modelling the interface zone between geosphere and biosphere. The Council wondered how and by whom SKB's planned studies of the last glaciation in Scandinavia will be done, given the fact that such work is already under way elsewhere.

No viewpoints on the climate programme are found in SSI's review of RD&D Programme 2007.

A study concerning an estimation of the erosion depth during the Quaternary Period was presented in RD&D Programme 2007 /19-1/. The study estimates an upper bound for for the average erosion of bedrock during a glacial cycle or a single glaciation. The erosion process had also been studied within the framework of SR-Can (Process Report), where current literature in the field is reviewed and conclusions are drawn regarding the impact on the repository. SKI emphasized that if considerable glacial erosion cannot be excluded, it should be taken into account in deliberations concerning an appropriate depth for the Spent Fuel Repository.

Newfound knowledge since RD&D 2007

SKB describes in its climate programme both a reference evolution containing a repetition of conditions reconstructed for the last glacial cycle and alternative future climate evolutions. The alternative evolutions that are studied are:

- A colder and drier climate than in the reference evolution, which results in deeper permafrost and longer periods with a periglacial climate domain, see Section 19.4. Several alternative evolutions are treated.
- A colder climate with greater precipitation quantities that can build up a thicker ice sheet or ice that lasts longer than in the reconstruction of the last glacial cycle, see below in this section. Traces of other glaciations than the last one are also utilized here. Several alternative evolutions are treated.
- A future climate dominated by global warming, i.e. warmer than during the last glacial cycle, see Sections 19.3 and 19.5. Several alternative evolutions are treated.

Glacial geological information

In order to get a better substantiated and more detailed picture of the reference evolution, several studies of the glacial history of the Weichselian have been done. The studies concern both the spatial and temporal extent of the ice sheet as described in this section, and involve quantitative climate reconstructions for selected periods, in order to study the bounds within which the climate in Scandinavia varied during the last ice age, see Section 19.5.

In September 2007, SKB arranged a small international workshop entitled "Fennoscandian palaeoenvironment and ice sheet dynamics during MIS 3" /19-2/. The purpose was to shed light on the state of knowledge concerning palaeoconditions during a part of the Mid-Weichselian called Marine Isotope Stage 3 (MIS 3), a long period that preceded the last glacial maximum. One of the purposes of the workshop was to gather the necessary input data for the climate modelling studies reported in the paragraph "Information from climate models" in Section 19.5. After the workshop, SKB published a special volume of the scientific journal Boreas, dealing with this important part of the Weichselian glaciation based on the information presented at the workshop /19-3/.

The climate simulations carried out within SKB's climate programme (see Section 19.5) focused on a period during MIS 3. Such a climate simulation requires a specific ice sheet configuration for the period in question. Pollen stratigraphic studies and datings of mammoth remains and sedimentary sequences have, however, often given disparate pictures of the size of the ice sheet during MIS 3. In order to permit an accurate assessment to be made of the configuration of the Weichselian during this period (as input data to the climate simulation, among other things), a study was conducted where all available datings of interstadial sediments from MIS 3 from Norway, Sweden, Denmark, Estonia and Finland were compiled in a database /19-4/. It is the first time such a compilation has been done where all available data are thoroughly and systematically assessed with respect to the quality of the datings. The results show that one possibility is that large parts of Sweden were ice-free early and during the middle part of MIS 3.

During the period in question, several studies have provided glacial geological information on the Weichselian of importance for the description of the reference evolution /19-2, 19-4, 19-5, 19-6/. The studies have, together with the studies described in Section 19.5, contributed information that necessitates a revision of the classic picture of the Weichselian glaciation, with a very long uninterrupted period of ice cover over Sweden from the start of the Weichselian around 74,000 years ago up to the deglaciation around 10,000 years ago /19-7/. The new picture of a more dynamic ice sheet, and thereby a more dynamic climate, with ice-free conditions in large parts of Scandinavia and Sweden during MIS 3, is also confirmed by other studies from the general scientific literature, summarized in /19-2, 19-4, 19-5/. Results from /19-4/ have also been published in a number of scientific articles /19-8, 19-9, 19-10/. In this context, the large body of scientific articles that have come out of /19-5/ are also of relevance, as reported in Section 19.5. The results, which describe a more dynamic picture of the ice sheet with ice-free conditions during parts of the Mid-Weichselian, concur with the picture of the evolution of the ice sheet that is used in SKB's reference evolution, see Figure 19-2.

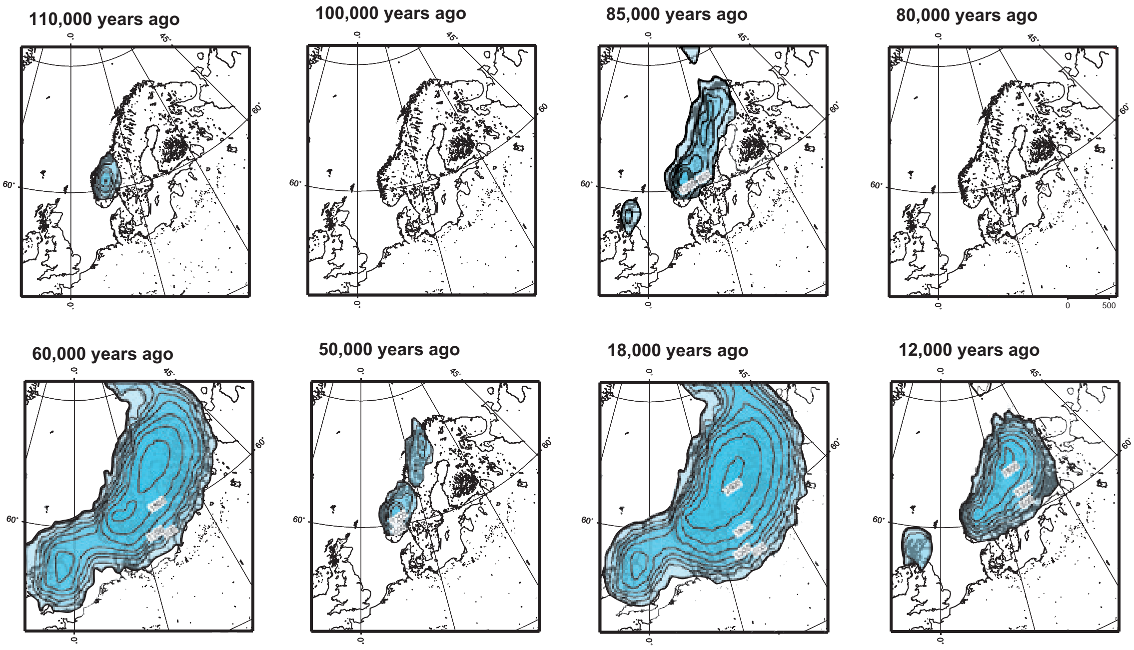


Figure 19-2. *Examples of ice sheet configurations and ice surface elevation (with a 300 metre contour interval) from the reconstruction of the Weichselian glaciation in SKB's reference evolution. It is worth noting that the extent of the ice sheet during the long period called Marine Isotope Stage 3 (MIS 3) is very limited above Sweden. This is exemplified in the figure by a picture of the ice 50,000 years ago, where the areas around Forsmark and Laxemar can be seen to be ice-free. In contrast to the classic picture of the Weichselian glaciation, where all of Sweden is covered with ice from MIS 4 until the end of MIS 2, much new information from SKB's climate programme and from other research suggests that a long ice-free period prevailed during MIS 3. Studies of the climate during an early and a late part of MIS 3 are described in Section 19.5.*

The results of the studies have reduced the uncertainties in the glacial history of the Weichselian and contributed important information on the long and important MIS 3 period during the Mid-Weichselian. As a result, the safety assessment has better input data for the reference evolution based on a repetition of the Weichselian glaciation, and a better picture of the variability that can occur during a glaciation.

Knowledge of the configuration and evolution of the ice sheet at the time of the last glacial maximum about 20,000 years ago and the subsequent deglaciation of Scandinavia is already relatively good.

Numerical ice sheet simulation

The numerical reconstruction of the ice sheet during the Weichselian (Figure 19-2), which was described in RD&D Programme 2007, is still used for SKB's reference evolution. The methodology and the results that described how extreme climates within the glacial domain are studied, for example concerning maximum ice thicknesses in Forsmark and Oskarshamn, are still relevant in the climate programme. The results showed that the maximum expected ice thickness above the sites is 2,600 metres in Laxemar and 3,200 metres in Forsmark.

As before, in order to describe a reference evolution where all climate-related processes of importance for repository performance are included, the variation in ice extent and ice load history from the ice sheet reconstruction (Figure 19-2) is used as input data in the model simulations of shoreline displacement and permafrost that are included in the reference evolution, see Sections 19.3 and 19.4.

In conjunction with glaciations, couplings between mechanical and hydraulic processes are also significant. Mechanical processes that can give rise to glacial earthquakes are, for example, dependent on the groundwater's pore pressure, which is affected during glaciations. The evolution of the ice sheet described in the reference evolution (Figure 19-2) has been used as input data to a 3D model study of how the stresses in the Earth's crust have changed during the Weichselian and how this affects the stability of faults /19-11/. The results of the study are described in Section 25.2.7.

Process studies of ice sheet hydrology

In addition to the previous compilation of the theoretical knowledge concerning how water flows in and underneath a glacier or ice sheet, and how this knowledge is applied in model simulations /19-12/, a compilation has been made of spatial and temporal variations in glacial hydrology /19-13/. The study summarizes extensive observations of glacial hydrology and basal hydrological conditions from Storglaciären in northern Sweden made during the period 1990–2006. The study also discusses how observed processes from small glaciers of this type can be used as analogues for the processes at ice sheets.

A challenge in this context is that our knowledge of glacial hydrological processes at ice sheets is inadequate, and that the processes in many cases are difficult to conceptualize in today's large-scale ice sheet models. The important link between the complex glacial hydrological system and the dynamics and the flow of ice sheets is therefore largely lacking in today's models. To address these issues, SKB, together with its sister organizations Posiva and NWMO (Nuclear Waste Management Organization, Canada), started a large research project on western Greenland (Greenland Analogue Project, GAP) to study hydrological processes at an existing ice sheet, see Section 19.6. This project will hopefully contribute information that makes the large body of glacial hydrological knowledge available from small glaciers much easier to use in discussions on an ice sheet scale.

The model studies of hydrogeology under glacial conditions that have been carried out within SKB's hydrogeological programme are described in Section 25.2.3. The hydrogeological model studies and the climate programme are integrated with each other due to the fact that the hydrogeological model runs are set up with data and assumptions based on the knowledge or data gathered within the climate programme (Figure 19-1). Possible examples are which ice profiles and associated hydrostatic gradients have been used, or the velocity at which the ice sheet arrives or departs above the model domain. The hydrogeological simulations also treat cases with permafrost and a combination of permafrost and ice sheet, which are also set up directly in cooperation with the climate programme.

Erosion

Since 2008, SKB has been conducting a project aimed at describing and, where possible, quantifying denudation (weathering and erosion) of the ground surface during long time spans (100,000 and a million years) in the regions around Forsmark and Oskarshamn. The method used is studies of the long-term morphological evolution of the bedrock. SKB is also keeping abreast of current literature in the field.

Programme

The scientific discipline that describes the hydrology at and beneath ice sheets is making rapid progress, via both the GAP project and studies by other research groups of how the Greenland ice sheet may respond to a warmer future climate. The current state of knowledge concerning ice sheet dynamics and glacial hydrology that was compiled and reported in /19-14/ will therefore be updated in connection with SR-Site and later again within the framework of the safety assessment for the extension of SFR. Information about the basal hydrology of ice sheets, for example how the hydraulic pressure situation varies on different spatial and temporal scales, is of great importance in setting up simulations of groundwater flow under glacial conditions. We intend to continue working within this area by studying how glacial hydrology can and should be conceptualized in hydrogeological studies, see Sections 19.6 and 25.2.3. The GAP project is important in this context. The results will give us a better understanding of the hydrological and geochemical conditions in and around a continental ice sheet, and specifically address issues concerning how an ice sheet affects the groundwater flow and chemistry around a final repository.

In connection with the GAP, SKB also intends to carry out a programme concerning ice sheet modelling, with a focus on conceptual questions and how basal conditions and basal hydrology are handled in ice sheet models. Field data collected from the study site for the GAP will be used (meteorological information on meltwater production on the surface, ice movement data, ice thickness, basal temperature, hydrological conditions, etc.). In addition, the model used for ice sheet simulations in SR-Can and SR-Site, and the assumptions on which the simulations are based, will be described in greater detail within the framework of SR-Site.

In conjunction with the GAP, SKB is also planning a study with the specific purpose of transferring the glacial hydrological knowledge obtained from the GAP to Scandinavian conditions, including sub-projects that deal with the area around Forsmark.

The project which SKB is conducting for the purpose of describing and quantifying denudation of the ground surface during long time spans in the areas around Forsmark and Oskarshamn is planned to be completed in 2010. We will also continue to monitor the literature in this field.

19.3 Isostatic changes and shoreline displacement

Shoreline displacement is the most important climate-related process for a repository within the temperate climate domain. The temperate climate domain is defined as a situation without an ice sheet or permafrost. In other words it consists of areas with a temperate climate in a very broad sense, and includes all conceivable climates in Sweden dominated by global warming.

In the SR-Site safety assessment, the description of a warmer future climate, caused by an increased greenhouse effect, has been broadened compared with previous assessments. Two cases with different durations of climate impact are now treated. In the first climate evolution, the first 60,000 years at Forsmark and Oskarshamn comprises, as before, a temperate climate domain. After that the climate becomes gradually colder, with at first short, but then increasingly longer, periods of permafrost. The first period of glacial conditions comes at the end of this approximately 100,000-year-long period. In the second climate evolution, dominated by global warming, a temperate climate domain exists during the entire initial 100,000-year period, i.e. the climate skips a whole glacial cycle before the onset of the next glacial period.

In reality, the evolution of a climate dominated by global warming could entail an initial period with temperate climatic conditions of a different duration than in the selected cases. A warmer climate globally could furthermore, at least theoretically, entail a cooler climate than today regionally across Scandinavia, caused by shifts in the thermohaline circulation pattern in the North Atlantic. How these complex issues are handled up to 2006 is described in /19-14/. This information will be updated as a part of the work with SR-Site.

Relatively good knowledge exists concerning conditions and processes in the temperate climate domain and their importance for repository safety and conditions in the biosphere.

Conclusions in RD&D 2007 and its review

SKI and the Swedish National Council for Nuclear Waste point out that a future ice sheet melting due to global warming, and the importance of this for the final repository, should be examined in future safety assessments, especially the consequences of infiltration of groundwater of higher salinity down into the Spent Fuel Repository. The Council finds the proposed programme for refined predictions of sea level rise in a warmer climate to be excellent, and points out that SKB needs to consider the problems which a higher sea level would entail during the repository's construction period. The Council writes that the impact of climate change on water chemistry needs to be taken into consideration.

Newfound knowledge since RD&D 2007

The current state of knowledge for theories regarding isostatic changes and shoreline displacement has been updated and reported in /19-15, 19-16/. A very large quantity of information on future changes in sea level has also been published in the general scientific literature since RD&D Programme 2007. In addition to the two studies referred to above, the scientific state of knowledge regarding isostatic changes and the impact of global warming on sea level variations will be updated in the work with SR-Site and the safety assessment for the extension of SFR.

One of the new studies /19-16/ describes the physics behind Global Isostatic Adjustment (GIA), how it affects sea levels and shoreline displacement, and the methods that are used to study and understand these processes. The study describes the scientific background of the processes and methods used in SKB's GIA simulations of shoreline displacement. The report provides a more in-depth understanding of GIA processes than that provided in /19-14/.

A study done in cooperation between SKB's biosphere and climate programmes provides a picture of possible future sea level changes at Forsmark and Laxemar up until the year 2100 /19-17/. The study, which is chiefly based on published information, includes processes such as eustatic changes (sea level), isostatic changes (land uplift) and regional and local extremes in water level today and in 2100. The results show that the maximum water level that could exist for short durations in the year 2100, given a scenario with substantial global sea level rise, is just over +2.5 metres for Forsmark and just under 3 metres for Laxemar-Simpevarp, see Figure 19-3. If additional uncertainties are included, the value for both sites is more than +3 m. For a description of the different estimates of possible extremes in global sea level rise, see /19-17/. In this context it is important to note that this scientific area is in a very intensive phase, and revisions of these figures are to be expected.

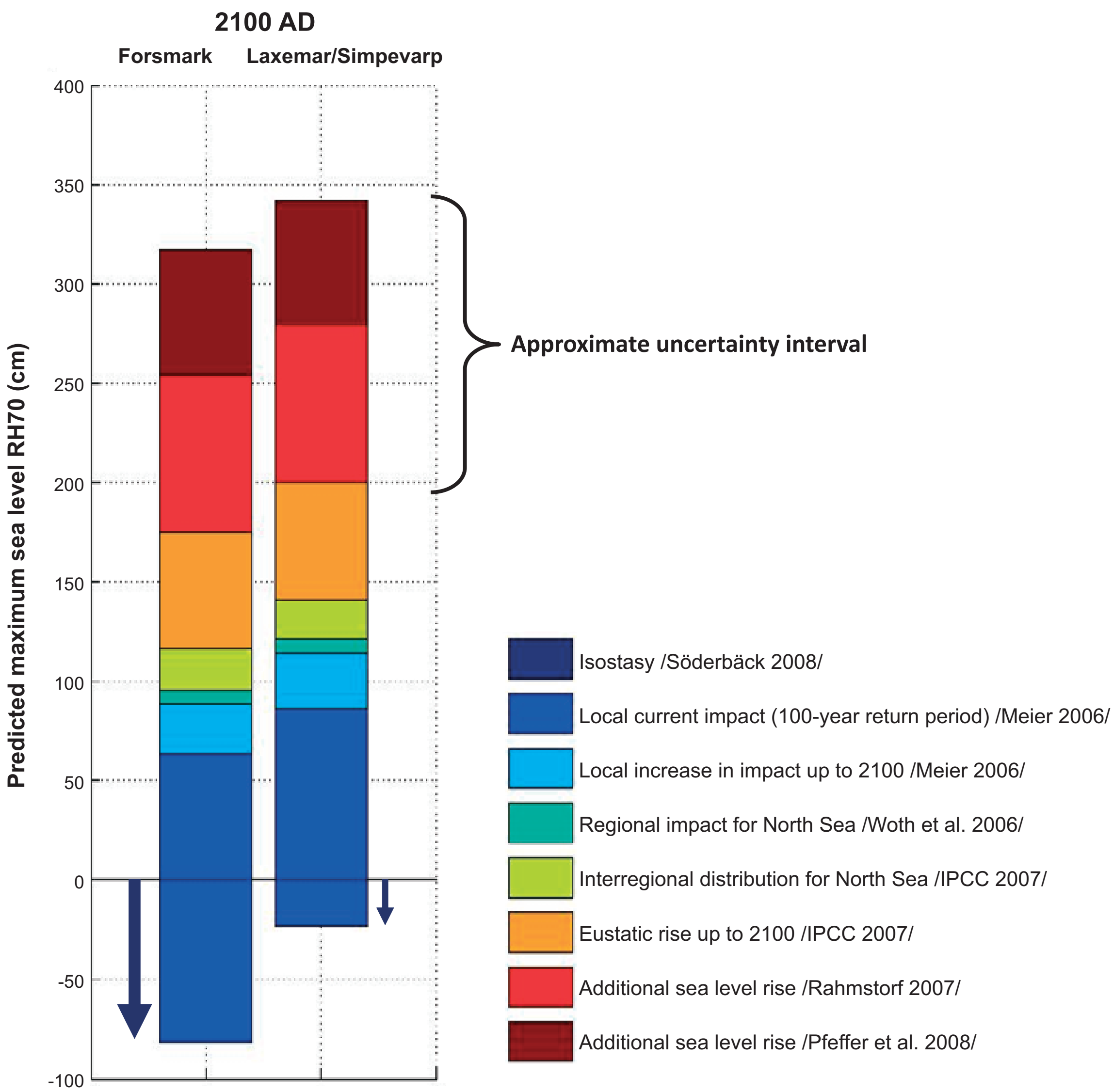


Figure 19-3. *Compilation of predicted contributions to extreme sea levels of short duration in Forsmark and Laxemar in 2100. Land uplift is indicated by downward arrows. As a rough measure of the uncertainty of the three eustatic contributions, this uncertainty has been indicated by a bracket centred on the highest value published by Rahmstorf /19-18/. For the references used in the figure, see /19-17/.*

Programme

The GIA simulations used to describe shoreline displacement in SR-Can will be used for the most part in SR-Site as well. They will, however, be supplemented by new information from published scientific literature, above all with respect to possible future sea level rises. The simulations, which were performed in 2D, will also be supplemented by some information from three-dimensional GIA simulations that include lateral variations in the thickness of the Earth's crust /19-16/.

The scientific literature concerning future changes in sea level (causes, mechanisms and consequences) will grow considerably in the years to come. SKB will therefore continue to follow this field of research closely, and like in the study /19-17/ use relevant results for assessment of the impact which changes in the shoreline at Forsmark would have on a Spent Fuel Repository, both in a short (up to 2100) and a longer time perspective (several tens of thousands of years).

A climate evolution with a warmer climate due to an increased greenhouse effect, with its very long initial period of temperate climatic conditions, was judged in SR-Can to be primarily positive for repository performance. It is likely that a similar judgement will be made in SR-Site. The impact which climate evolutions dominated by global warming may have on groundwater chemistry is being treated in SR-Site, see Section 25.2.10.

19.4 Permafrost

Aggradation (growth) and degradation (melting) of permafrost are the most important climate-related process for a final repository within the periglacial climate domain. In the reference evolution described in the safety assessments, a periglacial climate domain with permafrost prevails during much of the time, in SR-Can for about one-third of the next 120,000 years /19-14/. The occurrence of permafrost greatly affects the groundwater's flow pattern. Groundwater composition is also affected due to salt exclusion.

Conclusions in RD&D 2007 and its review

SKI points out that SKB's calculations of permafrost involve uncertainties in both models and input data that have not been reported clearly, and that SKB should report the consequences of the fact that buffer and backfill do in fact freeze. The Swedish National Council for Nuclear Waste finds that the programme for handling of permafrost is good, and that studies of freezing in the backfill material are particularly urgent.

Newfound knowledge since RD&D 2007

Detailed studies of permafrost growth in the Forsmark area have been carried out during the current RD&D period. The permafrost simulations in SR-Can were performed in 1D /19-14, Sections 3.4 and 4.4.1/. The supplementary studies for SR-Site are being done in a newly developed 2D model based on the former 1D model. The 15 kilometre long and 10 kilometre deep profile runs in the regional topography's gradient, through the planned Spent Fuel Repository. Detailed data from the site investigation programme have been used as input data with regard to, for example, surface topography, thickness and composition of the overburden, topographic moisture index (including vegetation type), shoreline displacement and future sea level change, locations of rock domains and fracture zones, and thermal, chemical and hydraulic properties. The purpose of the study is to provide a realistic picture of permafrost growth in the Forsmark area, and to what extent it affects repository performance. The results are also used to set up realistic conditions regarding permafrost in hydrogeological studies of groundwater flow under periglacial and glacial conditions. In these studies, dedicated groundwater simulations are performed under conditions affected by permafrost or the combination of permafrost and an ice sheet, see Section 25.2.3. In studies involving cooperation between climate, geohydrology and biosphere disciplines, the effect of the presence of potential future taliks (unfrozen zones in soil with permafrost) in the Forsmark area is also being studied, see Sections 25.2.3 and 26.6.

The new permafrost model also handles salt exclusion in connection with permafrost growth, something that was not included in SR-Can. Results describing salinity during and after permafrost conditions are planned to be used in SR-Site.

The results from the permafrost study demonstrate the site-specific process of permafrost formation and melting in the Forsmark area, given the local topography and the local properties of the ground surface, the rock and the groundwater. The results further show that given similar input data and setup of the model, simulations with the new improved model give maximum permafrost depths that are well in line with those presented in SR-Can. The preliminary results indicate, as expected, that conditions prevailing at the ground surface (temperature, vegetation and snow cover) are the primary controlling parameters for the spatial and temporal extent of the permafrost in Forsmark. The results further indicate that the multidimensional variation in thermal properties in the rock and the convective heat transport have only a minor effect on the permafrost, which supports the assumptions made in SR-Can's 1D permafrost simulations and a further use of these results in SR-Site as well.

Extensive sensitivity studies have been performed on input data with the new permafrost model, including on the temperature curve used for the temporally transient simulations. Like before, the reference evolution is analyzed, supplemented by climate cases more favourable for permafrost growth. The uncertainties in input data, and their consequences for the results, will be described in greater detail than in the permafrost study for SR-Can.

In this context it should also be mentioned that the properties of the buffer clay during and after freezing have been studied, see Section 24.2.4.

SKB has also carried out a study where measured borehole temperatures from boreholes in Forsmark and Laxemar have been compared with equivalent *simulated* temperature profiles (calculated based on thermal conductivity, thermal diffusivity, geothermal heat flow and assumed variations in palaeotemperature) /19-19/. The purpose was, among other things, to study the present ground temperature at the sites and how it has been affected by past climate variations, and to calculate values for the geothermal heat flow, corrected for a palaeoclimatic influence. The results show that the ground temperature used for the present-day situation in SR-Can's permafrost simulations was presumably too low. With the aid of results from this study, it has been possible to improve this in SR-Site's permafrost simulations, as described above. The calculated geothermal heat flow at Forsmark and Laxemar was 61 and 56 metre-watts per square metre (mW/m^2), respectively, with an uncertainty of +12 percent and –14 percent for both sites /19-19/. The results have been used as input data in the site-specific simulations of permafrost described above.

Programme

Even if permafrost is not included as a dedicated sub-project in the GAP, the GAP's programme is set up so that the project contributes important information in this area as well, see Section 19.6. In the region where the GAP is being carried out, there is extensive permafrost in the bedrock in front of the ice sheet today. Observations of, for example, salinity in the groundwater beneath the permafrost will exemplify what the geochemical composition of the groundwater may look like in crystalline bedrock under these conditions. In connection with the GAP, SKB plans to continue portions of the ongoing permafrost modelling programme.

To gain a greater understanding of the palaeotemperatures that have prevailed at Forsmark, we plan to possibly go further with a more detailed study of one of the deep boreholes in the area. The calculations of temperature curves for the borehole /19-19/ could be improved by reducing the uncertainties in data on the thermal properties of the rock, after which more detailed information could be obtained from a comparison with measured borehole temperatures.

Several of the approaches and methods used in the climate programme for the safety assessment of the Spent Fuel Repository can also be used for safety assessments of the repository for short-lived radioactive waste (SFR). However, there are in this context important differences between the two repositories: SFR is located much shallower, and the half-lives of the radionuclides that give doses are shorter. How this affects the validity of SKB's climate evolutions (the climate cases from SR-Can were used in SAR-08) will be further investigated within the framework of the safety assessment for the extension of SFR. Within the framework of this safety assessment we also plan to carry out dedicated permafrost/freezing simulations adapted to the special premises and issues that apply for SFR.

In connection with the GAP project (see Section 19.6), SKB plans, together with Posiva and NWMO, to conduct studies of permafrost and its importance for hydrology and groundwater composition.

19.5 Climate and climate variations

In addition to the studies of climate-related processes that take place in connection with ice sheets, permafrost and shoreline displacement described above, the climate programme also includes studies of the climate itself. The purpose is to provide more complete information and examples of what the climate may be like within the different climate domains and to provide a better background for estimating the magnitude of possible future climate changes. For this purpose, SKB uses both natural climate archives and climate models.

Conclusions in RD&D 2007 and its review

SKI supports SKB's plans to carry out projects aimed at obtaining a more detailed picture of the climate in Scandinavia during a glacial cycle. The Swedish National Council for Nuclear Waste points out that it is important to simulate the future climate evolution with an increased concentration of greenhouse gases, and that SKB's climate research should be developed according to three time scales: the next 100 years, the subsequent 1,000 years and the 100,000 years following that.

Newfound knowledge since RD&D 2007

The two planned projects described in RD&D Programme 2007 aimed at providing a more detailed picture of the climate in Scandinavia during a glacial cycle have been carried out as planned. A very large quantity of results have been obtained from each project, something which is not possible to present in this RD&D-programme. Most of the results are presented in the publications cited in the following summation in the paragraphs "Information from geological climate archives" and "Information from climate models".

The information being gathered within SKB's climate research programme is well suited to be used to describe climate and climate-related processes for the three time scales: the next 100, 1,000 and 100,000 years.

Information from geological climate archives

A palaeoclimatic study of a sediment core from the Sokli site in northern Finland /19-5/ focuses on the youngest interstadial during the Weichselian, MIS 3, where ice-free conditions on this site have been dated to approximately 50,000 BP (other parts of the study are described in Section 19.2). The locale from which the sediments have been taken is unique for Scandinavia in that the oldest sediments are around 130,000 years old and that sedimentation has occurred on the site during very long periods up to the present. An extensive reconstruction of conditions on the site has been carried out based on an analysis of a large body of multi-proxy data. The results have been surprising in many respects and call into question the classic picture of the glacial history of the Mid-Weichselian. The results show that this part of northern Scandinavia not only had ice-free conditions about 50,000 years ago, but that the climate during the ice age was very warm, with mean July temperatures as high as today (up to 13 ± 1.15°C). The warm climate is believed to be in part due to higher insolation during the summer at this latitude than today.

Studies showing ice-free conditions in a large part of Fennocandia 50,000 years ago (see /19-5/ and references in this) contribute important palaeoclimatic information that enables us to better understand within what limits the climate can vary naturally during a glacial cycle. The results are interesting because they show that the climate has shown great temporal and amplitudinal variability during the Weichselian. Such studies have never before been carried out so systematically and broadly for any locale in Scandinavia in terms of the quantity of multi-proxy data and parameters analyzed. The results of the study have also been published in a series of scientific papers /19-20, 19-21, 19-22, 19-23, 19-24, 19-25, 19-26/. The study has also been presented at a large number of scientific conferences /19-27, 19-28, 19-29, 19-30, 19-31, 19-32, 19-33, 19-34, 19-35, 19-36/.

More studies of relevance to the subject of climate and climate variations /19-4, 19-8, 19-9, 19-10/ have been described in Section 19.2

Information from climate models

SKB has conducted and concluded a multi-year climate modelling project whose overall purpose was to describe extremes within which the climate can vary on a 100,000-year timescale /19-37/. Based on the forcing conditions that have resulted in extreme conditions during the Weichselian as well as possible future conditions affected by anthropogenic emissions, climate simulations have been carried out to investigate and describe three different periods:

1. a period with a presumed periglacial climate (cold and dry) at the end of MIS 3 (about 44,000 years ago),
2. a period with glaciation (Last Glacial Maximum, LGM, about 20,000 years ago),
3. a future period dominated by global warming (several thousand years in the future).

In conjunction with the setup of the periglacial case, SKB arranged a small international workshop /19-2/ to evaluate the state of knowledge concerning parameters that need to be defined in the climate model (ice extent, sea level etc.) for the chosen period during MIS 3 and determine the best values that can be assigned to these parameters in the climate models.

The simulations of the three cases were carried out first with a global climate model (CCSM3), after which the results were scaled down in a regional climate model (RCA3), see Figure 19-4. The regional model produced detailed information about climate variables such as air temperature and precipitation over Europe. For the purpose of obtaining better data, vegetation simulations were also carried out in conjunction with the regional climate simulations, whereby vegetation and climate were allowed to influence each other (Figure 19-4). Relevant climate data for the three periods were then extracted from the regional model for the areas around Forsmark and Oskarshamn.

In addition to the modelling activities, an effort was made to collect different types of proxy data on climate parameters for MIS 3 and LGM. Some of the palaeoclimatic data were used to bound different parts in the model's setup, while other palaeoclimatic data were used to evaluate the model results.

The results of the comparison of the results from the global and regional model simulations with palaeoclimatic data show that the model results are in broad agreement with proxy data /19-37/. They are also in agreement with other model simulations. The resulting climates are also in qualitative agreement with the imposed extent of ice sheets and types of vegetation. Of particular interest is the fact that the results for the climate simulation for the MIS 3 stadial are consistent with the pre-imposed ice-free conditions in southern and central Sweden. The study also shows that a modern climate model for this late part of MIS 3 generates a cold and dry climate favourable for permafrost growth.

All simulations show that there is a wide span in possible climates in a 100,000-year perspective, exemplified for Forsmark and Oskarshamn. The difference in mean annual temperature between the simulation of a periglacial climate and a future climate dominated by global warming can be 15 degrees in the Forsmark area. The quantity of precipitation is around a factor of two higher in the

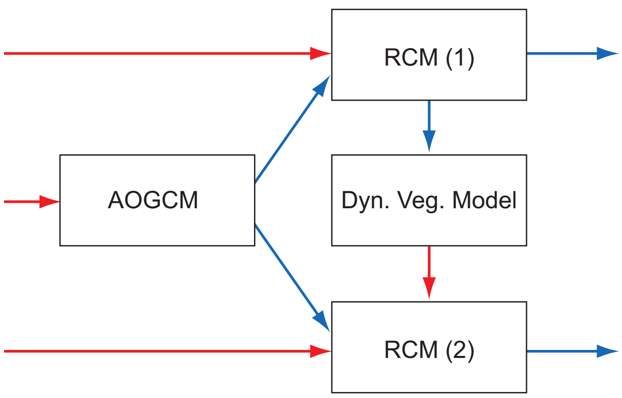


***Figure 19-4.** The model structure used in /19-37/. AOGCM and RCM are the global and the regional climate models, respectively. Dyn. Veg. Model is the vegetation model. Red arrows symbolize forcing conditions and blue arrows symbolize simulated climate.*

warm, wet climate compared with the periglacial climate simulated for the end of MIS 3 (not to be confused with the warm period at the beginning of MIS 3 studied in /19-5/).

Parts of the study /19-37/ have been published in scientific papers /19-38, 19-39/. The results have also been presented at a number of scientific conferences /19-40, 19-41, 19-42, 19-43, 19-44, 19-45, 19-46/.

The results from /19-37/ constitute a large body of data that makes it possible to exemplify in detail possible climate situations in the temperate climate domain dominated by global warming and in the periglacial climate domain. The results, in the form of a large number of simulated climate parameters on global, regional and above all local scales (Forsmark and Oskarshamn area), have been made available and are being used within both the climate programme and other research programmes at SKB, for example surface ecosystems, see Chapter 26.

The studies of geological climate archives and the studies with climate models contribute essential information to the description of the variability and properties of the climate in SKB's safety assessments. The studies have contributed to a greater understanding of the more extreme climate evolutions that are analyzed, and to investigating the realism of these climate cases.

Programme

In line with the glacial geological and palaeoclimatic studies that have now been done for the Mid-Weichselian /19-5, 19-4/, SKB is planning to study similar issues for an earlier, and presumably different, ice-free period during the early Weichselian and for the Holocene. The sediment core from northern Finland that has been analyzed in detail for the MIS 3 period during the Mid-Weichselian is planned to be analyzed in the same way for the interstadials MIS 5c–d during the early Weichselian. The purpose is, like in the previous study /19-5/, analyze and, if possible, quantify climate parameters for these earlier and presumably different periods during the Weichselian. The study will supplement the picture of what climates prevailed during the preceding glacial cycle and improve the background data for the descriptions of possible future climates in SKB's safety assessments. In this context, a study is also planned to analyze the Holocene sequence from the same sediment cores, partly to further evaluate the methods used for the climate reconstructions for the Mid-Weichselian /19-5/ and the early Weichselian (planned study), and partly to study Holocene climate variability. To further nuance the picture of possible future climates, we also plan to possibly go further with climate model studies to supplement the study in /19-37/.

19.6 Greenland Analogue Project (GAP)

In order to gain a better understanding of how climate change, and particularly glaciations, may affect a final repository in the long term, SKB has, together with Posiva and NWMO, initiated a project on western Greenland, the Greenland Analogue Project (GAP), where a modern ice age analogue is being studied. The expected results are of great importance for safety assessments of both the Spent Fuel Repository and for SFR.

Reconnaissance observations were conducted in the field area in 2005, after which the GAP was initiated in 2008 with introductory field investigations. The project is planned to be finished in 2013. The field investigations for the project are being conducted on western Greenland in an area east of the village of Kangerlussuaq. The rocks in the area exhibit great similarities to the rocks in Forsmark. This similarity is a prerequisite for the studies to be meaningful and provide the desired information on hydrology, hydrochemistry and permafrost adjacent to an ice sheet.

The following processes and general issues are being studied within the framework of the GAP:

1. How deep down in the bedrock can glacial meltwater penetrate?
2. What is the chemical composition of the meltwater if and when it reaches repository depth (around 500 metres)?
3. Where beneath the ice sheet are meltwater and groundwater generated?
4. How large a fraction of the oxygenated meltwater reaches repository depth?
5. What is the pressure situation beneath the ice sheet?
6. Can taliks act as discharge points for deep groundwater?

By using the Greenland ice sheet as an analogue for a future situation at Forsmark with a glacial climate domain, it is possible to make observations required for an integrated and better conceptual understanding of hydrological and hydrogeochemical processes during glaciation. The goal of the GAP is to provide better process understanding in order to be able to better create conceptual and numerical models of groundwater flow, groundwater chemistry and hydromechanical factors during glacial periods. One goal is that knowledge from the project can be used to better estimate the degree of pessimism in the assumptions made in today's hydrogeological simulations, and if possible reduce the degree of uncertainty in these assumptions.

The following organizations and universities are participating in the GAP: University of Wyoming (USA), University of Montana (USA), Aberystwyth University (UK), Waterloo University (Canada), Stockholm University (Sweden), Uppsala University (Sweden), GEUS (Geological Survey of Denmark and Greenland), GTK (Geological Survey of Finland), In2EarthModelling (Switzerland), TerraSolve (Sweden), Bergab (Sweden), SKB, Posiva and NWMO. The following universities are also participating indirectly in the project: Bristol University, Edinburgh University, Cambridge University, Swansea University, University of Washington, Princeton University and University of Indiana.

Conclusions in RD&D 2007 and its review

SKI regards with satisfaction SKB's plans to use the Greenland ice sheet as an analogue for glacial conditions at Forsmark and Laxemar. The Swedish National Council for Nuclear Waste considers the planned Greenland Analogue Project to be highly urgent, but says that the description of the project was vague and rudimentary.

Newfound knowledge since RD&D 2007

Comprehensive investigations were initiated in the field area on Greenland in conjunction with the GAP and have subsequently been carried out in RD&D Programme 2007. Information has been obtained on the permafrost depth in the area by means of cored boreholes and high-resolution temperature profiling, and the preliminary results suggest that there are active taliks in front of the ice. The results from bedrock and fracture mapping as well as geochemical analyses of the rocks in the Kangerlussuaq area show that the bedrock resembles the bedrock in Forsmark to a high degree. A network of GPS stations and automatic weather stations has been installed on the ice sheet in the field area, and a large quantity of ice movement data and meteorological data has been collected from these stations. Parts of the bottom topography beneath the ice have been studied by means of ground-penetrating radar surveys. These radar data, together with the results of the surface investigations on the ice, provide indirect information on how the subglacial hydrological system works.

SKB arranged a project workshop that was held in the autumn of 2009 in connection with a GAP modelling workshop that focused on ice sheet modelling and hydrogeological modelling. The results from the GAP have been presented at a number of conferences/19-47, 19-48, 19-49, 19-50, 19-51/.

Programme

The GAP consists of three sub-projects (A–C) working towards specific objectives.

Sub-project A is studying *indirectly* the subglacial hydrology of the ice sheet and groundwater recharge by means of glaciological and geophysical investigations. There is a full-coverage network of continuous GPS stations for measuring ice movement and automatic weather stations for providing data for calculations of water production on the surface. Furthermore, measurements are made via e.g. ground-penetrating radar, melt balance monitoring and seismic surveys.

The purpose of sub-project B is to make *direct* observations of subglacial hydraulic conditions (including spatial and temporal variations in water pressure) by means of hot water drilling through the ice sheet and placement of pressure gauges at the base of the ice.

Sub-project C studies the hydrogeochemistry and hydrogeology of the bedrock in the area outside and beneath the ice by means of rock drilling and monitoring programmes. A borehole down to the equivalent of repository depth is planned to be drilled and instrumented. Information on permafrost in the area will also be obtained from sub-project C.

Figure 19-5 is a schematic drawing illustrating in what areas the different sub-projects within the GAP are conducting investigations. Table 19-1 lists the different data being collected in the GAP, which are being used in the ice sheet and hydrogeological modelling work being pursued within, but also in parallel with, the GAP.

Constant cooperation takes place between the three sub-projects for planning, rationalizing and interlinking the results in the GAP. This also ensures that the end results from the project will provide an integrated picture of ice sheet hydrology, geohydrology and geochemistry within the study area. A number of disciplines within SKB's research programme derive direct or indirect benefit from the results of the GAP: 1. hydrogeology, the description of the hydrogeological conditions in and around an ice sheet (Section 25.2.3); 2. geochemistry, hydrogeochemical conditions in connection with glaciation/permafrost (Section 25.2.10); 3. thermal, hydrological and mechanical processes (THM), links between mechanical and hydraulic processes in conjunction with glaciations and permafrost (Section 25.3.2); and 4. biosphere, the evolution of the biosphere in a periglacial environment (Section 26.7).

Table 19-1. Data and information obtained within the GAP.

Sub-project	Information/data
A	Ice movement velocity (GPS), presence of free water at the bottom of the ice sheet, ice depth, bottom topography (ground-penetrating and airborne radar), meltwater production (meteorological data), water flow rates (tracer tests), pressure/outflow/flow rates in supraglacial lakes, seismic observations, observations of supraglacial hydrology (remote analysis data)
B	Spatial and temporal variations in water pressure beneath the ice sheet (ice drilling), borehole images (ice drilling), ice movement velocity (GPS), subglacial topography (low-frequency radar), water flow rates (tracer tests), meteorological data
C	Bedrock data, geological-structural data, geochemical data including isotope data on deep and surface groundwater, hydrogeological data (pressure, temperature and conductivity data), micro-balances, temperature profiles in boreholes

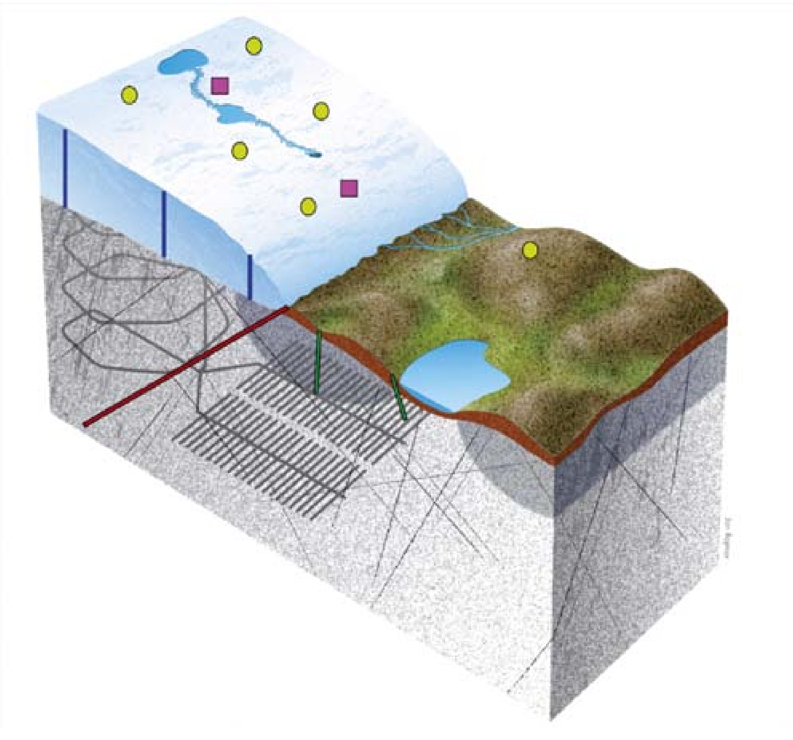


Figure 19-5. Schematic illustration of part of the field investigation area on Greenland. The shaded dark grey areas are permafrost. Beneath the proglacial lake is an unfrozen talik. A planned final repository in Forsmark is visualized beneath the ice sheet. Yellow circles are a network of continuous GPS stations and magenta-coloured squares are automatic weather stations for collection of data for sub-project A. The blue vertical lines are planned boreholes for sub-project B. Green lines show cored boreholes drilled to investigate taliks and the permafrost within sub-project C. The planned deep borehole is shown by the red line running at an angle beneath the ice.

20 Short-lived low- and intermediate-level waste

This chapter describes SKB's natural science research concerning the repository for short-lived low- and intermediate-level waste. Chapter 21 describes the research SKB plans to carry out in order to gain a better understanding of how the engineered barriers in the different repository parts in SFR are degraded during the post-closure period of 100,000 years. Chapters 25 and 26 deal with the research SKB is conducting to gain a better understanding of the geosphere and the biosphere. The research and development being pursued by the nuclear power plants (NPPs) regarding waste to be disposed of in SFR is presented in Part II (the LILW programme).

Conclusions in the review of RD&D 2007 and the supplement to RD&D 2007

Comments from the regulatory authority regarding general issues pertaining to the safety assessment of SFR are dealt with in Chapter 18.

In its review of RD&D Programme 2007, SKI finds SKB's efforts to determine the radionuclide inventory in SFR to be laudable. In particular SKB's studies of C-14, Ni-59 and Ni-63 provide important information. SKI also concurs in SKB's opinion that the use of correlation factors for certain nuclides should be replaced by other methods. SKI further considers it important that SKB come up with models for analyzing the impact of organic complexing agents on long-term safety.

In its review of the supplement to RD&D Programme 2007, SSM takes a positive view of the efforts still being made to gain better knowledge of the radionuclide content of operational waste from the NPPs, particularly regarding the important radionuclide C-14. SSM points out the importance of having SKB's work also include investigations of the C-14 content of waste delivered from the facilities in Studsvik.

SSM also takes a positive view of the fact that SKB has initiated a project aimed at finding out more about the degradation of cement and concrete over a long period of time, in view of the long time periods being considered for SKB's final repositories for low- and intermediate-level waste.

Safety assessment for the Extension Project

Since RD&D Programme 2007, the work with the safety assessment for the extension of SFR has been structured, permitting the research described in this RD&D programme to be presented in a similar manner to the research for the SR-Site safety assessment. This chapter therefore describes the research on the initial state of the waste, which is the starting point for the safety assessment, and the processes that are expected to affect the repository after closure. Since this is the first RD&D programme where the research on short-lived low- and intermediate-level waste is described in this manner, the focus is on description of variables and processes as well as programmes, rather than conclusions and newfound knowledge since RD&D Programme 2007.

20.1 Initial state

For a description of the origin of the waste and the distribution of radionuclides in the repository, see Part II (The LILW programme).

The initial state for SFR is defined as the state that exists in SFR at closure. In conjunction with closure, the repository's drainage pumping will cease and the repository will fill with water.

20.1.1 Variables

The initial state is the starting point for a safety assessment and is described by the initial values of a number of variables, see Table 20-1. The variables characterize the waste in a suitable manner for the safety assessment. The description applies not only to the waste itself, but also to the cavities that exist in the waste containers and between the waste containers and the engineered barriers into which water will penetrate. Processes such as advection and mixing will take place in the cavities. The variables are defined in Table 20-1 and are described in the following sections.

Table 20-1. Variables for the waste in SFR.

Variable	Definition	Section
Geometry	Geometric dimensions of all waste containers.	20.1.2
Radiation intensity	Intensity of alpha, beta, gamma and neutron radiation as a function of time and space in the waste.	20.1.3
Temperature	Temperature as a function of time and space in the waste.	20.1.4
Hydrovariables and hydrological boundary conditions	Water and gas pressures in the waste and the cavities in the waste containers, plus water flows from the surrounding area as a function of time and space.	20.1.5
Mechanical stresses	Mechanical stresses as a function of time and space in the waste packages.	20.1.6
Total radionuclide inventory	Total occurrence of radionuclides as a function of time and space in the different parts of the waste.	20.1.7
Chemotoxic inventory	Total inventory of chemotoxic substances, including contributions from the engineered barriers.	20.1.8
Material composition	The materials of which the different components of the waste are composed, excluding radionuclides.	20.1.9
Water composition and water saturation	Composition of water and water content (including any radionuclides and dissolved gases) in the waste and the cavities in the waste as a function of time and space.	20.1.10
Gas composition	Composition of gas (including any radionuclides) in the waste and in the cavities in the waste as a function of time and space.	20.1.11

20.1.2 Geometry

SFR contains waste of different types with different geometries. One geometry parameter besides the dimensions of the waste containers is the void volume, i.e. the volume of cavities in the waste containers. If the empty space between the waste containers is not backfilled, this will also affect advection and diffusion in the waste. The void volumes in the waste may also be of importance for advection and diffusion in the waste.

Programme

SKB will develop models for the influence of the void volume on hydrovariables. The influence of the void volumes on nuclide transport and long-term safety will be studied.

Backfilling will be studied with the aid of improved radionuclide transport models, see Section 21.3.

20.1.3 Radiation intensity

Radiation intensity is dependent the inventory of radionuclides and the geometry of the waste. The radiation intensity in the waste deposited in SFR is low and has been judged to have little influence on the evolution of the repository. Locally there may be waste packages with high activity levels.

Programme

Additional calculations may be needed due to a changed radionuclide inventory, for example if it is decided to dispose of reactor pressure vessels in SFR. If it turns out that decommissioning waste with high radiation intensity is to be disposed of, further calculations may be needed.

20.1.4 Temperature

The initial temperature in the repository is determined by the temperature of the surrounding rock. Initially there are no heat-generating processes in the waste. Corrosion of aluminium does not occur initially, since the aluminium is covered by a protective oxide layer that only dissolves when the repository is filled with water and at high pH /20-1/. The radiation intensity of the waste deposited in SFR is so low that temperature changes due to radioactive decay are negligible.

20.1.5 Hydrovariables and hydrological boundary conditions

The hydrovariables are water pressure and gas pressure. The water flows from the surrounding area comprise the boundary conditions.

The site of SFR was chosen in part because it is located in an area with a limited hydraulic gradient and limited fracture transmissivity. The placement of the repository was chosen in part for the purpose of ensuring low water flows through the different repository parts, which have been designed with different capabilities for limiting the water flow (see Section 21.1.7).

There are no water flows during deposition, since the repository is drained by pumping during the operating phase. When drainage pumping of the repository ceases, the tunnels will become water-filled and the water pressure in the surrounding rock will increase. The water pressure drives the flow through the repository parts and their surroundings. Water transport is described in Section 20.2.4.

20.1.6 Mechanical stresses

The waste containers are stacked in such a manner that they do not collapse under their own weight and/or due to overcasting. Expansion/contraction of the waste (Section 20.2.6), rock breakout (Section 21.2.8) and void volumes have an effect on mechanical stresses.

20.1.7 Total radionuclide inventory

At closure, the repository will contain radionuclides resulting from operation and decommissioning of the NPPs as well as smaller quantities from industry and research. SKB calculates and constantly monitors changes in the inventory to ensure that stipulated limits are not exceeded. Determination of the radionuclide inventory is done by means of measurements and calculations of annual generated activity at the NPPs and by correlation with key nuclides: Co-60, Cs-137 and Pu-239/240.

Conclusions in RD&D 2007 and the supplement to RD&D 2007 and their review

Both SSI and SKI, and later SSM, take a positive view of SKB's efforts to determine the radionuclide inventory in SFR.

Newfound knowledge since RD&D 2007

SKB's programme for better determining the radionuclide inventory in SFR has continued and additional efforts have been made to better assess the inventory of difficult-to-measure nuclides such as C-14. Some of the other difficult-to-measure nuclides – Cl-36, Ni-59, Ni-63, Mo-93, Tc-99, I-129 and Cs-135 – are now determined annually, based on operating conditions and sampling at the NPPs. We study and compile for each individual NPP the inventory that can be expected to arise and be deposited in SFR as a result of the decommissioning of the reactors.

Programme

Further measurements than those reported in /20-2/ are needed to obtain a more reliable body of statistics for deposited C-14 activity. In the expanded project, condensate filter demineralizer from all BWR units and operating ion exchange resins from all PWR units will be analyzed. This will provide additional information on deposited C-14 activity, in both organic and inorganic form, in SFR.

SKB's intention is to gain a better understanding of C-14, not just in the repository environment, but also in other parts of the chain from NPP to biosphere. Figure 20-1 shows the areas where SKB is conducting activities together with the NPPs.

Since the predominant future dose contribution in SAR-08 derives from organic C-14, we are pursuing work to gain a better understanding of how this organic carbon behaves in the chain from NPP to biosphere.

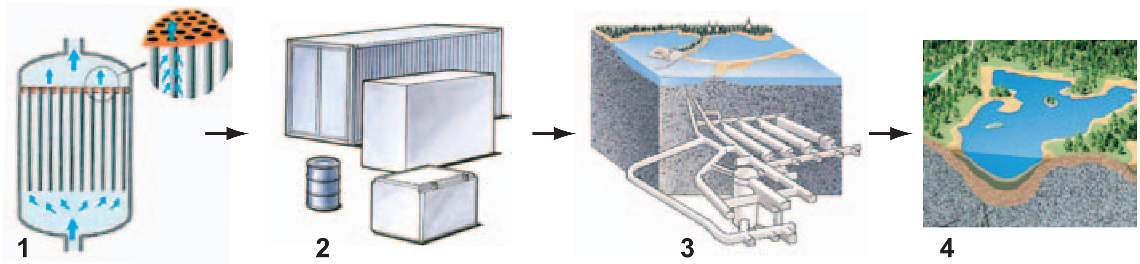


Figure 20-1. *Areas where SKB, together with the NPPs, is conducting activities to gain a better understanding of C-14. 1. Expanded analysis programme concerning uptake of C-14 on ion exchange resins. 2. Studies of actual deposited C-14 activity. 3. Retardation of C-14-containing species in the repository. 4. Consequences of C-14 release.*

In parallel with the measurement and sampling programme, work is being pursued to determine the structure of organic compounds with C-14 in ion exchange resins. There are indications in the literature that the majority of these compounds that are present in the water or the ion exchange resins may be simple organic acids, such as acetic and formic acid /20-3, 20-4/. The content may differ between PWRs and BWRs, so samples from both PWR and BWR reactors are being examined. This may lead to a better understanding of how organic compounds containing C-14 originate in the NPPs and how these compounds affect the long-term safety of SFR.

The purpose of the former and current measurement and sampling programmes is to find out how much C-14 activity is absorbed by the ion exchange resins in condensate cleanup and reactor water cleanup, respectively. Since the production rate of C-14 in the reactor water for each unit is known /20-3/, and knowing how much of the produced C-14 is absorbed on the ion exchange resins, the activity C-14 that is deposited can be calculated. This procedure can be somewhat misleading, since a certain amount of C-14 is expected to be lost during the conditioning of the waste, mainly at FKA, which dries its ion exchange resins before bitumen conditioning. As a part of its efforts to get a better idea of how much C-14 activity will really be deposited in SFR by FKA, SKB will, in cooperation with KTH and FKA, simulate the conditioning process in order to be able to measure the loss of C-14 in the process. There are indications that this loss is of importance at temperatures around 150°C. Furthermore, the effect of added evaporator concentrates during the drying of the ion exchange resins will be studied with respect to the loss of C-14.

Waste with ash belonging to AB SVAFO, sent to SFR from Studsvik, is only expected to contain a limited amount of C-14. Samples of 1,200 ash drums will be taken and analyzed with respect to their nuclide-specific activity. The results of these analyses should provide an indication of the C-14 content of existing waste of the type in SFR.

A waste type that has been studied previously is the 95 boxes of graphite from the shutdown R1 research reactor. This waste has not yet been deposited since it contains a relatively large amount of C-14, and it is not considered possible to dispose of this waste in the BTF repository in SFR.

The total amount of C-14 that has been shipped to and stored at Studsvik during the period 2002–2009 is less than 135 GBq. The work with NNW (non-nuclear waste) is expected to continue as before. Waste with C-14 that Studsvik Nuclear AB managed prior to 2002 has been incinerated. No study has been made of the amount of C-14 in the ash after incineration. The results from the above-described sampling programme should provide an indication whether C-14 is present in the ash.

Today the interim-storage at Studsvik holds solid and liquid waste with C-14. No conditioning has yet been done of the waste that has been received since 2002.

The uncertainty of the activity of the total radionuclide inventory, i.e. the activity of all radionuclides, is affected by the input data used to calculate the inventory and the models used in the calculations. Efforts are constantly being pursued together with the NPPs to reduce the uncertainties. The following points have thus far been identified as important in the continued study of uncertainties:

- The radionuclide activity in the waste is calculated for the majority of nuclides on the basis of measurements of, for example, a waste package's gamma spectrum. The calculations are often based on certain assumptions, such as the composition of the package. An inventory must be done of the uncertainties in the measurement methods. An analysis will be made of how the assumptions can affect the activity calculations.
- For short-lived key nuclides, the influence of the uncertainty of the production date on the calculation of the activity will be investigated.
- The activity of the majority of radionuclides is calculated from the activity of key nuclides using correlation factors. An inventory of the correlation factors will be done in order to estimate the uncertainty in the size of the correlation factors.
- The work of developing alternative methods to correlation will continue with a focus on the nuclides that are crucial for long-term safety.
- The uncertainty of existing methods will be evaluated, and the possibility of using new methods on historical measurement data will be examined.
- In certain cases the total amount of a certain nuclide contained in the waste can be determined. The methods for calculating the distribution of the activity between the repository parts will be further developed for these nuclides.
- Induced activity bound in certain materials, for example activity induced in reactor pressure vessels, is not available for transport out of a final repository until the material has been degraded. An inventory will be made of the availability of the radionuclides.
- There is waste, for example ion exchange resins, that has been collected in tanks over a long period of time. The uncertainty in production date for this waste is relatively great, which means that the uncertainty in the calculations of the activity of nuclides correlated to Co-60 can be relatively great. A study will be done of how best to calculate the nuclide content of this waste.

20.1.8 Chemotoxic inventory

Substances with chemotoxic properties are also deposited in SFR, for example small quantities of epoxy, lead and asbestos.

With the decommissioning of the nuclear power reactors, the quantity of substances such as asbestos, lead and hardened epoxy will increase in the repository. The quantity of chemotoxic waste from decommissioning will be estimated in the decommissioning studies that will be done, see Section 7.4.1.

20.1.9 Material composition

Much of the activity in SFR is in the wet waste. The wet waste consists for the most part of bead resin, powder resin, mechanical filter aids, evaporator concentrates and precipitation sludge. The ion exchange resins consist of organic polymers with acidic or basic groups, making them capable of cation or anion exchange. Non-radioactive materials are also captured in the ion exchange resins, examples being organic complexing agents such as citrate, gluconate and *N*,*N*-bis(carboxymethyl) glycine (NTA), and various cations such as Fe(II)/Fe(III), Ni(II) and Zn(II).

Much of the wet waste is conditioned, in other words solidified in cement or bitumen. The largest volume of raw waste consists of combustible solid waste. Due to incineration in Studsvik or local at-plant disposal, the volume remaining for disposal in SFR is comparatively small, however. The raw waste consists mainly of cellulose (paper, cotton and wood) and plastics (e.g. polystyrene, PVC, polyethylene and polypropylene).

A large portion of the waste volume in SFR consists of metals, above all carbon steel and stainless steel. Scrap metal arises mainly during maintenance outages when equipment is discarded, modified or renovated. Large metal components may be deposited in SFR in conjunction with the decommissioning of the reactors. In the case of large metal components, the corrosion rate is the limiting factor for metal dissolution.

Other materials occurring in the waste include mineral wool (used for insulation), concrete and brick. Various additional materials are also included in smaller quantities.

Newfound knowledge since RD&D 2007

Estimates have been made of the quantity of decommissioning waste that may eventually be deposited in SFR. The decommissioning waste consists primarily of metals and concrete, as concluded by the decommissioning study done for the Barsebäck NPP /20-5/.

A literature review has been done to try to determine the consequences of deposition of large quantities of oxalic acid derived from BKAB's decontamination of the two reactor systems. The conclusions drawn from this study are described in Section 20.2.10.

Programme

SKB has started a programme to quantify the amounts of non-radioactive cations deposited in SFR via ion exchange resins from condensate cleanup and reactor water cleanup at the NPPs. The purpose is to gain a better understanding of the future outbound transport of radionuclides with subsequent dose consequences.

In order to minimize the amount of organic complexing agents in SFR – which mainly enter SFR via cleaning agents that adhere to the ion exchange resins – we are investigating, together with the NPPs, the possibilities of developing a cleaning agent that does not contain any organic complexing agents, but is based entirely on inorganic complexing agents such as carbonate. This could ensure that organic complexing agents do not enter the repository in large quantities during continued operation and thereby minimize the risk of increased solubility of radionuclides. Carbonate is itself a good complexing agent, but the high concentrations of Ca(II) in the cement environment will dominate carbonate complexation.

20.1.10 Water composition and water saturation

The degree of water saturation of the waste is initially low, since the repository is drained by pumping during the operating phase. Water will enter the repository at closure, when drainage pumping ceases. The groundwater that enters the repository will be affected by the materials in the repository via various processes. The composition of the water will also be affected by the water flow through the repository, where waters of different compositions are mixed.

The groundwater that will flow into the repository after closure can be characterized as a saline groundwater. The reference composition of the penetrating saline groundwater after saturation of the repository is given in Table 20-2 /20-6/. The basis for the reference composition of the groundwater is measurements made during the construction of SFR 1 (1984–1986), data from the monitoring programme for SFR 1 (1989–1999) and geochemical calculations /20-7/. In the geochemical calculations, the water composition was adjusted so that the water is in thermodynamic equilibrium with commonly occurring minerals in the rock. In calculating concrete degradation, the content of calcium, magnesium and silicon was adjusted so that the water would be in equilibrium with calcite, dolomite and quartz /20-6/.

A hydrogeological model has been used to calculate how long it takes to fill and saturate the repository with groundwater /20-8/. The calculations show that the void (porosity) inside the silo repository is saturated last and that this can take 25 years. It only takes a few years to fully saturate BMA, BLA and BTF.

20.1.11 Gas composition

The initial gas composition will be controlled by the microbial activity attributable to SFR during the operating phase. Full-scale experiments in Olkiluoto (Finland) have shown that anaerobic conditions are created relatively quickly, and degradation of organic material (cellulose) leads to appreciable levels of e.g. methane and carbon dioxide /20-9/.The formed gas will then diffuse out of the waste and be expelled from the repository via the ventilation system.

When the repository has become water-filled, the processes begin that can lead to further gas formation, for example corrosion (Section 20.2.12), microbial activity (Section 20.2.16) and radiolytic degradation (Section 20.2.17).

Table 20-2. Composition of penetrating saline groundwater after saturation of the repository. Based on measurement data from SFR and geochemical calculations (ion concentrations in [mg/l]) /20-6/.

Parameter	Reference value	Min. value	Max. value
Redox [mV]	Reducing	–100	–400
pH	7.3	6.5	7.8
SO_4^{2-}	500	20	600
Cl^-	5,000	3,000	6,000
Na^+	2,500	1,000	2,600
K^+	20	6	30
Ca^{2+}	430	200	1,600
Mg^{2+}	270	100	300
HCO_3^- (alk)	100	40	110
Si as SiO_2(aq)	5.66	–	–
Charge balance in %	–0.04%		

20.2 Processes

A number of processes will eventually alter the state of the waste and in its cavities. Some take place under all conditions, while others are only possible if water has penetrated into the waste or anaerobic conditions prevail.

20.2.1 Overview of processes

The processes that affect conditions in the waste can be divided into four main classes: thermal, hydraulic, mechanical and chemical. Under each main class, a number of different processes can occur and they can interact with each other and affect or be affected by other processes that occur in the waste.

Thermal processes

There are few heat-generating processes in SFR. Corrosion of aluminium could be a possible exception, but has been shown to be of no importance /20-1/. Heat transport can essentially be expected to take place via heat conduction, which is governed by the thermal conductivity and heat capacity of the materials. The temperature of the repository, and thereby the waste, will be determined almost entirely by heat exchange with the surrounding rock and groundwater. The influence of the waste on the temperature is negligible. The influence of the temperature on waste containers of concrete and cement-embedded waste is not negligible, and freezing alters the integrity of the concrete /20-10/. Freezing is the process that is judged to affect the thermal evolution of the waste, see Section 20.2.2.

Hydraulic processes

The water flow through the repository and the waste is determined by the water permeability of different structural parts and the components in the repository, as well as by the hydraulic gradient. The water flows through the different repository parts are so low (two centimetres per year) /20-8/ that erosion of the waste containers will be negligible compared with chemical degradation. If gas is present at the same time, this gives rise to a two-phase flow, where both the water flow and the gas flow are affected by the relative degree of saturation of each phase. Two-phase flow can be of importance in the analysis of gas flows, but can be neglected in the analysis of post-closure repository saturation, since saturation takes place very rapidly /20-8/. High gas pressures caused by entrapped gas can give rise to a locally elevated water pressure and therefore be a driving force for the water flow out of these enclosures. Concentration gradients can also cause water flow via osmosis, but the process is of no importance except for the description of the degradation of bitumen. The magnitude of the water flow in the repository is determined to a high degree by the surrounding groundwater flow. The following hydraulic processes are dealt with:

- Water uptake in ion exchange resins and waste matrix, see Section 20.2.3.
- Water transport, see Section 20.2.4.
- Two-phase flow/gas transport, see Section 20.2.5.

Mechanical processes

The waste in the different repository parts will be subjected to external mechanical forces and internal volume changes. This will affect the stress distribution in the waste, which can in turn lead to cracking. If gas is generated that cannot escape, this can lead to considerable pressure and stress build-up. Finally, the waste will be affected by any deformations in the rock (falling blocks, rock movements, earthquakes etc.). The following mechanical processes are dealt with:

- Expansion/contraction of the waste, see Section 20.2.6.
- Fracturing, see Section 20.2.7.
- Rock breakout, see Section 20.2.8.

Chemical processes

The properties of different waste forms and waste containers are affected by numerous chemical processes such as recrystallization, water uptake, chemical and microbial degradation, corrosion of metals, dissolution/precipitation and formation of different corrosion products, leading to gas evolution. Water composition changes as a result of advection and mixing. Concentration differences are equalized via diffusion. Sorption of radionuclides is affected mainly by the water composition in the repository. The concentration of substances occurring in small quantities, such as complexing agents, can have a great influence on the sorption of cations dissolved in the water. The microbial activity in the repository is determined primarily by the availability of organic material, which is affected by the groundwater flow /20-11/. The following chemical processes are dealt with:

- Dissolution/precipitation, see Section 20.2.9.
- Degradation of organic compounds, see Section 20.2.10.
- Speciation, see Section 20.2.11.
- Corrosion, see Section 20.2.12.
- Diffusion, see Section 20.2.13.
- Advection and mixing, see Section 20.2.14.
- Colloid formation and transport, see Section 20.2.15.
- Microbial activity, see Section 20.2.16.
- Radiolytic degradation, see Section 20.2.17.

The research programme for the various processes in the waste is dealt with in the following sections. The interaction of the processes with each other is not discussed here, but will be described in the process report that will be submitted as a background report to the safety assessment for the Extension Project.

20.2.2 Freezing

Concrete waste containers and cement-embedded waste are considered to behave in the same way on freezing as the engineered barriers. This process is described in Section 21.2.3.

20.2.3 Water uptake in ion exchange resins and waste matrix

When the waste, especially ion exchange resins, takes up water it increases in volume and the entire waste matrix swells. With increasing water content, the processes that lead to a change in water composition are initiated. Increased water content enables the processes that lead to transport of nuclides from the repository.

Water uptake in a bitumen matrix takes place by diffusion in towards ion exchange resins and salt in evaporator concentrates, which are hygroscopic. The consequences of this could be to open up an interconnected porosity in the bitumen matrix and cracking of the matrix due to the stresses caused by the swelling. How rapidly water is absorbed depends largely on the fraction of ion exchange resins in the matrix. The low fraction of ion exchange resins in bitumen in SFR suggests slow water uptake. However, in the case of a few waste types with a higher fraction of ion exchange resins, water uptake could conceivably proceed relatively rapidly /20-12/.

The swelling of the bitumen matrix that takes place in connection with water uptake can also affect surrounding barriers if insufficient expansion volume is available. A calculation of the theoretically maximum swelling indicates that the volume increase could be greater than the void volume inside the waste package in the case of certain waste types in the silo repository and BMA /20-12/. In BMA, however, there are additional void volumes outside the waste packages if the packages are not grouted.

The main degradation of bitumen matrices that leads to release of radionuclides in the waste is expected to be water uptake and swelling. The time for water uptake and how this affects the matrix is very uncertain, however. Some indication of how effective a bitumen matrix is as a barrier for radionuclides can be obtained from leaching experiments. Extrapolation of results from such leaching experiments conducted over periods that are short in these contexts indicates that it could take several thousand years before all radionuclides have leached out of the bitumen matrix in a 200-litre steel drum /20-12/. The long-term material changes caused by water uptake and other processes will, however, probably permit the release of radionuclides from the bitumen matrix faster than what the leaching experiments indicate. A more reasonable time scale for release of radionuclides from bitumen-solidified waste is of the order of several hundred up to a thousand years /20-12/.

Water uptake in cement matrices is expected to proceed more rapidly than in bitumen matrices, and the cement in the cement-solidified waste in the silo is expected to be water-saturated within 25 years of closure /20-13, 20-8/, see Section 20.1.10. The changes that take place in the waste's concrete matrix are described in Section 21.2.10.

Water uptake for waste matrices in concrete moulds is controlled by the permeability of the concrete to water, which is dependent on the properties of the concrete and hydraulic pressure gradients.

20.2.4 Water transport

The water pressure gradient in the waste gives rise to pressure-induced flow. In general, the waste packages cannot absorb all water that enters the repository, leading to a build-up of water pressure. The water pressure drives the flow through the waste. The size of the flow is determined by the geometry and conductivity of the waste packages and the backfill material.

The size, direction and distribution of the water flow in different parts of the repository system and the geosphere affect radionuclide transport. Transport of other species, microorganisms and bacteria is also controlled by the water flow. Rock breakout and backfilling influence the water flow.

Programme

Hydrogeological calculations are planned to quantify expected water flows in the different parts of the repository and to analyze the water flow from the repository to the surface ecosystem. The programme for the latter is described in Section 25.2.3 and Section 26.6. The programme for modelling of water flows through the repository is described in Section 21.3.

20.2.5 Two-phase flow/gas transport

See Section 21.2.5.

20.2.6 Expansion/contraction of the waste

Previous analyses of the conditions in SFR have shown that the amount of sulphate that is obtained on complete degradation of all ion exchange resins in the silo repository and the volume increase that results when the sulphate reacts with cement and concrete can be taken up by the available expansion volume /20-14/. Volume increase of waste matrices and waste packages due to water uptake in ion exchange resins conditioned in bitumen and formation of expansive corrosion products can induce stresses in surrounding concrete barriers unless sufficient void volumes are available to take up this volume increase, see Section 20.2.3.

20.2.7 Fracturing

Carbonatization – caused by reactions with carbon dioxide and carbonate formed by decomposition of organic matter in waste – can clog pores in the concrete, which can lead to fracturing. This primarily affects cement and concrete in the waste packages if the waste is solidified with cement and/or packed in concrete moulds. Microbial processes can also lead to fracturing, see Section 20.2.16. The quantity of degradable organic matter in the silo repository is relatively small and the conditions in other respects are not particularly favourable for sustaining microbial activity in either the silo repository or the rock vaults that have concrete barriers /20-11/.

Corrosion products (see Section 20.2.12) have a larger volume than the original iron, which means they can exert a pressure on the surrounding concrete, which can give rise to fracturing /20-14/.

Expansion and contraction of the waste can also give rise to fracturing. See also Section 21.2.7.

Programme

Studies are under way on the effects of carbonate on concrete in SFR. The programme includes thermodynamic modelling of the long-term function of the barriers at elevated carbonate concentrations and assessment of the consequences of co-precipitation for relevant radionuclides.

A project for the purpose of studying the gas permeability of concrete and cement, which has a bearing on fracturing, is described in Section 6.3.4.

20.2.8 Breakout

In the repository parts that lack engineered barriers (BLA), rock breakout can directly affect the waste. In the repository parts that are surrounded by engineered barriers, breakout has no direct effect on the waste.

Programme

Regarding the research planned by SKB on rock breakout, see Section 21.2.8.

20.2.9 Dissolution/precipitation

Dissolution of the waste releases ions from ion exchange resins and mobilizes nuclides, which become available for transport. Processes such as water transport can alter the water composition, and speciation will change when chemical equilibria are established. This can lead to precipitation and immobilization of substances dissolved in the water. Dissolution of salt from evaporator concentrates releases chlorides, carbonates and sulphates, which in turn react with surrounding concrete barriers and ion exchange resins. How rapidly dissolution takes place depends on how the waste is conditioned and packaged, see also Section 20.2.3. It was assumed in the SAR-08 safety assessment that all radionuclides from cement-embedded waste are available for transport when the repository has been filled with water.

Water uptake in the bitumen matrix is slow, which means that dissolution and release of dissolved salts takes place over a relatively long period of time, see Section 20.2.3. This suggests that the impact on surrounding concrete should be small. However, the possibility cannot be ruled out that with the passage of time (above all after the first 1,000 years), these dissolved salts could form such high concentrations locally that nearby concrete could be affected, leading to porosity changes and possibly also fracturing. A more detailed analysis is required to test this hypothesis.

During the operation of SFR, corrosion products such as iron oxides and iron hydroxides are formed, which can sorb and possibly co-precipitate many elements. This suggests that precipitation of corrosion products is a process that could affect radionuclide transport in the repository.

Programme

The programme that has been initiated to gain a better understanding of this process is described in Sections 21.2.9 and 21.2.10. Further studies may be done to determine how rapidly radionuclides are released from waste conditioned in different matrices.

20.2.10 Degradation of organic compounds

Chemical degradation of organic compounds and materials in the waste or its matrix can generate products that affect the repository's long-term safety. Formation of products with complexing ability can under certain circumstances influence sorption and thereby radionuclide transport. Different radionuclides will be affected to different extents. Whether or not radionuclides form soluble complexes with organic complexing agents depends to a high degree on the oxidation number of the radionuclide and whether the resultant metal-organic complex is stronger than the hydroxide complex that is formed in the absence of the organic complexing agent.

An important mechanism for degradation of organic compounds and materials in the waste is hydrolysis at the high pHs that are generated in the cement pore water.

A number of organic components have been investigated with respect to alkaline degradation. The influence of the degradation products on the sorption and solubility of a number of radionuclides has been investigated /20-15/.

Most of these substances are not relevant to the long-term safety of SFR, while others may be of importance /20-16, 20-17/. The greatest influence is that of isosaccharinic acid (ISA), which occurs in two diastereomeric forms: α and β /20-18/, se Figur 20-2. ISA is produced by alkaline hydrolysis of cellulose and forms polydental complexes at high pHs when the hydroxyl group is deprotonated and bonds to radionuclides. Strong complexes are formed with tri- and tetravalent radionuclides /20-19/.

Degradation of ion exchange resins can give products with complexing properties. Alkaline hydrolysis of anion exchange resins can give rise to amines with complexing properties /20-16/. Degradation of cation exchange resins gives oxalate as the main degradation product. According to a number of studies, these degradation products from ion exchange resins have no significant impact on the sorption of radionuclides on concrete materials /20-17, 20-16, 20-20/. Sulphate and oxalate can be formed by the radiolytic degradation of ion exchange resins /20-21/. Sulphate is expected to form ettringite (a hydrated Ca-Al sulphate) on reaction with cement and concrete /20-14/. Oxalate affects the sorption of metals under neutral to acid conditions. Under the conditions prevailing in a concrete environment, the effect is limited due in part to the fact that the hydroxide ions compete with oxalate regarding complexation and in part to the fact that oxalate is precipitated as calcium oxalate.

Radiolytic degradation of bitumen at high pHs has been shown to result primarily in mono- and dicarboxylates and carbonates /20-22/. Of these, oxalate could be a potential complexing agent, but it is deemed to be negligible in a concrete environment, since its calcium salt is insoluble and will be precipitated. Furthermore, the radiation is deemed to be too low, except in the most radioactive packages, to give any radiolytic degradation of bitumen /20-12/.

Newfound knowledge since RD&D 2007

Refined rate constants for degradation of cellulose have been published since RD&D Programme 2007 /20-23/.

A study regarding the degradation of oxalic acid under repository conditions has shown that carboxylic acids with electron-attracting groups on the α carbon atom undergo decarboxylation under very mild conditions. Malonic acid and acetoacetate are examples of this, see Figure 20-3.

The first step in the reaction entails transfer of a proton and simultaneous cleavage of a carbon-carbon bond. The resultant enol then tautomerizes to the corresponding keto form (acetic acid), which in this case is the most stable of the two tautomers. The formation of the enol form is essential in order for the reaction to take place under mild conditions. Such a tautomer cannot be formed in connection with decarboxylation of oxalic acid. Decarboxylation of oxalic acid therefore requires more forced conditions /20-24/.

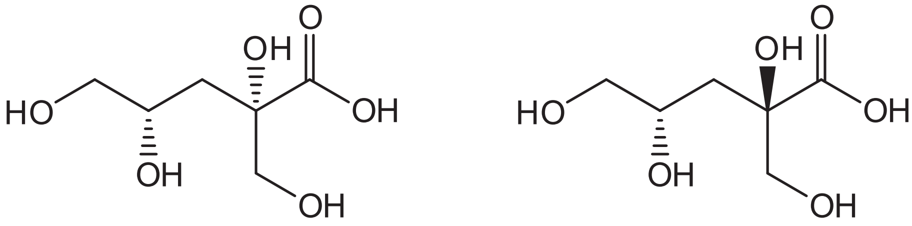


Figure 20-2. *α-isosaccharinic acid at left and β-isosaccharinic acid at right.*

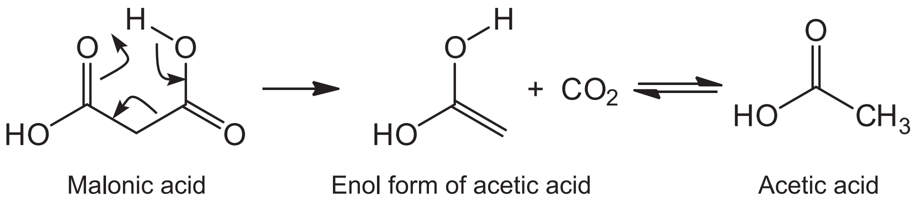


Figure 20-3. *Decarboxylation of malonic acid to acetic acid and* Co_2*.*

Carboxylic acids, such as malonic acid, are decarboxylated approximately 50 times faster under acidic conditions than under the basic conditions that will exist in the repository /20-24/. Both carboxyl groups will thus be deprotonated under the expected future conditions in SFR.

The solubility product for calcium oxalate is approximately 2×10^{-9} /20-25/. This means that at a concentration of 0.04 molar (M) oxalic acid, less than 0.1 percent will exist in solution and the rest as precipitated calcium oxalate, provided that the Ca(II) concentration is of the same order of magnitude as the concentration of oxalic acid. Oxalic acid, and to a greater degree oxalates, are very stable compounds. SKB's assessment is that they will be degraded at a very slow rate. The degradation that nevertheless takes place should give rise to formic acid and carbon dioxide. This means that the one carboxyl group in oxalic acid is oxidized from +III to +IV in carbon dioxide, and the other carboxyl group is reduced from +III to +II in formic acid. The reaction rate is difficult to estimate under the given conditions, however.

Programme

SKB has initiated a programme at Chalmers to study the degradation of cellulose to the two diastereomeric forms α- and β-isosaccharinic acid and their potential differences in complexing ability and solubility under the conditions expected to prevail in SFR. Moreover, further studies may possibly be needed concerning the degradation of the filter aid UP2 (polyacrylonitrile-based polymer) and the potential complexing ability of its degradation products, see Section 21.2.12.

20.2.11 Speciation

The speciation of the radionuclides that are present in SFR is dependent on the pH and the oxidation number of each radionuclide. The pH in the repository changes over time, whereby the speciation of certain pH-sensitive radionuclides will change and their sorption capacity will be affected.

The oxygen that is present in the repository at closure will quickly be consumed by corrosion of steel in containers and rebar (reinforcing steel bar), oxidation of dissolved iron(II) and sulphide in the water, and microbial processes. A low redox potential will be maintained in the different repository parts due to the release of Fe(II) ions in connection with anaerobic corrosion of steel and from corrosion products. When all iron has been corroded, the redox potential will be dominated by the composition of the groundwater and reducing conditions will be maintained.

Organic complexing agents may influence speciation and sorption for certain radionuclides, which affects radionuclide transport out of the repository. Since SFR is located relatively close to the surface, the possibility cannot be excluded that oxidizing conditions will arise in conjunction with a colder future climate, for example due to penetration of meltwater from permafrost or an ice sheet into the repository. If oxidizing conditions exist, redox-sensitive substances will oxidize. A change in the oxidation number of certain radionuclides can cause a change in their speciation, the complexes they form with organic complexing agents and sorbing capacity, which in turn affects radionuclide transport and water composition.

Programme

If modelling shows that uncertainties in input data have a great impact on the results of the modelling of the system, experimental studies may be called for to reduce the uncertainties.

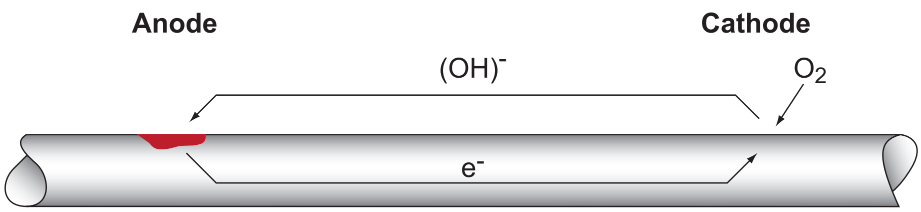
As a part of SKB's attempts to constantly improve and increase our understanding of the consequences of radionuclide releases from SFR, modelling is planned of the specific activity of the organic C-14 that migrates out of the repository. To start with, a model will be developed where the degradation of cellulose to isosaccharinic acid (ISA) affects the specific activity of organic C-14. In extension, this can lead to the development of a refined biosphere model for C-14. Potentially, this can result in less conservative assumptions regarding the future dose contribution from organic C-14.

20.2.12 Corrosion

As long as oxygen is present, i.e. during the operating phase and a short time after closure, corrosion of metals will be dominated by aerobic corrosion processes.

Rebar embedded in concrete exists in an environment with a pH > 12.5 due to the high alkalinity that exists in the concrete. In this environment, the rebar is put into a passive state and corrosion does not normally occur. But the passive state can be broken down, either by carbonatization of the cover layer due to penetration of carbon dioxide, or by penetration of chloride ions to the rebar. When the passive state is broken down, there is a high probability of corrosion in the structure. This corrosion of rebar limits the life of the structure. Rebar corrosion can be compared to a galvanic cell with an anode and and a cathode, where the anode is the negative terminal from which electrons migrate towards the positive terminal, i.e. the cathode. In order for the electric circuit to be closed, the pore water solution, which contains ions, acts as the galvanic cell's electrolyte. Corrosion occurs at the anode in the form of local pitting on the rebar when the iron oxidizes and forms iron oxide. See Figure 20-4 for a simple schematic drawing of the corrosion process.

When the oxygen in the repository has been consumed, reducing conditions will prevail in the repository and corrosion will be dominated by anaerobic corrosion processes. Anaerobic corrosion of metals in the waste, in waste containers and in rebar in concrete moulds and structures generates hydrogen. Anaerobic metal corrosion is thereby the process that is expected to contribute the largest quantities of gas. Factors that affect corrosion are the presence of water and the water chemistry, mainly pH, Eh and the concentration of dissolved salts.



Cathode: $3O_2 + 6H_2O + 12e^- \longrightarrow 12OH^-$

Anode: $4Fe \longrightarrow 4Fe^{2+} + 8e^-$

$4Fe^{2+} \longrightarrow 4Fe^{3+} + 4e^-$

Total: $4Fe + 3O_2 + 6H_2O \longrightarrow 4FeO(OH) + 4H_2O$

Figure 20-4. *Schematic illustration of aerobic rebar corrosion.*

A compilation of corrosion rates for anaerobic conditions similar to those in SFR shows corrosion rates for iron and steel in the interval 0.1–10 microns per year (µm/year) /20-1/. In the gas formation calculations, a corrosion rate for iron and steel of one µm/year has been assumed, corresponding to a hydrogen gas production of approximately three litres per square metre and year (l/m^2,year) and complete corrosion of a five millimetre plate in 2,500 years. The corrosion rate for aluminium and zinc has been assumed to be one millimetre per year, which is equivalent to corrosion of all material within a few years. A status report published by the EU on gas migration and two-phase flow in a deep repository provides an overview of studies and experiments published within the area of gas generation prior to January 1999 /20-26/. The compiled gas formation rates for different materials and environments are in accordance with the assumed corrosion rates.

Newfound knowledge since RD&D 2007

A corrosion rate of one µm/year is used for iron and steel in the safety assessment for SFR. This rate is based on data presented in /20-1, 20-27/.

Induced activity bound to corroding materials is released and becomes available for transport as the material corrodes. Previously SKB has conservatively assumed that all radioactivity is available for transport out of the repository when the repository is water-filled. This assumption can be considered to be realistic for radioactivity bound to ion exchange resins, or in oxide layers on metallic waste, but not for induced activity in various metallic structures, such as reactor pressure vessels, which could conceivably be deposited in a future SFR.

Reactor vessels contain two long-lived nuclides, C-14 and Mo-93, which could eventually constitute a significant fraction of the dose contribution from SFR. These nuclides are made available by corrosion. This means that C-14 and Mo-93 will be released gradually over a period of around 70,000 years (the estimated time it takes for the reactor pressure vessels to corrode assuming a corrosion rate of 1 µm/year /20-1/). Most of the Mo-93 will therefore have decayed before it becomes available, since its half-life is short compared with the time it takes for Mo-93 to be released. The SAR-08 safety assessment shows that C-14 will also contribute a maximum dose during the period 3,000–5,000 years after repository closure, which is the period when the discharge area consists of a lake. At the end of this period, only 5/70ths of the reactor pressure vessel will have rusted apart. This means that only about 5/70ths of the induced activity will be available at this time.

Programme

Since SFR contains large quantities of iron in waste containers and rebar, and since large metal components may be disposed of in SFR, it is important to make a more accurate estimate of the aerobic and anaerobic corrosion rate in an SFR-like environment. This can be done by renewed literature studies. If there is not enough data in the scientific literature, it may be necessary to conduct experimental studies. One such study has been initiated, see Section 6.3.4 in Part II.

20.2.13 Diffusion

Diffusion of solutes takes place in immobile pore water. The process affects the composition of the pore water. Diffusion is strongly coupled to nearly all chemical processes in the waste, since it accounts for transport of reactants to and reaction products from the processes. Diffusion is thereby a central process for the chemical evolution of the waste and radionuclide transport in the waste. In porous media, the diffusion rate is affected by sorption, see Section 21.2.12.

Programme

SKB is following developments within the discipline.

20.2.14 Advection and mixing

It is assumed that transport of solutes through the waste can take place by means of both advection and diffusion. Initially, diffusive transport is assumed to be dominant in the concrete-solidified waste, while advective transport is assumed to be negligible. When the concrete degrades, fractures will occur and advection will increase.

On a microscopic scale, mixing takes place by dispersion. Dispersion has an impact on the transport of radionuclides and other solutes, see Section 21.3.

Advection also takes place in void volumes and in spaces between waste packages that have not been backfilled. The process is handled within modelling of radionuclide transport, see Section 21.3.

Due to the absence of flow paths from BLA to the other rock vaults, any transport of ISA or other dissolved organic matter from BLA to the other repository parts must take place by diffusion through the intervening rock.

Programme

SKB is following developments within the discipline.

20.2.15 Colloid formation and transport

The quantity and type of colloids and particles in the water can affect the mobility of the radionuclides by acting as carriers. The occurrence of colloids in the water is dependent on the content of dissolved salts in the water and above all the concentration of positive ions, which destabilize the colloids. The colloid content is negligibly small in groundwater where the concentration of calcium ions is greater than 10^{-3} M (40 mg/l) /20-28/. Leaching of concrete also contributes to dissolved salts, which further hinders the formation of colloids in those repository parts that have concrete barriers /20-29/. This assumption is supported by the analyses that have been done of the groundwater in Maqarin in Jordan /20-30/. The groundwater in Maqarin has a pH of about 12.5 and a composition that otherwise resembles leachate from concrete. The colloid concentrations in this water are very low. Experimental results support the conclusion that at the low colloid concentrations that are obtained in alkaline cement pore water, sorption is negligible, even for strongly sorbing nuclides /20-31/.

Programme

SKB is following developments within the discipline. See also Sections 24.2.20, 24.2.28, 25.2.18, 25.2.19.

20.2.16 Microbial activity

The organic matter in the waste in SFR, especially cellulose, is a possible source of energy and nutrients for microorganisms. Microorganisms can degrade the organic matter to various components, which can create problems of varying scope. The microbial activity is dependent on the water flow through the repository, since this relates to the transport of nutrients to microorganisms as well as the removal of metabolites, which can otherwise poison the microorganisms. Since the water flow through the repository is low, the microbial activity decreases, and microbial effects on SFR are judged to be small /20-11/. Furthermore, the environment in SFR is special (for example high pH) and in no way optimal for microbial life.

Certain microorganisms can, especially in the absence of oxygen, form acids which could affect the cement and concrete structures in SFR. However, except during a short initial period after closure (and possibly in conjunction with a future ice age), conditions in SFR will be oxygen-free, and acid formation is therefore considered negligible.

Sulphate-reducing bacteria form sulphide, which has a corrosive effect on metals. Under special conditions, with local water flows near metal surfaces and in the presence of organic compounds from the repository, some pitting can occur.

Biodegradation of bitumen in SFR is not expected to significantly affect the bitumen matrix. The reason for this is that this type of degradation is generally a very slow process /20-12/ and that the environment in SFR is unfavourable for microbial activity /20-11/.

Microbial degradation of organic matter often leads to the formation of simple organic compounds (such as acetate and simple alcohols) or inorganic compounds (such as carbon dioxide/bicarbonate/carbonate) as end products. Some of these substances may have complexing properties. Microorganisms can also use organic compounds as an energy source and reduce the quantity of organic complexing agents. In anaerobic degradation, different inorganic sources such as nitrate and sulphate are used as oxidants. Water, nutrients and certain trace elements are required in order for degradation to take place. The end products of anaerobic degradation are hydrogen, methane and carbon dioxide. The chemical environment (pH and ionic strength) and other factors such as temperature, pressure and radiation level are of importance for the rate of microbial degradation. Of the organic materials found in the repository such as ion exchange resins, bitumen and cellulose, cellulose is the material that is expected to have the highest degradation rate.

Microbial degradation of organic materials under conditions expected to prevail in SFR after closure has been investigated /20-11/. Experiments cited there indicate that gas formation is initially fast but decreases after the initial phase. According to the study, the environment in SFR is not optimal for microbial degradation, but even as high a pH as 12 is no obstacle to microbial activity. Gas formation due to microbial activity in SFR is limited by the supply of oxidants and nutrients and the removal of reaction products. Cellulose degradation rates corresponding to complete consumption in less than 200 years have been assumed in the calculations of gas formation by microbial degradation. This is equivalent to a degradation rate of 0.2 mol per kg and year (mol/kg·year) and a gas formation rate of about two litres per kg and year (l/kg·year), assuming that 50% of the gases are inert /20-1/.

The few experiments that have been done on microbial degradation of bitumen, ion exchange resins and plastics indicate that these processes are very slow. For the sake of the safety assessment calculations it has been assumed that 0.002 mol/kg·year is degraded, corresponding to a degradation of all material in 15,000 years and a gas formation rate of 0.02 l/kg·year, assuming that 50 percent of the gases are inert /20-1/.

The question of which microorganisms are present in the groundwater, and what the chances are that they will be active in the environment prevailing in SFR after closure, has not been fully clarified but is not deemed at present to require any further study.

Programme

SKB is following developments within the discipline. See also Section 25.2.16.

20.2.17 Radiolytic degradation

When chemical substances are exposed to radioactive irradiation, chemical bonds can be broken and new substances can be created. Some production of hydrogen due to hydrolysis around waste fractions with high activity cannot be ruled out. This process has been deemed to be of little importance for the evolution of the waste.

20.3 Modelling – radionuclide transport

Modelling of radionuclide transport in both low- and intermediate-level waste and the engineered barriers in SFR is discussed in Section 21.3.

21 Engineered barriers in SFR

This chapter describes the natural science research which SKB plans to carry out to gain a better understanding of how the function of the engineered barriers changes during the post-closure period (100,000 years). Chapters 25 and 26 describe the research SKB is conducting to gain a better understanding of the geosphere and the biosphere.

Safety assessment for the Extension Project

Since RD&D Programme 2007, the work with the safety assessment for the Extension Project has been structured, so that the research described in this RD&D programme can be presented in a similar manner to the research for the safety assessment for the Spent Fuel Repository, SR-Site. This chapter describes the research on the initial state of the engineered barriers in SFR, and the processes that are expected to affect the repository after closure. Since this is the first RD&D programme where the research on the engineered barriers for short-lived low- and intermediate-level waste is described in this manner, the focus is on description of variables and processes as well as programmes, rather than conclusions and newfound knowledge since RD&D Programme 2007.

21.1 Initial state of engineered barriers

The purpose of the engineered barriers in certain repository parts in SFR is to retard the transport of radionuclides out of the repository, see Figure 21-1. Engineered barriers are present in the BTF, BMA and silo repositories. The silo repository is surrounded by two different engineered barriers: concrete and bentonite. The bottom bed at the silo consists of a mixture of 90 percent sand and 10 percent bentonite.

The shotcrete covering the rock around the rock caverns and the concrete floor slabs are counted (in SAR-08) as engineered barriers in all disposal chambers except BLA.

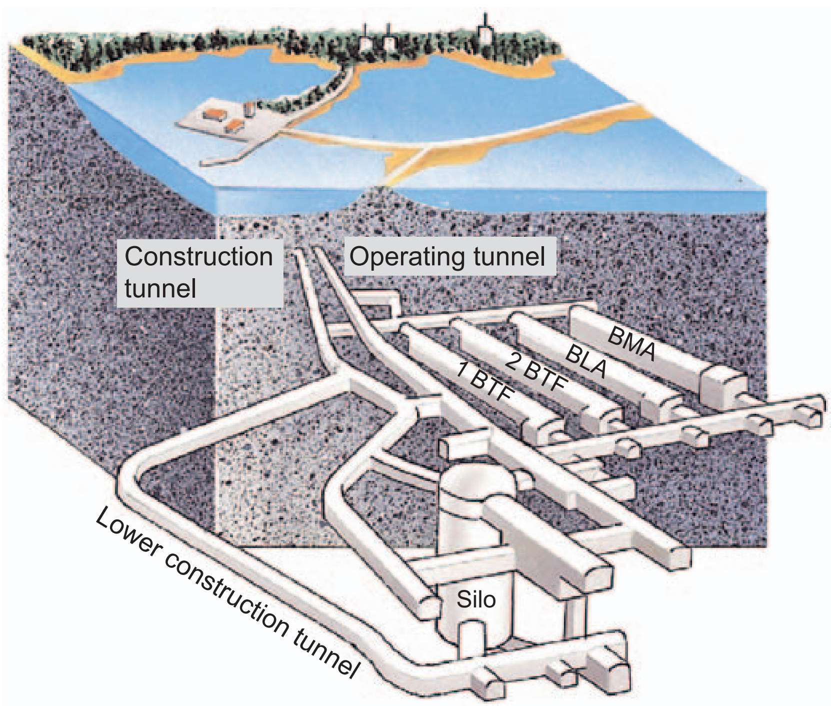


Figure 21-1. *Location of the different repository parts in SFR.*

The engineered barriers in 1- and 2-BTF consist of the concrete in the concrete tanks, the concrete floor slab and the overlay on the waste packages deposited in 1-BTF.

The research programme for the initial state in the concrete engineered barriers and the cavity that can arise between the waste and the concrete barriers is described in the following section. The research and development that concerns the bentonite barrier in the silo is described in Chapter 24. The point in time for the initial state of the engineered barriers is defined in the same way as for the short-lived low- and intermediate-level waste, see Chapter 20.

The programme also includes continuous monitoring of whether the initial state is affected by the maintenance of SFR that is included in SKB's programme, and if so how, see Section 5.2.2.

21.1.1 Variables

The initial state is the starting point for a safety assessment and is described by the initial values of a number of variables, see Table 21-1. The variables characterize the barriers in a suitable manner for the safety assessment.

21.1.2 Geometry

The geometry of the engineered barriers is determined based on the repository's configuration and the outside dimensions specified in the type descriptions of the concrete tanks credited as engineered barriers.

The configuration is dependent on the construction of the repository, any measures for maintenance during operation, and what measures are adopted at closure.

The stability of the rock nearest the rock vaults when rock support elements such as rock bolts and shotcrete no longer have any loadbearing strength is dependent on whether voids in the rock vaults have been backfilled, and if so how.

How the stability of waste, waste packages, and concrete and bentonite barriers evolves depends on a combination of mechanical and chemical processes.

The service life of rock support elements such as rock bolts and shotcrete is crucial for the long-term stability of SFR. As long as these elements retain their loadbearing strength, the changes in stress and deformations in the rock mass around the rock vaults are expected to be small. Rock support elements of the type used in the construction of SFR are normally expected to have a service life of 100–120 years, but it will probably take around 200–250 years for them to lose all their loadbearing capacity.

When the loadbearing capacity of the rock bolts in particular has declined, rock blocks can be expected to loosen and fall into the rock vaults. This process is described in Section 21.2.8.

Table 21-1. Variables for the concrete and cement barriers.

Variable	Definition	Section
Geometry	Geometric dimensions of the engineered barriers. A description of e.g. boundary surfaces inward towards the waste/barrier and outward towards the geosphere.	21.1.2
Pore geometry	Pore geometry as a function of time and space in barriers. The porosity, i.e. the fraction of the volume that is not occupied by solid material, is often given.	21.1.3
Temperature	Temperature as a function of time and space in barriers.	21.1.4
Water saturation	Water content as a function of time and space in barriers.	21.1.5
Mechanical stresses	Mechanical stresses as a function of time and space in the barriers.	21.1.6
Hydrovariables and hydrological boundary conditions	Water and gas pressures plus water flows from the surrounding area as a function of time and space in barriers.	21.1.7
Pore water composition	Composition of the pore water (including any radionuclides and dissolved gases) as a function of time and space in barriers.	21.1.8
Concrete composition	The chemical composition of the concrete (plus any radionuclides) as a function of time and space in the concrete. This also includes impurities and minerals other than portlandite.	21.1.9

Programme

The area is judged to require more research concerning the influence of the void and the backfill on hydrovariables, see also Section 20.1.2. The effect of backfill of different materials will be studied.

21.1.3 Pore geometry

The porosity in the structural concrete that constitutes the engineered barriers around the different repository parts has been calculated to be about 10 percent /21-1/. There is, however, no information on the size of the pores. The pore size can play an important role for the temperature at which the pore water in the concrete barriers is expected to freeze to ice. The pore geometry is affected by concrete degradation, see Section 21.2.10.

Programme

Drill cores have been taken from different parts of the barriers in BMA. These will be examined with regard to e.g. pore size. This can lead to a better understanding of the temperature at which the pore water in the concrete barriers is expected to freeze to ice.

21.1.4 Temperature

The initial temperature in the repository is determined by the temperature of the surrounding rock. Since, due to radiation attenuation, the waste does not generate any significant amount of thermal energy, the contribution of the waste to temperature changes in the engineered barriers is negligible.

21.1.5 Water saturation

Initially, water saturation in the barriers is low at closure due to the fact that the repository has been drained by pumping. Some water can occur initially via penetration from water-bearing fractures in the rock, and the water content of the different parts is dependent on their contact with the surrounding rock. The water content has an influence on the chemical processes in the repository, such as the tendency of the rebar to corrode.

The concrete in the barriers will become water-saturated after closure, when drainage pumping ceases. A hydrogeological model has been used to calculate how long it takes to fill and saturate the repository with groundwater /21-2/. The calculations show that the void (porosity) inside the silo repository is saturated last and that this can take 25 years. It only takes a few years to fully saturate BMA, BLA and BTF.

Newfound knowledge since RD&D 2007

SKB recently started an investigation programme for the structural concrete in BMA. This programme includes, among other things, measuring the relative humidity (RF) in the concrete at certain selected places in the structure.

During most of the year, large quantities of air are circulated to ventilate away radon and exhaust gases from vehicle operation. Since the supply air has a higher temperature for most of the year than the temperature in SFR, the relative humidity of the air reaches a high level due to the fact that the saturation vapour content is lower at a low temperature.

After a long period of exposure of concrete to an environment with high relative humidity, the concrete will reach equilibrium with that of the ambient environment in terms of relatively humidity. The measured values of relative humidity in specimens from BMA show that the relative humidity in certain parts of the concrete in BMA is high, see Figure 21-2.

Programme

The ongoing programme has not yet been concluded, and further investigations are planned for the BMA and BTF repositories. The purpose of the programme is to get a general idea of the state of the concrete structures in the BMA repository in particular.

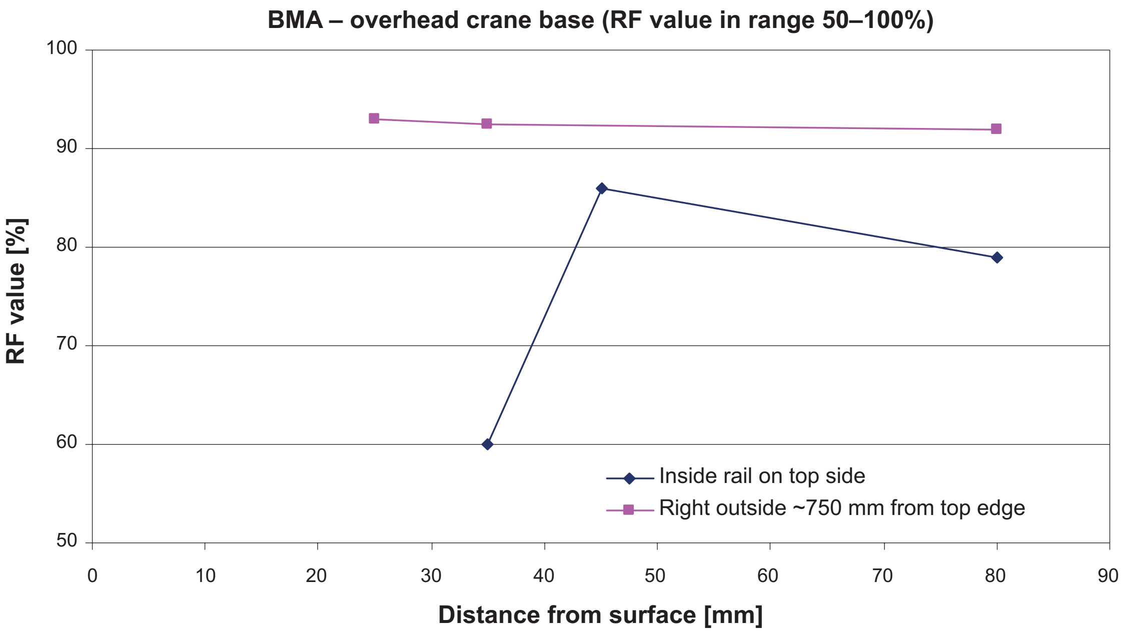


Figure 21-2. *Measured values of relative humidity (RF values) in drill cores in the overhead crane base in BMA.*

The degree of water saturation of the concrete can affect the corrosion rate and thereby the function of the barriers. It is therefore of interest to find out how the degree of water saturation changes during the operating time and the time after closure, before the repository has become fully water-saturated.

21.1.6 Mechanical stresses

Expansion/contraction of the waste and rock breakout affect mechanical stresses, see Section 20.2.6 and Section 21.2.8.

Corrosion of rebar in the concrete structures can give rise to mechanical stresses in the barriers both initially and later on, see Sections 21.2.6, 21.2.7 and 20.2.12. This can lead to a deterioration in the mechanical properties of the concrete.

Programme

Investigations have been started to assess the status of the concrete structures in the BMA and BTF repositories (concrete tanks that were emplaced during an early operating phase). The goal of the investigations is to get a picture of how the structures have been affected during the 22 years SFR has been in operation.

21.1.7 Hydrovariables and hydrological boundary conditions

The hydrovariables are water pressure and gas pressure and the boundary conditions consist of the water flows from the geosphere. The water flows do not occur initially in the engineered barriers, since the repository is drained by pumping during the operating phase.

The site of SFR was chosen in part because it is located in an area with a limited hydraulic gradient and limited fracture transmissivity. The placement of the repository was chosen in part for the purpose of ensuring low water flows through the different repository parts. The different repository parts have been designed with different capabilities for limiting the water flow.

The silo has the most advanced design; all waste is conditioned, all waste packages are embedded in porous concrete, there is an internal compartment structure of concrete, the outer walls are made of thick concrete, and above all the concrete silo is surrounded by bentonite. All waste in BMA is also conditioned. BMA has a simpler concrete structure and may be backfilled with material that has higher hydraulic conductivity than the concrete, which would lead to water flow through the backfill instead of through the more watertight concrete structure enclosing the waste. In 1BTF and 2BTF the concrete tanks are positioned so that they form an inner structure that is less permeable to water flow than the side and top fill of gravel or sand. BLA has no structure that limits the water flow through the waste.

During the operating phase of SFR, the repository will be kept drained and the repository will be open to atmospheric pressure. This gives a gas pressure (air) of one atm (approx. 0.1 megapascal (MPa)) and a water pressure of 0–0.1 MPa in the host rock, depending on the humidity of the air. On the other hand, there will be an initial negative pore water pressure in the surrounding concrete that draws in water, so that the water content of the concrete is in equilibrium with that of the air.

When drainage pumping of the repository ceases, the tunnels will become water-filled and the water pressure in the surrounding rock will increase. Water will flow into the repository via conductive fractures. In general, the barriers cannot absorb all water that runs through a fracture, so a positive water pressure arises. The water pressure drives the flow through the repository parts and their surroundings. This process is described in Section 21.2.4.

21.1.8 Pore water composition

Initially the pore water in the concrete will have a pH of 13.1 and its chemical composition will depend on the mineral composition of the concrete and how the concrete has been exposed to groundwater. The pore water composition has an influence on the chemical processes in the repository, such as the tendency of the rebar to corrode.

The concrete structures in SFR consist primarily of Degerhamn construction concrete. An analysis of cement pore water from fresh and leached cement is presented in Table 21-2. The analyses show that the highly soluble alkali hydroxides cause a high pH, which declines when they are leached out.

The porosity of a completely hydrated structural concrete of the type used in SFR is about 10 percent /21-1/.

The pore water composition is affected by all chemical processes in both the waste and the barriers, as well as the composition of the penetrating groundwater.

Programme

SKB has recently initiated a sampling programme for the structural concrete in BMA. This programme includes measuring the chloride concentration in the concrete at certain selected places in the structure.

21.1.9 Concrete composition

The concrete walls in the repository have a w/c (water/cement) ratio of about 0.47. Cement consists primarily of crystalline portlandite ($Ca(OH)_2$) and amorphous to crypto-crystalline calcium silicate hydrate (C-S-H gel). In addition to cement, concrete contains large quantities of aggregate, such

Table 21-2. Analysis of cement pore water from fresh and leached cement (ion concentrations in millimoles per litre [mmol/l]).

Parameter	Fresh cement[a)]	Leached cement[b)]
pH	13.1	12.6
SO_4^{2-}	0.04	0.02
Cl^-	<0.06	2 [c)]
Na^+	28	3 [c)]
K^+	83	0.1 [c)]
Ca^{2+}	0.9	20
Si_{tot}	0.8	0.003
Al_{tot}	0.04	0.002
OH^-	114	36

a) expelled pore water /21-3/,
b) crushed cement, analysis of leachate /21-4/,
c) same as concentration in leachate before leaching experiment

as sand, gravel and crushed rock. All cementitious materials contain some surplus material such as ettringite (hydrated Ca-Al sulphate), hydrotalcite (hydrated Mg-Al carbonate) and Friedel's salt (Ca-Al-Cl hydrate) /21-3, 21-5, 21-6, 21-7, 21-8, 21-9/.

In addition to pure inorganic minerals, the concrete in the engineered barriers also contains small quantities of organic additives. These additives are used in the casting of the concrete to give the concrete favourable casting properties, for example fluidifiers. When the waste in the silo has been backfill-grouted, cellulose is added to the cement. This will contribute about 9,000 kg of cellulose in the silo when it has been filled. The concrete composition is affected by the chemical degradation of cement and concrete, see Section 21.2.10.

21.2 Processes

A number of processes will eventually alter the state of the engineered barriers and their cavities. Certain processes take place under all conditions, while others are only possible when the barriers are water-saturated or when anaerobic conditions prevail.

21.2.1 Overview of processes

The processes that affect conditions in the barriers can, like the processes that affect the waste, be divided into four different main classes: thermal, hydraulic, mechanical and chemical. Under each main class, a number of different processes can occur that can interact with each other.

Thermal processes

The effect of temperature on the engineered barriers is not negligible, and freezing drastically alters the integrity of the concrete /21-10/. This changes the chemical and mechanical retardation properties of the concrete and affects radionuclide transport out of the repository. The temperature of the repository, and thereby the barriers, will be determined almost entirely by heat exchange with the surrounding rock and groundwater. The influence of the waste on the temperature is negligible, since the heat-generating processes in the waste are deemed to be negligible. The following thermal processes are dealt with:

- Heat transport, see Section 21.2.2.
- Freezing, see Section 21.2.3.

Hydraulic processes

The water flow through the repository is determined by the permeability of the various components in the repository and by the hydraulic gradient. If gas is present at the same time, this gives rise to a two-phase flow, where both the water flow and the gas flow are affected by the relative degree of saturation of each phase. The magnitude of the water flow in the repository is determined to a high degree by the surrounding groundwater flow. The following hydraulic processes are dealt with:

- Water transport, see Section 21.2.4.
- Two-phase flow/gas transport, see Section 21.2.5.

Mechanical processes

The barriers in the different repository parts will be subjected to external mechanical forces and internal volume changes in the same way as the waste, see Section 20.2.1. The following mechanical processes are dealt with:

- Expansion/contraction of the waste, see Section 21.2.6.
- Fracturing, see Section 21.2.7.
- Rock breakout, see Section 21.2.8.

Chemical processes

The physical properties of different concrete structures are affected by numerous chemical processes such as recrystallization, water uptake, chemical degradation, corrosion of metals, dissolution/precipitation and formation of different corrosion products, leading to gas evolution. Of these, it is above all water uptake – resulting in transformation of concrete, possibly with volume expansion – and metal corrosion that have any appreciable influence on the evolution of the barriers with respect to their content of solutes and colloids. Water composition changes as a result of advection and mixing. Concentration differences are equalized via diffusion. Sorption of radionuclides is affected mainly by the water composition in the repository. The concentration of substances occurring in small quantities, such as complexing agents, can have a great influence on the sorption of cations dissolved in the water. The following chemical processes are dealt with:

- Dissolution/Precipitation, see Section 21.2.9.
- Chemical cement and concrete degradation, see Section 21.2.10.
- Corrosion, see Section 21.2.11.
- Sorption, see Section 21.2.12.
- Diffusion, see Section 21.2.13.
- Advection and mixing, see Section 21.2.14.
- Colloid formation and transport, see Section 21.2.15.
- Microbial activity, see Section 21.2.16.

The research programme for the various processes in the barriers is dealt with in the following sections. The interaction of the processes with each other is not discussed here, but will be described in the process report that will be submitted as a background report to the safety assessment for the Extension Project.

21.2.2 Heat transport

When the temperature on the surface of the repository changes, this will affect the temperature of the barriers, with some delay. When the temperature changes the properties of the barriers may change, which can lead to a change in their mechanical and hydraulic properties.

Programme

The research conducted by SKB regarding the effect on the temperature in the barriers is described in Chapter 19.

21.2.3 Freezing

The frost resistance, or resistance to frost bursting, of the concrete is dependent on the pore structure in the concrete. The gel pores and capillary pores are filled very quickly with water during normal outdoor use of the concrete. When water is absorbed over a long period, the large pores (aeration pores etc.) are also filled with water. When the concrete freezes, some of the pore water turns into ice and the resultant expansion can create such great stresses inside the concrete that it is seriously damaged /21-10/.

The freezing point of water decreases with reduced pore size. For example, water does not freeze in a pore with a diameter of 150 angstrom until –20°C. At the normal freezing temperature, just under 0°C, water in the gel pores and in the finest capillary pores is therefore still not frozen. Besides pore size, the freezable water quantity is also dependent on whether the concrete has first been partially or fully dried-out before the pores are refilled with water.

Unaerated concrete, young concrete and concrete whose surface is exposed to weak saline solutions, for example from thaw salts or from sea water, is particularly susceptible to frost damage. The risk of damage increases with increased moisture loading.

In a final repository of the SFR Silo or BMA type, the moisture loading time is so long that not even the most effective air pore system is likely to be able to protect the concrete in this respect, since the repository will be water-filled when freezing occurs.

Frost-loaded concrete is subjected to two types of frost attack:

a) Salt frost attack, which is purely a surface attack caused by freezing and thaw salting, or freezing and sea water. Damage is manifested in the form of concrete flaking.

b) Internal frost attack and fracturing of the gel caused by the creation of high bursting pressures in the interior of the concrete due to expansion when the freezable water turns into ice. For this to occur, the concrete must be water-saturated above a critical level.

Internal frost attack is not expected under permafrost loading. The water in the facility will not be completely free of salt, but will have some salt content depending on the geochemical environment. Salt in the water reduces the freezing temperature and leads to less expansion of the water at a given freezing temperature than if completely non-saline water is assumed.

Damage due to internal frost attack can be observed in that the concrete shows signs of expansion and the concrete surfaces develop a dense pattern of wide and deep cracks running in all directions. The surfaces may be completely free of flaking, however. Damage due to internal frost bursting may also cause a loss in strength and modulus of elasticity, due to the fact that the porosity has increased.

The following parameters control the internal frost attack:

- The pore system in general, i.e. pore volume, pore diameter, pore size distribution and pore distance.
- The water/cement (w/c) ratio in the concrete.
- Moisture saturation in the pores.
- Quantity of freezable water.
- Availability of water.
- Time, i.e. available time to fill the pore system to the highest (critical) degree of moisture saturation that can be accepted if frost damage is to be avoided. This may be considerably lower than 100 percent.
- Whether there are microcracks due to previous drying-out or frost damage.

Conclusions in RD&D 2007 and the supplement to RD&D 2007 and their review

When RD&D Programme 2007 was being written, studies were still in progress concerning the impact of freezing on the engineered barriers. They only studied the impact of freezing on the engineered barriers.

Newfound knowledge since RD&D 2007

From the reflections and calculations that have been made within the framework of this study, some conclusions can be drawn regarding the barrier function of concrete after freezing and thawing. This knowledge is entirely based on literature studies and theoretical reasoning; no laboratory experiments have been conducted /21-10/. The following conclusions are drawn in the report:

- When internal freezing becomes a possibility, the concrete in the silo will have a water content that exceeds the critical value for avoiding internal freezing.
- At the first freeze, the concrete will suffer internal frost attack, which will break up the material to such a degree that its barrier function is largely lost. The remaining material has a barrier function similar to that of natural sand/gravel structures.
- After freezing and thawing, the degraded concrete barrier cannot be credited with any function as a moisture barrier beyond that of sand/gravel structures. At this point the concrete has a sand/gravel-like structure.
- The duration of the permafrost period will not be of importance for the frost heaves. What is decisive for the size of the heaves is the time it takes for the frost front to pass the repository. This is roughly the same for the different scenarios studied.

Programme

Studies of how the concrete's freezing properties change with time due to chemical degradation are currently in progress. At the same time, studies are being made of the pore size in the structural concrete of which the engineered barriers in SFR are made. Better knowledge of the pore size leads to better accuracy in the determination of the freezing point of the engineered barriers, as well as a better understanding of the impact of the temperature on the repository when permafrost prevails at the surface.

21.2.4 Water transport

The size of the water flow is determined by the geometry and conductivity of the engineered barriers and the backfill material, see Section 21.1.7 and Section 21.2.14. This process was also described in Chapter 20.

Programme

The area is mainly treated via modelling of water flows through the repository, and the work being done by SKB in this area is described in Section 21.3.

21.2.5 Two-phase flow/gas transport

In order for the gas formed in the waste and the barriers by e.g. microbial degradation and metal corrosion to escape, gas-conducting passages must be formed in the barriers. The gas transport and the quantity of water expelled from the silo repository and the rock vaults are determined by the design of the barriers and the properties of the barrier materials.

In the case of materials with a fine pore structure such as concrete, capillary forces are of importance for the pressure that needs to be built up to expel the gas. Water will be expelled from the concrete until a network of empty pores has been formed for transport of the gas. When gas-conducting passages have been formed in the barrier, the gas flows out as long as the pressure difference exceeds the capillary pressure. The capillary pressure in an intact structural concrete is of the order 1–2 MPa /21-11/. If the structural concrete has small fractures, less pressure is required to expel the gas. A pressure of 1–2 kilopascal (kPa) is required for a planar fracture with an aperture of 0.1 mm, while a pressure of about 15 kPa is required for a 10 μm fracture /21-12/. In concrete packages and concrete structures, a number of small fractures is sufficient to expel all the gas.

In the silo repository, the waste is surrounded by a porous concrete with low resistance to gas transport. Only a small gas pressure is required to open up gas passages in this concrete, and the quantity of water that is expelled has been measured experimentally to be 0.1–2 percent of the pore volume /21-13/. It has been assumed in the gas transport calculations that two percent of the pore volume in the concrete can be displaced. In order for the gas to find its way out through the gas evacuation pipes and the sand/bentonite barrier in the top, a gas pressure must be built up in the concrete silo equivalent to the opening pressure of the sand/bentonite barrier. Experiments show that a pressure of approximately 15 kPa is required to achieve gas transport through the sand/bentonite barrier and that the expelled water is only a few tenths of a percent of the total pore volume /21-14/.

In the case of a gravel fill, only small pressures are required to open up transport pathways and to drive the gas through the porosity in the fill. Gas that has flowed through gravel fill and made it to the void beneath the roof in the silo repository and the rock vaults can then be further transported in fractures in the surrounding rock.

Gas transport has been studied in the Lasgit test /21-15/ in the Äspö HRL. Lasgit (Large-Scale Gas Injection Test) is a full-scale experiment with KBS-3V geometry for studying the effect of gas transport in the bentonite buffer, see Section 24.2.7. Gas began to be injected in Lasgit in the spring of 2007. This could provide preliminary results regarding how the gas build-up and transport processes take place. Since the bentonite buffer has not reached its equilibrium state, however, the results from the first test series must be treated with some caution.

How gas is transported in the engineered barriers in a repository for intermediate-level waste has been studied on a field scale in the Gas Migration Test (GMT) /21-16/. The test started in 1997 in Nagra's underground laboratory in Grimsel, Switzerland. The field tests were concluded in 2004 and most of the evaluations and modellings have also been concluded. The test was conducted in a special-purpose concrete silo with a bentonite/sand barrier fitted with a gas valve. Two phases of gas tests were conducted in GMT. Before, between and after the gas tests, hydraulic tests were performed to see whether the properties of the engineered barriers were altered due to the gas breakthroughs.

Newfound knowledge since RD&D 2007

The following conclusions have been drawn from the GMT tests in Grimsel:

- Gas could be transported through the valve to the geosphere at relatively high flows and low pressures.
- The function of the bentonite/sand barrier was not adversely affected by the gas. No change could be detected in the hydraulic properties.
- The volume of water that was expelled from the silo during the gas tests was very small (10–13 litres, about one percent of the total injected gas volume).

Programme

The degree of saturation of the concrete can affect its freezing properties and thereby the function of the barriers. It is therefore of interest to find out how the degree of saturation changes over time, via two-phase flow and gas transport, as well as the freezing consequences of fully saturated concrete. The programme is linked to the programme for water saturation, see Section 21.1.5.

21.2.6 Expansion/contraction

There are a number of processes that could affect the concrete barriers and at worst cause them to crack. Volume increase of waste matrices and waste packages due to water uptake in ion exchange resins conditioned in bitumen and formation of expansive corrosion products can induce stresses in surrounding concrete barriers unless sufficient void volume is available to take up this volume increase, see Section 20.2.6. Volume increase due to corrosion of rebar can also cause internal stresses. Build-up of internal gas pressures and settlements and movements in the different barriers are other mechanisms that could eventually cause cracks in the concrete barriers. The consequences of these types of mechanical effects have not been analyzed in detail, and the possibility cannot be completely ruled out that they could eventually cause fracturing in the concrete barriers.

Programme

Parts of the research which SKB intends to conduct in this area fall within the framework of the initiated project "Chemical and mechanical properties of aged concrete", which is described in Section 21.2.10.

Rebar corrosion can contribute to altered mechanical properties of the concrete barriers, and it can therefore be of interest for SKB to study this influence and how it develops over time.

21.2.7 Fracturing

Fracturing in the concrete can be caused by a number of different processes, including expansion/contraction (Section 21.2.6), freezing (Section 21.2.3) and chemical concrete degradation (Section 21.2.10).

The reaction products that are formed when reinforcing bar in the concrete structures corrodes can lead to fracturing of the concrete around the rebar. The corrosion products have a greater volume than the original steel, which causes a pressure from the corrosion products on the surrounding concrete /21-17/. This is a well-known phenomenon that should be taken into consideration in determining the thickness of the concrete layer over the rebar. This mechanism is not expected to lead to penetrating cracks in the concrete structures.

Dissolution of salt from evaporator concentrates can free chlorides, carbonates and sulphates, which can in turn react with surrounding concrete barriers, see Section 20.2.9.

Fracturing can in turn affect conductivity, water flow and solute transport, which is dealt with in the section on water transport (21.2.4) and in Sections 21.2.14 and 20.2.14, which deal with advection and mixing.

Programme

Fracturing is being studied within the framework of the project "Chemical and mechanical properties of aged concrete", see Section 21.2.10. Further studies may be called for describing the influence of corrosion products on fracturing of the structural concrete in SFR.

Studies of precipitation of secondary phases and their effect on self-healing of fractured concrete are of interest, since precipitates could conceivably clog cracks in the concrete and thereby reduce transport out of the repository via cracks in the concrete.

The programme is linked to the programme for cracking in the waste, see Section 20.2.7.

21.2.8 Breakout

When the loadbearing capacity of the rock bolts in particular has declined, rock blocks can be expected to come detached and fall into the rock cavern. This process can continue until a naturally stable arch is formed around the rock cavern or over several rock caverns. A counterpressure can be created by the use of a stiff backfill, which reduces the size of the loosened area around the rock caverns.

This loosening probably proceeds in an empty rock cavern until the cavern has been completely filled with fallen rock, so that a counterpressure is exerted against the remaining rock. It is difficult to estimate how long the loosening process will proceed before stable conditions are obtained. Based on experience from cave-ins in abandoned mines, this is estimated to take 50–150 years.

With the existing rock support, the rock caverns in SFR are expected to remain stable for 200–250 years. After this a loosening of the rock around the rock caverns occurs that is expected to 50–150 years until stable conditions are obtained. A conservative estimate of the maximum size of the loose zones is presented in /21-18/. Provided the BMA and BTF repositories are backfilled with sand, the extent of the zone is estimated to be two metres, while the loose zone around BLA is estimated to be five metres if the rock vault is not backfilled /21-19/. The size and shape of the loose zone can vary by about 50 percent depending on the variation in rock quality and the direction of the fractures in relation to the rock caverns, see Figure 21-3.

Breakout changes the geometry of the disposal chambers, see Section 21.1.2. Breakout also affects the water flow, see Section 21.2.4.

Programme

The plans include constructing models that describe the consequences of possible rock breakouts as regards cross-contamination by radionuclides and other materials between rock caverns. It is of particular interest to study this cross-contamination between the BLA and BMA repositories.

The area is judged to require more research regarding the influence of breakout on hydrovariables and on mechanical stresses in the barriers.

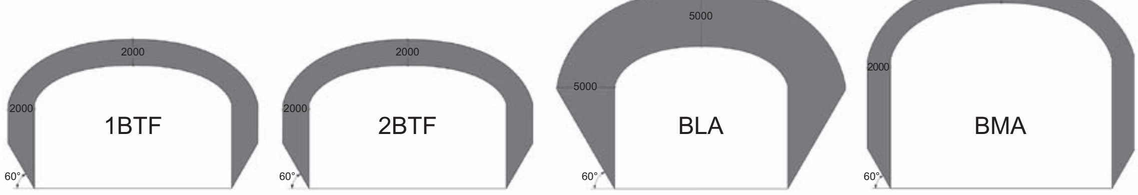


Figure 21-3. *The figure shows that in the case of rock caverns that are backfilled with sand (BMA and BTF), the depth of the loose zone (shaded) is generally expected to be about two metres. The size of the loose zone around BLA has been estimated to be five metres if BLA is not backfilled /21-19/.*

21.2.9 Dissolution/precipitation

Chemical compounds in the engineered barriers can react with each other and with substances dissolved from the waste and its matrices. These reactions can contribute to the precipitation and immobilization of certain substances. Conversely, this can lead to dissolution of substances, making them available for transport. Which reactions take place and which equilibria become established in the barriers depends on the concentrations of constituent substances in the barriers and of solutes from the waste and its matrices.

When substances in the waste and its matrices degrade, soluble substances can be formed which are thereby available for transport out of the repository. Substances can be precipitated on minerals in the concrete, affecting their concentrations in solution. In the same way, mineral phases in the concrete can be dissolved and precipitated and affect the concentrations in solution. The solutes that can form ions in solution may affect the water's ionic strength, which can in turn affect the sorption of radionuclides on the concrete. Since the concentration of solutes deriving from waste degradation products is not yet known, their effect on ionic strength cannot be quantified. The ionic strength of the cement pore water is mainly determined by the pH and the concentration of Ca(II).

Programme

SKB plans to study corrosion and the effects of degradation products on the barriers more closely in order to study how the sorption capacity of the barriers changes over time.

A programme has been initiated in the Äspö HRL where waste will be immobilized in concrete. An interim goal of the project is to study how the chemical and mechanical properties of the concrete are affected by degradation products from the waste.

Efforts in this area are also described in Section 21.2.10.

21.2.10 Chemical cement and concrete degradation

Along with the mechanical evolution of the repository, its chemical evolution is of importance for judging the durability of the engineered barriers in the repository with respect to the release of radionuclides and other species, and in particular for sorption.

The durability of the engineered barriers is affected by chemical reactions that take place when the barriers come into contact with the groundwater /21-1, 21-20, 21-22/.

The concrete and bentonite barriers in SFR are altered with time as a result of a number of chemical processes. The pore water in the concrete will at first have a pH above 13, due to the concentrations of KOH and NaOH. Interaction between concrete and groundwater leads to leaching firstly of highly soluble alkali hydroxides, and secondly of calcium hydroxide (portlandite), which is an important constituent in the cement paste in the concrete. When the portlandite is leached out, the pH of the pore water drops to around 12.5. When the portlandite has been leached out, an incongruent dissolution of calcium silicate hydrate (CSH) phases begins, and the pH of the pore water is further lowered to 10.5. The leaching process thus leads to a progressive lowering of the pH in the pore water in the concrete.

In summary, the calcium silicate hydrate phases degrade according to the following scheme:

CSH_1.8 → CSH_1.1→ CSH_0.8 (tobermorite)

CSH stands for calcium silicate hydrate, and the number after CSH stands for the ratio of Ca to Si. In other words, the calcium/silicate ratio decreases due to leaching.

Conclusions in RD&D 2007 and the supplement to RD&D 2007 and their review

It was concluded in the supplement to RD&D Programme 2007 that studies of concrete and bentonite degradation are included in SKB's programme.

In the review of the supplement to RD&D Programme 2007, SSM said that they take a positive view of the fact that SKB has initiated a project aimed at learning more about the degradation of cement and concrete over long periods of time.

Newfound knowledge since RD&D 2007

A new study of chemical degradation of the concrete barriers in SFR has been published /21-20/. The study is based on a thermodynamic model of the different stages of concrete degradation. Furthermore, the model deals with a longer time perspective (the time after permafrost) than previous models. The study also examines how varying porosity in the concrete and the bentonite affects long-term safety.

Programme

During 2009, SKB initiated two large projects for studies of the properties of cement and concrete and their interactions with the surrounding environment and embedded waste. These projects, for both of which the experimental work will start during 2010, focus on different time horizons and will be executed in different ways.

The project "Long-term durability of cement" is being carried out as an internal SKB project with most of the experimental activity taking place at the Äspö HRL and the cement laboratory at the Ringhals NPP. In addition, some activities relating to modelling of chemical processes will be executed, but the focus is not on the experimental design. The main goals of the project include studies of the interaction between different kinds of concrete and cement and bentonite or immobilized waste. Investigations of how a pH plume can develop in the bedrock around the test body are also included as a possibility, but are not the main purpose of the study.

The project "Chemical and mechanical properties of aged concrete" is being executed as a doctoral project in cooperation with Chalmers. The purpose of the project is to gain a better understanding of the chemical and mechanical properties of aged concrete and implement this knowledge in future safety assessments for SFR and SFL. The main goal of this project is to study property changes in a long time perspective, up to 100,000 years, in part by modelling, but mainly by means of experimental methods involving trying in different ways to manufacture cement with an aged structure and then investigating its properties. The project is expected to proceed during the entire upcoming RD&D period, and it is hoped that the results can be used in the future safety assessment for SFL.

In addition to the large projects described above, we also plan to commence studies of the gas permeability of concrete in 2010. Additional efforts related to this process are described in Section 21.2.9.

21.2.11 Corrosion

The oxygen that is present in the repository at closure will be dissolved in the inflowing groundwater. This oxygen will quickly be consumed by e.g. corrosion of steel in containers and reinforcing bar, oxidation of dissolved iron(II) and sulphide in the water, and microbial processes. A low redox potential will be maintained in the different repository parts due to the release of Fe(II) ions from anaerobic corrosion of steel and from corrosion products. The corrosion products of iron can contribute to buffering the redox conditions.

The research programme for this process is described in Section 20.2.12.

21.2.12 Sorption

Sorption of radionuclides is one of the most important retarding safety functions in SFR. Sorption takes place primarily on the cement in the barriers and the waste matrix and is dependent on the chemical composition of the water in the repository. A prerequisite for good sorption is a favourable chemical environment.

Sorption of radionuclides is expected in all repository parts and in the surrounding rock. The greatest potential for sorption exists in the repository parts that contain cement in the concrete walls, the backfill grout or the waste packages.

The importance of the safety function "sorption in concrete barriers" is strongly linked to the chemical properties of individual radionuclides. Some nuclides will not sorb under any conditions, in which case the safety function does not exist at all, but most radionuclides can be assumed to sorb very strongly under the conditions that exist in SFR. For many radionuclides, sorption is linked to the chemical environment in the repository. The different nuclides will, however, be affected in different ways by the chemical environment.

In general, cations sorb best at higher pHs, while the opposite applies to anions. In natural waters, however, where organic ligands and carbonates occur, it is not certain that sorption increases with pH, since these ligands bind cations and thereby reduce their sorption. Complexation also increases with pH. The chemical environment in the cement pores, high pH and high calcium concentrations, means that the carbonate concentration will necessarily be low, since it is regulated by the calcite equilibrium. High concentrations of free carbonate could otherwise contribute to increased radionuclide transport out of the repository. In the same way, high pH values and high calcium concentrations keep down the concentration of numerous other ligands, such as oxalate. Other ligands, such as α-isosaccharinic acid, sorb to cement, possibly by complexation with calcium-rich solid phases. The effect of additives in cement on sorption (cement and granite) has been studied jointly by SKB, Nagra, NUMO and Posiva /21-23/.

Some of the radionuclides are redox-sensitive, and for some sorption is much weaker under oxidizing conditions. A low redox potential in the engineered barriers is expected to be sustained by the presence of metallic iron and organic material. Sorption of U, Pu, Np and Tc decreases considerably under oxidizing conditions.

Organic compounds from degradation of the waste form (particularly cellulose) could form complexes with the radionuclides and in this way compete with sorption on solid surfaces. It is important to keep the quantity of complexing agents at low levels. The most important organic complexing agent is isosaccharinic acid, see Section 20.2.10 /21-22/.

Sorption takes place on the surfaces of solid phases. Cement has a relatively large porosity, and several of the solid phases which cement consists of are amorphous and have a large specific surface area, which favours sorption. Most of the cement's solid phases are transformed with time, which can change the sorption capacity. Degradation of cement that results in precipitates that reduce porosity can also contribute to a reduction in sorption capacity.

In summary, due to its unique chemical properties, cement has proved to be excellent for keeping down the concentration of ligands that can prevent the sorption of numerous radionuclides. As long as portlandite ($Ca(OH)_2$) is present in the cement, the pH of the cement pore water will be about 12.5. When the portlandite has been leached out, the pH will gradually be reduced to 10.5. Changes in pH change the speciation of the radionuclides and the formation of organometallic complexes, which affects the sorption properties.

Conclusions in RD&D 2007 and the supplement to RD&D 2007 and their review

RD&D Programme 2007 described how the uncertainties in different radionuclides' sorption coefficients in concrete, bentonite and sand/gravel have been estimated and determined. These data still apply and were presented in SAR-08. Furthermore, ongoing studies concerning complexation of radionuclides with organic degradation products such as cellulose were described. These studies are still in progress and have been expanded, see below.

Programme

SKB's research in the area is aimed at trying to model the sorption of the nuclides that are important for the safety assessment in the presence of organic complexing agents. The purpose is to construct a model where the suitability of new complexing chemicals for disposal in SFR can quickly be determined.

Other studies of importance are to evaluate the sorption properties of aged cement. The chemical composition of aged cement can be modelled with thermodynamic calculations, such as the program PHREEQC /21-20/. The hope is, based on this knowledge, to obtain new sorption coefficients for the different ageing stages of the cement. This can in turn lead to greater knowledge of the sorbing properties of partially degraded barriers, and how these properties affect future radionuclide releases from the repository.

When radionuclides are transported out through the concrete, either as hydroxide complexes or as organometallic complexes, they will be affected by a constantly changing chemical environment due to the fact that penetrating groundwater alters the chemical composition of the concrete at different depths in the concrete structure. This altered environment affects the speciation and solubility of the different radionuclides. How this affects the retardation of radionuclides may be the subject of further modelling studies.

Furthermore, a study is under way at Chalmers where degradation products from the filter aid UP2 (a polyacrylonitrile-based polymer) are being studied with respect to their influence on the sorption of Cs(I), Co(II) and Eu(III).

21.2.13 Diffusion

Solutes can be transported in stagnant pore water by diffusion. The process leads to redistribution of solutes in the pore water and thus affects the pore water composition, see Section 21.1.8.

The diffusion process is strongly coupled to nearly all chemical processes in the engineered barriers, since it accounts for transport of reactants to and reaction products from the processes. Diffusion is thereby a central process for the chemical evolution of, and radionuclide transport in, the barriers.

In intact concrete, the conductivity is so low that diffusion is the dominant transport mechanism. But concrete is not a self-healing material in the same way as bentonite, and the possibility cannot be excluded that there are cracks in the concrete walls, which could dominate the mass transport.

Based on the calculations that have been carried out regarding leaching of concrete, concrete moulds and cement matrices are not expected to be subjected to significant leaching of cement components for at least 1,000 years in BTF and 10,000 years in BMA /21-1, 21-20, 21-21/. No significant change is expected in the silo during the first 100,000 years /21-20/.

In porous media, the diffusion rate is affected by sorption, see Section 21.2.12.

Programme

When the concrete ages, its diffusion-related properties will change and the diffusion will therefore differ compared with “fresh” concrete. This affects the rate of outbound transport of radionuclides, making it important for SKB to determine the diffusion rate in aged concrete for relevant nuclides. This will be studied on concrete in various stages of ageing within the framework of the project “Chemical and mechanical properties of aged concrete”, see Section 21.2.10.

21.2.14 Advection and mixing

It is assumed that transport of solutes through the engineered barriers can take place by means of both advection and diffusion. Initially, diffusive transport is assumed to be dominant in the concrete barriers, while advective transport is assumed to be negligible. When the concrete degrades, fractures will occur and advection will increase.

Programme

When the concrete ages, its chemical and mechanical properties change. These changes can cause the size of the advective water flow to change with time. A changed water flow resulting from this can affect the rate of radionuclide transport, making it important to study the consequences of this in order to gain a better understanding of radionuclide transport out of the repository in the aged concrete.

21.2.15 Colloid formation and transport

This process is described in Section 20.2.15. See also Sections 24.2.20, 24.2.28, 25.2.18, 25.2.19.

21.2.16 Microbial activity

This process is described in Section 20.2.16. See also Section 25.2.16.

21.3 Modelling of radionuclide transport for SFR

The following text describes how processes in the barriers are linked with the transport of radionuclides and how models for radionuclide transport will be developed for the SFR repository.

21.3.1 Processes that affect radionuclide transport

Several processes affect the transport of radionuclides in and from the repository. Radionuclides are transported by advection and are also subjected to dispersion and mixing. In repository parts with little water flow, for example inside waste containers or inside the shafts in the silo or in BMA, diffusion is expected to be the most important transport mechanism. Sorption is the most important retarding mechanism. In previous safety assessments, the concentration of different radioactive substances has been considered to be too low to make precipitation an important retarding function, considering only the radionuclide concentration. On the other hand, co-precipitation with non-radioactive substances is probably an important retardation process. In the cement-rich environments in the repository, the calcium concentration is high, which counteracts the formation of colloids /21-24/. In previous safety assessments, colloid transport has not been considered to occur in these parts of the repository, since the concentration of Ca(II) in the groundwater is sufficiently high to prevent colloid formation. Chemical, mechanical and microbial degradation of bitumen affects radionuclide release from the bitumen waste. Gaseous radionuclides that are not dissolved in the water will be transported as gas, and with very little retardation.

Besides these, there are several different processes that affect radionuclide transport and that are handled by codes used for safety assessment calculations. The following sections explain how the identified processes are handled in radionuclide transport modelling, based on the process descriptions presented in Section 20.2 and Section 21.2.

21.3.2 Newfound knowledge since RD&D 2007

Extensive calculations of radionuclide transport for SFR were carried out within the SAR-08 safety assessment project /21-18/. The basis for these (division into elements and the hydrogeological model) was the transport calculations that were carried out in SAFE /21-25/, but the calculation models were implemented in another calculation program and the time for the calculations was extended so that the impact of climate change could be included in the analysis. After SAR-08, the development work has continued and SKB now has a functioning model in Ecolego, where both more complicated and more simplified transport models have been analyzed.

21.3.3 Programme for calculation codes for radionuclide transport

The calculation code that is intended to be used for probabilistic radionuclide transport calculations within the SFR Extension Project is Ecolego. In calculations carried out after SAR-08, the code has been shown to be suitable for solving the relevant problems and should be able to handle the process descriptions that are formulated in the research programme that is described in Section 20.2 and Section 21.2.

In addition, a study is planned to model near-field transport in a code that permits a higher room resolution, and a more detailed model for transport and flow in rock vaults may be developed. For this it is possible to use a multiphysics program, for example Comsol Multiphysics, where different processes are coupled together in a more complex way than has been done in the compartment-based codes used in previous safety assessments for SFR. The purposes of the new models are to be able to better evaluate backfilling and closure alternatives and better analyze the effect of fractures in the near-field, degraded barriers, short-circuiting between different repository parts and altered groundwater flows, as well as to gain a better understanding of the effect of different parameters on the result.

22 Fuel

The spent fuel that will be deposited in the repository comes from the nuclear power plants (NPPs). Based on the NPPs' operational planning, the quantity of BWR and PWR fuel that will be deposited has been estimated at just under 12,000 tonnes. In addition, 23 tonnes of MOX fuel, 20 tonnes of fuel from the Ågesta reactor and some residues from fuel investigations at Studsvik will be deposited. Burnup can vary from about 15 to 60 megawatt-days per kilogram of uranium (MWd/kgU). Differences in radionuclide content between PWR and BWR fuel are marginal viewed from a safety assessment perspective. MOX fuel has a higher decay heat than uranium fuel, which means that less fuel can be deposited in each canister. The differences between different fuel types are more important when it comes to assessing criticality.

22.1 Initial state

22.1.1 Variables

For the safety assessment SR-Site, the fuel is described by means of a set of variables which together characterize the fuel in a suitable manner for the assessment. The description applies to the fuel and the cavities in the canister, into which water can penetrate if there is a defect in the copper canister. The variables are defined in Table 22-1.

22.1.2 Total radionuclide inventory

Conclusions in RD&D 2007 and its review

It was mentioned in RD&D Programme 2007 that new calculations may be required in view of the burnup increase or new fuel types. No direct viewpoints were offered on this by the authorities.

Table 22-1. Variables in the fuel.

Variable	Definition	Comment
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.	No programme. Knowledge sufficient.
Radiation intensity	Intensity of α, β, γ and neutron radiation as a function of time and space in the fuel assembly.	No programme. Knowledge sufficient.
Temperature	Temperature as a function of time and space in the fuel assembly.	No programme. Knowledge sufficient.
Hydrovariables	Flows and pressures for water and gas as a function of time and space in the cavities in the fuel and the canister.	No programme. Not relevant (initially intact canister).
Mechanical stresses	Mechanical stresses as a function of time and space in the fuel assembly.	No programme. Knowledge sufficient.
Total radionuclide inventory	Total occurrence of radionuclides as a function of time and space in the different parts of the fuel assembly.	Section 22.1.2
Gap inventory	Occurrence of radionuclides as a function of time and space in gaps and grain boundaries.	Section 22.1.3
Material composition	The materials of which the different components in the fuel assembly are composed, excluding radionuclides.	Section 22.1.4
Water composition	Composition of water (including any radionuclides and dissolved gases) in the cavities in the fuel and canister.	See "Gas composition", Section 22.1.5
Gas composition	Composition of gas (including any radionuclides) in the cavities in the fuel and canister.	Section 22.1.5

Newfound knowledge since RD&D 2007

New inventory calculations have been done for SR-Site. The calculated radionuclide inventory is based on knowledge of currently existing spent fuel and a forecast of quantities and types of spent fuel that will exist in 2045 (the year when the last NPP is taken out of service according to the forecast). The inventory in the spent fuel was calculated with the code ORIGEN-S. The inventory for activation products in other material was calculated with the codes IndAct and CrudAct. The results of these calculations are presented within the framework of SR-Site.

Programme

The field is judged today not to require any further research, development or demonstration. New developments are being monitored and will be acted on when appropriate.

22.1.3 Gap inventory

During irradiation in the reactor, a certain fraction of the fuel's radionuclide inventory is segregated to the gap between fuel and cladding (fuel-clad gap) and to grain boundaries in the fuel. The fraction of the radionuclide inventory present in the gap, the gap inventory, is considered to be released much more quickly than the fraction embedded in the fuel matrix. It is thereby important for estimating the instant release fraction (IRF) of the inventory, which can give rise to pulse releases. Fission gases (for example Kr or Xe) are mobile and their behaviour is relatively well known and documented in a number of published studies. A fraction of the fission gases is segregated during operation and can be found after operation in the fuel-clad gap. The released fraction of fission gases is measured after operation when fuel rods are punctured and is called fission gas release.

Conclusions in RD&D 2007 and its review

SKI considers that SKB needs to review the supporting data and justification for the gap inventory.

Newfound knowledge since RD&D 2007

Certain mobile fission products behave like the fission gases and their segregation is generally considered to be comparable to fission gas release. Other radionuclides that can segregate are above all the fraction of the fuel inventory that is incompatible with the UO_2 matrix /22-1, 22-2, 22-3, 22-4/.

The analysis of data for the fraction of the radionuclide inventory that has been segregated to the fuel-clad gap or in grain boundaries that was done for SR-Can /22-5/ was based on fuel with an average burnup of 38 megawatt-days per kilogram of uranium (MWd/kgU) and a very limited number of fuel assemblies with a burnup of more than 50 MWd/kgU. In view of the plans to increase burnup of future BWR and PWR fuel up to 60 MWd/kgU, a new estimate of the gap inventory is being made within the SR-Site project.

Only a limited quantity of data exists for fission release from BWR fuel with a burnup of more than 50 MWd/kgU /22-6/. The available quantity of data for high-burnup (50–60 MWd/kgU) PWR fuel is slightly greater /22-5, 22-3, 22-4/. In view of the limited quantity of data for fuel with a burnup up to 60 MWd/kgU, fission gas release has been calculated for typical Swedish BWR and PWR fuels /22-7, 22-8/. These calculations take into account the reactor's power level, since the power level affects the fuel gas release more than the fuel's burnup /22-9/, although the two are often related. The plans for an increased power level in Swedish reactors entail an expected increase of fission gas release in the future.

Regarding fuel from light water reactors, there are more systematic studies of the release of fission gases than of other segregated radionuclides (for example Cs, I, Sr) /22-10, 22-11, 22-3, 22-12/. A study of the release of segregated radionuclides from four PWR fuel segments with high burnup (55–75 MWd/kgU) has been partially reported /22-13/ and is continuing in cooperation with Nagra and PSI (Switzerland).

Programme

The study of releases of segregated radionuclides from fuel segments from PWRs and fuel fragments with high burnup and measured fission gas release is continuing and is being supplemented with analyses of other radionuclides, such as I-129 and Se-79. The results are discussed regularly and will be published jointly with Nagra and PSI. Radiochemical analyses of C-14 and investigations of high-burnup BWR fuel are being carried out at PSI, along with method development by testing of a standard solution of Se-79, permitting determination of a detection limit for the analyses.

22.1.4 Material composition

Conclusions in RD&D 2007 and its review

Of the non-radioactive fission products, barium is of interest since co-precipitation of radium with barium can lead to reduced doses. The importance of further study of this was emphasized in the review of RD&D Programme 2007.

Newfound knowledge since RD&D 2007

The inventory of isotopes, including non-radioactive ones, has been determined by work reported in the framemark of the license application. The barium content can be estimated from this. The quantity of barium has a bearing on above all the estimation of co-precipitation of radium with barium, see Section 22.2.7.

Programme

A new literature review of co-precipitation Ra-Ba sulphate is being discussed, see Section 22.2.7.

22.1.5 Gas composition

Water in the canister's cavities occurs initially as vapour, which is why the variable "Water composition" is also treated in this section.

Conclusions in RD&D 2007 and its review

The quantity of radiolytically produced nitric acid can be limited by replacing the air in the canister with argon. With only very local areas with tensile stresses, the risk of stress corrosion cracking, which could reduce the strength of the canister, can be eliminated.

However, a remaining quantity of 600 grams of water in the canister can corrode the iron and produce hydrogen gas in a quantity that is not negligible. Investigation of the effects of hydrogen gas build-up on the material properties of the canister components was included in RD&D Programme 2007. See also Section 10.4.

The Swedish National Council for Nuclear Waste points out that time-dependent phenomena, such as the effect of hydrogen on nodular iron, must be explained.

Newfound knowledge and the programme concerning the effect of hydrogen on nodular iron are discussed in Section 23.2.2.

22.2 Processes in fuel/cavities

A number of processes will with time alter the state of the fuel and the canister's cavity. Some take place under all conditions, while many others are only possible if the isolation of the copper canister is breached and water enters the canister.

22.2.1 Overview of processes

The radionuclides in the fuel will eventually be transformed into stable substances by radioactive decay and nuclear fission. This process gives rise to alpha, beta, gamma and neutron radiation which, by interaction with the fuel itself and with surrounding materials, is attenuated and converted to thermal energy. The temperature in the fuel is changed by heat transport in the form of conduction and radiation, and heat is removed to the surroundings. The temperature change will lead to some thermal expansion of the fuel's constituents. This can, in combination with the helium formation caused by the alpha radiation, lead to rupture of the fuel's cladding tubes.

In an intact canister, radiolysis of residual gases in the cavity will lead to the formation of small quantities of corrosive gases, which could contribute to stress corrosion cracking (SCC) of the cast iron insert.

If the copper canister is not intact, water may enter the canister cavity, radically altering the chemical environment. Radiolysis of the water in the cavity will further alter the chemical environment. The water in the canister causes corrosion of cladding tubes and other metal parts in the fuel. If the cladding tubes' isolation should be breached initially or later by corrosion or mechanical stresses, the fuel will come into contact with water. This leads to dissolution of radionuclides that have collected on the surface of the fuel matrix and dissolution or transformation of the fuel matrix with associated release of radionuclides. The radionuclides may either be dissolved in the water, rendering them accessible for transport, or precipitate in solid phases in the canister void. This is determined by the chemical conditions in the canister cavity. On dissolution of the fuel, colloids with radionuclides may also form.

Radionuclides dissolved in water can be transported with mobile water in the canister (advection) or by diffusion in stagnant water. Colloids carrying radionuclides can be transported in the same way. Nuclides dissolved in water can be sorbed to the different materials in the canister. Certain nuclides can also be transported in the gas phase.

Finally, water can attenuate the energy of neutrons in the canister cavity. Low-energy neutrons can subsequently cause fission of certain nuclides in the fuel, releasing more neutrons. If conditions are unfavourable, criticality may be achieved, i.e. the process becomes self-sustaining.

Some of the fuel processes included in the SR-Site safety assessment are not judged to require any research programme. These processes are shown in Table 22-2.

The research programme for the other processes in the fuel is discussed in the following sections.

22.2.2 Radioactive decay

Conclusions in RD&D 2007 and its review

No research programme for this area was included in RD&D Programme 2007, and no direct viewpoints were offered in the review.

Newfound knowledge since RD&D 2007

The half-lives of the relevant radionuclides are generally well known. Se-79 is an exception, and its half-life has been discussed in the literature. The most recently published value is 3.77×10^5 years /22-14/. Previous estimates vary between about 1×10^6 years and 1×10^5 years, see for example /22-15/. The most recently published value of the half-life of Ag-108m is 438 years /22-16/. This differs only slightly from the value used in SR-Can (418 years).

Table 22-2. Fuel processes that are not judged to require any research programme. The processes are not commented on in the review of RD&D 2007.

Heat transport
Water and gas transport in canister cavities, boiling/condensation (see Section 23.2.4)
Thermal expansion/cladding failure
Advection and diffusion (see Section 24.3)
Residual gas radiolysis/oxygen formation (see Section 22.1.5)
Metal corrosion
Solution of gap inventory

Programme

SKB has no programme for studying half-lives, but keeps track of relevant research and updates the database when necessary.

22.2.3 Radiation attenuation/heat generation

Conclusions in RD&D 2007 and its review

It was observed in RD&D Programme 2007 that the consequences of increased burnup for the thermal evolution ought to be studied prior to SR-Site. No direct viewpoints on this were offered in the review.

Newfound knowledge since RD&D 2007

Work in the Fuel Line and in the SR-Site project has resulted in new calculations for the thermal evolution in and around the canister. Radiation attenuation and heat generation from the canister are limited by the requirement that no canister may emit more than 1,700 watts (W). The fuel content of the canister is then chosen based on this requirement. Heat generation in and around a deposited canister is modelled in SR-Site.

Programme

The field is judged today not to require any further research, development or demonstration. New developments are being monitored and will be acted on when appropriate.

22.2.4 Induced fission – criticality

Conclusions in RD&D 2007 and its review

In RD&D Programme 2007, new calculations were planned to examine the consequences of casting defects, and to study the criticality of MOX fuel.

In the review, SSI pointed out that SKB should determine what the scenario with buffer erosion and canister failure entails for the risk of criticality.

SKI was of the opinion that additional measures were required in the area to show that criticality due to changed geometry and redistribution of radionuclides is not an important process.

Newfound knowledge since RD&D 2007

The planned new criticality analyses, which include defective inserts and MOX fuel, have been carried out, see Section 10.5.

The calculations show that k_{eff} is below 0.95 inside a water-filled canister for all types of spent fuel. The calculations also show that an unirradiated modern PWR fuel bundle in a water-filled canister results in a k_{eff} above 0.95. Since all fuel that will be placed in the canisters for disposal will be spent, i.e. will have been exposed to irradiation, it is more realistic to allow credit for burnup in these calculations. When a burnup credit has been allowed for, i.e. reduced quantity of U-235 and increased quantity of fission products, this results in a k_{eff} below 0.95 even for PWRs, except at very low burnup.

The effect of a defective insert is a slightly increased k_{eff}, but the calculations show that k_{eff} is below 0.95 for spent fuel.

On water penetration the insert may be deformed and actinides may be transported and deposited outside the canister. The risk of criticality outside the canister has been judged to be very low due to the improbability of the courses of events that must be assumed in order for critical conditions to occur outside the canister /22-17, 22-18, 22-19/. This has recently been tested again in /22-20/, which evaluated the risk of criticality in Yucca Mountain and judged it to be negligible.

Programme

SKB, Posiva and Nagra plan a series of workshops to further investigate the consequences of changed geometry in the canister due to deformation and transformation of the insert. See also Section 10.5.

22.2.5 Water radiolysis

Radiolysis of water in the canister's cavities creates oxidants, which have an effect on corrosion of fuel and insert. The water chemistry in turn affects the quantity of dissolved oxidants.

Conclusions in RD&D 2007 and its review

RD&D Programme 2007 described the results of investigations of iron corrosion in initially oxygen-free water under gamma irradiation /22-21/. No further studies were included in the programme, and no direct viewpoints were offered by the regulatory authorities.

Newfound knowledge since RD&D 2007

Small quantities of dissolved hydrogen have been shown to reduce the concentration of produced oxidants in the water if the hydrogen concentrations in the water are above a certain threshold value /22-22/. Fuel leaching in the presence of metallic iron results in oxygen concentrations below the detection limit in the autoclave and formation of iron corrosion products such as magnetite and green rust, which are typical for strictly anoxic conditions /22-23, 22-24/. No radiolytic gas formation was observed in gamma-irradiated 5 M NaCl solutions containing 0.85 mM dissolved hydrogen /22-25/. Production of radiolytic gases started up again when bromide ions were added to the solution /22-26, 22-27/. These studies show that small quantities of hydrogen in the water keep down the quantity of radiolytically produced oxidants, and that bromide in the water can counteract this effect by reacting faster than the hydrogen molecule with the OH radical.

Programme

Studies of homogeneous alpha radiolysis at high pH and in the presence of bromide are planned at Chalmers.

22.2.6 Fuel dissolution

Conclusions in RD&D 2007 and its review

In RD&D Programme 2007, leaching tests were conducted with high-burnup fuel, as well as tests with alpha-doped material in the presence of reductants. The programme included collaborations with KTH, Chalmers and ITU for a better understanding of mechanisms and improvement of the modelling of the processes within the framework of the EU project MICADO.

In their review, the regulatory authorities find that SKB needs to conduct experiments and studies of high-burnup fuel in view of plans to gradually increase the average burnup of fuel at the Swedish NPPs. SKB needs to demonstrate fuel dissolution mechanisms by means of model studies. Furthermore, SKB needs to show that they have made a connection between the analyses of fuel dissolution and the repository's evolution, since e.g. buffer erosion can also affect the conditions for fuel dissolution.

Newfound knowledge since RD&D 2007

In view of the foreseen increase of the average burnup of the fuel in the future, a series of experiments was conducted with high-burnup fuel (average burnup higher than 50 MWd/kgU) under different conditions.

In high-burnup fuel, a HBU (High BurnUp) structure, or *rim*, is formed in the outer part of the fuel pellet, in which the grain size is smaller than one micron and the quantity of small pores (1–2 microns) is great. This structure is formed because neutron capture and Pu formation takes place in the outer part of the pellet, resulting in very high local burnup. The HBU structure is expected to have a higher dissolution rate than the rest of the pellet, due to the combination of an increased quantity of fission products and actinides, the porosity and the smaller grain size.

Experiments with leaching of fuel samples taken from different segments in a fuel rod with a burnup varying from 21 to 49 MWd/kgU /22-28/ showed a weak and almost linear increase of leaching (cumulative released fraction) up to 40–43 MWd/kgU, followed by a decrease.

This study has now been broadened with leaching data for four fuel segments with high burnup (55–75 MWd/kgU). The results for the whole burnup interval show that radionuclide release from the fuel does not increase proportionately with burnup /22-13/. The highest cumulative released fraction is observed for fuel of average burnup, followed by a slight decrease in release from the most high-burnup fuel pellets, see Figure 22-1.

Similar observations are made in two other systematic studies with fuel burnup in the intervals 29–60 MWd/kgU /22-29/ and 15–70 MWd/kgU /22-30/.

In fuel experiments conducted under the EU project NF-Pro, releases from the outermost part (about one millimetre, HBU structure) of a high-burnup fuel pellet (average burnup 65 MWd/kgU) were compared with releases from the central part of the pellet. The fuel samples from both parts of the pellet were in the form of powder with a similar size of the fragments. The leaching tests showed that releases of nearly all radionuclides were a little lower from the outermost part. Similar observations have been made previously /22-31, 22-23/, and have been interpreted as an effect of the larger quantities of fission products and other non-uranium atoms (dopants) present in high-burnup uranium oxide. The dopants reduce the dissolution rate of the fuel due to a surface effect (when the dopants are less soluble or have a lower dissolution rate) and a semiconductor effect, by altering the fuel's electron transfer capability and thereby reducing the oxidation of uranium.

A reduced dissolution rate of UO_2(s) due to dopants has also been observed in 1) experiments with oxidative dissolution of Gd-doped UO_2(s) /22-33/, 2) an electrochemical and spectroscopic study /22-34/ of simulated fuel (SIMFUEL) and 3) dissolution of natural uraninite (22-35/. Thermochemical measurements of UO_{2+x} containing different amounts of di- or trivalent oxides /22-36/ show that they are much more difficult to oxidize and/or dissolve than pure UO_2 due to the increased stability of the lattice which results from the energetic contribution of the dopants.

High-burnup fuel has also been tested under reducing conditions /22-37, 22-38/. The actinide concentrations were very low and in principle constant throughout the experiment, while the release of Cs decreased with time.

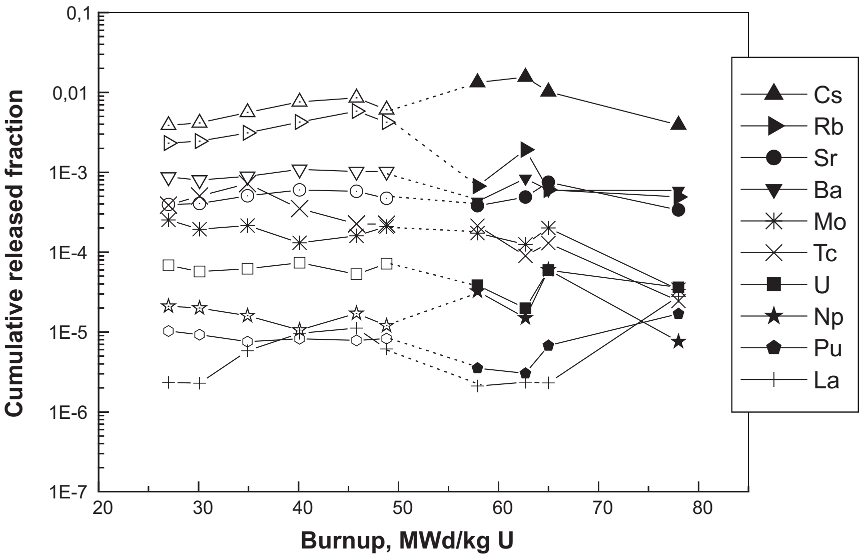


Figure 22-1. *Cumulative released fraction from high-burnup fuel /22-13/ (filled symbols) for a cumulative time of 182 days (four contact periods) compared with similar data from lower-burnup fuel /22-28/ (open symbols).*

The initial results from leaching of MOX fuel under reducing conditions /22-39/ show similar behaviour as in experiments with UO_2 fuel under a hydrogen gas atmosphere. Experimental data indicate less release from areas in the MOX fuel that contain Pu agglomerate and have a higher burnup than the rest of the UO_2 matrix.

A compilation has been made of new experimental studies of leaching of spent fuel and ^{233}U-doped UO_2 in the presence of 0.05–43 mM dissolved hydrogen, where the combined effect of actinide oxide surfaces, adsorbed water and radiation has been discussed /22-40/.

In leaching tests with spent fuel, the dissolution rate can be judged by analyzing releases of non-redox-sensitive fission products. A systematic reduction by more than two orders of magnitude of the released fraction of Sr or Cs during different time intervals was observed during the more than one-year-long experiments. With longer leaching periods, the number of intervals where no releases of Sr or Cs could be measured increased. The concentrations of nearly all redox-sensitive nuclides from a pre-oxidized fuel layer decline with time to values equivalent to the solubility of their reduced oxides. In certain cases the concentrations of actinides, such as Np and Pu, are much lower than the uranium concentrations and the actinide/uranium ratio is fairly close to that in the fuel, even though the solubilities of their tetravalent oxides are relatively similar /22-40, 22-38, 22-39. This indicates a possible co-precipitation of uranium with Np and Pu.

Fuel leaching tests that have been running for 1–3 years in sealed glass vials with Ar atmosphere show that the rate of fuel dissolution declines and approaches zero when the concentration of produced hydrogen reaches an order of magnitude of 10^{-5}–10^{-4} M. These data could be interpreted and modelled by including the effect of metallic particles on the hydrogen reduction of surface-oxidized uranium /22-41/.

Under the conditions prevailing in a final repository for the first ten thousand years, radiolytically produced oxidants are expected to be consumed by the high concentrations of reductants (Fe(II) and hydrogen) that are formed by anoxic iron corrosion. The residual alpha activity in very old fuel, i.e. after the first ten thousand years, is expected to be so small that oxidants are not produced in sufficient quantity to cause a measurable oxidation of the fuel /22-42, 22-43/. A limit value for specific alpha activity of 18–33 MBq/g UO_2 in deionized water (which has a very low redox buffering capacity compared with the geological medium) has been proposed based on experimental data /22-44/. This means that in the case of fuel that is several tens of thousands of years old, the fuel dissolution rates used in SR-Site are pessimistic and equilibrium with UO_2(s) can be used instead.

In studies within the framework of the EU project NF-Pro, which was done with alpha-doped uranium dioxide (which simulates the radiation field from several thousand year-old fuel), solutions with one part per million (ppm) sulphide have been used. The experiments were performed both in the absence and presence of metallic iron. The results show no measurable effect of either alpha doping level or test time on measured uranium concentrations. The uranium concentrations that have been measured in solution or by analysis of iron surfaces at the end of the experiment are very low. Estimation of the matrix dissolution rate confirms previous results /22-45/. Experiments with corrosion of alpha-doped pellets under anoxic or reducing conditions in saline solutions (0.1–1 M NaCl and (NaCa)Cl solutions) show no observable effect of alpha radiolysis and a slightly reduced dissolution rate with increased ionic strength /22-46/.

The effect of the presence of hydrogen has been investigated in studies performed in an autoclave with UO_2 doped with 10 percent ^{233}U under different hydrogen concentrations, which showed U concentrations that declined with time and oxidant concentrations in the autoclave below the detection limit. At the end of the experiment, the UO_2 surface was analyzed by XPS (X-ray Photoelectron Spectroscopy), which showed that the UO_2 surface was reduced /22-47, 22-48/.

Studies under an Ar atmosphere carried out with relatively low alpha activity showed a measurable but slow increase in the concentration of U. Studies of UO_2 doped with much higher alpha activity have recently been carried out. A UO_2 pellet with a very high doping level (385 MBq/g UO_2, equivalent to approximately 50 year-old fuel) showed a clear effect of alpha radiolysis under an Ar atmosphere, with U concentrations that increased rapidly in carbonate solutions. The same system was tested under one bar H_2, which led to a slight reduction of the U concentrations with time, instead of an increase /22-43/. During the EU project NF-Pro, experiments were also conducted with powder of highly doped UO_2 (285 MBq/g, equivalent to about 150 year-old fuel). Relatively high concentrations of U were measured in the solution at the start, but they declined slightly with time under the hydrogen atmosphere /22-49, 22-50/.

A number of studies have investigated the mechanism behind the effect of hydrogen gas on fuel dissolution, which has been linked to the catalytic effect of metallic ε particles (containing the fission products Mo, Pd, Tc, Rh and Ru) by activation of hydrogen on the fuel surface. The studies showed that the corrosion potential of SIMFUEL pellets decreased in proportion to the concentration of ε particles /22-51/ and the concentration of dissolved hydrogen /22-52/ to values well below the oxidation limit for $UO_2(s)$, indicating a complete inhibition of fuel corrosion. SECM (Scanning ElectroChemical Microscopy) was used to verify that these effects are due to a reversible splitting of dissolved H_2 on ε particles /22-53/.

The catalytic effect of metallic Pd particles on the reaction of hydrogen with hydrogen peroxide and uranyl ions was studied, and the results showed that the reaction is very fast, virtually diffusion-controlled and independent of the hydrogen gas pressure in the interval 1–40 bar /22-54, 22-55/. The influence of metallic particles on fuel dissolution has also been studied with UO_2 pellets containing different quantities of Pd particles /22-56, 22-57/. The results show that metallic particles catalyze both oxidation of UO_2 with hydrogen peroxide and reduction of surface-oxidized uranium with dissolved hydrogen. The results also show that the reduction of oxidized uranium on the uranium oxide surface, in the presence of hydrogen, is very fast and that only one ppm metallic particles would suffice to completely inhibit the dissolution of 100 year-old fuel under 40 bar H_2 /22-57/.

The effect of dissolved hydrogen on alpha radiolysis in bulk solution has also been modelled with respect to dose rate and presence of bicarbonate. The simulations show that the effect is strongly dose-rate-dependent and that the effect of hydrogen on radiolysis in solution is much less than the catalytic effect of metallic particles on fuel dissolution /22-58/. Metallic particles were extracted selectively from spent fuel with non-oxidizing reagent (phosphoric acid instead of nitric acid) in order to avoid dissolution of the most active metals and changes in composition. A solution with oxidized forms of redox-sensitive radionuclides such as Se(IV),Tc(VII), U(VI), Np(V) and Pu (VI) proved to be stable in the presence of a gas mixture (Ar with 10 percent H_2), but in the presence of metallic particles extracted from fuel the nuclides were rapidly reduced and precipitated from the solution. A slightly weaker catalytic effect, which was strengthened in the presence of β radiation, was noted for synthetic metallic particles /22-59/.

A model has been developed for radiolytically induced fuel dissolution that describes the geometric alpha and beta dose distribution based on the radionuclide inventory in spent nuclear fuel /22-60/. This model has been used to simulate radiation-induced dissolution of fuel when the surface concentration of hydrogen peroxide reaches almost a constant level (steady-state) due to production and consumption at the same rate /22-61/. The results of the calculations with the steady-state model have been successfully used to model experimental data from dissolution of 15-year old fuel /22-62/.

Studies of changes in the kinetics of the reaction between hydrogen peroxide, which is the main product of alpha radiolysis, and UO_2 suspensions have made it possible to determine how much hydrogen peroxide is consumed before the surface is saturated with oxidized UO_2. This provides a measure of the number of sites per unit surface area that are available for oxidation /22-63/. The influence of other factors, for example the solution's ionic strength, on the kinetics of UO_2 oxidation with hydrogen peroxide in the presence of different carbonates has also been studied /22-64/. Gamma and electron radiation were used to produce nanoparticles of uranium oxide, which showed high activity with regard to hydrogen peroxide consumption, mostly via catalytic degradation at the oxide surface /22-65/.

A study of the impact of different reactive species, such as Fe(II) or chloride ions, on the radiolytic dissolution of UO_2 was carried out in the presence of other gas mixtures. The results could be modelled with previously determined rate constants and show, among other things, that the presence of Fe(II) ions reduces the dissolution rate of $UO_2(s)$ /22-66, 22-67/. These experimental studies have been used to construct a model for fuel dissolution under conditions prevailing in a final repository that takes into account the relative importance of radiolytic oxidants and the influence of reactive species in groundwater, such as Fe(II) ions or dissolved hydrogen /22-68/.The results show that only 0.1 bar of hydrogen pressure is enough to completely inhibit the dissolution of 100 year-old fuel, while in the presence of 40 μm Fe(II), hydrogen gas from radiolysis alone is enough to stop the fuel dissolution.

Programme

Research activities are planned during the coming period both to obtain data on fuel dissolution under repository-like conditions and to shed light on the mechanisms of the different processes that contribute to fuel dissolution.

New tests are planned with an autoclave that will be placed in a hot cell for leaching of larger quantities of fuel than in previous leaching tests. Larger quantities of fuel increase sensitivity in the measurements. The tests will be performed under an H_2 atmosphere, and to avoid air contamination an autoclave with soft metal gaskets and other improvements has been purchased and prepared for use in a hot cell.

In view of the increase in average burnup for future fuel, leaching tests with high-burnup fuel (more than 50 MWd/kgU) are planned to be carried out under different conditions. Investigations of high-burnup fuel will continue, focusing on both the matrix dissolution rate and the release of the readily soluble fraction from the fuel. This also includes development of suitable analysis methods for analyzing nuclides such as Se or I in leachants.

New long-term tests in sealed glass vials, where fuel is leached in the presence of carbonate solutions, are being prepared and will be analyzed after extended periods in order to obtain new data and verify previous results.

Mechanistic studies of the dissolution of different UO_2 materials and spent fuel in the presence of deuterium instead of hydrogen is expected to provide a better understanding of the mechanism of the hydrogen effect. Studies of fuel dissolution under hydrogen with the addition of e.g. Cr(VI) in solution are expected to provide a better understanding of the reduction of oxidized radionuclides.

In order to investigate how fuel dissolution is affected by uranium sorption on bentonite particles in a damaged canister where bentonite has eroded, a study will be done with thorium oxide and bentonite slurry.

Studies that provide a better understanding of the mechanisms of the processes that take place during fuel dissolution and its modelling will continue during the period in cooperation with KTH. In cooperation with ITU, a mechanistic study has been initiated of the processes that take place at the fuel surface by investigating thin layers of actinide oxides with spectroscopic and electrochemical methods. In cooperation with Uppsala University, a theoretical modelling study has been initiated of the interaction between radiolytic oxidants and UO_2 surfaces.

Surface effects on measured dissolution rates will be studied in a collaboration with Stockholm University within the European network Delta-Min. The study aims at testing the hypothesis that measured dissolution rate is dependent on changes in the sample's surface morphology and the number of sites with higher dissolution potential (high energy sites). Linked to this, an international project is also planned in cooperation with VTT, Posiva, the University of Sheffield and Uppsala University.

22.2.7 Speciation of radionuclides, colloid formation

Conclusions in RD&D 2007 and its review

A literature review of co-precipitation of radium and barium was planned in RD&D Programme 2007. Studies of the redox kinetics of different oxidized forms of radionuclides on iron surfaces were to continue.

In its review, SKI considered it necessary that co-precipitation of radium with barium be studied further.

Newfound knowledge since RD&D 2007

In the scenario with a damaged canister, the redox conditions in the near-field are of very great importance. A research programme for studying the redox processes that are expected to occur in a damaged canister, especially their kinetics, has been under way for several years at SKB.

One study has shown that magnetite, which is the main corrosion product of iron under anoxic conditions, reduces oxidized U(VI) in solution to amorphous UO_2(s) at the corroded iron surface /22-69/. X-ray diffraction revealed the presence of UO_2(s), while XPS (X-ray Photoelectron Spectroscopy) confirmed the presence of U(IV). Sorption of U(VI) on magnetite surfaces has been studied by means of spectroscopic methods, including XPS and EXAFS (Extended X-Ray Adsorption Fine Structure), which showed that uranium sorbed on magnetite surfaces exists as both U(VI) and U(IV) /22-70/.

A study of the interaction between selenium and iron and iron corrosion products under anoxic conditions showed that both Se(IV) and Se(VI) can be immobilized by reduction as Se(0) and possibly FeS_2 on the iron surface /22-71/. Metallic iron reduces Se(IV) much faster than Se(VI), and the reduction proceeds faster with corroded iron. The effect of uranyl ions on reduction of Se(IV) on magnetite or iron surfaces has also been investigated. This study showed that the presence of uranyl increased the reduction rate of selenate at the same time as a reduction of uranyl occurred /22-72/. In another study, the metallic particles in spent fuel were simulated with Pd particles in UO_2(s), and the effect on reduction of Se(IV) in the presence of hydrogen was investigated /22-73/. The results showed that Se(IV) in solution was reduced to Se(0) by H_2 in the presence of Pd-doped UO_2(s).

Radium that is formed from uranium decay is expected to be one of the biggest contributors to the dose from a KBS-3 repository. The solubility of radium under deep repository conditions is determined by radium sulphate, but if co-precipitation of radium with the barium present in the fuel as Ra-Ba sulphate is taken into consideration, the Ra solubility decreases. A literature review has been made of the thermodynamics and kinetics of co-precipitation of Ra-Ba sulphate as applied to a KBS-3 repository /22-74/. Data from thermodynamics and kinetics show that a simultaneous release of Ra and Ba in sulphate-rich water causes co-precipitation of radium with barium as Ra-Ba sulphate. However, further kinetics studies are needed of how quickly radiogenic Ra is taken up by already precipitated barite ($BaSO_4$(s)). An experimental study of the kinetics of barite recrystallization showed that the process is rapid. The study of the kinetics of Ra uptake from solutions in two different barite crystals showed that the rate of uptake declines considerably over a period of 400 days /22-75/.

Programme

Studies of redox kinetics for different oxidized forms of radionuclides (including Se(VI), Tc(VII) Np(V) and Pu(V), Pu(VI)) on fresh and corroded iron surfaces will continue in cooperation with Studsvik, Nagra, ITU and PSI.

Studies of co-precipitation of Ra-Ba sulphate will continue during the coming period as well.

SKB continues to participate actively in the OECD-NEA project TDB (Thermochemical Data Base Project), where quality issues relating to the use of thermodynamic databases in safety assessments are discussed regularly.

22.2.8 Helium production

Helium nuclei from alpha decay in the fuel form gaseous helium that collects in gas bubbles. If the pressure in these bubbles becomes high enough, it can cause structural changes in the fuel. These structural changes can theoretically increase the fraction of relatively soluble radionuclides, which rapidly enter into solution on contact with water.

Conclusions in RD&D 2007 and its review

A possible need for further research was noted in RD&D Programme 2007. Since helium buildup takes place over long periods of time, this process is being studied by means of calculations and computer simulations.

Newfound knowledge since RD&D 2007

The consequences of helium buildup for the mechanical stability of the grain boundaries has been studied and discussed within the framework of the French PRECCI project (Research programme on the long term evolution of spent fuel waste packages, CEA) /22-76/. Continued work in this field has led to the conclusion that the helium buildup does not cause increased cracking in the fuel. Helium buildup will therefore not contribute to an increase of the instant release fraction /22-4/.

Programme

The field is judged today not to require any further research, development or demonstration. New developments are being monitored and will be acted on when appropriate.

23 Canister

23.1 Initial state

The initial state of the canister describes the properties which the canisters are expected to have when they have been emplaced in the deposition holes and will not be handled anymore in the final repository. The requirements and design premises related to the canister's barrier function in the final repository are described in Section 11.1 and in /23-1/.

23.1.1 Variables

In the assessment of long-term safety, a number of variables are used whose values vary with time in response to the long-term changes in the canister's shell and insert. The initial values of these variables can be taken from either information on properties of fabricated and inspected canisters or other documented information, see Table 23-1.

The values of the variables that define the initial state of the canister can be ascertained from the reference design and relate e.g. to the thickness of the copper shell (variable: geometry) and its oxygen content (variable: material composition). The initial radiation on the canister surface (variable: radiation intensity) also describes a part of the initial state. All initial values of the variables describe the properties of the canister and are closely linked to the design premises (Section 11.1). They are defined in such a way that the canister will withstand the loads (for example swelling pressure from buffer) that occur initially and could conceivably arise in the repository.

The canister must prevent criticality, which means that the design of the insert must be such that criticality cannot occur even if water has entered the canister. This relates to the variable "geometry" as well as to material composition, since certain elements that occur in nodular iron reflect neutrons better than iron. The temperature of the canister material affects the mechanical properties of the canister, which means that a maximum temperature is also defined in the canister's initial state.

Continued development work on canister design is described in Chapter 11. The research efforts that are planned regarding analysis of the initial state involve continued development of FEM (Finite Element Method) models for analysis of mechanical loads (see further Section 11.3). The research efforts that relate to the processes that affect long-term changes in the canister, for example deformation and corrosion, are discussed in Section 24.2.

Table 23-1. Variables for copper shell and nodular iron insert and related properties (design parameters), plus source where information on the variable is provided.

Variable	Definition	Comment
Geometry	Geometric dimensions of the canister components. This also includes a description of possible defects from fabrication, welding and the like.	Reference canister and initial state, Chapter 11.
Radiation intensity	Intensity of alpha, beta, gamma and neutron radiation as a function of time and space in the canister components.	Calculation of radiation intensity, Sections 11.1.1 and 11.1.3.
Temperature	Temperature as a function of time and space in the canister components.	Calculation and future inspections of temperature on the canister surface, Section 11.1.4.
Mechanical stresses	Mechanical stresses as a function of time and space in the canister components.	Residual stresses from fabrication of the canisters are of secondary importance compared to the mechanical stresses that arise in the components in the final repository.
Material composition	Material composition of the canister components, including possible corrosion products.	Reference canister and initial state, Chapter 11.

23.2 Canister processes

23.2.1 Overview of processes

Some of the radiation that penetrates out to the canister is converted to thermal energy by attenuation in the canister materials. Heat transport takes place by conduction within the insert and canister, and to a large extent by radiation between these two parts.

The insert and the canister can be deformed mechanically by external loads. The creep properties of the copper are an important property, especially initially when the copper creeps and the gap between the copper shell and the insert is reduced. The canister can also be deformed by internal corrosion products if water enters the canister through a hole in the copper shell.

The chemical processes mainly comprise different types of corrosion. External copper corrosion is a crucial factor for the integrity of the copper shell. Stress corrosion cracking is a possible process both in the copper shell and in the cast iron insert. Corrosion of cast iron is primarily of importance if water enters the canister through a hole through the copper shell. The materials can also be altered by radiation.

Transport of radionuclides in the near-field can take place via advection and diffusion in the aqueous phase or in the gaseous phase. Radionuclide transport can also be affected by sorption and solubility limitations.

The research programme for the different processes in the canister is dealt with in the following sections. The subdivision into processes follows the process description in the SR-Site safety assessment.

Some of the canister processes included in the SR-Site safety assessment are not judged to require any research programme. These processes are shown in Table 23-2.

23.2.2 Deformation of cast iron insert

Chapter 11 describes how the canister's reference design has been arrived at and evaluated for different loads, based on the requirements made on the canister. The basis for this is found in part in the design analysis for the canister. The canister's reference design is closely linked to the initial state of the canister, see Section 23.1.

Only creep in cast iron and the effect of hydrogen on material properties in cast iron are dealt with in this section.

Conclusions in RD&D 2007 and its review

No research programme for creep in cast iron was presented in RD&D Programme 2007.

The programme included investigation of the effects on the material properties of the canister components of hydrogen gas buildup from cast iron that corrodes due to residual water.

The Swedish National Council for Nuclear Waste points out that the creep properties of nodular iron need to be explained, along with other time-dependent phenomena, such as the effect of hydrogen on nodular iron.

Newfound knowledge since RD&D 2007

Cast iron from trial fabrication of inserts has been creep-tested at room temperature, 100°C and 125°C. The results of this testing demonstrate that the nodular iron exhibits logarithmic creep in the same way as unalloyed steel at these temperatures. Despite the fact that the stress on the test bars was of the same order of magnitude as the yield stress, the greatest creep deformation measured is only 0.025 percent after 10,000 hours (more than a year). Logarithmic creep entails that the creep strain between 100 and 1,000 hours is equal in size to that between 1,000 and 10,000 hours, and for each subsequent increase of the time interval by a factor 10. Even with the safety assessment's timescale of a million years, the creep in cast iron has a negligible effect on the properties of the canister. These results are being reported within the framework of SR-Site.

Table 23-2. Canister processes not judged to require a research programme. The processes are not commented on in the review of RD&D 2007.

Radiation attenuation/heat generation
Heat transport
Thermal expansion
Galvanic corrosion
Stress corrosion cracking of cast iron insert
Earth currents – stray current corrosion

Methods have been devised for hydrogen charging of cast iron from trial-fabricated inserts, whereby electrochemical charging is more suitable than thermal charging. Preliminary results from both tensile and creep testing show lower strength and ductility with increasing hydrogen content. However, these results are obtained from testing of short duration. The hydrogen concentration declines rapidly in the hydrogen-charged samples, which is a problem in long-term testing.

Programme

The work with creep in cast iron will continue with supplementary testing to obtain a more uniform body of data.

The studies of hydrogen-charged cast iron and its material properties will continue, primarily with analyses of what concentrations can be built up in the iron, and in particular by relating these concentrations to the quantity of hydrogen that can theoretically be formed due to corrosion by any residual water in the canister.

23.2.3 Deformation of copper canister under external pressure

The research on the creep properties of copper is reported in this section. Deformation of the copper by external pressure at different loads is treated integrated with the cast iron insert, see Section 23.2.2.

Conclusions in RD&D 2007 and its review

The programme for creep testing was supposed to continue during the three-year period, with a focus on the properties of the FSW (Friction Stir Welding) weld, effects of extremely slow loading and creep with a multiaxial stress state. Studies of how creep properties are affected by deformation hardening due to the cold working that can occur during handling were also included in the programme.

In the review of RD&D Programme 2007, SKI was positively disposed towards the analyses of the copper shell with regard to creep, but urged SKB to carry out further studies with other creep models. In addition, they pointed out that interaction effects between plastic deformation and creep deformation should be taken into account in the analysis of the integrity of the copper shell.

The Swedish National Council for Nuclear Waste called for a validated creep model so that different time-dependent processes can be better taken into account, such as possible shearing of the canister. The influence of phosphorus on the creep strength of pure copper, and the long-term stability and role of phosphorus must also be further investigated. Cold working effects must by analyzed with respect to creep properties.

Newfound knowledge since RD&D 2007

Both creep testing and model development have continued during the period /23-2, 23-3/. As far as modelling is concerned, the conclusions from 2007 on the effects of phosphorus /23-4, 23-5/ have in particular been included in new models /23-6, 23-7/.

To start with, only secondary creep was included /23-8/, but subsequently it has also been possible to include primary creep via an empirical model /23-9, 23-10/.

By means of slow tensile testing, stress-strain curves have been plotted for phosphorus-doped oxygen-free copper at temperatures between 20 and 175°C. From these, a fundamental creep model without fitting parameters has been constructed /23-11/. The model is based on an expression for the secondary creep rate. A direct relationship between creep and tensile testing with constant strain has been established.

The development of the models, along with an overview of the creep testing done at KIMAB (formerly the Swedish Institute for Metals Research) and KTH during the period 1984–2009, is presented in /23-12/. The report also contains an analysis of how uncertainties in the parameters in the creep models /23-6, 23-7/ affect the primary and secondary rate. Both of these models have been used in the modelling of creep deformation in the copper canister after deposition (in the design analysis of the canister), see Sections 11.3 and 23.2.1.

Studies are currently being conducted of cold working of copper and its effect on creep properties, and the first preliminary results for four types of cold working, both in compression and in tension, are presented in /23-13/. The effects that have been found are that creep ductility decreases (for both compression and tension) and that creep life increases and creep strength decreases due to cold working in the form of tension. The effect of cold working in compression is dependent on the compression direction. The area reduction is not affected in any case.

The importance of multiaxial stress states has been studied both experimentally and by modelling /23-14, 23-15/, whereby it was concluded that the phosphorus-doped oxygen-free copper is not sensitive to notches. Creep life can be estimated by FEM modelling of the steady-state creep stress.

Programme

SKB will continue investigating the creep properties of copper along several simultaneous lines of enquiry. Slow tensile testing will be done on FSW material for the purpose of obtaining fundamental equations for this material as well. The equivalent model devised for base material will be further developed so it can be used in FEM modelling of the canister under load.

The effect of cold working will be further studied, along with multiaxial stress states.

Another area where work is under way is the influence of hydrogen on the creep properties of copper. To start with, methods for controlled hydrogen charging of copper are being investigated, preferably by cathodic charging. Ab initio calculations (quantum mechanics modelling, "from the beginning") of solubility and diffusion of hydrogen in copper will also be carried out.

23.2.4 Deformation from internal corrosion products

Conclusions in RD&D 2007 and its review

The conclusion in RD&D Programme 2007 was that expansion caused by anaerobic corrosion in gaps does not occur, which is supported by both experimental studies and archaeological analogues. How the corrosion process proceeds when corrosion products gradually become compacted in the gap is being studied in miniature canisters installed in the Äspö HRL.

Newfound knowledge since RD&D 2007

The experiment with miniature canisters in the Äspö HRL (Minican) /23-16/ is continuing. The canisters have a diameter of 145 millimetres and a length of 315 millimetres, and have a design that essentially reflects the full-scale design. The shell consists of copper of the same grade as in the specification for the full-sized canisters, with a wall thickness of 7.5 millimetres. An insert of nodular iron is placed inside. The entire canister is then placed in a bentonite cage (two perforated cylinders of stainless steel with bentonite in between). The canister and the bentonite cage are galvanically isolated. Each canister in its bentonite cage is then placed in a borehole (diameter 300 millimetres) at about 450 metres depth in the Äspö HRL. The purpose of the experiment is primarily to study how water can get into the insert and how corrosion products, if any, develop with a hole in the copper shell. The miniature canisters therefore have holes drilled in the copper shell (one millimetre in diameter, 1–2 holes per canister). The canisters and the boreholes are also equipped with several sensors for measuring electrochemical potentials, which are used for calculation of corrosion rates, as well as devices for taking water samples.

The results so far show an increased concentration of iron and a decreased pH inside the bentonite, which could be due to microbial activity that affects the corrosion of iron and steel parts in the experiment. The measured potentials decline as oxygen is consumed, which is confirmed by analyses of dissolved gas.

Programme

The measurements on the miniature canisters in the Äspö HRL will continue. Planning has begun for retrieving at least one of the canisters in order to be able to compare the electrochemically measured corrosion rates with corrosion measured by weighing of the electrodes.

23.2.5 Radiation effects

Gamma and neutron radiation can affect the material properties of copper and cast iron.

Conclusions in RD&D 2007 and its review

In RD&D Programme 2007, SKB noted that the effects of radiation on the insert must be further studied, after calculations done by CEA had shown that radiation-induced segregations of copper in low-alloyed steel can give rise to embrittlement /23-17, 23-18/. Previous calculations have shown that the effects of radiation on copper are negligible /23-19/.

The Swedish National Council for Nuclear Waste points out that time-dependent phenomena, such as embrittlement of the nodular iron insert, must be explained.

Newfound knowledge since RD&D 2007

One study /23-20/ has been conducted, and the calculations on which the conclusions of the CEA studies are based have been verified. A review of the models that couple the precipitation of copper particles to the change in mechanical properties has, however, revealed shortcomings in the conceptual understanding. This does not alter the conclusions that can be drawn from empirical data, however. An initial analysis of the risk of radiation-accelerated diffusion of phosphorus to grain boundaries has also been done in the study. Segregation of phosphorus to grain boundaries can cause intergranular embrittlement in ferritic steels /23-21/.

The requirement of maximum 0.05 percent copper in the cast iron has been introduced in the specification for the insert.

Programme

The theoretical studies will continue during the period, above all with calculations of solubility and diffusivity of copper in iron, plus studies of the mechanisms of precipitation of copper at low temperatures, especially in interaction between copper clusters and vacancies. Experiments with irradiation of cast iron are planned to validate the models used.

23.2.6 Corrosion of cast iron insert

If water is in contact with the cast iron insert, anaerobic corrosion can occur. Water may remain inside the insert (in a very limited quantity, since the fuel is dried before being placed in the canister) or penetrate in to the insert if a breach occurs in the copper shell. There may also be air left in the canister, although in a very limited amount (see Section 11.1.3).

Conclusions in RD&D 2007 and its review

SKB has studied interactions between bentonite and corroding iron in the EU project NF-Pro. One of the conclusions was that the presence of bentonite increased the corrosion rate slightly, but reduced the volume of the corrosion products, compared with iron corrosion without bentonite. Raman spectroscopy showed that the corrosion products consisted of an inhomogeneous mixture of magnetite, hematite and goethite, where magnetite dominated. The experimental studies of the interaction between bentonite and iron were to continue during the period and be supplemented by geochemical modelling.

Gamma radiation can increase the corrosion rate, but in an intact canister the total corrosion is nevertheless limited by the quantity of water. The results that were presented in RD&D Programme 2007 have now also been published in a scientific article /23-22/.

Newfound knowledge since RD&D 2007

The continued studies of samples from NF-Pro have primarily investigated how the bentonite is affected by corroded iron, and have confirmed previous results showing that the bentonite's cation exchange capacity is reduced and its swelling capacity decreases /23-23, 23-24/. Detailed petrographic observations found no evidence for the formation of discrete iron oxide or iron oxyhydroxide phases, but showed that the clay particles had become enriched in iron, see also Section 24.2.17. The geochemical modelling studied iron transport in the bentonite and used the measured production of gas as a measure of iron corrosion /23-25/. Since ion exchange and sorption processes were not sufficient to explain the iron concentration in the bentonite, formation of solid phases or mineral transformations were proposed as partial explanations.

Programme

The field of corrosion of the cast iron insert is judged not to require any further research, development or demonstration.

23.2.7 Corrosion of copper canister

Conclusions in RD&D 2007 and its review

It was noted in RD&D Programme 2007 that research on corrosion of copper would continue within several subareas.

The mechanisms and kinetics of copper corrosion in anaerobic chloride and sulphide solutions were studied electrochemically to develop a corrosion model. The goal was to be able to use it in the SR-Site safety assessment. The transformation of an oxide film in a sulphide solution was studied as a basis for this. The properties of pure copper surfaces and copper surfaces with different surface films were studied with regard to sorbing and chemically bound carbonate and chloride species.

The goal during the period was also to obtain quantitative data on sulphide formation and diffusion in order to be able to include microbial activity in the safety assessment instead of, as before, excluding microbial sulphide formation due to environmental factors.

Posiva was in the process of updating SKB's and Posiva's joint compilation on copper corrosion.

In the review of RD&D Programme 2007, SKI said that SKB needs to gather updated information of relevance to the issue of copper corrosion in oxygen-free water, based on both experiments and theoretical calculations. The link to the issue of hydrogen embrittlement should also be studied.

SKI further believes that SKB needs to supplement its programme for corrosion by corrosion experiments on copper surfaces directly exposed to groundwater. The impact of microbial processes on copper corrosion needs to be further studies for cases both with and without bentonite. SKI considers that further studies of the buildup of sulphide films on the copper surface are urgent.

SKB should also consider broadening the account to make better use of existing knowledge concerning different forms of copper in nature.

SSI is of the opinion that uncertainties remain in the understanding of the microbial activity in the near-field in the repository environment and its importance for copper corrosion processes. The programme should take a broader approach to shedding light on the interaction between different microbes and the limitations defined by available energy sources and nutrients.

SSI believes that it is too early to dismiss the risk of corrosion in oxygen-free water and its impact on the canister's integrity. Since the process may potentially be of great importance for the repository's function, SKB should gather material to permit it to decide whether the process needs to be included in future safety assessments.

The Swedish National Council for Nuclear Waste believes that continued corrosion studies are required in different areas: accelerated long-term stress corrosion cracking experiments, general corrosion in chloride- and sulphide-containing water with bentonite, and microbial corrosion. Further, mechanisms of copper corrosion in oxygen-free water must be investigated experimentally to determine whether corrosion of copper by hydrogen evolution can take place in pure, deionized, oxygen-free water and in groundwater with bentonite.

Newfound knowledge since RD&D 2007

The study of mechanisms for sulphidation of copper oxide films has continued /23-26/, and has confirmed the previous conclusion regarding growth of copper sulphide film. Film growth takes place under sulphide transport control when the system approaches stagnant conditions. Growth of the copper sulphide film can proceed according to two different mechanisms. The initially formed film grows rapidly via an ion transport process. If the film remains intact, further growth takes place very slowly. If stresses in the film cause cracks to form in the film, film growth continues with the formation of a thick, porous film. In the presence of chloride, the copper sulphide film is destabilized and further growth similarly produces a thick, porous film.

In order to shed light on the properties of copper oxide films, thin copper films in contact with artificial groundwater and chloride, sulphate and hydrogen carbonate ions in aqueous solution have been studied in situ /23-27, 23-28/. Changes in the copper oxide films were observed by X-ray Absorption Spectroscopy (XAS) and Resonant Inelastic X-ray Scattering (RIXS). As expected, the corrosion rate increased at high chloride concentrations, while hydrogen carbonate in the solution increased the tendency towards passivation. In contact with groundwater, $Cu(OH)_2$ was formed as a corrosion product /23-28, 23-29/.

The question of corrosion of copper in pure, oxygen-free water has attracted attention in recent years. The overwhelming conclusion in the established literature is that copper does not react with water under hydrogen formation under the potential and pH conditions that exist in the repository /23-30/. This conclusion has been called into question in /23-32, 23-33, 23-34/, which describe experiments where gas pressure has been measured and corrosion products analyzed primarily by SIMS (Secondary Ion Mass Spectroscopy). The results have been interpreted as evidence of corrosion of copper under formation of hydrogen gas and a stable compound, first expressed as Cu_xH_y, later described as an amorphous copper(I)hydroxide. Comments with alternative interpretations were made on the first of the three publications in /23-35/, which was in turn commented on by the authors in /23-36/. Viewpoints on the interpretation in the third article were also offered in /23-37/ with rebuttal in /23-38/.

A critical review was done by Fraser King of experiments cited both in support of and as a counterargument against the proposed mechanism. The conclusions in the report are that the evidence in support of the mechanism is in part incomplete and contradictory, and that other researchers have not succeeded in reproducing the results. Furthermore, it is concluded that even if the mechanism did exist, it would not have any serious impact on the life of the canister in the repository.

The stability of possible compounds of copper(I), oxygen and hydrogen has been studied /23-39/ with ab initio calculations (quantum chemical calculations of electron structures), so that thermodynamic data can be calculated as a function of the temperature. The conclusions in the study are that despite the fact that a number of possible structures for solid phases of copper oxyhydride (Cu_4H_2O) and copper hydroxide (CuOH) have been investigated, no phase that is more stable than Cu_2O has been identified, which speaks against the idea that a stable copper-oxygen-hydrogen compound is the driving force behind corrosion.

Corrosion of copper in oxygen-free water was discussed at a scientific seminar arranged under the auspices of the Swedish National Council for Nuclear Waste in November 2009. The seminar discussed both our fundamental understanding of the process and the question of whether enough evidence exists to judge how important this process is or could potentially be in the final repository for spent fuel. The conclusion from the seminar states that the key question is what the partial pressure of hydrogen gas is at equilibrium in the reaction between copper and water. Thermodynamic calculations indicate an extremely low hydrogen gas pressure, which is the basis of the general standpoint that copper is resistant to corrosion in pure, oxygen-free water. If, on the other hand, another, as yet unknown reaction product exists, it must first be identified and characterized before it can be included in the thermodynamic calculations. Only then can its effect be judged. Furthermore, the corrosion rate of the reaction must be determined, and corrosion in a realistic final repository environment needs to be studied. The members of the expert panel also said that continued research is necessary to clarify the experimental results and the analytic methods used by Szakalos and Hultquist /23-40/.

Copper samples with FSW-welded material have been studied electrochemically /23-41/ to investigate whether a potential difference between the weld material and the base material could result in galvanic corrosion, which had been noted in the previous study of grain boundary corrosion /23-42/. The question of whether metal particles from the FSW tool could induce and sustain corrosion was also investigated in the study. The results of the experiments, which were carried out in a chloride environment, indicate very small galvanic currents between weld material and base material, and of a varying direction. Surface effects on the untreated weld surface are probably the cause of the potential differences, but these differences in the properties of the surface layer decrease with time as the oxide layer is built up. The FSW tool is cathodic compared with the weld material, and small fragments of tool material would have a very small cathodic surface area compared with the weld material. Potentials were also investigated under aerobic conditions, but there was no risk of reversed polarity (the potential of the copper increased, while the potential of the FSW tool remained largely unchanged).

During transport and handling of the canister, the material may be subjected to cold working in the form of blows or impacts. Locally, small areas of the copper might therefore have different corrosion properties. Potential decreases of 90–200 mV were measured in the studies of weld material /23-41, 23-42/, but they did not give rise to galvanic corrosion. Potential decreases of 7–12 mV have been measured on copper subjected to plastic strain /23-43/. It is therefore highly unlikely that small cold-worked areas on the canister surface would cause galvanic corrosion.

The update of SKB's and Posiva's joint compilation of copper corrosion is nearly finished.

Several studies have been conducted in order to gather data on sulphide formation and diffusion. The presence and survival of sulphate-reducing bacteria in commercial MX-80 bentonite was investigated in /23-44/, and the study showed that *D. africanus* can survive in a dormant, dried-out form in dry bentonite. When water is added, the bacteria are activated and can produce sulphide, at least up to a temperature of 40°C and a salinity of four percent. The bacteria could be activated even after a heat treatment of the dry bentonite at 100°C for 20 hours.

Further studies investigated the production of copper sulphide and resultant corrosion of copper in compacted bentonite of different densities (1.5–2.0 g/cm^3) and with different heat pretreatments /23-45, 23-46/. Copper coupons were placed in compacted bentonite, which was contacted by groundwater from 450 metres depth in the Äspö HRL. Radioactively marked sulphate (S-35) was added, along with different types of energy sources for the bacteria. Copper sulphide that formed on the copper coupons was examined by autoradiography, whereby the distribution of the radioactive sulphur could be studied. Using a diffusion model, diffusion coefficients were calculated for the sulphide in the bentonite and found to agree with data from the literature.

The results of these studies confirm previous results, that sulphide production decreases with increasing density and a corresponding increase in the swelling pressure /23-47, 23-48, 23-49/. Several factors have been discussed as a reason why it is difficult for bacteria to be active in compacted bentonite: high swelling pressure, low water activity and too little porosity for bacteria. Regardless of the exact combination of limiting factors, several independent studies show that only very limited sulphide production can take place in compacted bentonite. The analysis of the formed copper sulphide layer, together with the diffusion calculations presented in the most recent study /23-46/, show that the copper sulphide originated from sulphide formed outside the bentonite.

In the LOT experiment at the Äspö HRL, copper corrosion has been studied as a small part of the project. Analyses of the A2 test parcel revealed that corrosion products formed on the copper coupons mainly consisted of cuprite (Cu_2O) and paratacamite ($Cu_2(OH)_3Cl$) /23-50/. The average corrosion rate was found to be less than 0.5 μm/year, which agrees with the modelling that was done previously /23-51/ that conservatively estimated the corrosion rate under oxidizing conditions in the bentonite to be seven μm/year.

Real-time measurement of corrosion rates has been done by harmonic analysis, polarization resistance and noise resistance measurement on electrodes in the LOT test A2 parcel, but the different methods do not yield identical results /23-52/. Copper electrodes that have been mounted in situ have been transferred to the laboratory for further studies /23-53/. Electrochemical impedance spectroscopy has been used to further evaluate the corrosion measurements, but analyses of the actual electrodes are necessary in order to draw conclusions regarding the usefulness of the different electrochemical measurement methods /23-53/.

Programme

SKB's studies of corrosion on copper are continuing in order to provide additional knowledge of the details of the corrosion mechanisms.

Mechanisms of the reactions that can occur on copper surfaces and on copper oxides are being further studied by means of electrochemical measurements on copper in an oxygen-free environment and by means of quantum chemical calculations. The properties of copper(I)oxide are being further studied by means of ab initio calculations on bulk phases. Equilibrium reactions for the copper and water system are also being studied. Experimental studies have started, or will start, to measure gas evolution from copper. The experiments will in part be a repetition of the experiments in previously published studies.

The study of the properties of copper sulphide films is continuing, with a special focus on rate-determining steps in the growth of copper sulphide films.

The studies of microorganism activity in compacted bentonite have shown that there may be activity, but it is very limited. The interface between canister and buffer could possibly be a more favourable environment for microorganisms than the compacted bentonite, so a study of the possibility of formation of a biofilm of sulphide-forming bacteria on copper in compacted bentonite has started. In the study, copper pellets are placed in compacted bentonite of differing density, which then comes into contact with groundwater at 450 metres depth in the Äspö HRL together with different types of energy sources for the bacteria. Copper is a heavy metal and has an adverse effect on life processes at high concentrations, while at the same time copper is an essential metal that many microorganisms need in trace amounts. Titanium pellets are therefore also being studied for the sake of comparison.

In order to verify the corrosion properties of copper with cold-worked areas, measurements similar to those on welded material are planned.

The analyses of and with the electrodes from the LOT test will continue. Parts of the test will be interrupted and the electrodes analyzed with different surface analyses, while other electrodes will be kept in the bentonite and used for continued measurements, along with additional new electrodes.

23.2.8 Stress corrosion cracking of copper canister

Conclusions in RD&D 2007 and its review

RD&D Programme 2007 described studies of stress corrosion cracking in acetate-containing water, as well as a study of crack growth, concluding that the likelihood of cracking due to stress corrosion would be further studied.

In its review, SKI said that stress corrosion cracking could not be dismissed as a design-basis process in the repository. SKB should either show that even if a fracture is initiated, growth is so slow that the the integrity of the canister is not jeopardized, or describe the consequence of cracks caused by stress corrosion.

SSI made the judgement that the studies of individual bacteria can scarcely shed any light on the full extent of the role of the different microbial processes in canister corrosion. For example, the processes concerned with nitrogen circulation under increased concentrations of ammonia and nitrite may have a negative effect on stress corrosion cracking of the copper canisters.

The Swedish National Council for Nuclear Waste's conclusion was that continued corrosion studies are needed, specifically accelerated long-term stress corrosion cracking tests.

Newfound knowledge since RD&D 2007

A critical review of proposed mechanisms for stress corrosion cracking in copper is almost completed. The study describes the current state of knowledge regarding stress corrosion cracking in copper and copper alloys, and an assessment is made of the risk of stress corrosion cracking on the canister in a repository environment. A number of different proposed mechanisms have been examined in detail.

The conclusions in the study are that the probability of stress corrosion cracking under the initial oxidizing period is low due to the absence of necessary ions and that there is no well-substantiated mechanism for stress corrosion cracking under reducing conditions.

An experimental study with slow tensile testing of copper in sulphide-containing simulated groundwater has been presented /23-54/. Reduced ductility due to intergranular corrosion at a sulphide concentration of 0.01 moles per litre is reported in the study.

In the Canadian programme, stress corrosion cracking has been investigated in nitrite, ammonium and acetate environments in experimental studies /23-55, 23-56, 23-57, 23-58, 23-59/. The conclusions of these experiments are:

- except in the most aggressive environments, phosphorus-doped oxygen-free copper exhibits ductile behaviour in nitrite, ammonium and acetate solutions,
- only test bars with a clear oxide film exhibited cracking, and only at potentials and pHs that are consistent with stability in a Cu_2O film,
- chloride ions counteract stress corrosion cracking of copper,
- in nitrite solutions, susceptibility to cracking decreases with increasing temperature.

Calculations with the creep model that include both primary and secondary creep /23-9, 23-10/ show tensile stresses on the copper surface as well as parts where these stresses go through the whole copper thickness.

Programme

The research programme is continuing in the area of stress corrosion cracking for the purpose of further exploring the potential for stress corrosion cracking in copper.

An experimental study is being conducted to determine the conditions under which stress corrosion cracking can occur in a sulphidic environment. Preliminary results indicate difficulties in obtaining cracking to the extent Taniguchi and Kawasaki /23-54/ did.

In a joint project with Posiva, the behaviour of the copper material in simulated groundwater with ammonium is being studied. The studies are being carried out as slow tensile testing with potential monitoring (simulation of oxidizing environment in the repository) at different ammonium concentrations. Electrochemical measurements (impedance) will also be done.

23.2.9 Precipitation of salt on canister surface

Conclusions in RD&D 2007 and its review

The process was not specifically described as a canister process in RD&D Programme 2007.

Newfound knowledge since RD&D 2007

Precipitation of salts on copper surfaces has been studied in e.g. the LOT A2 test /23-50/. Salt precipitation could increase the concentration of chloride locally and thereby increase copper corrosion, but this effect is only noticeable at low pH, which is not the case in the presence of bentonite. Precipitates of calcium sulphate and carbonate are not electrically conductive, so the risk of local corrosion does not increase as a result of such precipitates /23-60/.

Programme

No research programme for the canister is planned in this area.

23.2.10 Radionuclide transport

Radionuclide transport in the canister's cavities and through a breached canister wall is dealt with (often pessimistically simplified) integrated with other processes. Radionuclide transport in the near-field is dealt with in Section 24.3.

24 Buffer and backfill

This chapter deals with research concerning the clay barriers SKB has developed for the final repositories. The backfill in the tunnels is now defined as a barrier in the KBS-3 concept /24-1/. The chapter also deals with the clay buffer in the silo in SFR 1. The requirements made on the buffer are described in detail in Part III. This chapter is concerned with the properties of the clay barriers that affect long-term safety.

24.1 Initial state

The initial state describes the properties of the engineered barriers when they have finally been put in place in the final repository. This is the starting point for the safety assessment and is described by the initial values of a number of variables, see Table 24-1. The initial state of some of the variables that are relevant to buffer and backfill is the result of the chosen reference design, fabrication and installation that have been arrived at based on the stipulated design premises, see Chapter 12. The expected initial state of other variables depends on how buffer and backfill evolve after installation and is determined by means of analyses. The research is focused primarily on the variables of the buffer, but the results can usually also be applied to the backfill.

When it comes to SFR, most of the bentonite buffer in the silo is already installed. In practice, this means that the initial state is already known. The information is needed in order to describe the long-term evolution of the repository.

The buffer's barrier function in the Spent Fuel Repository

The buffer's main purpose in the Spent Fuel Repository is to prevent flowing water from the rock from coming into contact with the canister and the spent fuel and thereby to ensure that transport of corrodants and radionuclides is dominated by diffusion. To achieve this purpose, the buffer's hydraulic conductivity must be low, the dimensions of the buffer must be maintained, and the buffer must be able to self-heal and be physically and chemically stable in a long time perspective.

Other important properties of the buffer relate to its possible effect on the canister in the Spent Fuel Repository. These properties are gas permeability, swelling pressure, deformability, thermal conductivity, colloid filtration and microbial activity. The requirements that are important from the perspective of long-term safety are compiled as design premises in Sections 12.1 and 13.1 as well as in /24-1/.

The variables described below are closely linked with the barrier functions and the performance indicators that are used to evaluate the performance of the barrier in the safety assessment. The values of the variables are changed by processes that take place in buffer and backfill after closure, see Section 24.2.

In order to achieve a desired barrier function, the water-saturated density, hydraulic conductivity, swelling pressure and temperature of the buffer must be ensured within certain limits, see further Section 12.1 and /24-1/. The research concerning buffer and backfill is therefore focused on the properties and processes that affect these variables.

Fabrication and installation of buffer are described in Chapter 12. Fabrication aspects are discussed in the following only insofar as they have a bearing on the presentation of the research programme for long-term buffer function.

Backfill in the Spent Fuel Repository

The backfill is necessary in order for the buffer and the rock to have the desired function. The requirements made on the backfill are described in Chapter 13. The most important functions of the backfill are to make the mass transport capacity comparable with that of the surrounding rock and to minimize the upward expansion of the buffer. Important properties of the backfill are thereby hydraulic conductivity and swelling pressure. The backfill may not have any adverse effect on the barriers in the repository, which imposes some requirements on its chemical composition.

Bentonite barriers in SFR

The use of clay barriers in SFR has more in common with the buffer in the Spent Fuel Repository than with the other barriers in SFR. The research programme for the bentonite in SFR is therefore presented together with the buffer in this section. The silo in SFR 1 is surrounded by a bentonite buffer that is placed between the concrete structure and the rock wall. The product name is GEKO/QI, and it is a soda-activated material. The functions and properties that are important for the silo buffer /24-2/ are similar to those of the buffer in the Spent Fuel Repository.

24.1.1 Variables

The variables that have been defined for buffer and backfill are presented in Table 24-1. These variables will be affected by the processes in the repository that ensue after closure. Work is being pursued to develop a similar set of variables for the silo.

24.1.2 Geometry

The geometry of the buffer is determined by the dimensions of the canister and the thickness of the buffer material required to obtain the desired function. The previously specified /24-3/ dimensions of 35 centimetres on the sides of the canister, 50 centimetres underneath the canister and 150 centimetres above the canister still apply to KBS-3V. The dimensions may be somewhat different for KBS-3H.

The buffer in the SFR silo consists of a bottom and a top bed plus a wall fill. See Figure 24-1.

Table 24-1. Variables for buffer and backfill.

Variable	Definition	Section
Geometry	Geometric dimensions of buffer and backfill. A description of e.g. boundary surfaces inward towards the canister and outward towards the geosphere.	24.1.2
Pore geometry	Pore geometry as a function of time and space in buffer and backfill. The porosity, i.e. the fraction of the volume that is not occupied by solid material, is often given. This variable also includes the density of the buffer.	24.1.3
Radiation intensity	Intensity of alpha, beta, gamma and neutron radiation as a function of time and space in buffer and backfill.	24.1.4
Temperature	Temperature as a function of time and space in buffer and backfill.	24.1.5
Water content	Water content as a function of time and space in buffer and backfill.	24.1.6
Gas contents	Gas contents (including any radionuclides) as a function of time and space in buffer and backfill.	24.1.7
Hydrovariables	Flows and pressures for water and gas as a function of time and space in buffer and backfill.	24.1.8
Load situation	Pressure as a function of time and space in buffer and backfill.	24.1.9
Bentonite composition	Chemical composition of the bentonite (including any radionuclides) in time and space in the buffer. This also includes impurities and minerals other than montmorillonite.	24.1.10
Montmorillonite composition	Chemical composition of the montmorillonite (including any radio-nuclides) in time and space in buffer and backfill. This variable also includes material sorbed to the montmorillonite surface.	24.1.11
Pore water composition	Composition of the pore water (including any radionuclides and dis-solved gases) in time and space in buffer and backfill.	24.1.12
Engineering materials	Composition of any engineering materials in the deposition holes. This no longer includes the bottom pad of cement, which is handled as a separate system part.	24.1.13

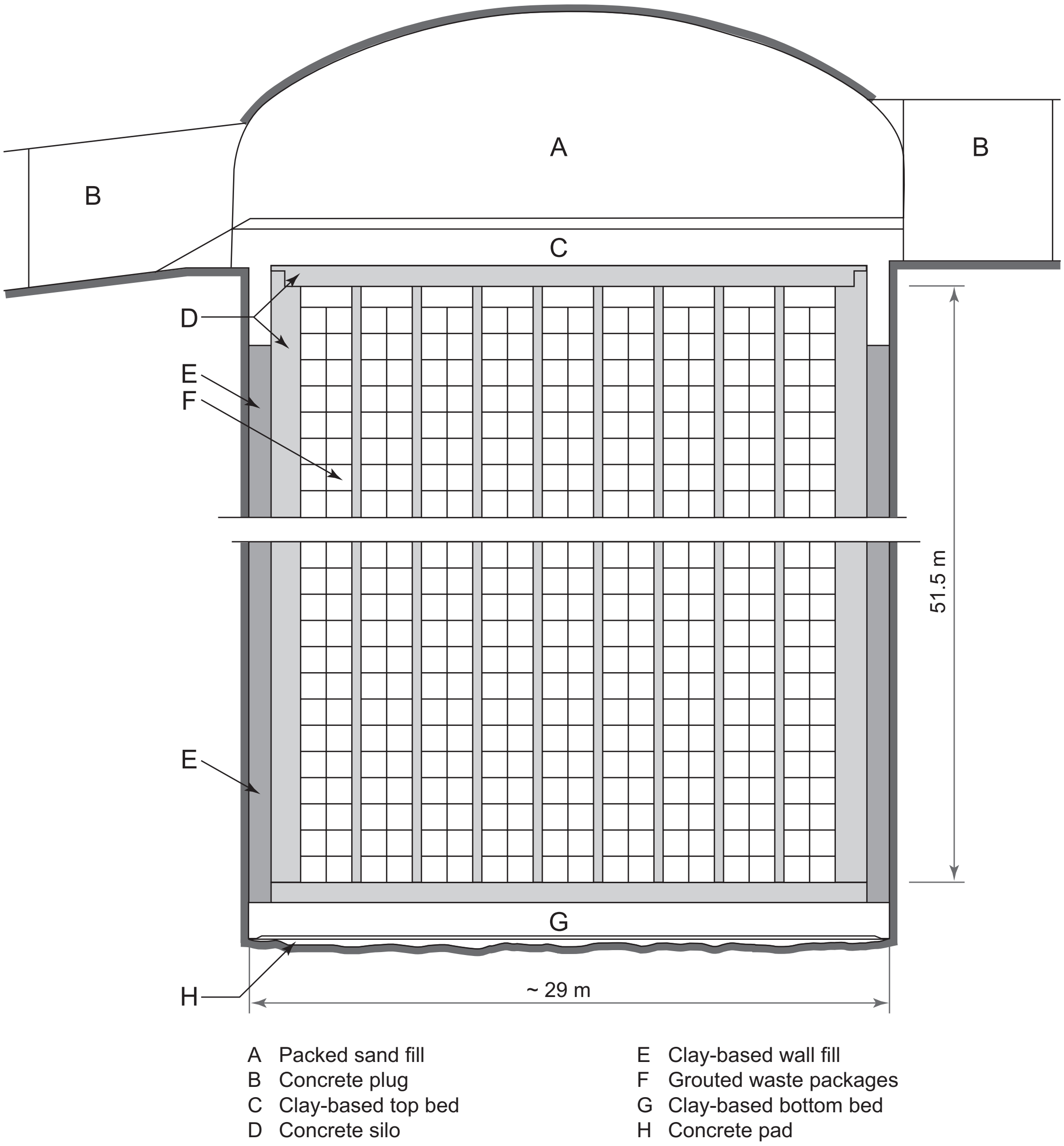


Figure 24-1. *Schematic section through SFR silo /24-2/.*

24.1.3 Pore geometry

In order to realize the defined buffer functions, there must be a specific counterion concentration the pore water. In a given buffer material, this concentration is controlled by the total water volume, which is determined by the porosity. The buffer's sealing properties, such as swelling pressure and hydraulic conductivity, are highly dependent on the porosity.

A design premise defined for the buffer in "Design premises" /24-1/ is that the buffer has a density in the saturated state of 2,000 ± 50 kilograms per cubic metre (kg/m^3). This is based on the properties of the reference material (MX-80), which has a grain density of 2,750 kg/m^3 and assumes that the water density is 1,000 kg/m^3. The buffer's dry density is thereby 1,570 ± 30 kg/m^3, which corresponds to a porosity of 43 ± 3 percent.

For an alternative buffer material, with the same montmorillonite content but another grain density, the density specification needs to be modified to achieve the same sealing properties. The relationship between porosity, density and grain density is described by simple geotechnical equations. Grain density has been determined for a large number of alternative buffer materials /24-4/.

The requirements on the wall buffer in the SFR silo include both low hydraulic conductivity to prevent water flow and a moderate swelling pressure so as not to damage the concrete structure. This imposes requirements on both a maximum and a minimum density in the material. The wall buffer has a dry density of around 1,000 kg/m^3 in the bottom 15 metres, 990 kg/m^3 in the interval 15–30 metres and 980 kg/m^3 at a height of 30–50 metres. This is equivalent to saturated densities of 1,650, 1,625 and 1,600 kg/m^3 /24-5/.

The bottom bed in the silo consists of a 10/90 mixture of bentonite and aggregate. It was installed with a dry density of 2,170 kg/m^3 (saturated density 2,370 kg/m^3). It is assumed that the top bed will have the same properties as the bottom bed.

24.1.4 Radiation intensity

The initial dose rate on the canister surface was calculated in SR-Can to be a maximum of 500 milligrays per hour (mGy/h). The radiation is dominated by the nuclide Cs-137. The dose rate is used to assess radiolysis of pore water and radiation-induced changes of the montmorillonite. However, the analyses in SR-Can show that the importance of these two processes is negligible.

24.1.5 Temperature

The buffer and the backfill are at ambient temperature at deposition. In Forsmark, the temperature in the rock is expected to be around 10°C. The temperature is dependent to some extent on the handling sequence, where the buffer blocks have been stored, heat from the deposition machine, season etc. An uncertainty of around 5°C is reasonable.

Determination of the initial buffer temperature is of trivial importance, in contrast to the heat transport in the buffer after deposition, see Section 24.2.3.

24.1.6 Water content

In SR-Site it is assumed that the compacted bentonite blocks have an initial water ratio of 17 percent, which gives a degree of saturation in the blocks of between 75 and 85 percent. The pellets in the gaps between the buffer and the rock are assumed to have an initial water ratio of 10 percent, which gives a degree of saturation of about 15 percent if the gap is not filled with water. The buffer-canister and buffer-rock gaps may be filled with water, but in SR-Site it is assumed that they are dry.

In the case of horizontal deposition according to the KBS-3H method, it is assumed that the water ratio is 10 percent in the rings around the canister and 17 percent in the cylindrical blocks, giving a degree of saturation of about 45 percent in the rings and about 75 percent in the cylindrical blocks.

The planned technology development related to the water content of the buffer is presented in Chapter 12.

24.1.7 Gas contents

The initial gas content is given by the water content and the porosity. The bentonite blocks have a degree of saturation of between 75 and 85 percent. This means that 75 to 85 percent of the pore volume is filled with water and the remainder with air. In other words, the gas content (counted as volume of gas divided by total volume of pores) is between 15 and 25 percent. The outer pellet-filled gap has a gas content of about 85 percent. The air in a deposition hole occupies approximately 16 percent of the total volume of buffer. The uncertainties in gas contents are not important for long-term safety.

24.1.8 Hydrovariables

The hydrovariables are water flow, water pressure, gas flow and gas pressure. Initially it is relevant to describe gas and water pressure. Flows do not occur initially in the buffer. At emplacement of canister and buffer, the deposition holes will be kept drained and the repository will be open to atmospheric pressure. This gives a gas pressure (air) of 1 atm (approx. 0.1 MPa) and a water pressure of 0.1–0 MPa in the surrounding host rock. There will, however, be an initial negative pore water pressure in the unsaturated bentonite blocks that drives the inward transport of water. This pressure is on the order of 40 MPa.

24.1.9 Load situation

Initially there is no swelling pressure. The initial pressures come from the weight of the overlying bentonite blocks and (for the bottom block) the canister. The swelling pressure develops when buffer and backfill come into contact with water.

24.1.10 Bentonite composition

Bentonite is the name of a naturally occurring clay that is rich in montmorillonite and varies in composition depending on how it is formed. Bentonite often occurs in several specific strata, which may vary in composition. In commercial products, such as MX-80, materials from different strata are often blended to meet specified quality requirements.

The buffer in the silo in SFR consists of approximately 6,000 m^3 of GEKO/QI, which is a sodium-converted calcium bentonite. There the requirement is that the montmorillonite content must be at least 65 percent.

Conclusions in RD&D 2007 and its review

SKI says that a requirement specification that only includes simple physical properties such as tightness and swelling pressure is insufficient. This is because bentonite has a complex and varying chemical and mineralogical composition which is important for the buffer's long-term evolution.

SKI says that SKB, besides specifying a minimum content of montmorillonite (which gives the bentonite favourable sealing properties), also needs to specify content intervals for other minerals that may have a bearing on long-term safety. It is, for example, well known that certain trace minerals are very important for the buffer's geochemical evolution. The Swedish National Council for Nuclear Waste comments that SKB should have a quality inspection of the bentonite that includes total mineral composition as well as the concentration of impurities.

SKI considers it positive that SKB has broadened its programme of testing of buffer materials. In connection with this, SKB should formulate a plan for which materials are to be included in new tests in order to make optimal use of resources.

Newfound knowledge since RD&D 2007

To meet the density requirement for the buffer, the montmorillonite content must lie in the range 75–90 percent. The smectite content (mostly montmorillonite) in commercial bentonites is normally 80–85 percent. It can be noted that virtually all bentonite deliveries which SKB has had analyzed lie within the latter range. Bentonite material for block fabrication will undergo comprehensive quality inspection prior to pressing, see Section 12.2.1, "Inspection and measurement methods". This inspection includes determination of the montmorillonite content.

All minerals beside montmorillonite are considered impurities in bentonite. They are also called accessory minerals. Usually, the impurities consist of minerals that are of little importance for the function of the repository (quartz and feldspar). Small amounts of e.g. amorphous silicon, calcite, pyrite, siderite pr gypsum may, however, be of some importance for the chemical evolution in the repository's near-field, see Section 24.2.18. The possibility that sulphide minerals (for example pyrite) in the buffer will influence canister corrosion cannot be excluded, and there is therefore a criterion for how much sulphide the buffer may contain. This criterion states that < 0.5 percent of the mass of the buffer may consist of sulphur in the form of sulphide. If the buffer contains carbon, sulphide can also be formed from sulphate by sulphate-reducing bacteria. A limit has therefore been defined on total sulphur and total organic carbon. Both of these limits are less than one weight percent.

As long as the requirements on montmorillonite and sulphur content as well as organic carbon content are met, there are currently no requirements on the other accessory minerals.

Previous studies /24-4/ have shown that there are small differences between different bentonites with regard to swelling pressure and hydraulic conductivity in the density intervals that are relevant for the buffer. It is, however, clear that different bentonites can have different properties in many other respects. They are not necessarily of any importance whatsoever for the bentonite's function in a final repository. However, SKB is conducting a programme to study the different properties of different types under relevant repository conditions. The work is being pursued in the project "Alternative Buffer Materials", which is presented in Section 24.2.17.

Programme

The programme with studies of alternative materials is continuing, mainly in the ABM project (Alternative Buffer Materials), which is presented in Section 24.2.17.

The characterization of GEKO/QI was done in the 1980s. It could be of value to conduct a new mineralogical investigation.

24.1.11 Montmorillonite composition

The mineral montmorillonite is characterized by nanometre-thin mineral layers with the ideal structural formula:

$M^{z}_{(x+y)/z}\ (Al_{(4-x)}\ Mg_{x})\ (Si_{(8-y)}\ Al_{y})\ O_{20}(OH)_{4}\ n(H_{2}O)$

where M represents positively charged counterions and z is the mean valence of the counterions. The sum of x and y can by definition vary between 0.4 and 1.2 units (charge per $O_{20}(OH_4)$), and x > y. A certain fraction of the aluminium (Al) can be regarded as exchanged for (Mg), and a smaller fraction of silicon (Si) is exchanged for aluminium (Al). The exchange of trivalent aluminium for divalent magnesium leads to a negative net charge in the mineral layers, which is balanced by the counterions (M). Other substitutions also occur in natural systems, for example iron can replace aluminium to some extent. A varying quantity of water molecules (*n*) can be incorporated between the mineral layers.

Conclusions in RD&D 2007 and its review

The programme included plans for further developing the structural formula of the swelling mineral, with a focus on determining the valence of the iron. Element analysis with a transmission electron microscope (TEM) was planned to statistically study the variation in the montmorillonite structure in a given buffer material.

Newfound knowledge since RD&D 2007

A relatively large number of bentonites have been investigated with XANES (X-ray Absorption Near Edge Structure) to determine the iron(II)/iron(III) ratio. The study was carried out at MAX-lab in Lund and will be published in 2010. TEM studies have been carried out in the LOT project to determine the distribution of elements in the montmorillonite, see Section 24.2.17.

Programme

An investigation of statistical variation with respect to layer charge, ionic species distribution and layer size is planned for montmorillonite from several different bentonites.

24.1.12 Pore water composition

When water is taken up in the bentonite, it will preferably hydrate the montmorillonite's counterions, see Section 24.1.11. It is thereby not possible to distinguish counterions from other positively charged ions in the pore water. The actual ion content of the pore water thus depends on the montmorillonite's counterions and on the solubility of the accessory minerals. In principle it is possible to determine the activity of the ions by equilibrating the material with an external solution (mechanically separated, for example via a metal filter). However, at equilibrium there are great differences in the concentration of all ions between the pore water and the external solution as a consequence of the fact that the montmorillonite's layer charge must be compensated for. See further Section 24.2.14.

At delivery the water ratio will be no more than 12 percent according to today's specification. Before the bentonite is pressed into blocks, deionized water is added to achieve a water ratio of 17 percent. The buffer material will be analyzed with respect to constituent minerals, as well as ions in the supernatant in dispersed material (the aqueous solution above a slurried and centrifuged sample). This quantification provides an opportunity to calculate the initial composition of the pore water.

The programme related to pore water composition is described in Section 24.2.18.

24.1.13 Engineering materials

At the present time it is not expected that any engineering materials will be left in the deposition holes. However, a bottom slab of cement and copper is still planned as a bed at the bottom of the holes, see Section 12.2.1. This is treated as a separate system part in SR-Site.

24.2 Processes in buffer and backfill

24.2.1 Overview of processes

Buffer and backfill

The processes that are expected to influence the evolution of the buffer after closure can be divided into radiation-related (R), thermal (T), hydraulic (H), mechanical (M) and chemical (C). A compilation of the processes is presented in Table 17-1. The chemical evolution in buffer and backfill is determined by a number of transport and reaction processes.

Since the backfill, like the buffer, will consist of highly compacted blocks of 100 percent bentonite with a gap towards the rock that is filled with bentonite pellets, all processes will be very much like those in the buffer. The biggest difference compared with the previous concept with an in-situ-compacted bentonite/mixture is that a number of critical processes such as self-healing of erosion pipes, compressibility on swelling of buffer, hydraulic conductivity at high salinities and freezing are less critical due to the fact that density and bentonite content are higher. However, the difference between the properties (above all compressibility) in dry and water-saturated conditions is greater than for in-situ-compacted mixtures.

Four different materials have been used to describe possible properties and models of different backfill materials. These materials are Friedland Clay, Asha 230 (an Indian sodium-dominated bentonite with about 80 percent montmorillonite), IBECO RWC-BF and MX-80. For further description of these materials see /24-6, 24-4/.

The research programme for the different processes in buffer and backfill is dealt with in the following sections. Many processes in buffer and backfill are coupled and need to be studied integrated. They are described in Section 24.2.11.

Silo buffer

During the filling phase, the rock cavern around the silo in SFR is kept drained and water uptake in the buffer will be insignificant. The bottom bed is compressed as the silo is filled. The top bed is installed at closure. When the drainage is shut off, water saturation begins in the buffer.

When the buffer is water-saturated, full swelling pressure is exerted against the concrete structure. This pressure can increase when the groundwater becomes more diluted. Calcium from the concrete will be exchanged for sodium in the bentonite. The pH of the concrete pore water will affect the montmorillonite in the bentonite, which means that new minerals will be formed. The silo may freeze under permafrost conditions.

24.2.2 Radiation attenuation/heat generation

Gamma and neutron radiation from the canister are attenuated in the buffer. The magnitude of the attenuation is dependent above all on the density and water content of the buffer. The result is a radiation field in the buffer that can lead to radiolysis of water and have a marginal impact on the montmorillonite. The radiation that is not attenuated in the buffer penetrates out into the rock. Our understanding of this process is deemed to be sufficient for the needs of the safety assessment.

24.2.3 Heat transport

Heat transport is an important process in the buffer that affects the temperature evolution in the near field. The process is of less importance in the backfill and in the silo buffer, so the following text focuses on the buffer.

Conclusions in RD&D 2007 and its review

RD&D Programme 2007 described the connection between the temperature criterion for the buffer and the thermal design of the Spent Fuel Repository, as well as the plans for continued work with the simulations of the Thermo-Hydro-Mechanical (THM) processes. In its review of RD&D Programme 2007, SKI asked for a better justification for the temperature criterion of 100°C, since the temperature is an important factor for many processes in the buffer. SKI points out that a thorough analysis of the thermal evolution requires good knowledge of links to the hydraulic and rock mechanical evolution.

Newfound knowledge since RD&D 2007

The calculation model for temperature evolution in buffer and rock has provided thermal design criteria for tunnel and canister spacing. The results are presented in the so-called dimensioning (thermal design) report /24-7/. The thermal design is based on the hypothetical case that all canister positions are dry, so that no credit is taken for any increase in the thermal conductivity of the bentonite blocks or the pellet-filled gap due to water uptake during the roughly ten years between the time of deposition and the time of the temperature maximum. Consequently, it is also assumed that the insulating air-filled gap between the canister and the bentonite blocks is still in place at the time of the temperature maximum.

Regarding heat transport through the buffer, it is assumed that the buffer's effective thermal conductivity, i.e. the weighted block-pellets conductivity, is about one watt per metre and degree Kelvin (W/mK). The design calculations take into account uncertainties in the buffer's thermal conductivity resulting from, for example, uncertainties in degree of saturation and void ratio. The uncertainty margin is about three degrees, which corresponds to a lowest effective thermal conductivity of about 0.85 W/(mK). For the design calculations, the uncertainty in the buffer's effective thermal conductivity is the dominant contribution to the total uncertainty in the calculation of the maximum buffer temperature for the hottest canisters in the repository. Verifying calculations with numerical Code Bright models show that dehydration effects, i.e. moisture redistribution due to vaporization and vapour diffusion from the hottest buffer parts to cooler parts and to the backfill material above the buffer and in the tunnel, do not affect the effective buffer conductivity enough to justify a greater margin.

In the THM report for the geosphere /24-8/, an analysis is made of the consequences that would have to be allowed for (in terms of number of affected canisters) if the total uncertainty margin should for some reason turn out to be insufficient. The results are discussed in greater detail in Section 25.2.2.

The question of which parts of the buffer that could be affected if the margin should prove to be insufficient is analyzed in /24-7/. Figure 24-2 shows temperature contours around the upper part of a canister with an insulating air-filled gap, i.e. a dry canister. The analysis is done for several different assumptions regarding what the air-filled gap looks like around the flanges on the canister lid. To analyze the effects of uncertainties in local heat transport, it is enough to show just the contribution from the local canister, as is done here. Regardless of what the air-filled gap looks like in detail, i.e. regardless of how the thermal contact between flange and buffer works, any excess temperature will be localized to a small buffer volume in the pocket that is formed inside the flange and to even smaller volumes immediately adjacent to the flange. 2.5 centimetres above the flange the buffer temperature will be below the design temperature, even if the margin in the design calculations has been underestimated by 4–5 degrees.

Provided the thermal properties of the buffer material will be as in the Prototype Repository, there does not appear to be any reason to increase the uncertainty interval in the description of the buffer's effective thermal conductivity. For example, thermal simulations of different holes in the Prototype Repository show that the thermal conductivity of the blocks is above 1.25 W/(mK) in both dry and wet holes /24-9/. The accuracy of the calculations and of the measurement data used for comparison is, however, not sufficient to enable the uncertainty interval to be reduced. Minor deviations from conditions in the Prototype Repository, i.e. minor variations in water saturation and void ratio, will be accommodated within the interval now used in the design calculations for layout D2.

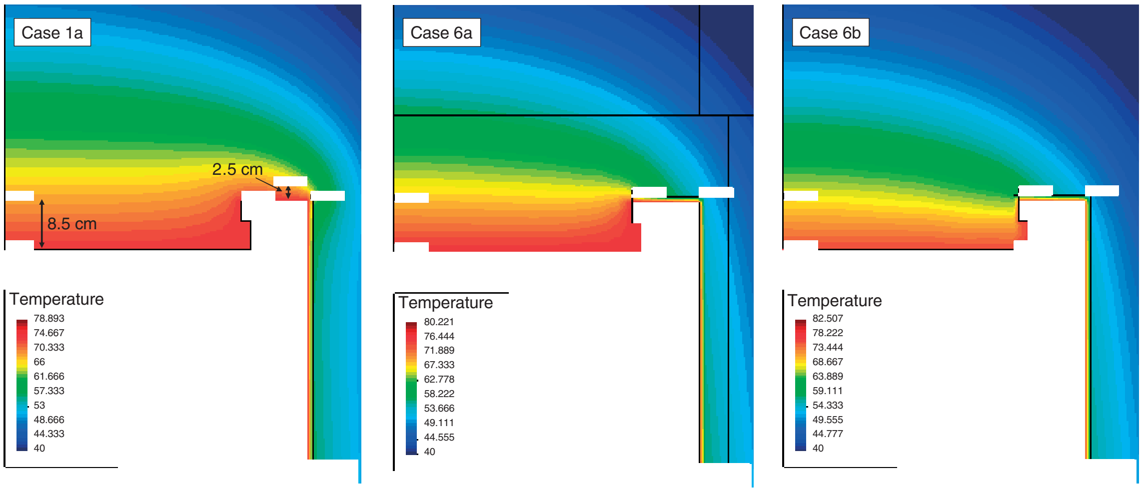


Figure 24-2. *Temperatures around the canister lid for canisters deposited in dry holes. Case 1a: The insulating air-filled gap extends along the vertical canister surface while the top edge and inside of the flange are in direct thermal contact with the buffer. Case 6a: The insulating air-filled gap extends along the vertical canister surface and along the top edge of the flange, while the inside of the flange is in direct thermal contact with the buffer. Case 6b: The insulating air-filled gap extends along the vertical canister surface and along the top edge and inside of the flange.*

Programme

Developments with regard to material choice, changes in design etc. must be monitored so that changes do not occur in effective buffer conductivity entailing that the temperature margin must be updated for future design steps.

The thermal conductivity of the pellet-filled gap may be sensitive to small changes in e.g. material and in the size, density and degree of saturation of the pellets. The effective buffer conductivity assigned in /24-7/ is based on the assumption that the pellet-filled gap has roughly the same properties as the pellet-filled gap in the Prototype Repository.

24.2.4 Freezing

When water turns to ice, its volume increases by about 9 percent. If water in the bentonite buffer freezes, the pressure is therefore expected to increase, which could damage canister and rock.

The freezing point of bentonite is dependent on its water content (density), but is generally lower than the freezing point of pure water, which is 0°C at atmospheric pressure. The freezing point lowering in bentonite is analogous to that in ordinary saline solutions. The freezing point of the water in the buffer is between −5°C and −10°C. All water does not freeze at the same temperature, and temperatures below –40°C are required to freeze all water in the buffer.

However, the buffer is also affected by the fact that the surrounding groundwater freezes because the swelling pressure decreases. The pressure decrease is roughly linear with the temperature, and the swelling pressure is zero at the freezing point of the bentonite. This effect is also completely analogous to the decrease in osmotic pressure in a saline solution with decreasing temperature below 0°C.

Harmful pressure increases due to freezing are thus not a problem in an intact buffer, since the temperature will not fall below the freezing point even in the most extreme permafrost scenario (–2°C) /24-10/. Furthermore, the swelling pressure lowering above the freezing point is not a problem, since this effect requires that surrounding groundwater is frozen.

However, freezing of the buffer cannot be ruled out if its density is somehow reduced, for example by erosion, see Section 24.2.20. But the maximum pressure at the lowest temperature in the most extreme permafrost scenario can be assumed to be 26 MPa.

Freezing occurs in the same way as in all bentonite, and the process is therefore relevant for the backfill as well. The lower montmorillonite content of the backfill results in a higher freezing point compared with the buffer. A backfill closer to ground level will also be exposed to lower temperatures than those predicted for the buffer. The possibility that parts of the backfill may freeze can therefore not be excluded.

The temperature in SFR will fall to below –5°C after about 45,000 years. At that temperature the bentonite may freeze. After freezing and thawing, however, the bentonite will retain its swelling properties and will act as a diffusion barrier during the periods when the area is not frozen /24-11/. The bentonite in the vertical gap between the concrete silo and the rock will exhibit ice lens formation /24-11/. Ice lens formation entails that water is absorbed by the freezing bentonite, where it accumulates in ice layers that grow progressively in thickness. The ice layers normally grow in a direction parallel to the heat flow. In this case, where the frost front is horizontal, the ice layers will grow in thickness vertically. The driving force for this process comes from the unfrozen water, which has a potential that is lower than that in the surrounding free water. This lower potential is a consequence of the fact that the water does not freeze at 0°C, but at a lower temperature. The phenomenon is common in natural soil layers and is then termed frost heave. Ice lens formation can cause fractures to open in the surrounding rock. Swelling bentonite is then squeezed out into the fractures as it thaws. This can lead to some material loss and reduced density in the bentonite. The duration of the permafrost will not be of any importance for the heaves caused by ice lens formation. What is decisive for the size of the heaves is the time it takes for the frost front to pass the repository.

Conclusions in RD&D 2007 and its review

Continued theoretical studies and laboratory tests are needed in order to verify the theories and the conclusion that no water freezes at temperatures above –5°C.

SKI says that SKB needs to evaluate remaining uncertainties to establish a safety margin against freezing. SKI also says that the consequences of freezing, including in a partially eroded buffer, need to be analyzed.

In its review of SAR-08, SSM points out that better evidence is needed for the assumption that the bentonite around the silo regains its protective properties after a freeze/thaw cycle and is not affected by the degradation of the surrounding concrete barriers.

Newfound knowledge since RD&D 2007

A theory has been developed for the temperature dependence of the swelling pressure. The theory is based on the same conceptual bentonite model that has also been applied to e.g. ion diffusion, the dependence of the swelling pressure on salinity and gas breakthrough /24-12/. According to the theory, the freezing process in bentonite is analogous to that in an NaCl solution. Specifically, the theory predicts that no pressure increases occur when the temperature passes 0°C and that the swelling pressure decreases roughly linearly with the temperature below 0°C, see Figure 24-3. The loss in swelling pressure is on the order of one MPa/°C and is determined primarily by the entropy difference between water in solid and liquid form. This means that the freezing process is more or less identical in all bentonite materials – the only controlling parameter is the measured swelling pressure at zero degrees. One consequence of this is that the freezing point in a given bentonite material varies greatly with the water content (between 0 and < –40°C).

Extensive laboratory studies of a number of bentonite materials have in principle verified all predictions of the theory. For example, the reversibility of the process has been demonstrated in tests with up to seven freeze/thaw cycles. The test method has further proved useful for determining fundamental thermodynamic properties in the bentonite.

A supplementary study has been conducted which bounds the pressure for a given temperature in a frozen bentonite system. The results show that the pressure on the canister does not become harmfully high even in the case with a strongly eroded buffer.

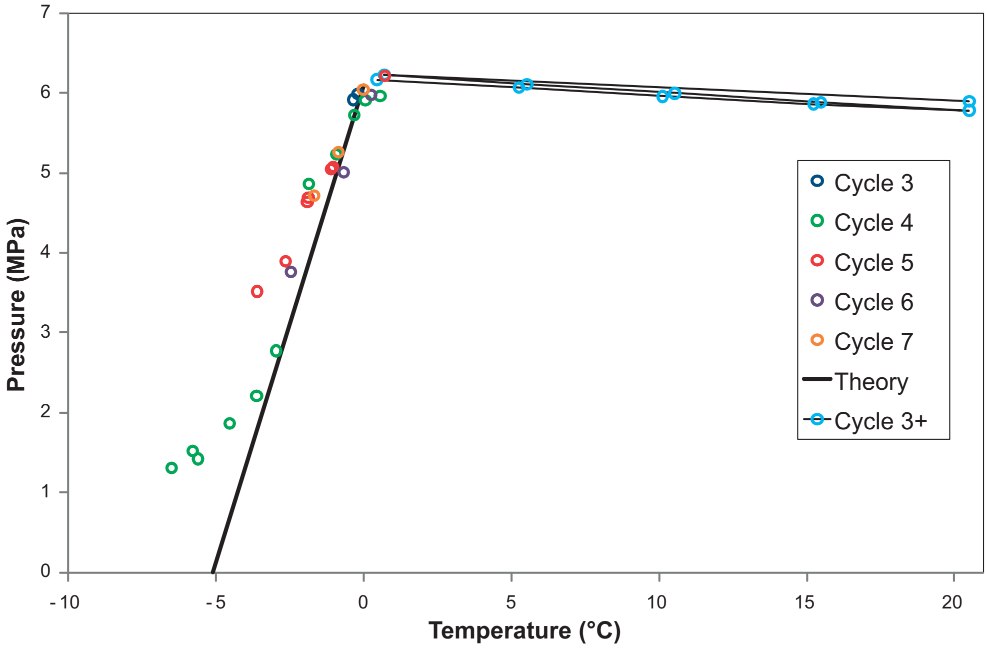


Figure 24-3. *Equilibrium pressures at different temperatures in one and the same bentonite sample (MX-80 at approximate buffer density). The sample has been subjected to a total of seven freeze/thaw cycles during a period of about two years. The thick solid line shows the prediction made by the theory. The slight negative slope at temperatures above 0°C is completely in accordance with the theory. From /24-12/.*

Programme

Further studies of freezing of the buffer are not considered necessary from a safety assessment perspective. Using the temperature dependence of the swelling pressure to quantify fundamental thermodynamic properties of the bentonite can, however, be fruitful in gaining a better understanding of many other processes such as swelling, ion exchange and colloid formation. Possible pressure effects in the backfill due to freezing in rock caverns that extend vertically need to be investigated, for example for combinations of repository parts such as backfilling tunnels and shafts.

The description of the freezing of the silo will be updated. The work will be based on the study results described above /24-12/.

24.2.5 Water transport under unsaturated conditions

When the buffer blocks and the pellet fill have been installed in a deposition hole, the buffer will absorb water from the surrounding rock. During the saturation phase, the buffer will develop a swelling pressure that exerts a mechanical force on the rock, the canister and the backfill. Water transport in the unsaturated buffer is a complicated process that is dependent on, among other things, the temperature, density, smectite content and water ratio in the different parts of the buffer. The most important driving force for achieving water saturation is the relative humidity in the buffer, which can be regarded as a negative capillary pressure in the buffer pores leading to water uptake from the rock. The hydraulic conditions in the rock surrounding the deposition hole determine the course of the saturation process. With an unlimited supply of water, full water saturation will be achieved between canister and rock within a few years. A number of conditions in the rock are of importance for the water supply.

The same model can be used for buffer and backfill and the same parameters are needed as input data to the modelling, but the values differ since the materials are not identical. Water transport under unsaturated conditions is also dealt with in Section 24.2.11.

In SFR, the operating phase's drainage pumping will cease when the repository is closed. The hydrogeological model /24-13/ has been used to calculate how long it takes to fill and saturate the repository with groundwater. The calculations show that the void (porosity) inside the silo repository is saturated last and that this can take 25 years.

Conclusions in RD&D 2007 and its review

Continued laboratory and model development work was planned in order to gain an even better understanding of, and improve the models for, the wetting process. Model calculations of the different field tests in the Äspö HRL and comparisons with measurement data were planned, see also Section 24.2.11. Since the hydraulic interaction between the buffer and the rock is crucial for the wetting process, a field test in the Äspö HRL was also proposed.

SKI said in its review that SKB needs to analyze the resaturation process in rock with very low permeability. SKI also said that SKB should better justify the temperature criterion for the buffer and more thoroughly study the risk of dehydration and the implications of the buffer remaining unsaturated for a long period.

Laboratory tests of Friedland Clay in particular were planned in RD&D Programme 2007. Modelling of the wetting process in the backfill under different hydraulic conditions was also planned, mainly for blocks and pellets of Friedland Clay.

SKI called for clearer information on how boundary conditions will be handled for modelling of the early hydraulic evolution and how the hydraulic properties of the rock will be represented. SKI also says that the interaction between backfill of the buffer needs to be studied for the case where the bedrock is relatively watertight (impermeable) and that studies of the backfill in "dry rock" will be more important if Forsmark is selected.

Newfound knowledge since RD&D 2007

Laboratory tests of water transport in bentonite under different temperature gradients at different densities have been carried out. The results of these tests will be used to update the material model.

Modelling of laboratory tests and field tests and comparisons with measurement data that have been done within TF EBS (Task Force Engineered Barrier Systems) have provided experience and a better understanding of unsaturated water transport. The modelled field tests are ITT (IsoThermal Test) and BCE (Buffer/Container Experiment) in the URL in Canada and CRT (Canister Retrieval Test) in the Äspö HRL, where modelling of the latter is still in progress.

The wetting calculations for SR-Site /24-14/ have been preceded by a process of qualification of the parameter values that would serve as input data to the different models. The procedure was to compare previously used material relationships and parameter values with evaluated relationships from new measurements (hydraulic conductivity) or to show agreement with independent measurements (thermal conductivity and water retention properties).

The time scale for buffer and backfill hydration was investigated for a number of models where different parameter studies were done. The calculations differ from the models in SR-Can with respect to:

- The description of vapour transport (only buffer hydration), which is formulated directly in the SR-Site project as diffusion (vapour concentration gradient rather than temperature gradient).
- The boundary conditions (and the model dimensions) used, which are more realistic and justified in the SR-Site project in view of site data.
- Mechanical effects (of homogenization) have been indirectly accounted for in the SR-Site project by modelling of both original and homogenized states.

The properties of the rock (hydraulic conductivity, fracturing etc.) proved to be of great importance for the saturation time. Variations in the buffer's hydraulic properties (water retention and permeability) did not result in as great changes as when the properties of the rock were varied.

Water redistribution has been studied for a situation with dry (watertight) rock, in order to obtain critical thermal conductivities for different areas in the buffer (for use in temperature calculations). In the Finite Element Method (FEM) model, the rock was assumed to be hydraulically inactive. After 15 years of water redistribution, the lowest buffer saturation was found at the top of the canister and the pellet-filled gap had increased its degree of saturation /24-14/.

Another analysis concerned ventilation of the deposition tunnels and possible dehydration of the surrounding rock /24-14/. The air-filled pore space in the rock could potentially constitute a sink for the water that accompanies the buffer at installation. The purpose of this study was therefore to investigate the extent of dehydration as a function of the properties of the rock, in particular its water retention properties. The method used can identify parameter combinations in the water retention law that can lead to extensive dehydration in conjunction with ventilation of the deposition tunnels. However, the analysis shows that the gas-filled pore volume is much smaller than the available water volume in the installed buffer for cases with reported parameter values for water retention and relative humidity.

Laboratory investigations of water absorption properties and retention have been done on Friedland Clay, Asha 230 and IBECO RWC-BF. Figure 24-4 shows retention curves determined for these materials and for MX-80 /24-15, 24-16/.

Water uptake properties were measured under constant volume conditions by leaving material with different densities in contact with water for different times, after which the tests were interrupted and the degree of saturation and distribution of the void ratio with distance from the water intake were measured. These tests are intended to be used to calibrate and validate the model for water transport in the unsaturated state.

Model calculations of the wetting process have been done for SR-Site for different hydraulic conditions in the rock /24-14/. At the extreme of free availability of water at the rock margin, the time to full saturation is about 80 years, and at the opposite extreme of no fractures and a hydraulic conductivity in the rock of 5×10^{-13} metres per second, the time to full saturation is 1,200 years.

Extensive tests with large-scale models of a tunnel backfilled with blocks and pellets have been conducted on Äspö. These tests have provided valuable information on how the backfill reacts to water inflow, mainly during the backfilling phase /24-17, 24-18/. See also the programme for backfilling technology development, Chapter 13.

Programme

Questions relating to water transport under unsaturated conditions still require large initiatives during the coming period. Continued modelling of CRT, modelling of TBT (Temperature Buffer Test), which is being mined in 2010, and modelling of the outer section of the Prototype Repository, which is planned to be mined in 2011, are included in the programme. The laboratory tests with temperature gradient will be modelled and used to calibrate the parameters in the material model, and further tests may be done.

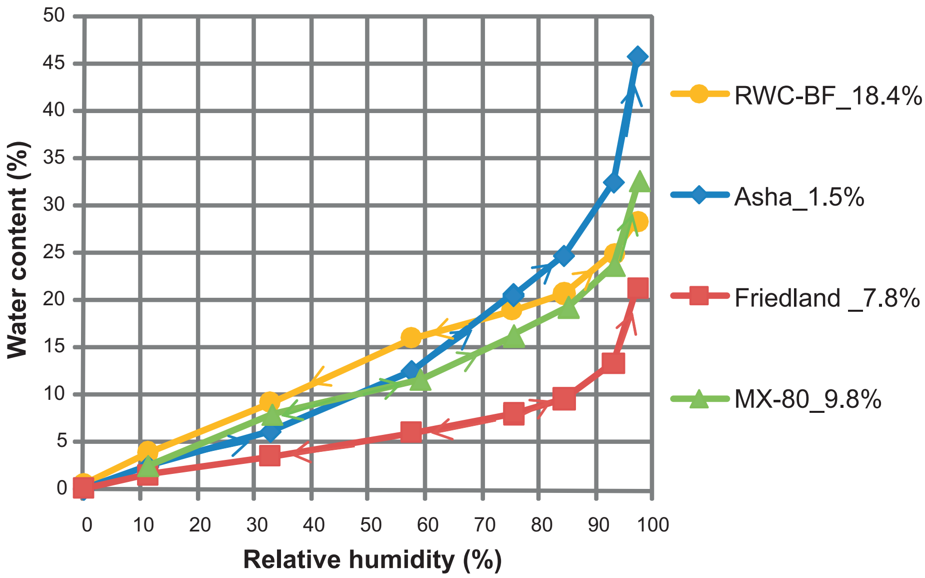


Figure 24-4. *Retention curves for investigated candidate backfill materials /24-15, 24-16/.*

Within TF EBS, the deposition holes in the outer section of the Prototype Repository will be modelled and the new experiment BRIE (Buffer Rock Interaction Experiment), which is intended to study the hydraulic interaction between rock and bentonite under unsaturated conditions, will be modelled in collaboration between the two task forces Engineered Barrier Systems (TF EBS) and Ground Water Flow and Transport of Solutes (TF GWFTS). The intention is to study both dry and wet conditions.

Further studies of dehydration scenarios in vary dry rock will be conducted in the form of investigations of the possible occurrence of natural convection in pellet-filled gaps and its importance, if any, for dehydration.

Further studies of fundamental mechanisms in support of different transport laws and selection of parameter values will be done.

Proposals have been made to investigate the water retention curve for both purified material, i.e. pure Wyoming Na-montmorillonite (WyNa) and pure Wyoming Ca-montmorillonite (WyCa), and material that has been exposed to high temperature. The influence of grain size distribution, hydration-dehydration cycles, areas with high relative humidity (RH) and the magnitude of the RH change during both hydration and dehydration are among the questions to be investigated.

The laboratory studies of different backfill materials should continue, especially some material that is considered to be representative of the material that is intended to be used. Evaluation of the parameters for the material model will be done with the water uptake tests as a basis for both used materials and possible new ones. The calculations will be updated with the new models.

The intention is to model the water saturation process in SFR in a similar manner as for the fuel repository.

24.2.6 Water transport under saturated conditions

The basis for the sealing properties of the bentonite buffer is the montmorillonite's affinity for water. When the pore space in the buffer is filled with water, this affinity leads to an effective distribution of the water to an approximately one-nanometre-thick water layer between the clay particles. This distribution and the direct interaction of forces between the water and the counterions in the montmorillonite is an effective barrier to water movements. Normally the flow resistance for water is given in the form of hydraulic conductivity (K). The reference bentonites studied in SR-Site project have a hydraulic conductivity of about 10^{-13} metre per second, which is of the same order of magnitude as fracture-free granite.

No doubts exist about the buffer's ability to limit the water flow in the deposition holes in accordance with its safety function, provided that no extensive transformation or loss of buffer takes place. The process is critical, however, since it is dependent on other less studied processes, above all loss of buffer due to colloid formation, see Section 24.2.20.

The same model can be used for buffer and backfill and the same parameters are needed as input data to the modelling, but the values differ since the materials are not identical. This reasoning also applies to the wall buffer in the silo in SFR. But the density is much lower there and the hydraulic conductivity is therefore higher.

Conclusions in RD&D 2007 and its review

According to RD&D Programme 2007, measured hydraulic conductivities were to be evaluated after mining of the Backfill and Plug Test. This point remains since the test has not yet been excavated.

Measurements of hydraulic conductivity in backfill candidates were included in RD&D Programme 2007.

In its review of RD&D Programme 2007, SKI says that the activities identified and started by SKB are suitable for gathering sufficient data for assessment of the hydraulic evolution of the backfill. SKI also says that SKB should reconsider the scope and focus of the activities at the Äspö HRL aimed at gaining more knowledge about the implementation of technology for buffer and backfill.

Newfound knowledge since RD&D 2007

Hydraulic conductivity has been measured on field-exposed material in tests in the Äspö HRL (the Canister Retrieval Test and "parcel 1" in Alternative Buffer Materials). The results show, in accordance with earlier results from the LOT test, a slight tendency towards lower values in comparison with the reference material.

Hydraulic conductivity has been measured on Friedland Clay, Asha 230 and IBECO RWC-BF. Figure 24-5 shows hydraulic conductivity as a function of the dry density of these materials. As expected, the results show that sufficiently low hydraulic conductivity can be achieved in the backfill materials if either sufficiently high density or a suitable material is used. For example, a dry density higher than 1,500 kg/m³ is required if Friedland Clay is used, but only about 1,300 kg/m³ if Asha 230 or IBECO RWC-BF is used.

Programme

The laboratory studies on different buffer and backfill materials will continue, especially on materials considered to be representative of the material intended to be used in the Spent Fuel Repository. This is being done to broaden the body of knowledge of the properties of different bentonite grades.

Verifying measurements of reference material and material from the Canister Retrieval Test at the Äspö HRL with different water pressure gradients are being made.

An extensive testing programme will be conducted on material from the TBT field test (Temperature Buffer Test) in the Äspö HRL.

Hydraulic conductivity in purified montmorillonite will be studied in the two projects Bentonite Erosion and Forge for a large set of densities, water pressure gradients and counterions. The results thus far show that the hydraulic conductivity for a given counterion is largely determined by the density distribution in the sample.

The very low hydraulic conductivity of the buffer (despite the fact that its porosity is about 40 percent) suggests that the distribution of the pore volume is of crucial importance for the properties of the bentonite. Laboratory tests will be performed for the purpose of investigating the fundamental principles (advection/diffusion/dispersion) for water transport in bentonite under saturated and volume-limited conditions.

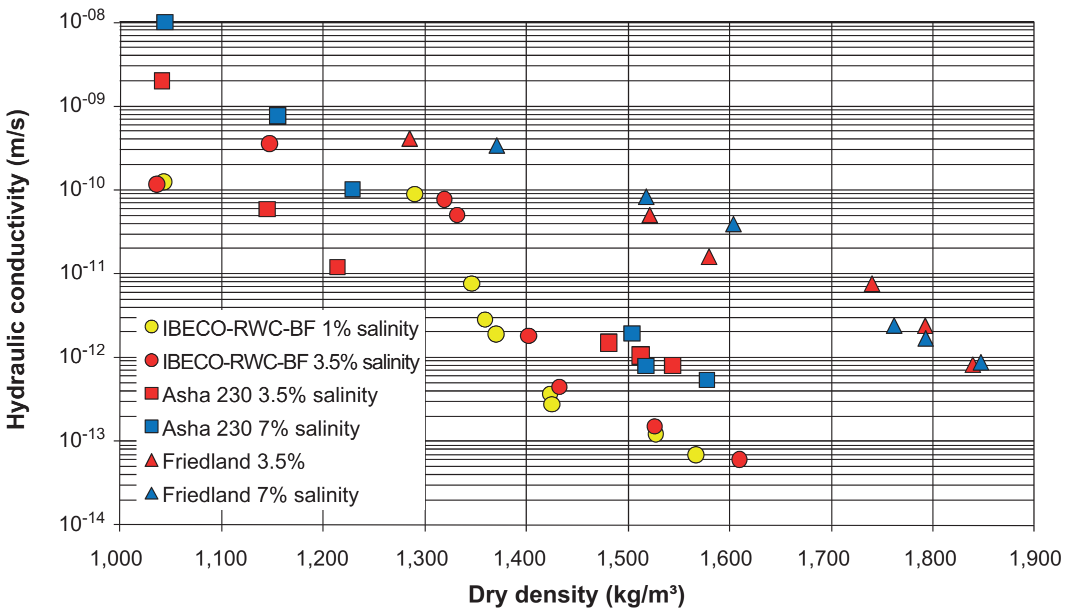


Figure 24-5. *Relationship between hydraulic conductivity and dry density for investigated backfill candidates /24-15, 24-19/.*

After excavating the Prototype Repository and the Backfill and Plug Test, an evaluation will be made of densities and measured hydraulic conductivities.

Data on the hydraulic conductivity of GEKO/QI, which is used in SFR, are based on measurements performed more than 25 years ago. It may be of value to repeat the measurements and verify the properties of the material.

24.2.7 Gas transport/dissolution

Chemical reactions inside a damaged canister may cause gas formation and buildup of a gas pressure inside the canister. It is important to be able to show that this pressure will not lead to any negative consequences for the performance of the repository. This means that the gas must be able to escape without damaging buffer or rock.

In order for the gas formed in waste packages and concrete structures in SFR to escape, gas-conducting passages must be formed in the barriers. The gas transport and the quantity of water expelled from the silo repository and the rock vaults are determined by the design of the barriers and the properties of the barrier materials. In the silo repository, the waste is surrounded by a porous concrete with low resistance to gas transport. Only a small gas pressure is required to open up gas passages in this concrete, and the quantity of water that is expelled has been measured experimentally to be 0.1–2 percent of the pore volume /24-20/. It has been assumed in the gas transport calculations in SAR-08 that two percent of the pore volume in the concrete can be displaced. In order for the gas to find its way out through the gas evacuation pipes and the sand/bentonite barrier in the top, a gas pressure must be built up in the concrete silo equivalent to the opening pressure of the sand/bentonite barrier. Experiments show that a pressure of approximately 15 kPa is required to achieve gas transport through the sand/bentonite and that the expelled water is only a few tenths of a percent of the total pore volume /24-21/.

Conclusions in RD&D 2007 and its review

SKI considers that SKB should report as much as possible of the results of the demonstration tests on Äspö before the applications, and that SKB should consider the risk that the Lasgit experiment will not lead to the expected results.

Newfound knowledge since RD&D 2007

During the period 2007–2009, all efforts in the area have been focused on the Lasgit test on Äspö. Lasgit (large scale gas injection test) was started up in February 2005 and has not been in place for more than five years. In May 2007, a hydraulic test was started for the purpose of determining the hydraulic conductivity of the bentonite before the gas tests started. The first gas injection test was started in August of that year /24-22/. In the test, the flow of gas increased sharply when the gas pressure on the pump side was slightly higher than the local total pressure measured on the rock wall, but slightly lower than the total pressure on the canister surface. The measured axial stress was also slightly higher than the gas pressure. The gas pressure then rose to a maximum of 5.66 MPa after approximately two months' gas injection. This was followed by a short negative transient, after which the pressure stayed in a quasi-steady-state around 5.5 MPa, which is illustrated in Figure 24-6. The behaviour resembles that observed in previous laboratory tests /24-23/. However, it should be pointed out that the maximum pressures measured in Lasgit are considerably lower than those reported in /24-23/.

In order to test whether gas injection and gas flow affect the hydraulic properties of the buffer, hydraulic tests were performed before and after the gas injection test. The tests show clearly that the hydraulic properties have been altered very little, if at all, by gas injection /24-22/. Based on available data, there is nothing to indicate that the gas transport pathways that are formed have any effect whatsoever on the hydraulic function of the buffer.

At installation of Lasgit, bentonite blocks with a high water content were used to speed up the water saturation. The natural inflow of water to the deposition hole for Lasgit is also relatively high. This means that the amount of water that is needed to saturate the buffer is already present in the deposition hole. During the test in 2007, however, the bentonite was not in mechanical equilibrium. There were parts of the buffer where the pore water pressure was negative. This means that some caution is needed in drawing conclusions.

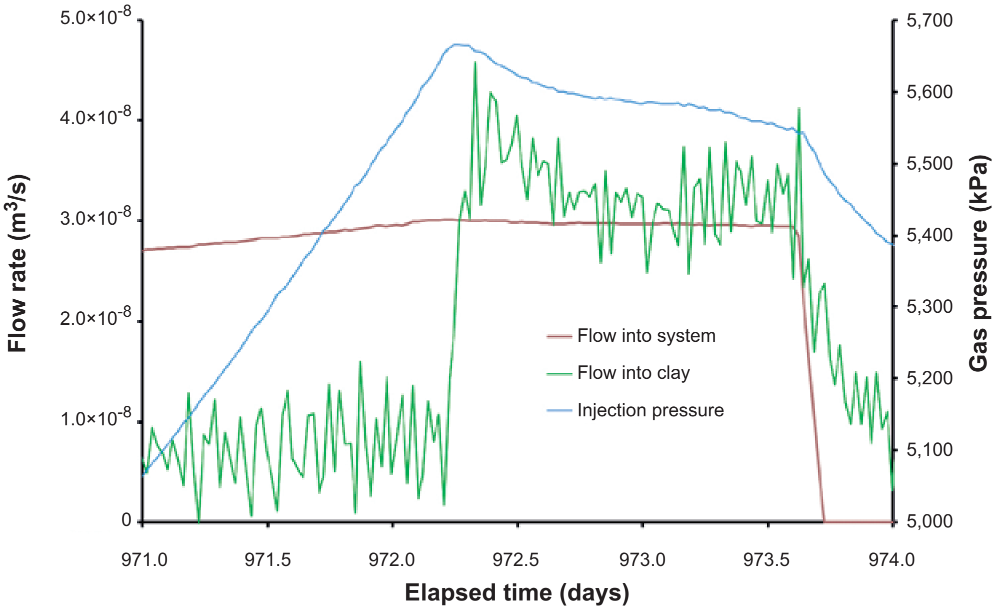


Figure 24-6. *Flows and pressures around the maximum gas pressure in the first gas injection test in Lasgit. The gas flow into the bentonite increases rapidly after the peak, which is then followed by a small spontaneous negative transient /24-22/.*

In order to equalize the pressure conditions and allow the buffer to "mature in peace", no gas injection was done in Lasgit during all of 2008.

A second series of gas injection tests started in 2009. It is still in progress (February 2010) and the results have not yet been published. One of the purposes of the test is to investigate whether there are time-dependent effects in the gas transport process. Gas injection started in May 2009 in the same filter as the test in 2007. The gas pressure has been ramped in steps and has been kept constant for long periods. The last ramp started on 12 November 2009 with a pumping rate of 500 microlitres per hour. The pressure rose to 5,872 kPa after 25 days when a gas flow could be measured. The pressure then fell and a steady state was established at the turn of the year, see Figure 24-7.

The following can be observed from the second gas test:

- The gas breakthrough came close to the total pressure, which was the same in the 2007 test. This agrees with the traditional picture, but disagrees with observations in laboratory tests /24-23/. It is evident that size is of importance for the process in question.
- The behaviour of the gas after breakthrough is complicated. This is partially illustrated in Figure 24-7, where pressure and flow oscillate. The process is therefore a challenge to model.
- The instruments in Lasgit work well and are sufficiently sensitive for the gas transport process to be followed.

The observations that have been made in Lasgit support the treatment of the process presented in SR-Site in all respects.

Programme

Lasgit is part of the EU project Forge, which is running during the period 2009–2012. The tests will continue during this time. The results will be evaluated in 2012 and a new decision will be made on the future of Lasgit. To supplement Lasgit, laboratory tests will be carried out to study critical parameters that are difficult to evaluate in a full-scale test.

No further studies of gas transport in the silo in SFR are planned.

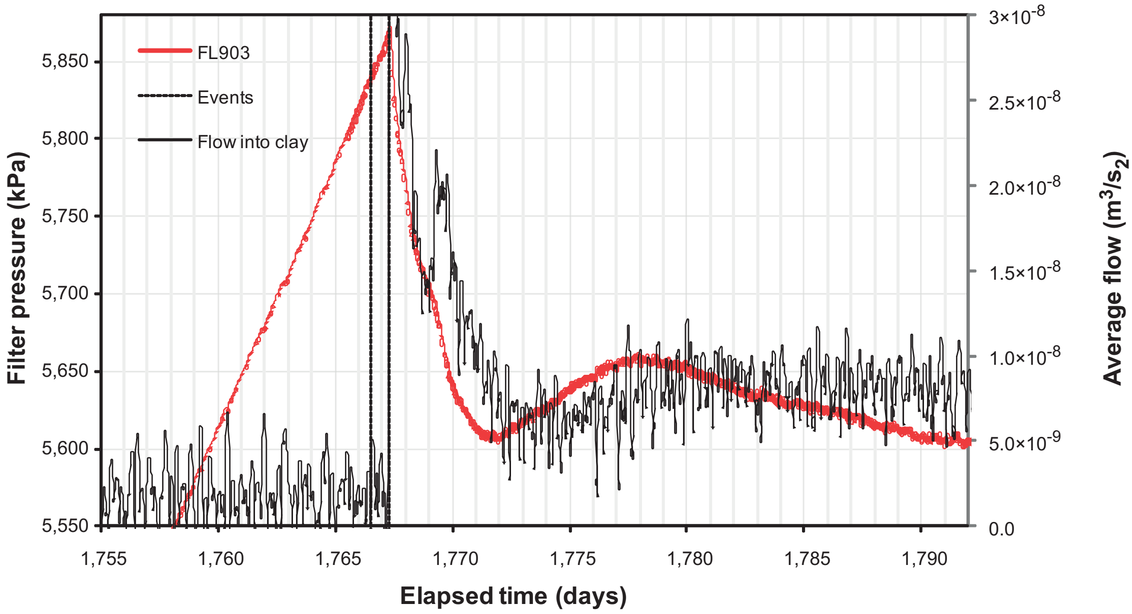


Figure 24-7. *Gas pressure and flow in Lasgit in the 2009 test (preliminary data). FL903 = pressure in gas injection filter FL093.*

24.2.8 Piping/erosion

Water inflow in the deposition holes and tunnels in a final repository mainly takes place through fractures in the rock, leading to wetting and homogenization of buffer and backfill. But in general the buffer and the backfill cannot absorb all water that runs through a fracture, resulting in the build-up of a positive water pressure when the flow of water is obstructed. If the counterpressure and the strength of the buffer and the backfill are not great enough, piping may occur with accompanying erosion.

The process is the same for buffer and backfill, but the effect may be more serious for the backfill since most candidate materials have slightly poorer healing capacity. On the other hand, the safety consequences are mitigated due to the distance to the canister and the fact that the mass of backfill in a section is much greater than the mass of buffer.

The processes and consequences associated with piping and erosion have been studied in several projects, involving numerous laboratory test series on different scales. The tests have led to an erosion model that is used in calculations of how much bentonite can erode from the buffer. The piping and erosion referred to here mainly occur before full water saturation has been achieved and will heal after full water saturation has been achieved and the water flow has been stopped. Homogenization is dealt with in Section 24.2.9.

The process is probably of very little importance for SFR. The water pressure around the silo will be restored quickly when the drainage has been shut off, providing a brief period during which piping can occur.

Conclusions in RD&D 2007 and its review

Piping tests in the buffer were planned in RD&D Programme 2007, mainly within the KBS-3H project. The problem for KBS-3H was that each distance block is supposed to seal so that water cannot run between two canister sections during the water saturation phase. Erosion was to be studied mainly for backfill materials. The results will be studied in the field in connection with the mining of hole 1 in the Prototype Repository, where the inflow at installation was 0.08 litre per minute.

Regarding the buffer, the following comments were offered by SKI:

- SKB needs to improve its theoretical understanding of piping/erosion and describe the countermeasures in greater detail.

- Criteria for choosing deposition holes so that piping/erosion is avoided need to be clarified prior to a licence application.
- The Prototype Repository can probably provide information on piping/erosion, but this single experiment is not enough.
- The plans for piping/erosion are vague for KBS-3V compared with the work for KBS-3H.

As far as the backfill is concerned, ongoing laboratory tests of piping and erosion were supposed to continue and be supplemented by similar tests with other materials. The goal was to bound the quantity of backfill that can erode away before complete sealing with the end plug has been achieved and by internal redistribution of eroded material.

SKI said that the differences between buffer and backfill should be taken into account.

Newfound knowledge since RD&D 2007

The piping tests with KBS-3H showed that the distance blocks cannot seal against piping and cannot prevent water from running between the sections. The basic design has therefore been abandoned in favour of two other variants, one of which (Dawe) is the new main concept. In Dawe, the gap between buffer/distance block and rock is filled artificially with water immediately after a certain length of deposition tunnel is finished. One conclusion drawn from the tests is that piping can never be stopped by the buffer (either in KBS-3H or in KBS-3V) if the inflow is so great that the bentonite is not able to absorb the incoming water /24-24/. A plug in the end of the deposition tunnel is needed to stop the water flow and create a stagnant water without hydraulic gradient so that the bentonite can seal and heal.

Since the bentonite cannot itself seal, it must be assumed that the deposition holes and the deposition tunnel must be filled with water before erosion is stopped, which means that the total erosion from a deposition hole can be calculated. Several series of erosion tests have therefore been carried out and led to an empirical erosion model.

In the model, accumulated eroded bentonite mass is related to accumulated mass of eroding water. The model takes into account erosion proneness, which is affected by geometry, the salinity of the eroding water, material composition and flow direction. An important observation is that erosion declines with time, regardless of the circumstances. This is of great importance for the total erosion.

The erosion model is used to bound the inflow requirements on deposition holes in relation to inflow into the tunnel /24-14/. For KBS-3H, there is so little erosion due to the small void volume in the deposition tunnel and due to the fact that this void is filled artificially that no special requirements are needed.

As far as the backfill is concerned, erosion tests have been carried out both on blocks of different backfill materials (Friedland Clay, Asha 230 and IBECO RWC-BF) and on different kinds of pellet fills (MX-80, Cebogel, Minelco and IBECO RWC-BF /24-15, 24-25, 24-26/. The results show that the model developed for the buffer can also be used for the backfill.

Sample calculations of how much erosion can occur in the backfill can easily be done with the model. At a block filling degree of 80 percent, the total volume in a 300 metre long tunnel that can be filled with water is about 750 cubic metres. If, as an extreme case, the unlikely assumption is made that all water filling the tunnel runs in at the same point, the total erosion at this point is between 12 kg and 1,200 kg of dry bentonite. The tightness of the plug then determines whether further erosion can occur, but just as for the buffer, it must be assumed that piping/erosion in the backfill does not cease until the water pressure gradient is taken by the plug. Since 1,200 kg of bentonite only causes the average density in a cross-section of the tunnel to decrease by one or two percent, it can be concluded that erosion of backfill material is not a problem if a tight plug is in place. A problem can only occur in the buffer under similar extreme circumstances.

Fracture sealing capacity when water runs through a pellet fill and out into a fracture in the rock has been investigated in the laboratory in a number of tests. The results obtained thus far indicate that fractures with a width (aperture) of up to 150 millimetres self-heal under these circumstances. If the results can be shown to be generally valid, water leakage out of the tunnel is improbable. However, the results are probably very sensitive to the geometry, and the pellet fill probably has to be in direct contact with the fracture. More investigations are needed.

Programme

The plans in RD&D Programme 2007 have been more than fulfilled, and SKI's comments have been taken into account so that a tool is available for formulating inflow requirements on deposition holes and tightness requirements on the plug with regard to buffer erosion. However, it is desirable to gain a deeper understanding of piping and erosion and of how different factors affect the parameters in the model.

Additional erosion tests will be carried out, which will include variation of geometries etc. to gain a better understanding of how different factors affect the parameters in the model.

The tightness requirements on the tunnel plug are rigorous and are controlled by buffer erosion, which means it is important to investigate how and when the flow and the erosion cease.

The question of at which inflows and geometries piping does not occur will be investigated, since it can be difficult in the case of tight rock to impose requirements on the ratio between inflow in deposition holes and inflow in the tunnel.

As regards the backfill, further major initiatives are planned to gain a better knowledge of

- when piping ceases,
- the validity of the erosion model,
- self-healing of fractures.

These questions will be further explored in a new project concerned with water saturation, self-healing and homogenization by conducting scenario analyses of the entire installation and wetting process for buffer, backfill and plug in rock that is dry and has different fracture constellations. See also Section 24.2.11 and Chapter 13, which describes technology development for the backfilling line.

24.2.9 Mechanical processes

The swelling process has here been combined with other stress- and strain-related processes that can cause mass redistribution in the buffer such as thermal expansion, creep movements and a number of interactions with canister, near-field rock and backfill. The emphasis is on processes after full water saturation, but many modelling runs have been simplified and done under the assumption that the final result is fairly independent of whether swelling and homogenization take place before or after full water saturation. In recent years, the models have been improved, and some new homogenization calculations have been done and will be done without this simplification, i.e. under unsaturated conditions. The results are reported in Section 24.2.5.

Water uptake after deposition in the buffer and the backfill, which are inhomogeneous at emplacement, will lead to swelling. This causes all gaps in the buffer, between rock and buffer and between canister and buffer to disappear, and the buffer to be homogenized. However, some inhomogeneity will remain due to friction in the bentonite. In the buffer, heating will furthermore lead to thermal expansion of the pore water. If swelling is prevented, a swelling pressure will instead develop.

In the interface between the buffer and the backfill, an interaction arises due to the fact that the buffer exerts a swelling pressure against the backfill, and vice versa. Since there is a difference in the swelling pressures, a net pressure arises against the backfill, whereby the buffer and the backfill are compressed. The size of the upswelling depends on the original densities of the buffer and backfill, as well as the expansion and compression properties of the materials and the friction against the rock. Calculation models, both analytical and numerical, exist for analysis of this interaction.

A mechanical interaction between buffer and canister arises due to the fact that the buffer generates both compressive and shear stresses. The interaction also arises through the pore water, which only generates compressive stresses, and through gas in the buffer, which also only generates compressive stresses. These three variables change during the water saturation process. The weight of the canister acts on the buffer, while the weight of the buffer on the canister only has a negligible effect. Rock movements that occur in fault planes, for example after earthquakes, give rise to stresses on the canister, which are transmitted from the rock through the buffer. The processes associated with the mechanical interaction between buffer and canister after water saturation are relatively well understood. The uncertainty mainly concerns the evenness of the wetting, the irregularities of the deposition hole, and the pressure build-up caused by any gas formation. Another uncertainty stems from creep movements of the canister caused by the weight of the canister.

The interaction between buffer and rock is caused by e.g. swelling pressure from the buffer, convergence of deposition holes and shear movements in the rock. Convergence is dealt with in Section 25.2.9. In the case of horizontal deposition, the bentonite will first extrude through the outer perforated supercontainer and then into the space between the rock and the container. In the long term the container will corrode. The transformation from iron to magnetite entails a volume increase and an increased pressure against the rock and the canister.

The swelling causes clay to penetrate into the fractures in the rock. Due to the swelling properties of the bentonite, any damage that occurs in the buffer – for example due to piping and erosion, gas penetration or rock movements – will swell shut and self-heal.

In the long run, chemical changes in the buffer can lead to changes in its swelling and deformation properties, see Section 24.2.16. A model for swelling under water-saturated conditions has previously been developed for the finite element code ABAQUS. Swelling occurs during the water saturation phase as well, see Section 24.2.11.

The swelling and compression properties of the backfill are important for the function of the final repository. The design with blocks and pellets in the backfill imposes requirements on the homogenization capacity both between the blocks and the pellet fill and for healing of erosion pipes, but swelling pressures and compression properties are also important for e.g. buffer upswelling and impact on the plug.

Two types of movements are being studied for the silo in SFR: settlement of the bottom bed and movements of the silo top. This is discussed in detail in /24-5/. The measurement programme for the movements of the silo top started in 1987. In 2002 the top had moved 18.1 millimetres, which is a little less than the prediction of 24–27 millimetres. The main cause of the movements is compression of the bottom bed.

After closure the bentonite buffer in the silo will become water-saturated and exert a swelling pressure against the concrete structure and the surrounding shotcrete.

The silo in SFR could also be affected by rock movements that occur in fault planes created by earthquakes. They could give rise to stresses on the concrete structure, which are then transmitted from the rock through the wall buffer.

Conclusions in RD&D 2007 and its review

It was observed in RD&D Programme 2007 that the mechanical processes in the buffer are important for the function of the Spent Fuel Repository and that both understanding and models needed to be improved. The following proposals were made for further research:

- Studies of swelling pressure buildup in bentonite.
- Programme of laboratory tests to study and model swelling.
- Modelling and large-scale test in BB (Big Bertha) to study the extrusion of bentonite through the perforations in the supercontainer in KBS-3H.
- Evaluation and modelling of the homogenization in the mined Canister Retrieval Test.
- Further studies and modelling of canister shear.
- Studies of the brittle behaviour of certain parts of the buffer in the LOT tests.
- Continued studies of the effect of high water pressures on swelling pressure and deformation properties.

SKI has the following comments on the programme:

- The transition between non-eroded and partially eroded buffer, i.e. a state still without advection, needs to be evaluated.
- SKB should study what conditions in the rock could lead to intrusion of the buffer up into the tunnel and how this could be dealt with.

RD&D Programme 2007 described plans for continued studies of the swelling and compression properties of different backfill materials in the Baclo project. The same applied to the investigations of the mechanical interaction of the materials with buffer and rock. The interaction between the emplaced backfill blocks and the pellet-filled gap was also to be studied so that the initial state after full homogenization would become known.

Newfound knowledge since RD&D 2007

Swelling pressure measurements done on bentonite after mining of the Canister Retrieval and LOT tests do not show any changes, i.e. the field conditions and high temperatures to which the bentonite is exposed have not affected its swelling capacity.

Swelling pressure measurements have been done on Ca-bentonite (mainly Deponit CaN, but also MX-80Ca, which is MX-80 ion-exchanged to Ca) because the possibility cannot be ruled out that the buffer will be ion-exchanged to Ca in the long term. The measurements have been performed by different methods and on different occasions, and they give a concordant picture of the relationship between swelling pressure and void ratio. Figure 24-8 shows a compilation of the measurement results.

Studies of swelling and homogenization properties that were planned as a series of laboratory tests have been carried out to some extent, but these studies are not yet finished. Continued studies aim at improving the current state of knowledge and the modelling capacity for self-healing and homogenization of partially eroded buffer that was called for by SKI.

Modelling of bentonite extrusion in KBS-3H has been done /24-27/, but the large-scale test in BB has not yet been done because the equipment was being used for other tests.

Modelling of homogenization in the Canister Retrieval Test is under way, mainly within TF EBS. Homogenization of the bentonite rings, the pellet-filled gap and the canister-ring gap has been studied in the SR-Site project for different widths of the pellet-filled gap /24-14/. See also Section 24.2.11.

A great deal of work has been done with canister shear. New material models for rapid shear have been developed for both MX-80 and MX-80Ca. Triaxial and uniaxial compression tests have been done on MX-80Ca and Deponit CaN at different densities and shear rates. Figure 24-9 shows a compilation of measured maximum stresses as a function of the strain rate and the relationship used in SR-Site for the calculations of the rock shear case. The rate dependence in the new model is less than that used for SR-Can.

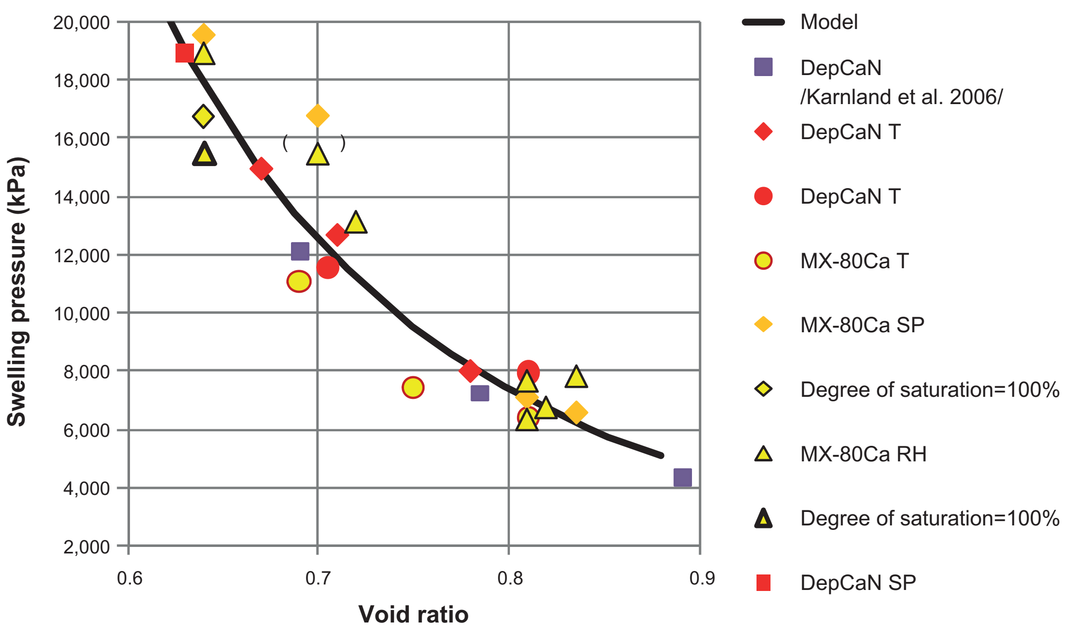


Figure 24-8. *Compilation of swelling pressure measurements on Ca-bentonites.*

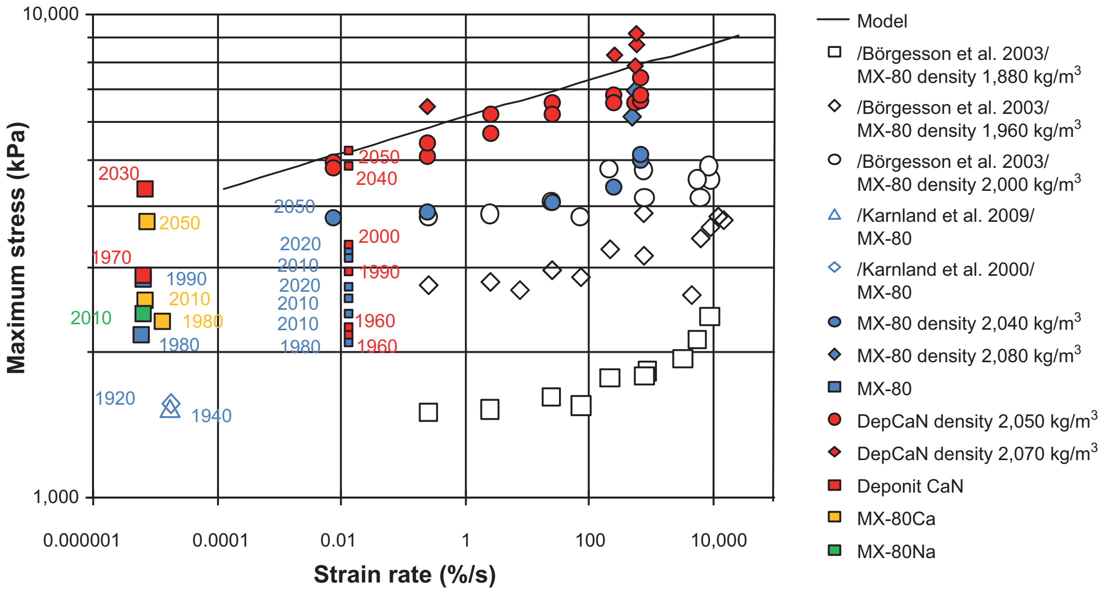


Figure 24-9. *Stress at failure measured as a function of the strain rate on Ca-bentonites and the relationship used in SR-Site.*

The calculation method has also been improved by including the rate dependence of the strength of the bentonite as a strength that is determined individually for each element in the buffer and that is a function of the strain rate (s^{-1}) instead of the shear rate (m/s). (Previously the strength of the bentonite has been taken into account by means of a constant rate corresponding to the total shear strength.) With the new material model and the new calculation method, the impact on the canister of a rock shear has decreased and will, according to the calculations, not induce critical stresses in the canister. A final report of the results of this work will be made within the framework of SR-Site.

Equivalent calculations for KBS-3H show that the effect of the perforated supercontainer on the strains in the canister after a rock shear is insignificant and that the results of the calculations for KBS-3V can also be used for KBS-3H.

Studies have been made of which THM processes can cause increased brittleness and increased strength in the buffer /24-28/.

The continued studies of the effects of high water pressures on swelling pressure and strength show that an increase in the swelling pressure caused by a high water pressure (under glacial conditions) do not appear to increase the shear strength, which supports an old assumption that it is the void ratio that controls the strength rather than the swelling pressure.

Modelling of buffer upswelling against a backfilled tunnel has also been done /24-14, 24-29/. The results show that the upswelling is greatest if the tunnel is dry. At a buffer density of 2,000 kg/m^3, the maximum upward swelling is about ten centimetres and the critical buffer density on top of the canister is 1,960 kg/m^3. Figure 24-10 shows the vertical swelling for this case. If the buffer density is only 1,950 kg/m^3, the upward swelling is about seven centimetres and the buffer density on top of the canister is at least 1,920 kg/m^3. These calculations have been done assuming that the buffer is water-saturated and homogenized and has access to water for swelling, while the backfill consists of blocks and pellets without access to water. Supplementary calculations with the new material models for unsaturated buffer that also take the wetting and homogenizing processes into account are desirable (see Section 24.2.11).

The backfill materials Friedland Clay, Asha 230 and IBECO RWC-BF have been investigated in laboratory tests with respect to swelling, homogenization, self-healing and compression /24-15, 24-19/. Figure 24-11 shows a compilation of results from swelling pressure measurements on these materials. Gap width, water quality and pellet fill have been varied in the tests. The results will be used in modelling and evaluation of the homogenization of the backfill. The compression properties of water-saturated backfill materials have been measured by means of oedometer tests.

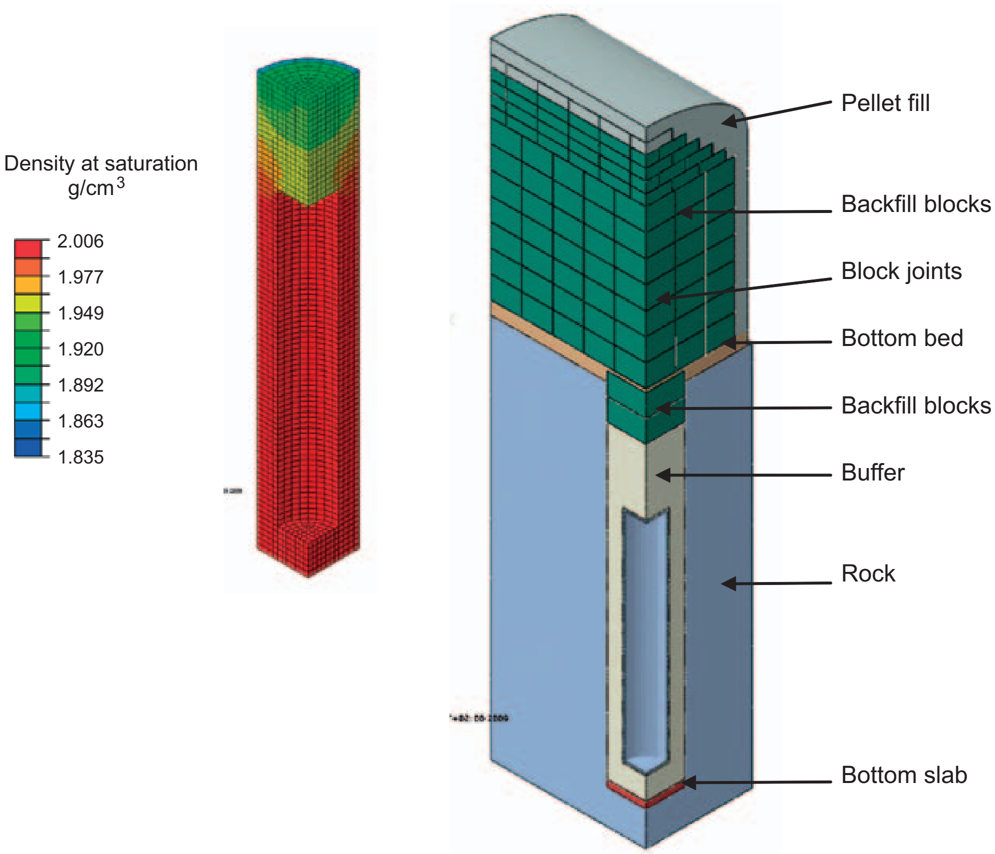


Figure 24-10. *FEM model of unwetted backfill and examples of results of upward swelling in the reference case /24-14/.*

The self-healing capacity of different backfill materials after piping and erosion have occurred has been investigated in a number of tests. In the tests, hydraulic conductivity was measured before and after drilling of a small-diameter hole in the water-saturated material. A compilation of the measurement results shows that Asha and IBECO RWC-BF homogenize well at dry densities above about 1,400 kg/^{3}, while Friedland Clay needs a dry density above 1,700 kg/m^3 to homogenize /24-15, 24-19/. Pellet fills give poor homogenization due to the low density, and a 30/70 mixture of crushed rock and bentonite also gives unsatisfactory homogenization, despite very high dry density. The pellet fill needs help from the bentonite blocks to homogenize.

Modelling of homogenization of the system of blocks and pellets in the deposition tunnels has been done within the SR-Site project. Different rock breakout cases have been studied. The results show that for the extreme case of maximum breakout, a residual inhomogeneity in the pellet-filled gap is obtained that gives a minimum dry density in the gap of just under 1,400 kg/m^3. But in the absence of material data, MX-80 has been used as a calculation example.

Programme

Homogenization and self-healing are important processes for repository function, and much of the current work will continue.

The laboratory tests with different types of swelling will continue, and some of them will be used as calculation cases for TF EBS. They will be used to verify, calibrate and possibly update the material model. The technique for modelling this swelling with FEM will also be developed for both ABAQUS and Code Bright. Initiatives are required to handle the large outswelling that can occur, which often leads to deformed assemblies. In addition to the simple tests with axial or radial swelling, more complicated geometries corresponding to eroded sections in deposition holes will also be simulated in the laboratory on a scale of 1:10 and be modelled with FEM.

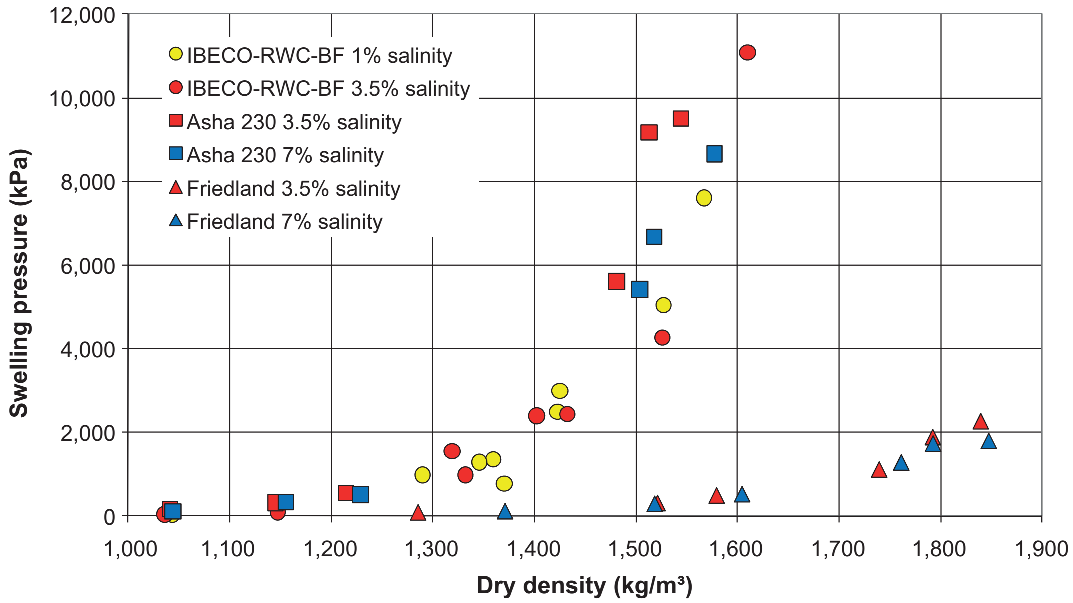


Figure 24-11. *Relationship between swelling pressure and dry density for investigated backfill candidates /24-15, 24-19/.*

An important factor that helps limit the upswelling of the buffer and bentonite extrusion in fractures is the friction between bentonite and rock wall and between bentonite and canister. This friction will be studied by shear tests between bentonite and different materials.

Modelling of the density distribution after different types of bentonite losses will then be carried out with the new updated models and the improved calculation techniques. A full-scale test to verify one or more critical bentonite losses is being considered.

The BB (Big Bertha) test, where outswelling occurs through the perforated container with full-scale hole and gap, is planned to be carried out during the coming year. After full outswelling, the density distribution will be determined and compared with modelling results.

The studies concerning what processes may cause enhanced strength and increased brittleness will continue. They will be conducted in an EU project called PEBS. In addition to a literature survey, a large number of laboratory tests and above all uniaxial compression tests will be done in this project. Other tests that may be performed are triaxial tests and plotting of water retention curves.

Preliminarily, these uniaxial compression tests will be focused on effects of the following:

- field-exposed material from other projects than LOT and other materials than MX-80,
- elevated temperature in the laboratory at different degrees of saturation,
- cycles with elevated temperature,
- compacting direction, anisotropy and pore pressure during sample preparation,
- structure of the material at preparation, for example pellets, powder, grinding, occurrence of small cracks or stress concentrators.

Continued measurements of swelling, homogenization, self-healing and compression properties are planned for the different backfill materials.

Similar laboratory tests and parameter evaluations as those done for the buffer need to be done for the backfill to update the models for backfill materials.

Modelling of the homogenization process will also be done with the updated models. The models will also be used in the studies of how the backfill affects the plug.

The intention is to update the entire mechanical description of the silo in SFR. This includes THM modelling of the water saturation phase and collection of swelling pressure data for GEKO/QI.

24.2.10 Thermal expansion

This process is included in the THM model, see next section.

24.2.11 Integrated studies – THM evolution during the water saturation phase

When buffer and pellets have been installed in a deposition hole, the buffer will absorb water from the surrounding rock. Water uptake also takes place in the backfill after installation. The water uptake affects and is affected by a number of coupled thermal, hydraulic and mechanical processes. Temperature differences between different parts of the buffer lead to pressure differences, which in turn lead to movement of the pore water to equalize the differences. This process could create high pressures against rock and canister if water saturation occurs before the temperature maximum. Thermal expansion is included in the coupled THM model.

Extensive experimentation and model development work within integrated THM evolution are being conducted in SKB's programme. The least certain part of the coupled model for unsaturated bentonite is the mechanical submodel. Mechanical submodels are included in both ABAQUS and Code Bright.

Conclusions in RD&D 2007 and its review

In RD&D Programme 2007, SKB intended to carry out extensive work with these integrated studies. The FEM codes ABAQUS and Code Bright were to be further developed and a better choice of parameters was to be evaluated. SKB also said that there was a need to further develop Code Bright, both with regard to the code's selection of mechanical and hydrodynamic constitutive relationships, and with regard to its selection of element types.

Modelling and comparisons with measurement data from first the two field tests in URL and then the Canister Retrieval Test in the Äspö HRL were planned within the international project TF EBS (Äspö Task Force on Engineered Barrier Systems). The field studies of the Prototype Repository and TBT were to continue, while the results of the Canister Retrieval Test were to be used to evaluate the models. We also intended to consider new field tests of both KBS-3H and KBS-3V.

Work within the Baclo project was planned to focus on piping and erosion during the backfilling operation. Investigations of the processes for healing of pipes and water saturation and homogenization of the backfill were also planned.

Newfound knowledge since RD&D 2007

The coupling of the THM processes during the saturation phase is not crucial for the safety of the final repository, but is important for an understanding of how the buffer is wetted, swells and is homogenized under the influence of temperature changes. It is also important for the understanding and evaluation of the field tests in the Äspö HRL.

Coupled THM processes during the water saturation phase and their interaction with the rock, the canister and the backfill have been studied by development of material models and modelling of different experiments and scenarios, by measurement of the THM processes in large-scale tests in the field and by small-scale laboratory tests. See also Section 24.2.3.

Model studies

The homogenization process in the unsaturated buffer and backfill was not investigated in SR-Can. A big step has been taken with regard to parameter determination for the mechanical model in Code Bright. This has also included modifications of the constitutive laws, for example the "elastic" swelling module has been coupled to a void-ratio-dependent swelling pressure curve. This modification has been implemented in the source code for Code Bright and has proved to be relevant. Furthermore, the hysteresis effects connected with swelling/compression cycles have been taken into account and have been shown to contribute to a persistent heterogeneity during the homogenization process.

In the investigation of buffer homogenization in SR-Site /24-14/, an analytical model was used to study what controls homogenization in the buffer. This investigation shows that the wetting sequence of, and the difference in total pressure in, the block and the pellet-filled gap has a considerable effect on the final difference in the average void ratio in the block or the pellet-filled gap.

The response of FE models (developed for the Code Bright solver) is being compared with sensor data from the Canister Retrieval Test in order to determine whether different parameter sets are suitable. The effect of different pellet-filled gap widths (i.e. different deposition hole diameters) and different wetting sequences are also being investigated here.

Calculations have been done using the FE programme ABAQUS, in which the mechanical model has been refined, to model the homogenization process in a deposition hole during the water saturation phase with a 3D model (actually an axisymmetric 2D model) in order to study the coupled THM evolution of the whole buffer until full water saturation has been achieved /24-14/. Geometries and block densities etc. equivalent to those used in the Canister Retrieval Test and the Prototype Repository have been assumed. The results show that the final densities will vary locally between 1,990 kg/m^3 and 2,050 kg/m^3, regardless of the consequences of upswelling.

In TF EBS, THM modelling has been done of the two Canadian field tests ITT and BCE and of the Äspö CRT test. Modelling of CRT shows that considerable improvements have been achieved of coupled THM modelling with the updated material models for ABAQUS and Code Bright, especially for the mechanical model, although considerable uncertainties still remain.

The level of activity in the modelling of TBT and the Prototype Repository has been low due to other priorities but will increase in connection with the excavation of these tests.

Laboratory tests

In order to get input data for the THM models, several series of oedometer tests and suction tests have been carried out on unsaturated bentonite. In addition, a series of temperature gradient tests is under way.

The risk that the bentonite blocks will fracture during the installation phase and the consequences of this have been investigated in a number of test series. The results show that the blocks fracture under high temperature gradients, when water begins to enter the pellet-filled gap in KBS-3V, and when the blocks in KBS-3H are exposed to high relative humidity /24-24/. The consequences of fracturing of the blocks can be serious and limit the time the blocks can be exposed before the backfill has been emplaced and limits the acceptable inflow in the deposition holes.

Field studies

The results of measurements of temperature, moisture content and pressure and the results from the excavation have been compared with the modelling results. They are in good agreement although questions remain, mainly with regard to the evolution of the total pressures /24-14/.

Follow-up and measurements in TBT and the Prototype Repository have continued and have yielded data that will be used in modelling when these tests are excavated.

Backfill

The properties that are of importance for the evolution of the backfill after installation are mainly the hydraulic and mechanical properties of unsaturated materials. The hydraulic properties are handled in Section 24.2.5 and piping/erosion in Section 24.2.8, while the mechanical properties of the unsaturated backfilling system of pellets and bentonite blocks are dealt with here.

In order to be able to calculate the mechanical interaction between buffer and unsaturated backfill, a number of mechanical parameters of the bentonite blocks and the pellet fill have been determined for Friedland Clay, Asha 230 and IBECO RWC-BF both on blocks and pellet fills /24-16, 24-15/. Furthermore, the effects of different relative humidities in contact with blocks of these materials have been investigated in several laboratory series.

The homogenization of backfill blocks and pellet fill during wetting was analyzed for SR-Site using several axisymmetric models /24-14/. The models show that the backfill material will not be fully homogenized. The remaining difference is so great that the difference between the inner and outer parts is slightly more than 0.2 in void ratio, and the reason for this is friction in the backfill material and hysteresis effects during the swelling/compression cycles.

Programme

The next few years' work will be dominated by the excavation of TBT and the outer section of the Prototype Repository. In addition, a number of laboratory tests are proposed as a basis for improvements.

Model studies

Modelling of the Canister Retrieval Test, TBT and the Prototype Repository will be prioritized model studies for the next few years. Both Code Bright and ABAQUS will be used.

The Canister Retrieval Test will be modelled both as a task in TF EBS and prior to the final reporting of the test in 2010. An evaluation of the results is expected to provide valuable information on both the reliability of the material models and the capacity of the calculation tools.

TBT will be mined during 2010 and differs from the Canister Retrieval Test primarily in that very high temperatures have occurred and in the interaction with the sand shield and the sand filter. The results of modelling of this test will be compared with measurement results both from sensors and from physical determinations after excavation.

Mining of the outer section of the Prototype Repository is planned to take place in 2011. This test differs from the Canister Retrieval Test and TBT in that wetting of the buffer occurs naturally via the rock and there is a mechanical interaction with the backfill. Modelling of the Prototype Repository will be done in part due to the fact that one of the deposition holes has been proposed as a task in TF EBS.

Another task in TF EBS is to perform a sensitivity analysis of simple THM modelling cases to identify the most important processes and parameters and determine how they affect the results.

In order to improve the state of knowledge regarding homogenization, detailed studies are planned of the hydromechanical process in buffer and pellet-filled gaps. Knowledge gained from these investigations will be used to calibrate and develop models.

New relationships in the mechanical material model have recently been successfully formulated and implemented in Code Bright. These new properties have increased the generality and robustness of the numerical tool. Further development is planned with regard to similar modifications.

Both experimental data and thermodynamic relationships indicate a strong connection between the water retention properties of, and the mechanical processes in, the buffer materials. This aspect appears to be one of the controlling mechanisms and should therefore be implemented in Code Bright.

There are processes that cannot be described correctly by the mechanical material model that is currently used. Further theoretical development will take place to improve the generality of the model. More general versions of the model will then be implemented in the THM code (Code Bright).

The work of developing material models should strive for models that:

- simulate the measured behaviour of the materials,
- have a good basis in the controlling mechanisms,
- have parameters that are closely related to "well defined" material characteristics,
- are generally applicable,
- are numerically stable,
- are "simple" to use.

Laboratory experiments

The laboratory experiments for improving the material models will continue with e.g. temperature gradient tests. The mechanical model will be further developed with laboratory experiments to investigate the following effects:

- volume change (undrained oedometer tests with constant water ratio),
- hysteresis (influence of initial water ratio and temperature under constant volume),
- anisotropy (influence of compacting direction on swelling pressure and possibly on strength),
- water uptake (RH measurement and determination of gradients in water ratio and density).

Field studies

Follow-up and excavation of TBT and the outer two deposition holes in the Prototype Repository.

Backfill

The laboratory experiments and the model calculations will be updated with new backfill candidates. A critical parameter in the swelling calculations is the size and mechanical properties of the horizontal joints between the backfill blocks. If possible, experiments will be done to permit better modelling of these joints.

The coupled THM evolution as described above will be studied in a general project concerning water saturation, self-healing and homogenization. See also Chapter 13.

The goal of the project is to clarify, by analysis of experimental data, that we have a sufficiently good understanding of the processes of piping, erosion, water saturation, self-healing and homogenization in buffer and backfill. This understanding is required to determine the evolution of density and swelling pressure in the interacting system of buffer, backfill and plug for the time between installation and repository closure as well as for the time after closure. When this development work is finished, the design premises for these system parts can be updated. This also contributes to a better long-term process understanding.

In detail, these studies involve describing the water saturation process in the interacting system of buffer, backfill, plug and rock according to the following points:

- examine different scenarios for water saturation of the backfill as a function of the tightness of the plug,
- investigate the possible extent and redistribution of material,
- collect data as a basis for more practically applicable design premises by evaluating the requirements on inflow to deposition holes and deposition tunnels as well as the combination of these.

Further development is needed of the modelling of the deformation that occurs in the backfill due to upward swelling of the buffer.

The long-range conceptual and numerical model for the entire system of buffer, backfill, plug and rock requires further studies as follows:

- Determine tightness requirements and requirements on how much water may be allowed past the plug during construction and hardening, based on scenarios of different cases of piping, erosion, water saturation and homogenization, and based on laboratory tests and analyses, optimize the tightness requirements on the plug.
- Determine design premises for the plug in the form of unevenly distributed load during water saturation of backfill and the bentonite part of the plug.
- Evaluate the possibility and effects of saturating the backfill through the plug.

24.2.12 Advection

Solutes can be transported in pore water by pressure-induced flow. The process is of importance in the buffer during the unsaturated period when a net flow of water takes place into the buffer. The most important requirement on the buffer material is that it should prevent flow around the canister under saturated conditions. Solute transport in the pore water is then dominated by diffusion, see Section 24.2.13. Water flow in the buffer under unsaturated and saturated conditions is dealt with in Sections 24.2.5 and 24.2.6. It was stated in SR-Can that advection could be a dominant transport process when the density of the buffer is low, as a result of for example erosion, see Section 24.2.20.

Advection in the backfill is dealt with in the section on water transport under saturated conditions (Section 24.2.6).

24.2.13 Diffusion

Diffusive transport tends to equalize differences in the chemical potential of the constituent molecules or ions, usually with the result that the substances can be transported from areas with higher concentration to areas with lower concentration. However, in the special electrochemical environment that exists in compacted bentonite, concentration differences may remain even after equilibrium has been reached. This is especially true for ions.

The diffusion process is strongly coupled to nearly all chemical processes in the buffer, since it accounts for transport of reactants to and reaction products from the processes. Diffusion is thereby a central process for the entire chemical evolution in the buffer.

The diffusive transport capacity is normally described with diffusion coefficients that are unique for each type of molecule or ion and is also dependent on virtually all present molecules and ions. Typically, the diffusion coefficients in the buffer are reduced by a factor of 50 or more in relation to pure water. See also Section 24.2.25.

Diffusion in the backfill and diffusion in the wall buffer in SFR can be handled in the same way as diffusion in the buffer.

Conclusions in RD&D 2007 and its review

SKI commented that other chemical and physical properties of the buffer, besides tightness and swelling pressure, need to be specified. The Swedish National Council for Nuclear Waste urges SKB to devise models for transport through the bentonite for the most important radionuclides.

Newfound knowledge since RD&D 2007

The material model for bentonite, which is also used to calculate the dependence of the swelling pressure on temperature (Section 24.2.4) and salinity (Section 24.2.14), has been applied to ion diffusion and resulted in a simple mathematical framework that describes diffusion for both negatively and positively charged ions as well as for neutral species. The framework entails a simplified handling of diffusion in bentonite and provides an explanation for the dependence of measured effective ion diffusion coefficients on background concentration /24-30/.

Certain concepts in this framework have been utilized in the diffusion part of the quantification of mass loss due to colloid formation /24-31/.

Diffusive outbound transport of sulphate has been studied in purified montmorillonite with a geometrically well-determined gypsum source and in through-diffusion tests with saturated gypsum solution /24-32/. The results show that the same ion equilibrium exists between mineral and clay as between solution and clay. Furthermore, a considerably higher concentration gradient was found in the case with gypsum solution in sodium-dominated montmorillonite than in pure calcium montmorillonite. This indicates that the solubility of gypsum is dependent on the sodium/calcium ratio in the montmorillonite. The results also show that the controlling chemical principles in this type of test are the same as in equivalent tracer diffusion tests and indicate that the above-described material model and interpretation are correct. This has implications for how bentonite should be modelled geochemically as well as mechanically.

Programme

In the LOT modelling project, laboratory tests are performed to measure diffusion in pure calcium and sodium montmorillonite of chloride and sulphate solutions for the main purpose of developing the material model for bentonite.

Measurements of diffusion rates in montmorillonite on a short timescale (nanoseconds) by means of neutron scattering experiments show that this diffusion is very similar to the one measured in pure water for the low water ratios that exist in the buffer as well. At the same time, diffusion coefficients measured in bentonite on longer timescales, relevant for laboratory and field tests, have much lower values than those in pure water. In order to link diffusion in bentonite on different timescales, molecular dynamic simulations of diffusion in montmorillonite are being carried out on a short timescale (ns) along with work with analytical models within TF EBS. These efforts will continue during the coming period.

Continued theoretical and laboratory studies will be carried out for the purpose of improving our fundamental understanding of transport properties in bentonite. Evaluation and further development of existing reactive transport codes with respect to bentonite-specific properties will be carried out.

24.2.14 Osmosis

The buffer's sealing properties, primarily high swelling pressure and low hydraulic conductivity, are intimately linked to the bentonite's affinity for water. For a given bentonite material, this affinity declines as the quantity of absorbed water increases. The relationship can be measured and is usually described with a so-called water saturation curve, see also Section 24.2.5. Other components in a repository system also have a varying affinity for water, giving rise to competition for the water. The swelling pressure of the bentonite is thereby affected in a way analogous to the change of osmotic pressure in a saline solution. The process of osmosis is therefore intimately linked to diffusion, see Section 24.2.13.

In highly compacted bentonite, the charged individual montmorillonite layers have very limited mobility due to their size and the large number of layers. The prerequisite for equilibrium in such a system is that the product of the chemical activities of the diffusible ions is equal in the groundwater and in the pore water between the bentonite layers, and that electrical neutrality exists in both compartments. An increase in the groundwater's ion concentration leads to a new osmotic equilibrium, which entails a reduction in the bentonite's swelling pressure. The pressure changes can be calculated, since the activity of counterions and ions in the groundwater can be determined. A similar swelling pressure reduction occurs when the groundwater freezes, see Section 24.2.4.

Conclusions in RD&D 2007 and its review

In its review of RD&D Programme 2007, SKI expressed a desire that SKB would explain the criterion for maximum salinity in the buffer.

Newfound knowledge since RD&D 2007

Measured swelling pressure effects due to freezing have been shown to be completely analogous to swelling pressure effects due to salinity, see Section 24.2.4.

Programme

SKB considers that the process has been well illuminated with respect to sodium-dominated systems and that further research is not warranted.

Certain additional measures are planned for calcium-dominated systems because the calculation method is considered to underestimate the counterion concentration at the mineral surfaces for divalent ions, which in turn leads to an overestimation of the pressure reduction in the calcium case. SKB believes that it is essential to ensure a correct conceptual picture and therefore intends to continue with more advanced modelling tools and apply improved theories (inclusion of correlation effects) in order to describe calcium-dominated systems.

Within TF EBS, continued molecular dynamic simulations are planned of ion distributions in the pore water, as well as of how concentration differences with an external solution are maintained in equilibrium. These results are compared with fundamental thermodynamics and analytical continuum models, where these are applicable (the Poisson-Boltzmann equation).

For the material in the silo in SFR it may be of value to conduct further studies of hydraulic conductivity and swelling pressure as a function of salinity.

24.2.15 Ion exchange/sorption

In montmorillonite, electrical charge neutrality is maintained by the fact that positively charged counterions in the pore water compensate for the negatively charged mineral layers. Under non-saline conditions, the counterions will therefore accumulate around the montmorillonite despite the fact that they are diffusive. In contact with a saline solution, however, a charge-preserving diffusive exchange will take place between ions associated with the montmorillonite and positively charged ions in the external solution until equilibrium prevails. Redistribution of counterions in this manner is called ion exchange. The total ion exchange capacity is determined in the laboratory by completely ion-exchanging the clay to a specific ion.

In a buffer, the set of counterions will vary with time and be dependent on the groundwater chemistry and the set of accessory minerals in the bentonite. Different sets of counterions lead to different physical properties in the bentonite.

In addition to ion exchange, the bentonite is also able to sorb certain ions by surface complexation on the constituent minerals, especially on the edges of the montmorillonite layers. Ions bound to the clay in this way are fixed and therefore not diffusive. Surface complexation is strongly pH-dependent, with greater fixation at higher pHs.

Sorption in the backfill and in the wall buffer in SFR can be handled in the same way as sorption in the buffer.

Conclusions in RD&D 2007 and its review

In the review of RD&D Programme 2007, SKI expressed that SKB should pay greater attention to cementation processes, the link between ion-exchange processes, alterations of smectite and the risk of structural decomposition of the clay. Furthermore, SKI said that the swelling properties of the calcium bentonite need to be characterized more, and that the impact of copper ions on buffer should be studied.

Newfound knowledge since RD&D 2007

The ion exchange equilibrium between calcium and sodium has been studied in pure montmorillonite as a function of density /24-12/. No appreciable differences were noted for the selectivity coefficient. It was also demonstrated how the swelling pressure declines with an increasing calcium fraction for lower densities as well as how the swelling pressure change as a function of the calcium/sodium ratio is thermodynamically reversible.

A compilation of experimentally measured equilibrium constants for ion exchange and sorption reactions with Fe(II) has been published /24-33/.

Laboratory studies of montmorillonite gels have shown that the ability of montmorillonite to form gel is drastically impaired in pyrophosphate solutions /24-12/. This is interpreted as indicating that gel formation ability is dependent on the amount of positive charge on the edges of the montmorillonite layers, which is eliminated by sorption of negatively charged pyrophosphate ions. The results are of great importance for our understanding of how colloids are formed, see Section 24.2.20.

Programme

The programme includes measurements of swelling pressure and hydraulic conductivity in Cu(II)-ion-exchanged montmorillonite.

The ion exchange and diffusion processes are strongly linked, and a common research programme is described in Section 24.2.13.

24.2.16 Montmorillonite transformation

The desirable physical properties of the buffer, especially swelling pressure and hydraulic conductivity, are dependent on the montmorillonite's affinity for water. A mineral with a layer charge near zero, for example pyrophyllite, has an insignificant affinity for water, providing radically different properties than in montmorillonite. A minor increase in the montmorillonite's layer charge, and thereby more balanced cations, leads to a greater affinity for water. But if the charge increases enough, the ions will be fixed to the mineral layers, resulting in reduced affinity for water. The end product of such a process is mica minerals, whose affinity for water is also insignificant. The typical properties of the montmorillonite are thus a consequence of a medium-high layer charge.

Potassium ions are fixed at a lower layer charge than sodium ions, which are in turn fixed at a lower charge than calcium ions. The mineral illite has a layer charge between those of montmorillonite and mica. Potassium ions are fixed to some extent in an illite clay, but not sodium or calcium ions. Fixing of multivalent ions, usually iron or magnesium, can also take place via a bridge of hydroxide, which gives a chlorite mineral.

In order for a montmorillonite to be transformed towards illite or chlorite, there must be an increase in the layer charge, which can be brought about by release of silicon, exchange of aluminium or change of valence in the structure.

In the event of a transformation, secondary processes may be of importance for the performance of the buffer. Release of silicon would probably lead to precipitations of different silicon minerals (see Section 24.2.18), which can affect the rheological properties of the buffer.

In SFR, the saturated bentonite will be in direct contact with concrete, and the pH of the water may therefore be very high. There is great uncertainty regarding how bentonite reacts at high pH. The expected reactions are described below.

Conclusions in RD&D 2007 and its review

In its review of RD&D Programme 2007, SKI points out that:

- SKB should better justify the temperate criterion for the bentonite buffer and more thoroughly investigate the risk of dehydration. In addition, the consequences of the buffer remaining unsaturated for a long time needs to be studied further.
- Prior to a licence application, SKB should report as much as possible from the demonstration tests on Äspö. SKI comments that SKB has a good set of in situ experiments and that it is important that these be completed and the results used to support the licence application. However, SKB should take into account the risks that long-term experiments will fail, develop models for interpretation, take into consideration when decisions have to be made, etc.
- SKB should better account for cementation processes, the link between ion-exchange processes, alterations of smectite, and the risk of structural decomposition of the clay. SKI assumes that samples that exhibit cementation will be account for in the safety assessment.
- SKB needs to study the effect of copper ions on the buffer.

Newfound knowledge since RD&D 2007

Bentonite from the A2 test parcel in the LOT series at the Äspö HRL has been further studied /24-34/. This test parcel has been exposed to accelerated conditions (maximum temperature 130°C). A small but significant increase in ion exchange capacity in the hottest parts has been verified by two independent laboratories, see Figure 24-12. This increase in ion exchange capacity is correlated to an increase of magnesium ions in the clay. Minor rheological changes have also been measured in the hot parts (cementation). Studies relating to KBS-3H have also included how different metals (iron, titanium, copper) affect the buffer (see Sections 16.1.1 and 24.2.18).

High pHs (around 13) have a negative effect on the stability of the montmorillonite. However, water with a pH in the range 10–11 reacts very slowly with montmorillonite. One of the performance indicators for the buffer is therefore that the pH of the groundwater should be lower than 11. Since the reaction rate between water and bentonite at pH 11 is so slow, it is difficult to study the process in laboratory experiments. The focus in the research programme is therefore on natural systems and field tests. SKB is participating together with Posiva (Finland) and the NDA (UK) in a project to study long-term reactions between natural bentonite and hyperalkaline water from the Troodos ophiolite on Cyprus. The pH values in this groundwater lie in the range 10.0–11.9, which is also typical of a low-pH cement. The conceptual model for the reaction model is presented in Figure 24-13. So far the project has focused on identifying sites that agree with the conceptual model.

Cement-clay interactions have been studied in the EU ECOCLAY II project /24-35/. This project shows that dissolution of montmorillonite in an alkaline environment increases the concentration of Al, Si, Mg and Na in solution. These elements can be precipitated in the form of various silicates and aluminosilicates. If the environment is also rich in potassium, the non-swelling mineral illite can be formed. If the pH increases even more close to the surface of the concrete, illitization may be followed by precipitation of phillipsite-K (an aluminosilicate).

The reactions that are expected to occur in the bentonite when the pH increases (in a calcium-poor environment) are summarized below /24-35/:

Montmorillonite	clay mineral of the smectite type that has shrinking and swelling properties and a high ion exchange capacity
↓	
Beidellite	clay mineral of the smectite type
↓	
Saponite	clay mineral of the smectite type
and	
Clinochlore	rock mineral. Clinochlore is the most common chlorite
↓	
Zeolites	rock minerals, hydrated aluminosilicates such as analcime, chabazite, mordenite and phillipsite-NaK
↓	
Gismondine	rock mineral of the zeolite type
and	
Gyrolite	rock mineral, a silicate without aluminium

The chemical interaction between concrete barriers and bentonite has been studied for SFR 1. The purpose has been to estimate both the timescale and the magnitude of potential changes in the chemical composition of the barriers and consequential changes in the transport properties of the barriers /24-35, 24-37/. The main purpose of the latter study /24-37/ is to see the impact of climate change, especially permafrost, farther in the future. The calculations have been carried out with coupled chemistry and transport models (PHAST in /24-35/ and PHREEQC-2.13 in /24-37/). This enables an assessment to be made of the combined effect of the processes that take place simultaneously in the system. Porosity changes are not included in the version of PHAST that is used in the modelling /24-35/.

The geochemical calculations show that quite a few changes are expected in both concrete and bentonite barriers in the silo repository over a period of 100,000 years. Figure 24-14 illustrates the expected mineral transformations in the silo buffer after 10,000 years. The montmorillonite is gradually transformed on contact with high-pH water, and after 10,000 years a third of the total quantity of montmorillonite in the bentonite has been dissolved and calcium silicate minerals, zeolites and new clays have been formed at the interfaces to the concrete silo and the shotcrete. The new minerals have somewhat different properties than the original montmorillonite, including poorer swelling properties and a higher molar volume /24-35/.

Silica sol is a potential fine sealant for construction of the Spent Fuel Repository and is being investigated in situ in the Äspö HRL. After admixture of ten percent NaCl (5:1 silica sol:NaCl), silica sol is pumped into boreholes and fractures and normally gels within an hour. During the normal construction process, it should not be possible for the sealant to come into contact with bentonite or other buffer/deposition materials. An unforeseen leak of ungelled silica sol could, however, adversely affect the bentonite barrier. Under normal conditions, the silica sol particles gel irreversibly with each other by colloidal aggregation. In the event of contact with bentonite, there is a risk that the same process will occur with the montmorillonite.

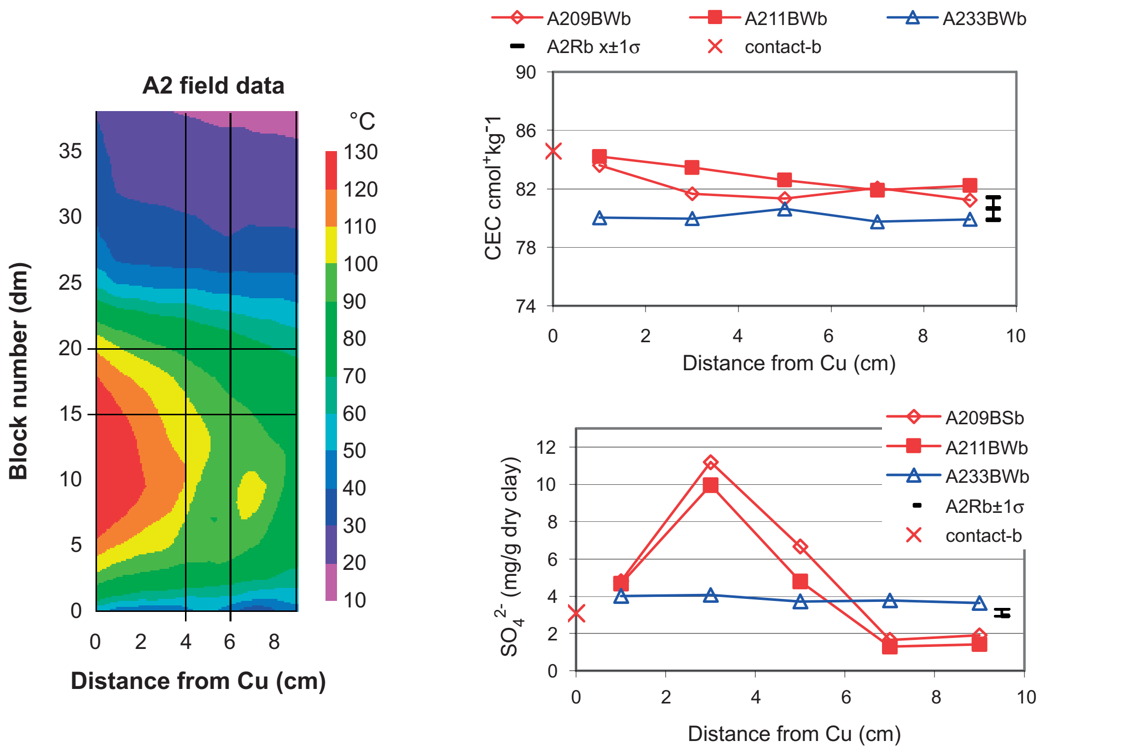


Figure 24-12. At left: Temperature profile during operation in the A2 test parcel in the LOT series. Top right: Measured cation exchange capacity (CEC) in blocks exposed to high temperature (A209BWb and A211BWb) and low temperature (A233BWb) and for a sample in direct contact with the heater in the hottest part (contact-b). The diagram also shows the variation in cation exchange capacity among 5 reference samples (A2Rb). A small but noticeable increase in the cation exchange capacity can be noted in the hot parts of the test parcel. Bottom left: The sulphate content in the same positions. A redistribution has occurred in the hot parts. From /24-34/.

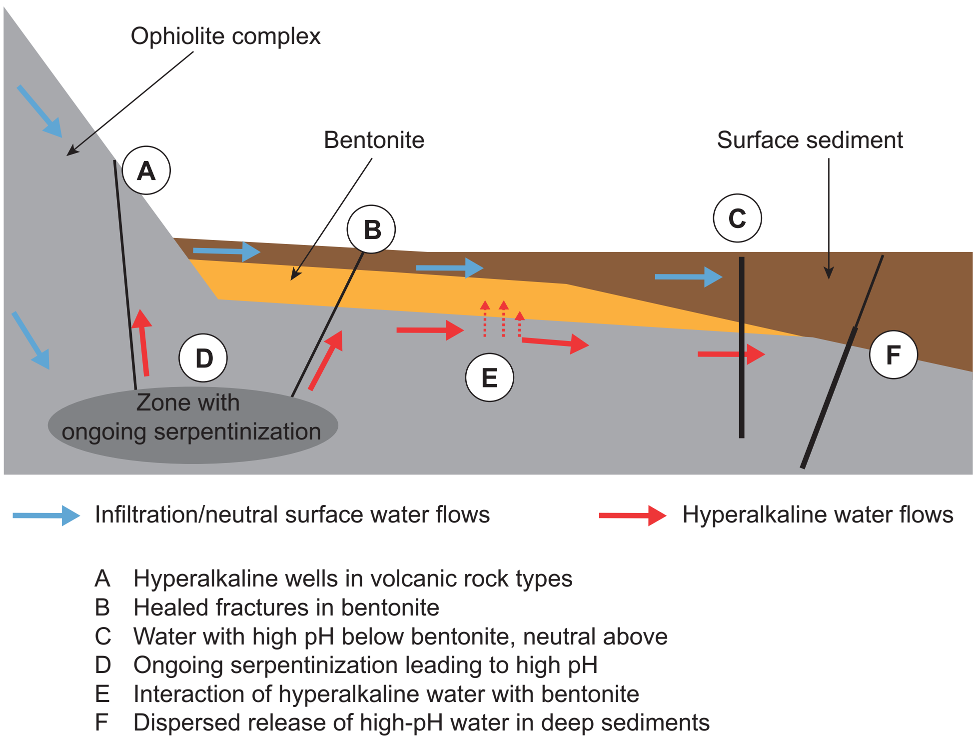


Figure 24-13. Conceptual model of the reaction between hyperalkaline groundwater and bentonite on Cyprus /24-36/.

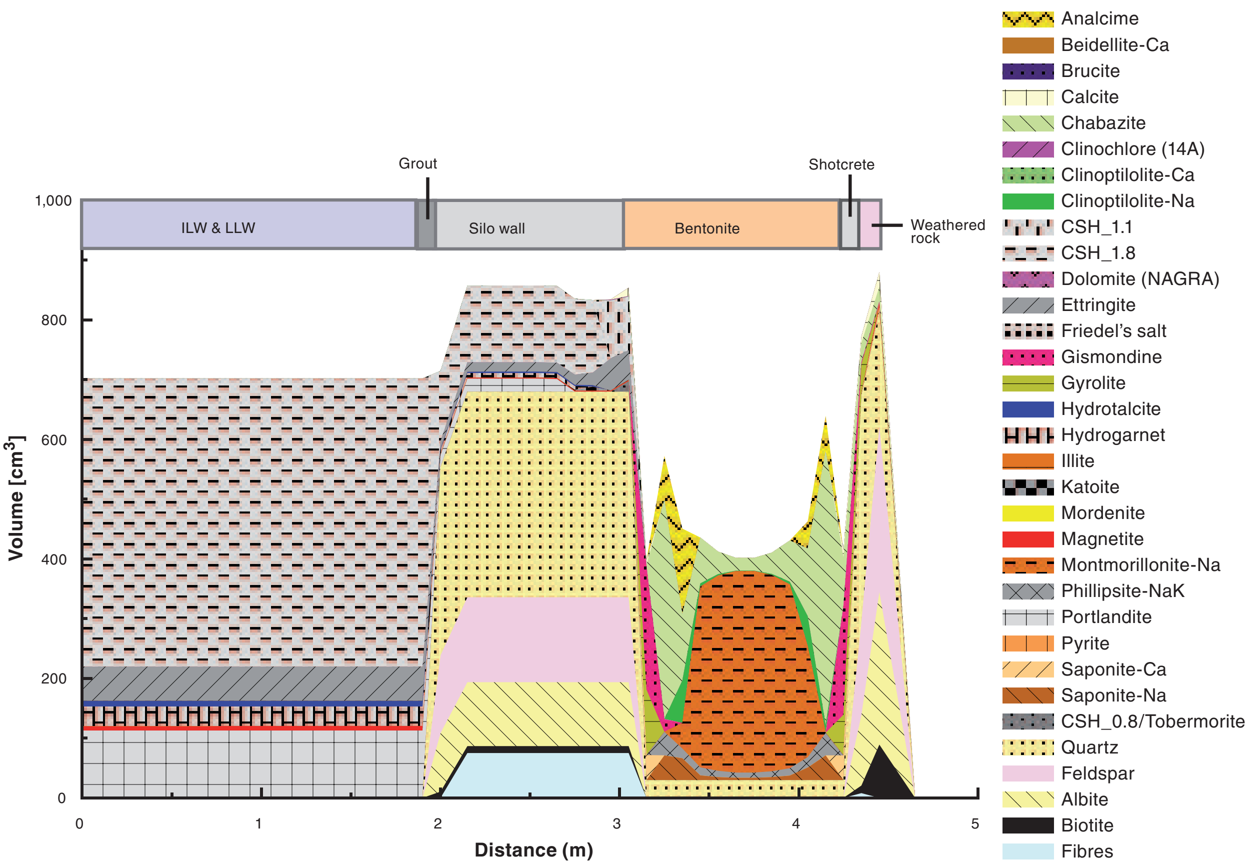


Figure 24-14. *Mineralogical composition of the system of waste matrix – grout – silo wall – bentonite – shotcrete – rock after 10,000 years of leaching (simulation with normal pore diffusion, weathered rock and saline groundwater) /24-35/.*

In order to estimate the effects of silica sol on bentonite at a qualitative level, investigations of the sealant and different mixtures of silica sol and bentonite have been performed using different methods, including Photon Correlation Spectroscopy (PCS), Zeta potential measurements, free swelling and sedimentation tests, Scanning Electron Microscopy (SEM) and Atomic Force Microscopy (AFM).

The preliminary conclusion is that silica sol can aggregate mainly after dehydration (or at low water contents), as well as at high ionic strengths, whereby larger aggregates are formed. However, comparable quantities (by weight) of silica sol and bentonite are required for significant effects on the bentonite as a bulk material.

Programme

A collaboration has been initiated with Nagra to gain greater knowledge of montmorillonite transformation. In this collaboration, a laboratory programme has been devised for the main purpose of verifying existing models and parameter values for montmorillonite transformation under as repository-like conditions as possible. The results of the laboratory study will be used in modelling to study coupled processes, for example dissolution/transport of silicon in a thermal gradient.

SKB will investigate the possibility of transformation under dry conditions via a literature review and possibly a laboratory programme.

Further studies will be conducted on material from the LOT test for the purpose of clarifying the causes of the measured rheological changes. Planning for mining of LOT is subject to the needs of the SR-Site safety assessment and will be finalized in conjunction with the evaluation of the completed safety assessment.

Hopefully the Cyprus project will continue with sampling and analysis of bentonite that has been in contact with high-pH water. But a prerequisite is that a suitable and relevant site can be identified.

Reactions between low-pH cement and bentonite will also be studied in a new experiment on Äspö, see Section 21.2.10. The intention is to install parcels on the same scale as ABM (see Section 24.2.17), but with a centre pillar of cement instead of a heater. The experiment is a part of SKB's long-range programme in the Äspö HRL. No excavation of the parcels that contain bentonite is expected to occur within the next 10-year period.

The interaction between various kinds of concrete and cement (OPC, sulphate-resistant and low-pH cement) and bentonite will be studied within the project "Long-term durability of cement," see Section 21.2.10. Coupled chemistry and transport models of substances in the silo barriers may be developed for another software than the one used today. The geochemical evolution of the bentonite will then be included in the models.

The microscope studies of the interaction between silica sol and bentonite have left some unanswered questions, which will be given closer study during the coming period. Tests will be done with Ca montmorillonite, since Ca ions have a known effect on silica sol aggregation. These investigations can hopefully provide more specific answers to how colloids of silica sol and bentonite aggregate under different conditions.

24.2.17 Iron-bentonite interactions

Metallic iron in an anaerobic environment normally corrodes very slowly, forming iron(II)hydroxide and hydrogen, see Section 23.2.6. In an environment with bentonite clay, it is conceivable that Fe(III) in montmorillonite could interact with the metallic iron by formation of Fe(II). Reduction of Fe(III) to Fe(II) in montmorillonite could potentially affect the mineral's layer charge and thereby also its interaction with water and its tendency towards mineral transformation. Corrosion of iron can moreover lead to a higher pH, which affects the solubility of silicon as well as the stability of the montmorillonite.

These interactions were not described in RD&D Programme 2007.

Newfound knowledge since RD&D 2007

The underground test "Alternative Buffer Materials" (ABM) was started in 2006 at the Äspö HRL. The test includes eleven different types of clay of varying smectite content, type of counterion and iron content. Compacted clay blocks are in direct contact with a central heater of metallic iron. The heater has had a target temperature of 130°C. Studies of the reference clays have been conducted with a focus on mineralogical content, but microbiological characterization has also been performed. The first experimental parcel (of three in all) was mined in the spring of 2009. Sampling and storage of the clay has been optimized to preserve the redox conditions in the clay, permitting studies of the iron's redox chemistry and its effects on the properties of the clay. Figure 24-15 shows a bentonite sample of calcigel (block 5) from the retrieved ABM parcel. Distinct colour differences can be seen from the heater, which has been located to the left of the sample. High Fe(II)/Fe(III) ratios were measured with XANES in the area nearest the heater.

Figure 24-15. *A sample from a block in the ABM test at the Äspö HRL, with a distinct reaction zone between iron and bentonite. The sample is about 4 cm long.*

SKB has also participated in the analysis of an iron-bentonite sample from VTT (Technical Research Centre of Finland). The mineralogical content and swelling properties of the clay (in pure water, frozen and in glycol) were studied with XRD (X-ray diffraction). No significant changes could be seen in the properties of the montmorillonite. Higher Fe(II)/Fe(III) ratios were measured with XANES in exposed material compared with the reference material.

Programme

Studies have been initiated with XANES on the synchrotron at the MAX-lab in Lund for the purpose of quantifying the Fe(II)/Fe(III) ratio in the clays from ABM. The focus has been on three clays: MX-80, Asha 505 and Deponit CaN. Testing of swelling pressure and hydraulic conductivity and characterization of microbiological content have been started. Cation exchange capacity (CEC), exchangeable counterions (EC) and water-soluble salts will be determined for all clays. Phase analysis with XRD will be performed on bulk material of all clays from ABM, and on the three prioritized clays also as purified clay fraction ion-exchanged to Mg(II) and saturated with glycol. This is a standard technique for identifying transformations of montmorillonite to illite.

No more ABM parcels are expected to be excavated during the coming 3-year period. However, it is clear that the first parcel will deliver interesting results, and there are therefore plans to install a fourth parcel to obtain greater flexibility in the excavation in the future.

24.2.18 Dissolution/precipitation of impurities

The fraction of the buffer material that is not montmorillonite consists of other common minerals such as quartz, feldspars, gypsum and calcite and small amounts of organic material. The accessory minerals are included here among the impurities in the material, since they do not contribute to the sealing properties of the buffer. In the repository environment, these minerals can dissolve and sometimes re-precipitate, depending on the prevailing conditions. Mineral precipitations in the bentonite can naturally also occur as a result of e.g. montmorillonite transformation (see Section 24.2.16) or interaction with groundwater or a damaged canister.

A redistribution of minerals can alter the sealing properties of the buffer, as well as the rheological properties so that the material becomes stronger and more brittle. See also Section 24.2.19.

Dissolution and precipitation of minerals also affects the ion exchange and sorption processes, see Section 24.2.15, by affecting the local pore water chemistry, see also 24.1.12.

Conclusions in RD&D 2007 and its review

The conclusions from the review of RD&D Programme 2007 that were presented in Section 24.2.16 also apply to this process. SKI also points out that SKB should test VTT's chemical measurement methods for compacted bentonite /24-38/.

Newfound knowledge since RD&D 2007

Dissolution and diffusive transport of gypsum has been studied in calcium and sodium montmorillonite, see Section 24.2.13.

Reactive transport modelling has been done of the A2 test parcel in the LOT project. According to this, the characteristic anhydrite profile measured in parts of the test parcel exposed to high temperatures was formed during the water saturation process.

Reactive transport modelling has been done to bound the influence of iron corrosion on the bentonite in a KBS-3H repository /24-33/. The results indicate large magnetite precipitations immediately adjacent to the iron source, as well as an effect on the bentonite in a limited area around the iron source. The results of laboratory experiments and field tests in ABM (Alternative Buffer Materials) indicate that the affected area is more extensive. A possible reason for the difference between modelling results and experimental results is an initial aerobic corrosion phase.

Programme

The programme includes an updating of the reactive transport modelling of iron corrosion in a KBS-3H geometry. Furthermore, further studies are planned of how accessory minerals are leached out, which is relevant to the risk of colloid formation, see Section 24.2.20. Cementation effects will also be studied during the period, see Section 24.2.19.

24.2.19 Cementation

Cementation is a collective term for a set of processes that lead to rheological changes or deterioration of swelling properties in the buffer. The overall effect is important for the buffer and is gathered under a single heading. Typically, cementation is caused by mineral precipitations in bentonite, and there are therefore strong links to other processes, such as montmorillonite transformation (Section 24.2.16), dissolution/precipitation impurities (Section 24.2.18), diffusion (Section 24.2.13), iron-bentonite interactions (Section 24.2.17) and ion exchange/sorption (Section 24.2.15). There are two main reasons why the effect of cementation is important to take into account in a bentonite buffer:

- Elevated hydraulic conductivity
  Cementation caused by mineral precipitations can reduce porosity. In non-swelling material, this usually leads to a reduction of hydraulic conductivity. In bentonite, however, it is not porosity but the interaction between water and montmorillonite that primarily determines hydraulic conductivity, see Section 24.2.6. A cementation process that reduces porosity can therefore lead to elevated hydraulic conductivity if it also counteracts or reduces the water-montmorillonite interaction, for example as a consequence of montmorillonite transformation. Elevated hydraulic conductivity can, in turn, lead to advective conditions in the buffer, see Section 24.2.12.
- Elevated shear strength
  Cementation can lead to elevated shear strength in the buffer. A displacement of the surrounding rock (for example an earthquake) can then lead to the transmission of excessive stresses to the canister, which can increase the risk of canister damage, see Chapter 23. Precipitations or other processes can moreover lead to a more brittle buffer or reduced strain at break.

The bentonite's swelling pressure is a strong function of density, but also exhibits large hysteresis effects, i.e. the swelling pressure is also dependent on the direction in a swelling/compression cycle, see Section 24.2.9. It is therefore conceivable that the rheological properties have a similar dependence so that a compressed bentonite exhibits higher strength in comparison with a swelled sample at the same density.

Newfound knowledge since RD&D 2007

Minor rheological changes have been measured in the A2 test parcel in the LOT series, see Section 24.2.16.

A comprehensive study has been carried out to investigate the cementation that was identified in the regular investigation programme for the A2 test parcel in the LOT series /24-28/. A large number of series of uniaxial compression tests were carried out on prepared cylindrical bentonite pellets with a height of two centimetres and a diameter of two centimetres exposed to different temperatures, see Figure 24-16. In addition to temperature, density, water saturation and type of bentonite material were also varied.

A similar cementation effect as was noted in the LOT material was measured in laboratory tests after heat exposure (150–200°C) for only 24 hours, see Figure 24-17. The same failure behaviour could also be obtained by means of elevated density or reduced water saturation in the samples.

The material resumed its original rheological properties after drying, grinding and recompaction.

Programme

SKB will continue to study rheological effects of cementation within the EU PEBS project.

Figure 24-16. Example of a uniaxial compression test. The sample is loaded with a constant deformation rate in the axial direction to failure. Axial stress and strain are measured. From /24-28/.

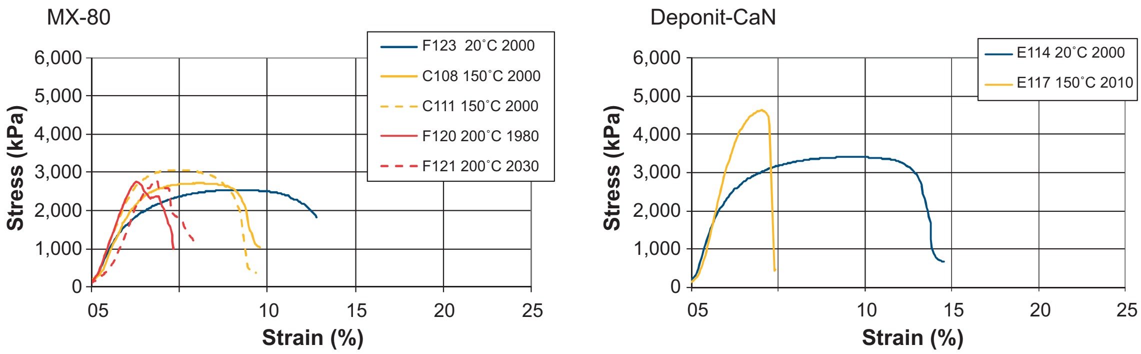


Figure 24-17. Example of cementation effect: material exposed to 150°C or 200°C shows reduced strain at failure in comparison with material of similar density prepared at room temperature. The behaviour is the same for both sodium-dominated (MX-80) and calcium-dominated (Dep-Can) bentonite. Density is given at the end of the ID string in units of kg/m³. From /24-28/.

24.2.20 Colloid release/erosion

The buffer and the backfill in a KBS-3 repository consists for the most part of microscopic smectite particles. During water saturation the particles in the highly compacted blocks will be subjected to very strong repulsive forces. This causes the buffer to swell and the empty spaces in the deposition holes to be filled up. The buffer can also swell short distances into the fractures in the walls of the deposition hole. Water that flows into the fractures could shear off the outermost particles and expose new particles to the flowing water. This problem is exacerbated if the water has very low ionic strength.

Conclusions in RD&D 2007 and its review

SKI says that SKB needs to show that knowledge of buffer erosion is sufficient before a licence application is submitted, that buffer erosion and its consequences need to be bounded in the safety assessment, that a solid basis is needed for calculations of consequences, and that it needs to be shown that buffer erosion can be reduced with alternative designs. SKI also observes that the transition between non-eroded and partially eroded phases, i.e. a condition still without advection, needs to be evaluated.

In the review of SAR-08, SSM points out that SKB should better substantiate the main scenario as a basis for fulfilment of requirements with regard to SSM's risk criterion. An additional calculation case is needed showing the effect of reasonable barrier degradation on the characterization of risk, or alternatively a more detailed explanation of why the effect of all reasonably likely barrier degradation processes can be ruled out for a period up to 40,000 years for the repository parts BMA and Silo.

SSM also notes that in the longer term, such as in connection with updating of future safety assessments, in-depth studies may also be needed of how the bentonite barrier is affected by dilute groundwater. SR-Can showed that the near-coastal Forsmark area is affected by long periods of temperate conditions due to land uplift and infiltration of precipitation water. This gradually leads to more diluted groundwaters, and SSM therefore finds that SKB needs to model the evolution of ionic strength and determine whether there is a risk that the bentonite surrounding the silo may be affected by such waters. Such a study should be based on updated knowledge from ongoing SKB projects regarding which groundwaters can potentially affect bentonite stability.

Newfound knowledge since RD&D 2007

SKB has carried out a project (Bentonite Erosion) to study erosion of bentonite in dilute waters. The purpose of the project was to construct a quantitative model for judging the extent of the erosion process in the SR-Site safety assessment. The project ran from 2007 to 2009. The different phases of the project were literature studies, modelling and experiments. These phases are described below.

Literature studies and information gathering

The goal of the literature studies and the information gathering was to gain understanding and acquire data as a basis for formulating and developing a dynamic force balance model for how montmorillonite clay expands in water and how the particles can form a stable sol (colloidal suspension). The following forces and mechanisms were accounted for: van der Waals forces, forces caused by the electric diffuse double layer around the particles, gravity, forces that cause Brownian motion and the force of friction that opposes motion. A number of models used to quantify these forces under different chemical conditions were summarized and discussed /24-39/.

Modelling

One of the most important parts of the project was to develop a dynamic model for how a bentonite gel expands in fractures and absorbs dilute water during the expansion /24-31/. The model serves as a basis for the quantitative handling of the process in the safety assessment.

The model is based on a description of a force balance between and on the smectite particles in the bentonite, which move in the water. Attractive van der Waals forces, electric diffuse double layer forces, gravity and forces on the particles caused by the gradient of chemical potential act to move the particles in the water. Friction between particles and water can restrain these movements. The diffuse double layer forces are strongly dependent on the type and concentration of ions in the water surrounding the particles. The dynamic model of gel expansion has been successfully tested against gel expansion experiments in test tubes in time and space.

Another part of the quantitative model is a model for the viscosity of dilute gels, which accounts for the influence of the ion concentration as well as the volume fraction of smectite in the gel. The model accounts for the presence of a diffuse double layer, which makes the particles seem larger so that they can interact at low particle densities. The viscosity model uses experimental data to obtain certain necessary fitting parameters, but is otherwise based on established theories of suspension viscosity.

These two models form the core of the quantitative model. Both show a strong dependence on the ionic strength of the pore water. Simulations have been performed for a case where the gel expands outward into a fracture that intersects the deposition hole. Dilute groundwater approaches and passes the gel/water interface. Smectite colloids can move out into the water due to the repulsive forces between the particles and Brownian motion (included in the dynamic model). The dilute gel/sol is mobilized and is transported downstream in a thin region where the viscosity is low enough to permit flow. Sodium will diffuse from the compacted bentonite into and through the expanding gel towards the gel/water interface and further out into the flowing water. Mass transport resistances for ions as well as smectite particles are accounted for in the simulations. The sodium concentration profile in the gel influences the repulsive forces between the particles as well as the viscosity of the expanding gel. Under the most unfavourable circumstances, i.e. at high flow rates and large fracture apertures, considerable loss of buffer can be expected. Results of simulations are presented in Table 24-2. This is the model that is used to quantify bentonite erosion under conditions with dilute groundwaters in SR-Site.

Calculations have also been done to determine under what combinations of flow, groundwater composition and initial composition of the bentonite the water composition in the gel/water interface could be higher than the critical coagulation concentration (CCC). At the same time, the distribution between calcium and sodium as counterions in the clay in the interface was studied. The reason for this is that it has been found that if more than 90 percent of the counterions are divalent, the properties of the clay are altered radically, and there are indications that such a clay would not release any colloids at all.

An alternative model that uses effective stress theory to judge how far bentonite can penetrate out into fractures is presented in /24-12/. The model is partially based on empirical expressions for hydraulic conductivity based on laboratory tests and on rheological measurements. That model only handles expansion up to an aqueous/solid phase ratio of 30. After that the material is regarded as "lost".

A study with Monte Carlo simulations of the clay-water system has been performed to study the importance of the ionic species for the swelling /24-40/. The simulations predict a powerful swelling with monovalent counterions, whereas in the presence of divalent counterions the swelling is limited with an approximately one nanometre thick water layer between the clay layers. The results of simulations of a lamellar clay system are in perfect agreement with small angle x-ray scattering (SAXS) measurements, but in disagreement with dialysis experiments. Apparently there is a lamellar swelling and an "extra-lamellar" swelling.

Experiments

Numerous laboratory studies have been carried out within the "Bentonite Erosion" project. Most of them are presented in /24-12/. Experimental, as well as theoretical, studies have been performed with respect to:

- Free swelling ability.
- Rheological properties.
- Rate of bentonite loss into fractures.
- Filtering.
- Ion exchange.
- Sol formation ability.
- Ion diffusion.
- Mass loss due to erosion.

Table 24-2. Estimated mass loss of buffer as a function of water velocity /24-31/.

Water velocity, m/yr	Smectite release for a 1 mm fracture aperture, g/yr	Penetration of gel into centre of fracture, m
0.10	11	34.6
0.32	16	18.5
0.95	26	11.5
3.15	43	7.0
31.50	117	2.1
315.00	292	0.5

Based on the experimental results, numerous observations have been made, but the most important issue has been to determine under what conditions erosion of bentonite can occur. In SR-Can, a critical coagulation concentration (CCC) of one mM [M^{2+}] or 100 mM [M^{+}] was used as a criterion for when bentonite erosion could occur. The results of the experiments show, however, that this is far too simple a picture. In a system with only monovalent ions, Na(I), the clay can expand "infinitely", and approximately 25 mM Na^{+} is needed to stop colloid formation. If, on the other hand, the clay only contains divalent ions, Ca(II), no colloids are formed even with distilled water. In a real system with both sodium and calcium there is no clear-cut value of CCC, since it is dependent on both the counterion content of the clay and the composition of the water.

Colloid release and erosion at low ionic strengths has been studied experimentally in a swelling pressure cell where initially pure NaCl solutions have been circulated in contact with a mixed Ca/Na montmorillonite. At low ionic strengths, the NaCl solution is very close to equilibrium, since virtually all calcium stays in the clay. Results from such a test are shown in Figure 24-18, and a summary of results from several tests is provided in Table 24-3.

Table 24-3. Summary of erosion experiments with Ca/Na montmorillonite from /24-12/. The X column shows the charge fraction of calcium in the clay. The column "Limit of stability" shows the lowest NaCl concentration for which no erosion could be detected.

Sample	Clay type	X	Limit of stability [Na^{+}]
CaNaErosion01	Milos	0.5	2 mM
CaNaErosion02	Milos	0.5	0.5 mM
CaNaErosion03	Milos	0.25	1 mM
CaNaErosion04	Milos	0.75	3 mM
CaNaErosion05	Wyoming	0.5	4 mM
CaNaErosion06	Kutch	0.5	1 mM

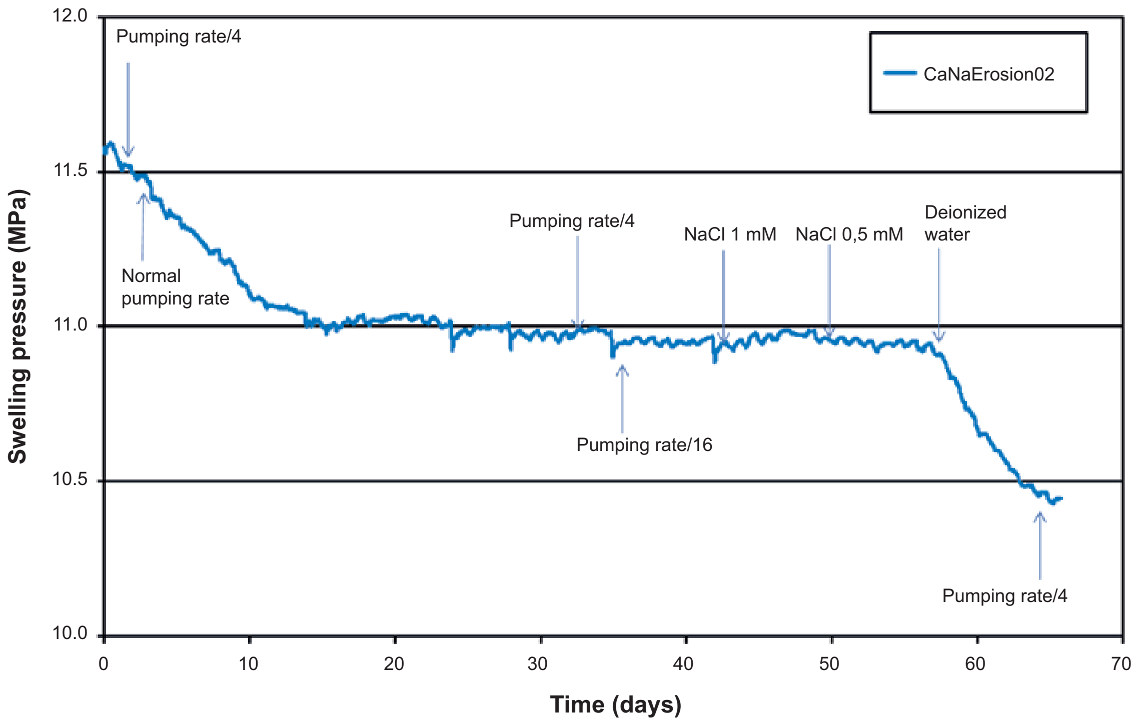


Figure 24-18. *Swelling pressure evolution of eroding test CaNaErosion02 /24-12/. The montmorillonite is of Milos type and the ion population is 50/50 Ca/Na. The solution consists of two mM NaCl(aq) at the beginning of the displayed time period. Note that the erosion stops spontaneously at day 15 without a change in either pumping rate or circulating solution. At day 57 the solution is replaced with deionized water and the loss of montmorillonite begins again, demonstrating the relationship between colloidal sol formation and ionic strength.*

The erosion tests show that erosion does not occur in a mixed calcium/sodium montmorillonite in which at least 20 percent of the counterions consist of calcium when the external solution contains more than four mM charge equivalents. These results are in harmony with the proposed conceptual theory for sol formation and measured equilibrium properties in mixed calcium/sodium montmorillonite. It is not probable that there will ever be a lower fraction of divalent counterions in the buffer than 20 percent.

In another study /24-41/, NMR (Nuclear Magnetic Resonance) was used to study swelling of different clays under different conditions. Examples of results are shown in Figure 24-19 and Figure 24-20. The results show that both calcium and sodium-dominated clays swell quickly and that even calcium clay can expand to four times its original volume. The results of the NMR experiments have been used to test the dynamic model /24-31/ with successful results.

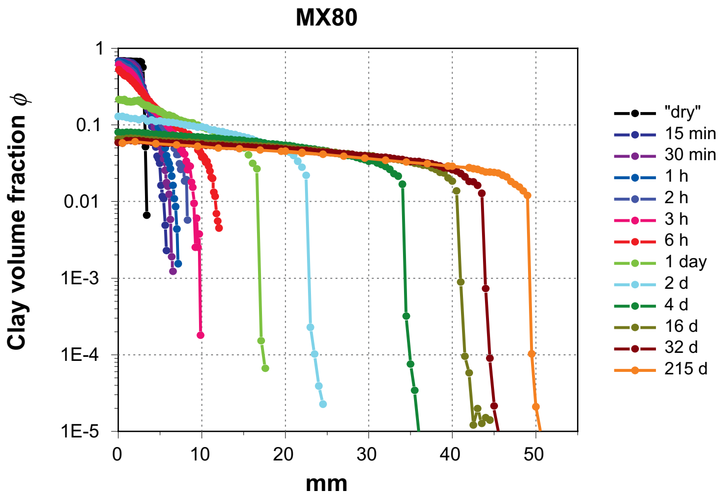


Figure 24-19. Swelling of natural MX-80 in distilled water (note the log scale on the y axis) /24-41/.

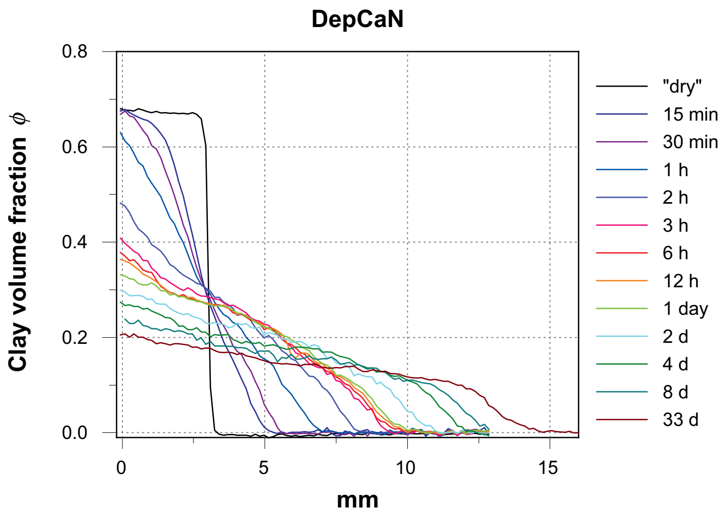


Figure 24-20. Swelling of natural Deponit Ca-N in distilled water (note the linear scale on the y axis) /24-41/.

The project has also included studies of the possibility of stopping colloid release with natural or synthetic filters. The ability of the bentonite clay to self-heal during leaching with deionized water was studied in /24-42/. The investigation focused on the formation of a filter cake made of accessory particles of MX-80 and the separation of solid material for a solution containing 1 percent smectite that passes through the filter cake at a pressure difference of five bar. The experiment showed good separation of smectite particles from the solution when it passed through the filter cake. In all tested cases, the separation was almost complete after a long enough time, indicating that the cake has small enough pores to act as a geometrical obstacle for the small particles. The results of /24-12/ also show that filter pore sizes of less than 0.5 μm effectively stop all montmorillonite particles from getting through, while a filter pore size of two μm has no appreciable effect on erosion. This is illustrated in Figure 24-21.

Programme

Despite great efforts, the results from the "Bentonite Erosion" project do not give a clear picture of how the process can be quantified. The model presented in /24-31/ is based in principle on a pure Na system and probably overestimates the erosion. In order to get more realistic, less conservative results in future safety analysis reports, it is therefore important to continue studies in the area.

Uncertainties still exist regarding how calcium affects the process, especially in mixed Na/Ca systems. A doctoral project has therefore been started for the purpose of constructing a model that can describe swelling and erosion in calcium bentonite.

The actual erosion takes place in a fracture, while most experiments have been done in open pipes or through filters. Additional experiments with erosion in gaps of different widths will therefore be conducted. They will be combined with experiments with different flows to determine the importance of the flow rate for the process.

The models and descriptions that are being prepared for the buffer are also applicable to the backfill and the bentonite in the SFR silo.

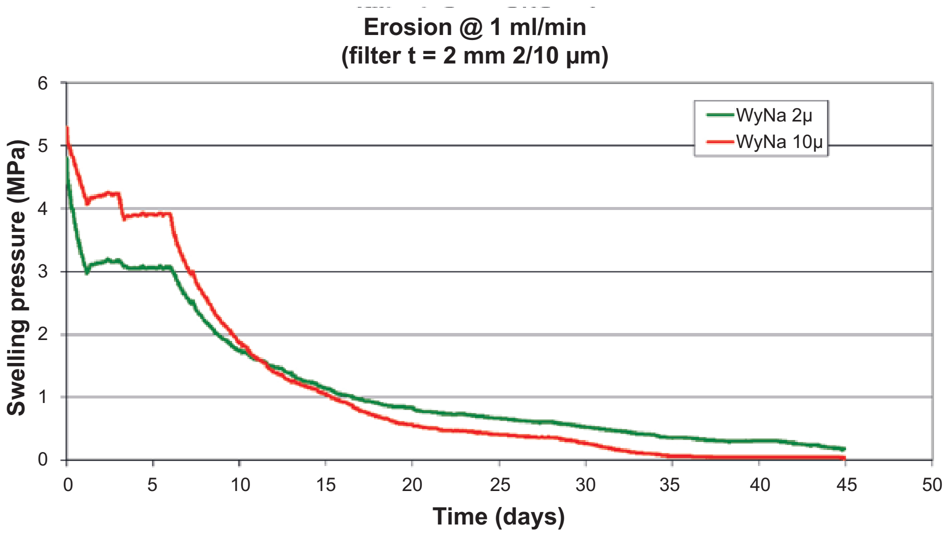


Figure 24-21. *Erosion as a function of filter pore size (2 and 10 μm) for pure Wyoming Na-montmorillonite (WyNa) in distilled water /24-12/.*

24.2.21 Radiation-induced montmorillonite transformation

Montmorillonite in the buffer can be broken down by ionizing radiation. The result of this is a decrease in the montmorillonite concentration and a change in the properties of the bentonite. However, experiments have shown that the accumulated radiation dose to which the bentonite will be exposed in a final repository does not cause any measurable changes in the montmorillonite concentration or the properties of the bentonite. The handling of this process in the SR-Site project is based on the same body of data as was used in ANDRA's safety assessment Dossier-2005. No further studies are planned in the area.

24.2.22 Radiolysis of pore water

Gamma radiation that penetrates through the canister can decompose pore water by radiolysis, forming OH radicals, hydrogen, oxygen and several other components. The oxygen is consumed rapidly by oxidation processes that affect the redox potential, while the hydrogen is transported away. The canister's wall thickness is, however, sufficient so that the effect of gamma radiolysis on the outside is negligible, see Section 23.2.5.

24.2.23 Microbial processes

Microbial processes can give rise to the formation of gases and sulphide under certain conditions. Gas formation could result in mechanical stresses in the repository, while sulphide could corrode the copper canister. In order for microbial formation of sulphide to be of any importance for the life of the canister it must take place very close to the canister's surface. Since the process is primarily of importance for the properties of the canister, it is dealt with in Chapter 23.

The potential for bacterial activity increases in the backfill material with decreasing density and increasing water content. There are also greater amounts of iron and organic matter there.

24.2.24 Radionuclide transport – advection

The main function of the buffer is to guarantee that diffusion is the dominant transport mechanism around the canisters. With an MX-80 buffer with a water-saturated density of 2,000 kg/m^3, the transport capacity for diffusion is at least 10,000 times higher than that for advection.

It is assumed that radionuclides can be transported both advectively and diffusively through the deposition tunnels. Diffusive transport is normally assumed to be dominant and advective transport negligible. But if the conductivity of the backfill should for some reason be higher, advective material transport may be of importance for the safety assessment of a KBS-3 repository.

Transport in the backfill is included in the integrated description of transport in the rock, see Section 25.3.4.

In SR-Can, radionuclide transport is also calculated for a case where the buffer has been lost and the mass transport is dominated by advection. The process is handled in the same way as in SR-Can.

24.2.25 Radionuclide transport – diffusion

The transport of radionuclides through the buffer is mediated by different diffusion mechanisms. It has been established that certain cations can have high diffusivities (be transported more efficiently).

There are different ways of looking at how diffusion takes place in bentonite. Simply expressed, they are based either on a model where the bentonite is regarded as a homogeneous medium or a model where there is one porosity inside the clay particles and another between them, and transport of negatively charged species can only take place between the particles.

For SKB's safety assessments, a more practical approach is used based in principle on measured diffusivities. See also radionuclide transport in SFR, Chapter 21.

Conclusions in RD&D 2007 and its review

SKI proposes that SKB should carry out sensitivity analyses to identify particularly important nuclides and parameters for sorption and diffusion in bentonite and that the choice of parameters for sorption and diffusion needs to be reviewed when buffer material has finally been selected and the water chemistry on the sites has been determined.

Newfound knowledge since RD&D 2007

The body of data for radionuclide-specific diffusivities that was gathered for SR-Can /24-43/ will also be used for SR-Site. Some modifications have been made, however, to fit the requirements made by SR-Site's data report.

The experimental programme that was planned to verify the model for transport between the immobile water and the buffer and the flowing water in a fracture has not delivered any useful results yet. This is due in part to the fact that equipment has been used in the bentonite erosion project and in part to the fact that the geometry may perhaps not have been the most suitable for the task.

Programme

When it comes to data for both sorption and diffusion of radionuclides, SKB believes that the material is adequate to the needs of the safety assessment. The transient transport of nuclides through the buffer is of relatively little importance in the most important cases in the nuclide transport calculations.

The programme for verifying the model for transport between the immobile water and the buffer and the flowing water in a fracture will continue. The idea is to conduct experiments on a relatively small scale in order to quickly be able to produce results and make parameter variations. Figure 24-22 shows the results of preliminary small-scale tests with diffusion of a dye in a gap of variable aperture. A status report describing the conceptual model for flow resistance (Qeq) together with the uncertainties that exist in the model is being prepared within SR-Site.

24.2.26 Radionuclide transport – sorption

The surface of smectitic clays has a permanent negative charge. The charge imbalance is neutralized by an exchange of cations between the layers. When the clay is water-saturated, the exchangeable cations are hydrated and an electric double layer is formed in the water-clay interface. The charge-compensating cations can easily be exchanged for other cations from the solution that is in contact with the clay surface. The sorption of cations in smectite minerals can be described as ion exchange reactions and modelled with thermodynamic equilibrium constants or selectivity coefficients. Ion exchange is the typical sorption mechanism for alkali and alkaline-earth metals. Many transition metals are also sorbed via ion exchange.

Radionuclides can also be sorbed via reactions with the surface and form surface complexes. Most actinides and lanthanides form surface complexes. Nuclides sorbed as surface complexes cannot be transported by surface diffusion.

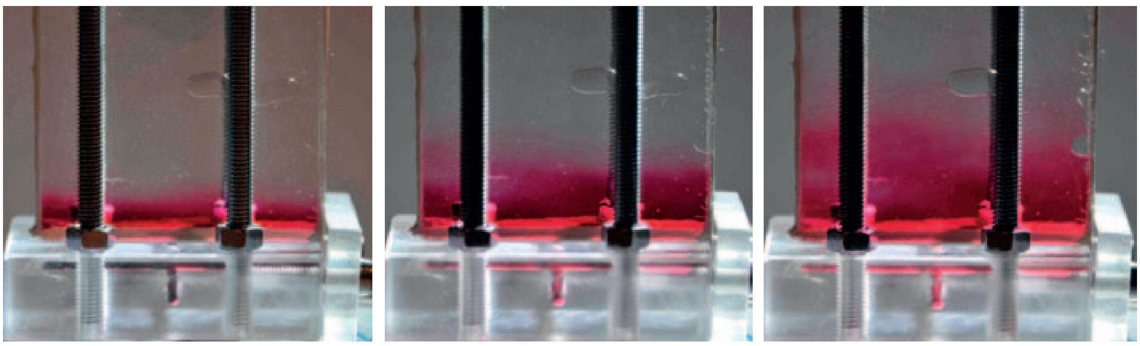


Figure 24-22. *Results from preliminary tests with diffusion of a dye in a gap with an aperture of 50 millimetres. The photos are taken after 41 minutes, 18 hours and 67 hours, respectively.*

SKB does not consider sorption of radionuclides in bentonite to be a prioritized research area. However, existing data will be updated with new information prior to each new safety report.

The programme for ion exchange/sorption (Section 24.2.15) is strongly linked to the programme for diffusion, see Section 24.2.13.

The backfill material has been changed between SR-Can and SR-Site. The data used for radionuclide transport in SR-Site's data report are adjusted according to the new composition. Otherwise, sorption in the backfill is not a prioritized research area and no new initiatives are planned during the upcoming period.

24.2.27 Speciation of radionuclides

The speciation of radionuclides is of importance for sorption and diffusion in the buffer. It is influenced by what speciation the nuclide had at the boundary to the buffer, i.e. inside the canister, but also by the chemical conditions in the buffer. The speciation process is discussed in Chapter 22.

24.2.28 Radionuclide transport – colloid transport through bentonite

If colloids can be transported through the bentonite, this transport pathway can contribute to the total radionuclide transport. Previous experimental studies have shown that small organic humic colloids can diffuse through compacted bentonite, and that they increase the transport of radionuclides that are strongly sorbed to bentonite. However, the compacted the bentonite has filtering properties, and these properties are studied more closely here.

Conclusions in RD&D 2007 and its review

In RD&D Programme 2007, a study was planned of colloid diffusion of gold colloids for the purpose of investigating whether there is a specific filtration limit for colloids in bentonite.

In its review of RD&D Programme 2007, SKI does not specifically comment on the programme for colloid transport through bentonite. SKI points out that bentonite erosion can affect the ability of the bentonite to filter colloids.

Newfound knowledge since RD&D 2007

Colloid filtration in compacted bentonite has been studied /24-44/ to determine whether colloids are effectively filtered by compacted bentonite, and to find out what diffusion pathways may exist. Another underlying purpose has been to study whether inorganic and organic colloids have different diffusion properties. Small gold colloids (two, five and 15 nm) were used as colloidal tracers in three colloid filtration experiments, Figure 24-23. It was concluded from these experiments that the bentonite's microstructure effectively filters even very small inorganic colloids. Transport of gold colloids could only be detected at such a low dry density (0.6 g/cm^3, see Figure 24-24) that the average distance between the montmorillonite layers exceeded the size of the smallest colloids, two nm. These results indicate that the real bottleneck for any colloid diffusion in bentonite is transport between the montmorillonite layers.

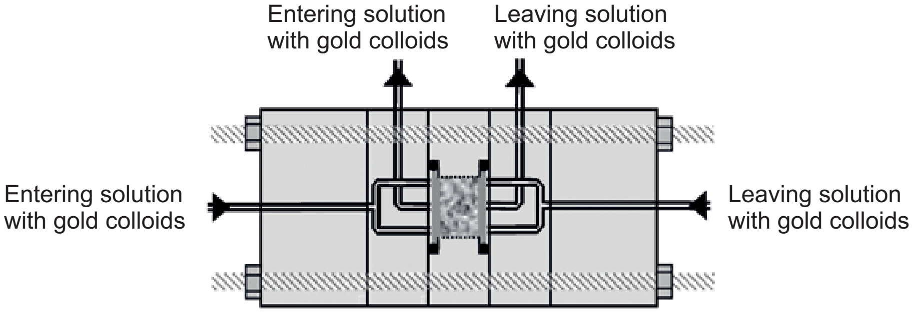


Figure 24-23. *Experimental setup for the colloid filtration experiment.*

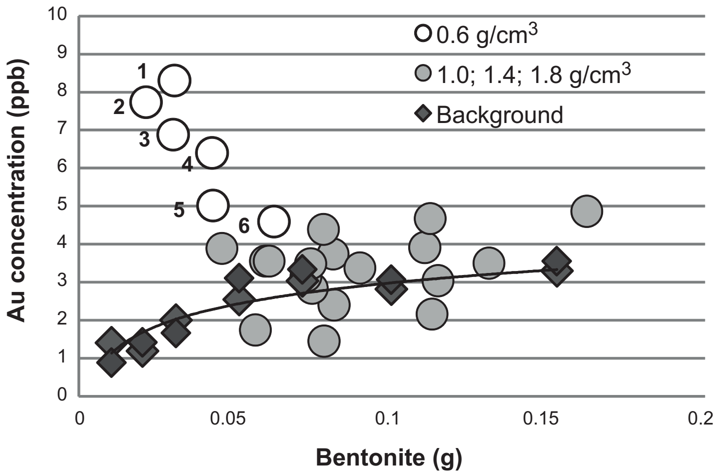


Figure 24-24. *Gold concentrations measured with ICP-OES for the 2 nm experiment. Only at the lowest compaction were significant gold concentrations found.*

Programme

The results of the completed programme are unambiguous. No further initiatives are planned.

24.3 Integrated modelling – radionuclide transport in the near-field

A number of cases of radionuclide transport in the near-field are being studied in the SR-Can safety assessment and in the ongoing SR-Site assessment. The most important risk-contributing case in SR-Can was one with partially eroded buffer around the deposition hole. Other cases that have been analyzed in the near-field include one where the canister is sheared apart due to an earthquake and one where the backfill in the tunnel collapses and water can flow freely at the top of the tunnel. A hypothetical case with a canister defect that gradually expands is also analyzed.

All of these cases are modelled with the numerical model Compulink /24-45/. The model incorporates fuel dissolution, precipitation in the canister's internal water-filled cavities, transport through a defective canister, diffusive transport in the buffer and advective and diffusive transport in the tunnel backfill. The result of the modelling is release of radionuclides from the near-field, which in turn comprises input data to equivalent modelling in the far-field, see Section 25.3.4.

Minor modifications of the existing Compulink program code for the near-field are being made in the SR-Site safety assessment.

25 The geosphere

25.1 Initial state of the geosphere

The analysis of long-term safety starts with the initial state, the state that exists when the repository has just been closed and sealed. This in turn requires knowledge of the state that existed before the repository was built and how it has been affected since. The results of the site investigations and the detailed characterization, plus the results of the rock excavation works, are the primary basis for determining the post-closure state of the geosphere.

In general, conditions in the rock that are favourable for long-term safety are also conducive to good constructability and good occupational safety. Good constructability and a stable hard rock facility are also advantageous for safety during the operation of the facility. The requirements and premises that apply to the underground openings are described in the programme for technology development (Chapter 15).

The initial state of the geosphere is determined in part by the methods described in Section 15.4, which deals with the detailed characterization programme and the further development of methods that is planned for mapping and measurement of the thermal, mechanical and hydraulic properties of the rock. The initial transport properties and hydrogeochemical state of the rock are also a part of the initial state. These initial properties and conditions are changed over time by the processes described below.

25.2 Processes in the geosphere

25.2.1 Overview of processes

Heat that is generated in the fuel is conducted out via the canister and the buffer and heats the host rock. The groundwater is redistributed in the geosphere's fracture system by groundwater flow. Gas migration may also occur. A mechanical state exists initially in the geosphere which is determined by the natural rock stresses and fracture systems on the repository site plus the changes to which construction of the repository has given rise.

The mechanical evolution of the repository is determined by how the geosphere responds to the different mechanical loads to which it is subjected. The loads may consist of the thermal expansion caused by the heating of the repository, the pressure from swelling buffer and backfill, effects of earthquakes and the large-scale tectonic evolution. Changes in the geosphere may include fracturing, reactivation (sudden movements in existing fractures) or rock creep (slow redistributions in the rock). Movements in intact rock, i.e. compression or expansion of otherwise intact rock blocks, also occur, along with erosion, i.e. weathering of the surface rock, particularly in conjunction with glaciations.

The post-closure chemical evolution of the repository is determined by a number of transport processes and reactions. The predominant transport process over long distances is advection, while diffusion plays a great role over short distances.

In advection, solutes accompany the flowing water. The process leads to mixing of different types of water from different parts of the geosphere. Reactions occur between the groundwater and fracture surfaces, giving rise to dissolution and precipitation of fracture-filling minerals. Moreover, very slow reactions occur between the groundwater and the minerals in the rock matrix. Microbial processes, degradation of inorganic materials from repository construction, colloid formation and gas formation take place in the groundwater. During a glaciation, methane ice formation and salt exclusion can also occur.

Diffusion can be important if the water is immobile or moves very slowly. An important aspect of this is matrix diffusion, where radionuclides diffuse into the stagnant water in the microfractures in the rock and are thereby retained and transported more slowly than the flowing water. Sorption, where radionuclides adhere (sorb) to the surfaces of the fracture system and the rock matrix, is also crucial for radionuclide transport. Matrix diffusion and sorption are the two most important retention processes for radionuclides in the geosphere. Another factor that can be of importance for retention is sorption on colloidal particles and transport with them.

The chemical environment in the water determines which speciation (chemical form) the radionuclides will have, which is particularly crucial for sorption phenomena. Certain nuclides can be transported in the gas phase. Radioactive decay influences the groundwater's content of radionuclides and must therefore be included in the description of transport phenomena.

The research programme regarding different processes in the geosphere that can influence long-term safety is discussed in the following sections. When it comes to the mechanical processes, they are coupled to each other in reality, which makes it difficult to describe them separately in the following sections. The trend in modelling of handling processes in an integrated fashion is presented at the end of the chapter.

25.2.2 Heat transport

Conclusions in RD&D 2007 and its review

A thermal analysis of the canister-buffer system was presented in RD&D Programme 2007 /25-1/. In addition, a method was presented for upscaling of laboratory determinations of the thermal conductivity of different rock types to scales that are relevant for calculating the temperature evolution of individual canisters in typical deposition geometries /25-2/. The upscaling method takes into account variability within rock types and between rock types. Based on idealized mineral compositions for igneous rock types, a general correlation was demonstrated between density and thermal conductivity /25-3, 25-4/.

Boundary conditions for thermo-hydro-mechanical models (THM models) of the evolution of the buffer towards water saturation were defined with the aid of studies of the temperature evolution in the rock mass in the Prototype Repository on Äspö /25-5/.

SKI observed that SKB has paid better heed to the impact of blasting on both the rock mechanical and thermal initial state. Furthermore, SKI said that SKB had been successful in developing thermal models for Forsmark and Laxemar and the SKB had shown that it is possible to calculate the temperature evolution in rock mass and buffer, including the canister-buffer gap, with the aid of data from the Prototype Repository on Äspö. SKI expressed its support for SKB's programme concerning geostatistical methods to estimate and limit uncertainties in temperature calculations.

The Swedish National Council for Nuclear Waste considered further development of efficient and inexpensive field methods for determination of thermal properties, particularly full-scale methods, to be urgent.

Newfound knowledge since RD&D 2007

Work has continued on the principles of thermal modelling of the Forsmark and Laxemar sites. Spatial variation and upscaling to relevant modelling scales have been coupled to the geological site models and analyzed by means of geostatistical methods /25-3, 25-6, 25-7, 25-8/.

The work of quantifying and limiting uncertainties in temperature calculations has also continued, along with the development of both field and laboratory methods for determination of thermal properties /25-9, 25-10/. The focus has been on methods for determinations of canister spacing in deposition tunnels, which is dependent on e.g. thermal conditions. This means that it is now possible to adequately take into account the distribution and spatial correlation of thermal conductivity properties in the design calculations, i.e. the calculations that serve as a basis for the canister spacings in the different rock domains for layout D2 /25-11/. The purpose of these calculations is to ensure that the buffer temperature does not exceed 100°C for any canister at any time, at the same time as the available rock volume is utilized efficiently. An important part of the thermal design calculations is to determine an adequate, but not excessive, uncertainty margin.

In the THM report for the geosphere /25-12/, the consequences, in terms of number of affected canisters, if the total uncertainty margin in the design calculations should for some reason turn out to be insufficient are analyzed. The results are exemplified in Figure 25-1 and relate to the temperature distribution at the time of the peak temperature for 6,000 canisters deposited according to layout D2 in Forsmark. The figure illustrates the excess temperatures of canisters for different assumptions of how much the necessary margin has been underestimated in the design calculations. The margins may be dependent on variations in the thermal conductivity of the rock.

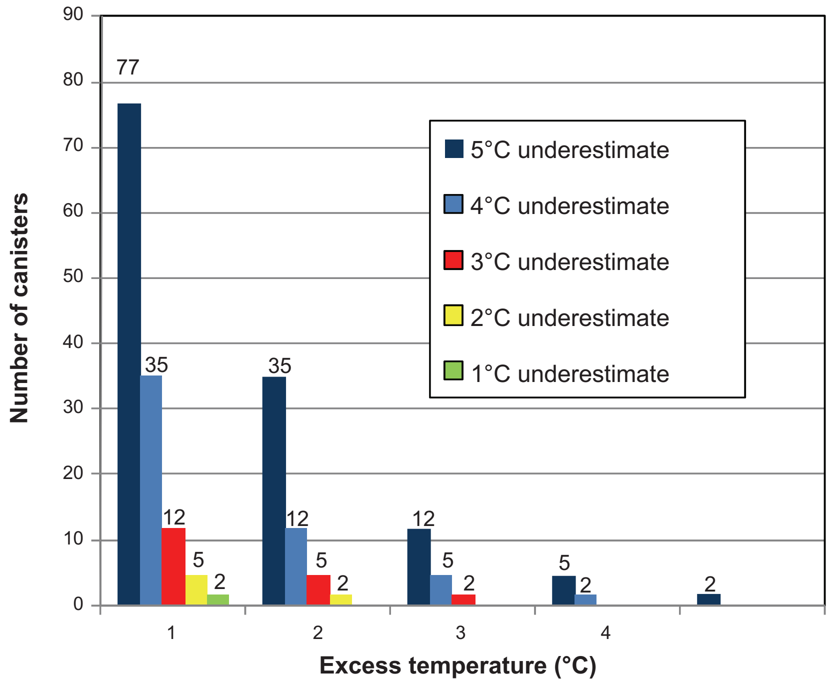


***Figure 25-1.** Excess temperatures of canisters deposited in accordance with layout D2 in Forsmark for different assumptions regarding how the necessary margin has been underestimated in the design calculations. If, for example, the margin has been underestimated by 3°C, two canisters will be about 3°C too hot, five canisters about 2°C too hot and twelve canisters about 1°C too hot. The estimate applies if all these deposition holes are completely dry at the time of the peak temperature (from /25-11/).*

Temperature data in boreholes from Laxemar and Forsmark have been used to determine the temperature at repository depth and to confirm water flow from hydraulic measurement methods. The temperature distribution contains more information, however. A separate project has investigated the spatial variability of the thermal conductivity on a larger scale and analyzed traces of historical climate changes /25-13/. See further Chapter 19.

Programme

The development work on measurement of thermal properties will primarily be pursued within the Detailed Characterization Programme, see Section 15.4.

Some model development for thermal processes will take place within the framework of analysis of the Prototype Repository on Äspö.

25.2.3 Groundwater flow

Conclusions in RD&D 2007 and its review

The programme presented in RD&D Programme 2007 mainly involved modelling studies and development of the codes ConnectFlow, DarcyTools and MIKE SHE.

In its review, SKI commented that the programme presented by SKB for groundwater flow had an appropriate level of detail and was generally suitable. Furthermore, SKI felt that SKB had made progress in the hydrological modelling and gained important practical experience via application of the models in the site modelling and the SR-Can safety assessment. However, SKI pointed out the absence of links from the account of groundwater flow to other disciplines and sections in the RD&D programme, for example links to glacial hydrology, geochemistry, piping/erosion of buffer and backfill, the biosphere chapter and rock mechanical processes.

Newfound knowledge since RD&D 2007

During the current period (2007–2009), a large number of modelling studies have been carried out within the framework of site descriptive modelling, and a number of studies have been initiated within the SR-Site safety assessment project. These studies have mainly provided a better understanding of groundwater flow modelling with site-specific data and modelling in the safety assessment with a detailed description of the Spent Fuel Repository with high resolution.

The most important studies in site modelling are /25-14, 25-15, 25-16, 25-17/ for Forsmark and /25-18, 25-19, 25-20/ for Laxemar. A high-conductive layer has been included in the upper part of the final groundwater model for Forsmark /25-14/. The high-conductive layer represents the water-bearing bedding planes encountered in the bedrock. The high-conductive layer dampens the importance of the topography for the groundwater flow pattern at great depth. This answers an explicit question in SKI's evaluation of previous RD&D Programmes.

In SR-Site and for site selection, the site models for Forsmark and Laxemar have been applied for the different time periods included in the assessment, i.e. for the period with an open repository /25-21/, for the temperate period, and for the glacial period /25-22/. The code DarcyTools has been developed with the aim of being able to study the resaturation process in an initially unsaturated backfill material /25-21, 25-23/. The simulations for the glacial period – consisting of the combinations permafrost, combined permafrost and ice sheet, and only ice sheet – have contributed considerable new knowledge of groundwater flow under glacial conditions compared with the knowledge level at the time SR-Can was done. Specifically, the study /25-22/ shows how an ice sheet combined with permafrost and taliks (openings in the permafrost) can affect the groundwater flow in both magnitude and direction at a hypothetical repository depth in the rock. Site-specific input from the climate programme has been used wherever possible /25-24/ with regard to ice profiles and permafrost evolution in these simulations. However, the models are still based on a number of assumptions that cannot at present be fully verified. An improved understanding of these assumptions is expected to emerge from the Greenland Analogue Project (GAP), which is described below under "Programme" and in Section 19.6.

During the period (2007–2009), a number of hydrogeological studies have been carried out that have provided a better understanding of the conditions around SFR. A modelling study was conducted for SAR-08 to determine uncertainty factors for previously done simulations and to analyze flow paths from SFR /25-25/. This study complemented the simulations that were done for the SAFE project /25-26/. Before the work with the SFR Extension Project started, a study was done where flow through tunnels and future flow paths from a possible extension of SFR were analyzed /25-27/. The report concluded that it was advantageous to build an extension south of the existing SFR facility due to longer future flow paths and another recipient than for the existing SFR facility. As a first step in the hydrogeological modelling in the SFR Extension Project, a model was set up in DarcyTools /25-28/ with a similar parameterization as that used in a previous modelling with the code GEOAN /25-26/. One important conclusion of the simulations was that the future water divides used to define the model area need to be studied in more detail and possibly updated. In a second step, /25-29/ used the same hydraulic set of data as reported in /25-26/ to parameterize a preliminary version of the new deformation zone model for the SFR area /25-30/. A modelling study /25-31/ was conducted with the parameterization from /25-29/ to investigate whether the pier above the existing SFR facility will act as a water divide in the future as well, in other words whether the groundwater table can be maintained inside the pier. This study showed that a description of the future surface hydrology (future rivers, lakes etc.) is of greater importance for the near-surface groundwater flow in the future than the hydraulic properties of the pier. The study showed that code development to handle the surface-hydrological processes in DarcyTools may be suitable when shallow repositories such as SFR are studied. It was further found that the algorithm for particle tracking in DarcyTools needed to be updated. The update was carried out in 2009 and resulted in a more flexible and user-friendly method, particularly for steady-state solutions.

During the current period, SKB has continued the supraregional simulations that were already in progress. An in-depth follow-up study with a focus on conceptual uncertainties has been published /25-32/. Another supplementary study has also been conducted /25-33/. The purpose of the supplementary study has been to present an in-depth evaluation of conceptual simplifications and model uncertainties in connection with supraregional groundwater modelling, for example model depth, topography, location of the groundwater table and model boundaries. Furthermore, local site characteristics for a final repository were studied in the perspective of regional flow paths. The modelled region coincides with that in the previous study, i.e. eastern Småland. In the previous model /25-32/, the position of the groundwater table was assumed to be the same as the topography of the ground surface, with the exception of areas with glaciofluvial deposits and areas beneath the sea. A flux boundary condition with surface water runoff on the top of the model is, however, preferable to prescribed levels of the groundwater table, since a flux boundary condition is a better approximation of the system's actual behaviour. With flux boundary conditions, it can be concluded that the specific flows at the starting points of the flow paths at repository level decrease compared with a model based on prescribed levels of the groundwater table, and that both the lengths of the flow paths and the particles' travel times increase slightly. When it comes to the depth of the flow cells in a supraregional model, a sensitivity study shows that it is largely the heavy salt water located at greater depths than repository depth that prevents flow paths from reaching down to even greater depths. In /25-33/, simulated properties for flow paths from a hydraulically favourable area were also compared with simulated properties of flow paths from the regional model area in Laxemar. The selected area is located in the Virån River's catchment area about 30 kilometres from the coast. In reality, the area is dominated lithologically by basic rocks and does not necessarily fulfil all requirements and criteria made on a repository. Many of the flow paths from the selected area are long and extend towards the coastline and reach down to great depth. When the flow paths approach the coastline, they tend to turn upward from great depth. The comparison between the two areas has shown that it is above all the travel times that are much longer from the selected area, but that the variation in travel times is extremely great. The vertical boundary conditions of a supraregional model and the importance of surface water divides in the simulation of flow paths have been studied in a separate project /25-34/.

During the current period, the surface-hydrological modelling has been developed above all within site descriptive modelling /25-35, 25-36/ and the ongoing SR-Site safety assessment, see also Section 26.6. The main purpose of the surface hydrological modelling is to gain a better understanding of the groundwater in the soil layers and how it is related to surface water, atmosphere and groundwater in rock, as well as to furnish the biosphere model with input data. The development of numerical models in MIKE SHE /25-37, 25-38/ and calibrations of these models against measured water levels and flows have been important components in this work. The results of these modelling efforts have been used as a basis for describing boundary conditions and the properties of the soil layers in the models of groundwater flow in the rock. Similarly, descriptions of the properties of the rock have been transferred to the surface hydrological models. This means that the connection between surface hydrology and the hydrogeology of the rock in site descriptions and safety assessments has been strengthened during the period.

The hyperconvergence concept, originated in /25-39/, has been applied in /25-40/. The study indicates that groundwater flows in a sparse channel network just above the percolation threshold, and further that the actively flowing channels are only a small subset of all available channels. The results of this study indicate that "channelling" should be incorporated in the modelling tools that are used. Channelling is discussed in Section 25.2.11.

Programme

Some further development of the groundwater flow codes DarcyTools and ConnectFlow are planned for the coming period. However, this development will not affect the conclusions in SR-Site. A version of DarcyTools based on the Navier-Stokes flow solution is being developed. It has already been used for grouting, but is being developed to describe surface water flows based on the Navier-Stokes solution. This new Navier-Stokes version should be able to be coupled together with the traditional current version of DarcyTools based on a Darcy flow solution so that coupled groundwater-surface water problems can be handled.

A separate line of development is being pursued where the description of surface water hydrology in the current version of version DarcyTools based on a Darcy flow solution is being improved. This version will specifically be used within the SFR Extension Project. The hydrogeological model versions 0.2 and 1.0 will be finished in 2010. The statistical distributions of parameters that describe the hydrogeological properties of the bedrock will be based above all on the site-specific data gathered in the SFR Extension Project. With these model versions as a basis, a number of modelling studies will later be conducted to determine a suitable siting and design of an extension of SFR. A number of modelling studies of the near-field hydrology will also be done in this project, i.e. flows in the tunnel system will be studied in greater detail in order to look at properties of plugs and fill materials, for example. An important component in the parameterization of the models is calibration against groundwater inflow and measured changes in the groundwater level in boreholes near the existing SFR facility. It will also be investigated whether the ventilation air leaving SFR contains significant quantities of water, which could affect the calibration of the flow model. For the purpose of satisfying the modelling needs within the SFR Extension Project, some improvement is also being made of the handling of surface water hydrology in the current version of DarcyTools.

For ConnectFlow there are plans to investigate the prospects for making a more formal coupling with geochemical processes, and to develop of the concept of discrete fracture network modelling, see Section 25.3.1. In the current version of ConnectFlow (as well as DarcyTools), it is only possible to describe the transport of chemically conservative substances, which is a limitation when the evolution of the groundwater chemistry over long periods of time is to be simulated. Chemical reactions that are of interest for e.g. calculating redox and pH cannot be incorporated. This limits which chemical constituents can be used for calibration or for confirmatory testing, which has been used extensively within site modelling. It was also learned from the site modelling that the methodology used to generate and parameterize stochastic fracture networks, based in part on data from Posiva flow logging, has limitations. Specifically, the methodology used to determine the size distributions of the fractures has probably overestimated the intensity and size distribution of the interlinked (connected) fracture network by overcorrecting (reducing the power law exponent) the size distribution of the actual fracture set. The reason for this is that if the Terzaghi correction factor used to evaluate simulated fracture frequencies in an absolutely vertical borehole is too low, the modelled intensity must be increased beyond the real intensity so that the fracture frequencies measured for vertical fractures in inclined holes will be reached in the model. This – in combination with the fact that all modelled fractures, regardless of size, are assumed to have constant properties – means that the flow capacity has in all probability been overestimated. These lessons will be re-examined with fresh eyes and various approaches will be tried, see also the discussion in Section 25.3.1. An initial study for alternative parameterization based on flow log data has been done in /25-41/.

Some development can also be expected regarding certain processes such as flow in unsaturated fractures and resaturation processes. A new Task 8 is planned within the Äspö Task Force for Modelling of Groundwater Flow and Transport of Solutes (TF GWFTS) to study the hydraulic interaction between rock and bentonite at a deposition hole. The resaturation process is at the centre of focus here. The effect of rock stresses on water permeability and thereby on repository function may be studied with coupled hydraulic-mechanical models.

Studies to gain a better understanding of groundwater flow under glacial conditions are planned within the Greenland Analogue Project (see Section 19.6), where the connection between the ice, the subglacial layer and the rock will be conceptualized. The experimental results obtained in the GAP will be modelled with DarcyTools. Improvements in the handling of permafrost and description of the subglacial layer in DarcyTools are expected to be made in the GAP.

Development of methodology and experiments, as well as their evaluation, is planned during the coming period in support of the detailed characterization programme, see Section 15.4. A project has already been initiated for this purpose. Evaluation of the correlation between measurable properties and the post-closure groundwater flow will take place in conjunction with detailed characterization and will be based on the results of the analysis in /25-21/. The conclusions are expected to be useful in a possible revision of the design premises for acceptance of deposition holes.

25.2.4 Gas flow/dissolution

The programme regarding gas flow/dissolution coincides with the programme described in Section 25.2.3 concerning rewetting processes, and specifically Task 8 within the Äspö Task Force for Modelling of Groundwater Flow and Transport of Solutes (TF GWFTS).

25.2.5 Movements in intact rock

Conclusions in RD&D 2007 and its review

RD&D Programme 2007 said that SKB had further developed the technique for characterizing the mechanical properties of the rock and applied this technique to the investigation areas in Forsmark and Laxemar in order to derive site descriptive rock mechanical models.

In the site investigations both before 2007 and since then, extensive field work has been done on rock stress measurements. Overcoring and hydraulic methods have been applied on both sites. Both methods have their limitations at great depth, especially in Forsmark, where the rock stress level is higher compared with the Oskarshamn area. However, it was noted in RD&D Programme 2007 that the two methods have provided a fairly concordant picture of the stress situation down to levels around or slightly above the depths studied for the Spent Fuel Repository. However, the stress magnitudes measured by the two methods exhibited certain differences.

The results of the APSE experiment in the Äspö HRL showed the importance of small confinement pressures to stabilize the deposition hole walls where the thermally induced increase of the tangential stresses could lead to spalling /25-42, 25-43/.

SKI agreed with SKB in the assumption that if deposition tunnels are oriented parallel to the major horizontal stress direction, the risk of spalling during the construction phase is less than for orientation in other directions. SKI emphasized that there is evidence /25-44/ indicating that spalling failures can occur at lower stresses (more superficial) than SKB states in its report on spalling /25-45/. SKI further stated that the swelling pressure from the bentonite can retard the development of rock breakouts, but that this can lead to fracture initiation and increased fracture length for tension fractures in the main stress direction. SKI would like SKB to study this process further.

The Swedish National Council for Nuclear Waste thought that general studies should be conducted regarding what effect existing and altered rock stresses may have on hydraulic conductivity in fractures in different directions and the consequences for the detailed design of the repository. Furthermore, the Council thought that coupled issues regarding changes in connection with an open repository should be examined when it comes to the integration of altered rock stress conditions, groundwater-conducting zones and the possible influence of groundwater chemistry.

Newfound knowledge since RD&D 2007

A project with a special focus on examining more closely the effect of confinement has been carried out in the form of a field experiment in a number of boreholes with a diameter of about 500 millimetres. The field experiment CAPS (Counterforce Applied to Prevent Spalling) used LECA pellets to simulate the confinement effect of non-water-saturated bentonite pellets. The experiments indicate that the small counterforce exerted by the LECA pellets is sufficient to limit spalling and prevent the formation of a highly conductive zone of fractured rock in the 500-millimetre holes /25-46/.

A programme has also been carried out within the framework of the site investigations to quantify the degree of stress-induced microfractures in drill cores from several depths from at both Forsmark and Laxemar by means of triaxial loading and microscopy studies.

Programme

SKB will primarily follow and participate in the development work on spalling within the framework of the POSE project being run by Posiva. The conclusions from SR-Site on the importance of thermally induced spalling for long-term safety will also be of importance for the level of ambition in the continued research. At present, this spalling is judged to be of only limited importance for safety.

25.2.6 Thermal movement

Conclusions in RD&D 2007 and its review

RD&D Programme 2007 reported results concerning thermally induced spalling that were obtained based on data from the APSE experiment, see Section 25.2.5.

SKI agreed with SKB that the APSE experiment at Äspö has given SKB new knowledge about spalling in deposition holes with credible conclusions about the properties of the rock when it is exposed to thermomechanical loading. However, SKI believed that there is a risk that both shorter and longer fractures in rock around a deposition hole may, due to thermomechanical load, start to propagate from the end of a fracture and combine with other fractures. This can in turn lead to an increase of the groundwater flow around deposition tunnels and deposition holes.

Newfound knowledge since RD&D 2007

The analytical thermomechanical solution for the KBS-3 repository that was presented in /25-47/ has been further developed so that cases with several different deposition areas can be analyzed. The original equations have been coded to easy-to-use mathematical calculation sheets and have been used in sensitivity analyses and analyses of how the deposition sequence can affect the near-field evolution /25-12/. For example, it has now been verified that the approximation that is normally made in analyses of the thermomechanical evolution of the near-field, namely that all canisters are deposited simultaneously, is valid provided that the time difference between rock extraction and deposition in nearby tunnels is not too great. With the aid of this analytical solution, it can easily be checked that the approximation is also valid for a given proposed deposition sequence. The analytical solution as such has been verified previously /25-48/. The recoded version has been verified by comparison with numerically obtained results /25-12/.

The thermomechanical evolution in the Prototype Repository is currently being analyzed using a method similar to that previously employed for the APSE experiment /25-49/. The analysis is based on the verified thermal analysis now available, in other words with a careful description of how the effect has varied in the different canisters. The compilation work is under way, but the analysis shows preliminarily that the tangential stresses in the deposition holes do not equal the nominal spalling strength. It is therefore not possible to draw any important conclusions regarding spalling and confinement effects from the Prototype Repository.

The possible pressure dependence of the coefficient of thermal expansion of typical repository rock types will be determined, mainly via a literature study. The purpose is to assess the risk that the thermal stresses in the repository are underestimated due to the fact that the parameter values are determined by tests on unloaded specimens. The literature study shows preliminarily that this is not the case. In order for such an underestimation to be possible, extreme assumptions must be made regarding the anisotropy in the microporosity of the rock samples.

Programme

General development of codes for modelling will be done to gain a better understanding of large-scale thermomechanical processes, see further Section 25.3.2. However, current knowledge is judged to be sufficient to bound the importance of the thermally induced movement for long-term safety.

25.2.7 Reactivation – movement along existing fractures

Movement along existing fractures is related to structural geology and tectonics, i.e. it is mainly a question of earthquake effects and effects of postglacial faults.

Conclusions in RD&D 2007 and its review

RD&D Programme 2007 described analyses with the distinct element code 3DEC /25-50/ of the effects of earthquakes on the repository. Two different earthquake geometries were analyzed: one where the fault reaches one kilometre below the ground surface and one where the fault breaches the ground surface. In both cases the earthquake had a magnitude of six. The purpose of the study was to determine the magnitude of the secondary shear movements that can be induced in fractures at different distances from an earthquake. The possible shear movement of certain fractures in conjunction with seismic activity in nearby zones is one of the reasons the canisters should not be deposited in holes intersected by long fractures. The analysis showed that the maximum calculated shear movement of fractures at a distance of 200, 600 and 1,000 from the fault is well below the canister damage criterion (100 millimetres). The results – i.e. the distance limitations and fracture size rules – were judged to be valid for larger earthquakes as well, for example magnitude seven earthquakes.

In order to be able to apply the conclusions in /25-50/, a general structural geological characterization project was initiated with a focus on quantifying the occurrence and extent of fractures and minor deformation zones on the order of 100–500 metres /25-51/. The work also discusses different methods for estimating the location and size of the structures during different investigation and design phases for a final repository. A methodology for mapping fractures /25-52, 25-53/ based on fracture mapping in tunnels and use of DFN modelling was also developed within the project.

On behalf of SKB, Uppsala University had conducted two studies describing different mechanical effects in the Earth's crust of a glaciation similar to the Weichselian glaciation /25-54, 25-55/. Both studies were based on numerical two-dimensional modelling with the finite element code ABAQUS. The first study concerned the stability of the crust and the potential for earthquakes during the deglaciation period, as the ice sheet retreats. The other study concerned the stress evolution at a depth of 500 metres in Forsmark and Laxemar and was carried out with more developed crustal models and for two assumptions regarding the temporal evolution of the Weichselian glaciation.

One study that utilized the "Differential SAR Interferometry" (dInSAR) method showed that no movements had been observed during the studied period in any regional lineament or in any fracture zone in the Forsmark area /25-56/. Several cases of local subsidence in loose sediments were identified, however.

SKI would have liked RD&D Programme 2007 to have included general studies of large fractures and fracture zones, which they consider urgent in view of the regional and local modelling that is needed for analysis of stress states and questions concerning coupled processes. SKI also thought that SKB should in the future present sensitivity analyses regarding modelling where the mechanical properties of fractures and fracture zones are varied to a greater extent. Further, SKI expressed a desire for additional and more in-depth studies regarding the fact that the repository may constitute a plane of weakness and thereby serve as a fracture initiator in connection with future earthquakes.

SKI and SSI found that SKB has a good approach for handling earthquakes in SR-Can, where SKB assumed an estimated annual probability for earthquakes of magnitude six or greater. However, the regulatory authorities found that the analysis in SR-Can was based on a number of insufficiently justified assumptions, which need to be reconsidered for SR-Site. For example, a thorough discussion was lacking concerning the safety-related importance of repeated earthquakes of small magnitudes.

Like SKI, SSI thought that SKB should, based on a coherent problem description, derive and present a programme for the following: continued work to shed light on the development of models to assess the effects of an earthquake of magnitude six or greater, methods for identification of fractures and deformation zones, further work with discrete network models, and development of respect distances and criteria for choice of deposition positions.

The Swedish National Council for Nuclear Waste said that SKB should carry out measurements of any rock movements on the selected site for the Spent Fuel Repository during the period for processing of the site application and then for an extended period during construction and operation the repository.

Newfound knowledge since RD&D 2007

The work of developing and analyzing models of stability conditions and stress evolution in the Earth's crust during a glacial cycle has continued, mainly within the framework of the SR-Site safety assessment. The models analyzed previously have been two-dimensional and based on simplified assumptions. A three-dimensional situation has now been analyzed by means of extensive modelling work with the computer code ABAQUS. The overall goal has been to determine boundary conditions in a regional perspective for the evolution of rock stresses in Laxemar and Forsmark during a glacial cycle. Furthermore, the stability of faults in Laxemar and Forsmark have been analyzed assuming a seismogenic depth of 9.5 kilometres /25-57/.

A study has been carried out of the potential for hydraulic jacking in conjunction with a glaciation /25-58/. Hydraulic jacking occurs when the pore pressure in a fracture becomes so great that the fracture's tensile strength is exceeded and the fracture aperture increases rapidly, with difficult-to-foresee consequences for transmissivity and flow. A number of different circumstances have been identified that could theoretically lead to the generation of high water pressures beneath the ice that could also affect the pressure situation outside the ice. However, hydraulic jacking is not judged to be able to reach depths below about 250 metres.

A broadened numerical analysis has been done of the effects of earthquakes on fractures in the repository rock. The principles of the analysis are essentially the same as those that were also used for SR-Can /25-50/. The conclusions are presented within the framework of the SR-Site safety assessment.

The main purpose of the earthquake simulations has been, as before (see e.g. /25-59, 25-50/), to obtain relevant estimates of the secondary movements that can be induced in the target fractures in the vicinity of the active earthquake zone.

In order to assess the transmissivity changes in the fractures in the near-field during construction, thermal loading and during a glacial cycle, thermomechanical near-field models /25-60/ have been developed in agreement with the updated layout and site descriptions /25-12/. Stress data obtained from new three-dimensional analyses of the latest glaciation are used for the stress evolution during the glacial cycle /25-57/. A more realistic stress-transmissivity model has been constructed. The updated model does not differ qualitatively from the one used in the SR-Can analyses, in other words the transmissivity of a fracture has the same type of exponential dependence on the effective normal stress. The model's parameter values have, however, been determined from results of measurements of the dependence of fracture stiffness on the normal stress in fracture samples from Forsmark. The measurements were performed as cyclical loading tests /25-61/. The results of the second load cycle have been used for the evaluation, since it can be assumed that the first load cycle has been disturbed by sampling, installation etc.

A literature survey /25-62/ has been conducted to obtain a better perspective on the validity of the relationship between stress and transmissivity used in the model /25-12/. In summary, the results show that the model is fundamentally right, but that the uncertainties are great, partly because there are so few documented and relevant field experiments that can be used for control. In the field experiments that do exist, the variations are great. Two sets of parameters values are therefore used in the application /25-12/: one for an expected average impact of the transmissivities and one that gives an estimated maximum impact.

The deformation zones in and around the deposition areas will affect the thermal volume expansion in the near-field and thereby the thermal stresses around deposition holes and deposition tunnels. In an attempt to bound this effect, large-scale 3DEC models of the sites have been analyzed /25-12/. The results of the large-scale 3DEC models, which include a representative set of minor deformation zones inside the repository, are then used to determine time-dependent boundary conditions (first expansion, then contraction) for near-field models of different parts of the repository. The influence of major deformation zones at a little greater distance is analyzed in the same way. The results show that both the few deformation zones inside the repository and a single major zone a little farther away have a negligible impact. In the modelling done of the near-field to quantify the potential for thermally induced spalling, the impact of the deformation zones is therefore ignored.

A stability analysis has been made of a plane with a set of tunnels with dimensions similar to those of KBS-3's deposition tunnels /25-63/. The analysis was done with the two-dimensional distinct element code UDEC. The stability of the plane was analyzed for different assumptions regarding future loads, both predicted probable loads and purely hypothetical extreme loads. For the thermal load during the temperate period, the horizontal plane through the repository is in principle the principal stress plane, i.e. there are no shear stresses. The vertical stress between and next to the tunnels is sufficiently high throughout the thermal period so that any planes of weaknesses that may exist are in high compression. There are thus no indications that the local expansion that exists around the tunnels at the start of the thermal phase could contribute to a horizontal fracturing between the tunnels. For hypothetical load cases with large shear stresses in the horizontal plane, the UDEC models show that the stability margin is hardly affected by the tunnels. In order for the tunnels to have more than a marginal impact, the tunnel spacing must be small: of the order of 15–20 metres. Another hypothetical process is the formation of sheeting joints as a consequence of large additional horizontal stresses in rock formations with a convex surface topography. With the rock properties at Forsmark, vertical tensile stresses cannot be generated at greater depths than about 100 metres in UDEC models with a convex topography, even if the model is compressed so much that the horizontal stresses grow by 40 MPa. At repository depth, the analysis shows that the vertical stress is sufficient for the horizontal fracture plane to be in compression with ample margin, even if it is assumed that the pore pressure is elevated due to residual effects of a retreating ice cover. Since the vertical effective stress at repository depth is thus compressive even under conditions that cause sheeting joints at shallower depths, there cannot be any horizontal fracturing in the repository plane, with or without tunnels. In conjunction with earthquakes, fracturing can theoretically take place in horizontal planes at repository depth in areas with large stress concentrations at the edge of the active earthquake zone. In all probability this is an effect of the fact that fracture propagation in the active zone is caught up abruptly by the surrounding elastic continuum in the dynamic model that is used to obtain the stress evolution on the plane. But the loss of stability is virtually independent of the repository's cavities. In summary, the analysis in /25-63/ shows that the repository will not act as a plane of weakness.

Programme

Within the framework of the detailed characterization programme, tectonic and construction-related aspects of the rock are being treated on the canister hole and deposition hole scale, see Section 15.4. The knowledge level regarding the importance of future earthquakes is now judged to be sufficient for the SR-Site safety assessment, but in order to further improve the state of knowledge and better quantify remaining uncertainties, further research initiatives are planned within structural geology, seismology and tectonics according to the following programme.

SKB plans to learn more about the stress state at depths greater than one kilometre in the Earth's crust in Sweden. Knowledge of the stress field is a fundamental component of all understanding of dynamic processes in the Earth's crust, and in particular for models of the mechanisms that generate earthquakes, both postglacially and today. An updated state-of-the-art description is needed based on deep boreholes, earthquake mechanisms, deformation data (GPS) and the seismic structure of the crust, along with an evaluation of stress models from topography, the last glaciation and plate tectonics. Furthermore, models of stress evolution at depth in the lithosphere will be analyzed. The study will take account of how the stress field can maximally vary in space, something that is of the greatest importance for stress concentration and thereby major earthquakes, including for estimates of the maximum magnitude of an earthquake. This should be done both by means of theoretical/numerical models and with data from areas in the world that are more active than Sweden today.

The national digital seismic network has now been in operation for about ten years. The database includes more than 3,200 earthquakes. The earthquakes have been analyzed separately, but an integrated analysis has not yet been done. An integrated analysis (called a multi-event analysis) would probably provide better certainty in the localizations, in particular with regard to depth determinations. Further, such an earthquake interpretation methodology would provide a refined velocity structure in 3D that feeds back to the localizations and provides a better understanding of more coherent earthquake mechanisms and how the earthquakes may follow structures down in the Earth's crust. The interpretation methodology will also provide a better understanding of what today's large-scale rock stress looks like.

Three-dimensional (3D) determination of the velocity structure in an area can be done today by means of earthquake tomography. In applying nano- or microseismic monitoring of the Spent Fuel Repository, it is of great importance to have as accurate a picture as possible of the seismic velocity structure in order to be able to correctly localize (induced) earthquakes (see also plans for monitoring in Section 15.4). Seismic velocity is affected by temporal variations in state variables such as pressure, temperature and pore pressure. By means of repeated tomography studies (4D) it is possible to provide a picture of the evolution of the state in a changeable rock mass. But the methods need to be refined to achieve better resolution, which can be done by using more of the waveform than just the first arrivals, so-called waveform techniques. Such method development and application is intended to be tested in a mine environment, or possibly in conjunction with extension of the Äspö HRL.

Estimates of the stresses that caused the great postglacial faults in northern Fennocandia require knowledge of the geometry of the earthquake planes. Investigations are being conducted at the Pärvie fault of microearthquakes to see whether they define a larger earthquake plane below three kilometres' depth. Seismicity at Pärvie is relative low, however, which makes the analysis difficult. The Skellefteå area has the highest seismic activity in Sweden, and the earthquakes appear to be located along the two postglacial faults that have been identified there. SKB is considering broadening the programme by densifying a seismic network in the area, which is expected to result in nearly 1,000 well-determined earthquakes. This would provide information on the fault's geometry and stress state and the mechanisms that drive earthquake activity today. With both the Pärvie and Skellefteå faults thoroughly investigated, the intention is that we should better be able to evaluate what the earthquake activity at a presumptive postglacial fault in other parts of the country should look like.

In order to obtain greater information on recent movements in the postglacial faults, studies of the Pärvie fault are planned using modern satellite radar technology (dInSAR). With this method, ground movements along the faults can be analyzed in a way that covers a much larger area than single GPS measurements. With modern interpretation technique using large quantities of data, it should be possible to identify even small fault movements.

It should be noted in this context that several Swedish universities, with Lund University as the principal, have been awarded funds by the Swedish Research Council (VR) to purchase a drilling rig making it possible to carry out different types of scientific investigations of conditions down to about three kilometres in the Fennocandian Shield. The drilling rig will be a vital resource in the Swedish Deep Drilling Programme (SDDP). Several drilling projects for the rig are in the planning phase and they are concerned with such subjects as structural geology, seismology and tectonics, but hydrogeological, thermal and hydrogeochemical activities are also planned. SKB intends to become involved in suitable aspects of the SDDP when it comes to general geoscientific knowledge accumulation of conditions in the Swedish bedrock.

Coupled to the SDDP and the International Continental Scientific Drilling Program (ICDP), preparatory work is currently under way to drill investigation holes in connection with postglacial faults (e.g. the Pärvie fault).

SKB intends to become involved in the PFDP (Postglacial Fault Drilling Project), which is planned as a part of the SDDP and is aimed at acquiring general knowledge of postglacial faults in northern Scandinavia. Both deep cored boreholes (about 1,000 metres) and an occasional hole to 3,000 metres are planned with the SDDP rig. With its experience from the site investigations and the drilling of KLX02 to a depth of 1,700 metres, SKB can contribute drilling- and investigation-related advice along with quality-assured handling of investigations and large databases.

The deformation measurements with GPS (Global Positioning System) technology were initiated in the Forsmark area in 2005 and are continuing. The goal of the study is to obtain additional experience of the technique with fixed GPS stations and measure any ongoing bedrock movements (mainly horizontal) along the most prominent regional deformation zones: the Singö Zone and the Forsmark Zone. Raw data have not yet been analyzed from a tectonic perspective.

25.2.8 Fracturing

Conclusions in RD&D 2007 and its review

RD&D Programme 2007 referred to the APSE experiment in the Äspö HRL and to other activities regarding questions related to intact rock. Safety-related aspects of the excavation-damaged zone (EDZ) had been treated within the framework of SR-Can. According to SR-Can, an EDZ with up to one and a half orders of magnitude higher conductivity than the surrounding rock would not entail any major safety problems. A fundamental prerequisite is, however, that tunnelling is carried out using quality-assured practices with controlled blasting.

In the review of RD&D Programme 2007, SKI said that SKB needs to further develop methods that reveal the possible existence of flow paths in the excavation-disturbed zone alongside a drill-and-blast tunnel. The Swedish National Council for Nuclear Waste had a similar comment and wished for a better account of the extent and properties of the excavation-damaged zone in connection with controlled blasting.

Newfound knowledge since RD&D 2007

SKB has been involved in several projects that have dealt with properties and safety-analytical aspects of the excavation-damaged zone (EDZ). Among these can be mentioned THM modelling within the framework of the international DECOVALEX programme and a targeted programme with several sub-projects called ZUSE (Swedish acronym for "Mechanical and Hydraulic Properties of the Disturbed Zone") /25-64, 25-65, 25-66, 25-67, 25-68, 25-69, 25-33/. An account of the state of knowledge concerning EDZ matters is provided in Chapter 15.

Programme

Measurement data from the Prototype Repository in the Äspö HRL (acoustic emission data, stress measurements, thermal data) will be evaluated together with results from thermomechanical calculations in order to investigate to what extent fracture propagation can occur around a deposition hole. For the thermomechanical calculations, modelling of the thermal properties is proposed via an inverse modelling based on temperature data where a conditional stochastic modelling of the rock mass and its thermal properties is first carried out. This provides a spatial distribution of the thermal properties in the Prototype Tunnel. The Prototype Repository will be opened in 2011.

Questions regarding the extend of the disturbed and/or damaged zone around a tunnel will be handled from now on within the framework of the Rock Line's activities and the detailed characterization programme, see Section 15.4.

25.2.9 Time-dependent deformations

Conclusions in RD&D 2007 and its review

It was observed in RD&D Programme 2007 that the ratio of stress to strength must exceed certain threshold values in order for any creep to take place /25-70/.

SKI noted that SKB had heeded the request to establish a special programme focused on time-dependent deformation and fracture dynamics, and in particular thermally induced microfractures and time-dependent processes such as subcritical fracturing and creep.

Newfound knowledge since RD&D 2007

Time-dependent deformations are generally linked to fracture dynamics. On behalf of SKB, a coordinated study has been compiled dealing with the occurrence of microfractures, subcritical fracturing and creep /25-71/. The mechanical response of a crystalline rock to very long-duration compressive loading is of interest for a final repository that must be isolated from the biosphere for very long periods of time. The big question is whether strength decreases without limit, or is there a minimum threshold strength below which the rock will cease deform. The investigative work has been based on interpretation of results from short-term testing of creep in rock samples, numerical model analyses of the effect of reduced fracture toughness due to stress corrosion on the strength of the rock, evidence from plate tectonic processes and observations of rock stresses in quarries.

To sum up, the conclusions of the publication are:

- A stress threshold, i.e. a deviatoric stress that can be borne without limit, exists for crystalline rock types. This threshold is probably a significant fraction of the fracture-initiated stress (i.e. 40–60 percent of the uniaxial compressive strength).
- Exponential extrapolation of the results of short-term creep tests entails a realistic time-independent strength corresponding to a Driving Stress Ratio of about 0.45. It is is totally unwarranted to consider linear extrapolation to a final zero strength /25-72/.

Based on this publication, SKB draws the conclusion that time-dependent deformations of the rock can be dismissed as a process of importance for long-term safety.

Programme

For development of transient rock mechanics modelling, see Section 25.3.2.

25.2.10 Advection/mixing – groundwater chemistry

This section deals with the effect of the mixing that occurs due to the fact that the water moves at varying velocity in the fracture system in the rock and how the process affects the chemistry of the groundwater. Section 25.2.11 deals with the importance of advection and dispersion for radionuclide transport.

Conclusions in RD&D 2007 and its review

Verification of the updated code M3 and calculations of the evolution of salinity in connection with climate change were planned in RD&D Programme 2007.

In its review, SKI takes a positive view of SKB's plans. At the same time, they ask for an explanation of how advection and mixing affect transport of corroding substances to the repository, especially sulphide, methane, oxygen and organic matter. This question is discussed in Section 25.2.11.

Newfound knowledge since RD&D 2007

The new data that have been collected within the site investigation programme have been analyzed and site models have been constructed /25-73, 25-74/. These models have utilized both hydrogeochemical and hydrogeological data, as well as our knowledge of the palaeohistory of the sites. The uncertainties in the mixing models have been investigated in /25-75, 25-76, 25-77, 25-78/. The models will be used in the SR-Site safety assessment for detailed modelling of groundwater chemistry during the temperate period following repository closure. Similar calculations were performed for SR-Can /25-79, 25-80, 25-81/.

The computer code M3 /25-82/, which is used for statistical processing by multivariate analysis, has been developed with support from SKB. The code has been updated and verified /25-83, 25-84, 25-85/.

Programme

The development of models including advection and mixing process is described in Section 25.3.3. Calculations of the evolution of salinity in connection with major climate change over long timespans will be reported within the framework of SR-Site. Data and models for transport of substances such as methane and oxygen will also be reported within SR-Site.

In order to study the effects of mixing and reactions in detail, permanent groundwater stations will be installed and new sampling methods will be used on Äspö. The purpose is to study the changes in time and space related to a disturbed hydrogeochemical and hydrogeological tunnel system, analogous to the construction and operation of a final repository. It will then be possible to study the effects on water composition, Eh, pH, gas, microbes and colloids in conjunction with mixing processes in Äspö's tunnel system, where surface water or sea water meets deeper water.

25.2.11 Advection/mixture – radionuclide transport

Conclusions in RD&D 2007 and its review

In RD&D Programme 2007, SKB planned to evaluate the need for detailed studies of advection in single fractures for modelling with DarcyTools. An evaluation of the difference between models based on dispersion and models based on mixing was also identified as a possible need.

SKI took a positive view of SKB's plans to evaluate the difference between mixing and dispersion in the flow models. Furthermore, SKI thought that SKB should show that the models for non-reactive transport that are used to explain the geochemistry, for example calculations of the evolution of the salinity of the groundwater, are in agreement with the models used for the radionuclide transport calculations.

Newfound knowledge since RD&D 2007

The planned simulations of flow in a single fracture with the Navier-Stokes version of DarcyTools have been postponed and are dealt with under "Programme" below. However, a review of the causes of channelling and its effects has been done in /25-86/. Advective transport in two-dimensional fracture networks has been studied in /25-87/.

Programme

Channelling of the groundwater flow can occur due to spatial variability of a fracture's aperture and due to the manner in which the fractures are interconnected in a network structure. The hydraulic boundary conditions also affect channelling. Channelling can affect flow-related transport parameters such as advective travel time and flow-related transport resistance (F). In the safety assessment, F is customarily reduced (by a factor of 10 in SR-Can) to take into account the fact that homogeneous fractures have been used in the flow simulations. Previous more detailed analyses /25-88/ show, however, that this approach is overly conservative.

Simulation of flow and transport in a single fracture plane with variable aperture is planned. Both the planned Navier-Stokes version of DarcyTools and possibly other models will be used. Studies are in progress and further studies are planned to analyze how the stagnant water volume that is created in connection with channelling affects the diffusion of transported solutes from the mobile water to the adjacent immobile aqueous phase and further into the matrix. Solutions for this problem have been devised previously /25-89/ and have even been analyzed as a part of the site modelling /25-86/ and within SR-Site. The results of the analysis within SR-Site indicate that there is reason to reduce F. The reduced flow-wetted surface area resulting from channelling is offset by the additional surface area that becomes available in the stagnant parts of the fracture plane. After SR-Site, additional more systematic simulations using available methods will be carried out to arrive at a more general strategy for handling channelling in different applications.

The programme also includes simulation of flow and transport in single fracture planes with variable aperture. Both the planned Navier-Stokes version of DarcyTools and possibly other models will be used.

Within the framework of SR-Site, the question of how advection affects the transport of corroding substances such as methane, oxygen and organic matter to the repository is being analyzed /25-22/. The question of the difference between mixing and dispersion in the flow models is also being analyzed within SR-Site, where the effect of dispersion on dilution and resulting radionuclide transport is being analyzed using the code MARFA.

A project is planned to examine what different types of porosity (fracture apertures) are used (different conceptualizations) and how they are measured and used in different applications. Studies during the past 15 years (for example the TRUE experiments in the Äspö HRL) have resulted in an improved conceptual description and quantification. What is missing, however, is an integrated description that spans relevant scales and combines utilized concepts in a number of disciplines. The planned project has links to several other processes besides advection (specifically diffusion) since, depending on which type of process is being studied, porosity on different scales is involved. Different porosities emerge, ranging from the individual pores of the matrix rock (matrix diffusion, matrix pore water), via the porosity of single fractures with their filling materials (in situ-tracer tests), to the porosity of large conductive deformation zones and the rock's large-scale bulk porosity (regional flow modelling, particle tracking, palaeohydrogeochemistry, etc.).

25.2.12 Diffusion – groundwater chemistry

This section is about the effects of molecular diffusion of groundwater components. Its importance for nuclide transport is dealt with in the next section. Diffusion of the groundwater's gas components is described in a later section.

The interaction between the matrix pore water/fluid and the groundwater in the fractures takes place mainly by diffusion. Chemical reactions mainly take place with fracture-filling minerals, although reactions with the rock matrix's primary minerals can in the long term take on increased importance for the chemistry of the groundwater.

Conclusions in RD&D 2007 and its review

Studies of matrix pore water in the Äspö HRL (the Matrix Fluid Chemistry Experiment) showed that there is no reason to suspect that there is matrix pore water of high salinity at repository depth in an environment such as that on Äspö /25-90/. The lessons learned from the Matrix Fluid Chemistry Experiment were then used in the site investigations. The sampling and analysis techniques were further developed /25-91, 25-92/.

In its review, SKI says that SKB should study the evolution of the groundwater stratification over time. In the review they support SKB's plans for continued investigations of matrix fluid. However, they say that the plans lack studies of how chemical reactions in the rock matrix affect radionuclide transport.

Newfound knowledge since RD&D 2007

The investigations of Forsmark and Laxemar have in general confirmed accepted hydrogeochemical knowledge /25-73, 25-74/. New matrix pore water data has emerged /25-93, 25-94, 25-95/. Pore waters of postglacial origin in the uppermost 500–600 metres of the rock matrix and accumulation of glacial meltwater by diffusion during the last glacial retreat have been identified. In some locations, close to conductive fracture zones, the composition of the pore water suggests some in-diffusion of sea water after the latest ice age. A pore water of warmer meteoric origin has also been identified. The age of this component is unknown, but it must have originated before the last glacial cycle. These pore waters have been preserved untouched over long periods of time due to a low fracture frequency. They therefore serve as a palaeohydrogeological archive and help in interpreting the circumstances surrounding the evolution of the Baltic Sea and its impact on the near-coastal sites. The trends occur to a varying degree in both Laxemar and Forsmark; the biggest difference lies in the location of the different pore water types, which reflects the different hydraulic properties.

The studies have also permitted interpretation of diffusion properties for the main rock types. The results show that transport of solutes in the rock matrix is driven by diffusion and that matrix diffusion takes place over distances of several tens of metres. Even though the quantity of data is limited, these preliminary observations support the models for radionuclide transport calculations in the safety assessments.

Programme

There are no immediate plans for further extensive analyses of matrix pore water. However, SKB is planning several laboratory and modelling studies to further improve the analysis technique and the interpretation of existing data. New developments are being monitored and will be acted on when appropriate.

25.2.13 Diffusion – radionuclide transport

Conclusions in RD&D 2007 and its review

The programme that was presented in RD&D Programme 2007 includes analyses and final reporting in the project Diffusion Experiments Sorption-Diffusion (LTDE-SD) in the Äspö HRL. Diffusion tests in the laboratory on LTDE-SD material were also planned in order to permit comparison of the results in LTDE-SD with results from the site investigations.

In its review, SKI says that SKB should address questions relating to in situ measurements of electrical resistivity and supplementary laboratory measurements. It needs to be shown that the measurements reflect diffusion of ions in pore water and are not affected by conductive minerals in the rock.

SKI also calls for a clearer connection between quantification of matrix diffusion depth and the properties of the rock, for example different types of microfractures, the degree of mineral alterations and the accessibility of the rock matrix for diffusion.

SKI also asked for an analysis of the Single Well Injection Withdrawal (SWIW) tests in order to increase confidence in the parameterization of the transport models.

Newfound knowledge since RD&D 2007

Data on diffusion properties from the site investigations have been presented for Forsmark /25-96/ and Laxemar /25-97/. Further interpretation of these data can be found in /25-86/ and /25-98/ for the respective sites. Studies of the diffusion properties of the rocks yield formation factors (i.e. how much lower the diffusivity in the rock is compared with the diffusivity in pure water). Results of both through-diffusion measurements in the laboratory and electrical resistivity measurements in the laboratory and the field are presented. Laboratory measurements of the formation factor by means of electrical methods gave higher values than equivalent in situ measurements, indicating that destressing and disturbances in the samples due to the drilling can cause an overestimation of diffusivity in the laboratory measurements.

A study has also been done to evaluate the uncertainties associated with the electrical resistivity method for measurement of the formation factor /25-100/. In the study, the geometric formation factor obtained from through-electromigration tests and through-diffusion tests is compared with the apparent formation factor obtained with electrical methods. The results indicate that the formation factor may be overestimated, due to surface conductivity effects, if the apparent formation factor is used directly. This applies specifically to superficial conditions, where the groundwater is typically of low salinity. At high ionic strength, the difference between the apparent formation factor and the geometric formation factor is small /25-100, 25-99/, indicating that electrically conductive minerals do not appreciably influence the results. The results here satisfy the wish of the regulatory authorities to show that conductive minerals do not affect electrical measurements of ion mobility.

A long-term test for determination of diffusion and sorption properties in situ has been carried out at typical repository depth in the Äspö HRL. The experiment proceeded without interruptions or other disturbances and was concluded according to plan after just over six months. The rock around the test sections in the packered-off fracture and in the matrix rock behind it was taken out by overcoring. Small sample cores with a diameter of 24 millimetres were taken out from the big drill core from overcoring. A total of 18 cores, about 17 centimetres in length, were drilled from the fracture surface on the stub. Around the test section in the small-diameter borehole (diameter 36 millimetres), 16 cores were taken out in the matrix rock. The sample cores were scanned with a scintillation detector and measured by gamma spectrometry. The sample cores were mapped in detail and photographed through a stereo magnifying glass. They were then cut into thin slices and measured by autoradiography to determine the distribution of radioactivity in the slices. Nuclide-specific analyses of the slices were done to determine penetration profiles. Autoradiography shows that the radioactivity (i.e. the trace element content) is unevenly distributed, especially in the rock adjacent to the fracture surface, which can be linked to the spatial distribution of porosity and mineral. Penetration depths of up to six centimetres for weakly sorbing elements (Na, Cl) and a few centimetres for Cs were observed. The elements that form surface complexes (e.g. Gd and Ni) are sorbed on the fracture surfaces and the internal surfaces in the rock matrix. For Ni, the observed maximum penetration depth is three millimetres in the rock matrix and 5–6 millimetres in the rock adjacent to the fracture plane. These results satisfy the regulatory authority's request that SKB should demonstrate the relationship between diffusion depth and the properties of the rock.

Supporting laboratory experiments were performed on samples from the in situ experiment's rock matrix and fracture surface (from a nearby parallel investigation borehole) and the same nuclide cocktail as in the in situ experiment, with the addition of tritium. The measurements were made on whole pieces of rock and crushed rock material in three size fractions. Measurements of sorption on whole pieces of rock show the same relative sorption strength, i.e. the ratio between the nuclides, for the laboratory measurements as for the in situ experiment. Comparison of in situ results with results from the laboratory measurements on rock samples is not fully possible yet, since the results from the in situ measurements are under evaluation.

The effect of biofilm on matrix diffusion is addressed in the section on microbial processes (Section 25.2.16). Results of analyses of matrix pore water are presented in Section 25.2.14.

Programme

Regarding measurements of the formation factor by means of electrical resistivity methods, a correction method for dealing with this problem will be proposed in SR-Site and presented in the Data Report for SR-Site.

Some supplementary measurements and quality checks of experimental data remain to be done within the LTDE-SD project. Evaluation, modelling and reporting also remain to be done.

A simpler setup of LTDE-SD will be tested as a possible method for investigation of transport properties at repository depth in conjunction with detailed characterization. This will be done in the Äspö HRL during the current period.

25.2.14 Reactions with the rock – groundwater chemistry

Dissolution and precipitation of fracture-filling minerals is a constantly ongoing process. Most of these minerals have been formed under hydrothermal conditions (which means they are older than 400 million years), but low-temperature minerals also occur, for example calcite. They can be utilized to understand the evolution of the groundwater, for example with respect to redox conditions. Reactions with the minerals in the rock matrix occur to a limited extent and the processes are slow, but in the long term these reactions with the primary minerals of the rock matrix may take on increased importance for the chemistry of the groundwater.

Oxidizing conditions at repository depth can be broken down into two sub-problems:

- The repository will be oxygenated during construction and operation. Some oxygen will thus probably remain in and near the repository after closure.
- Concern has been expressed that oxygenated water might penetrate down to repository depth during periods of greatly changed hydrogeological conditions, for example in conjunction with a glaciation.

Conclusions in RD&D 2007 and its review

The account in RD&D Programme 2007 included further experiments with weathering of ferrous fracture-filling minerals, mainly for the purpose of ascertaining the occurrence of microbial processes in laboratory experiments. Dating with the aid of iron(III) oxides and uranium series analyses was planned, along with measurements of traces of organic molecules, so-called biomarkers.

SKI pointed out in the review that the modelling that had been done within the framework of SR-Can should be supplemented with an analysis of uncertainties in both the rock's reducing capacity and the models for infiltration of glacial meltwater.

Newfound knowledge since RD&D 2007

As a result of previous EU projects (EQUIP and PADAMOT), a methodology has been developed for fracture mineral investigations /25-101/ and this method has been applied in the site investigations for Forsmark and Oskarshamn. Studies of stable isotopes and tracers are reported in /25-102/ and /25-103/. The ages of fracture-filling minerals and the subsequent tectonic evolution on the sites have been documented by means of direct and indirect dating methods /25-104, 25-105/. The use of fracture-filling minerals to gain a better understanding of palaeohydrogeology has also been discussed in other international constellations /25-106/. All in all, experience of the employed methodology is positive and the methodology can be considered to produce the results that are needed for an understanding of the site with respect to both past and present interaction between minerals and water. In addition to careful analysis work and microscopy, the availability of drill cores of high quality, permitting sampling of undisturbed material, is of crucial importance for the results. Integration of hydrogeochemical and hydrogeological information is also an important component.

In order to be able to evaluate the risk that glacial meltwater will reach repository depth, the distribution of redox-sensitive fracture-filling minerals, such as pyrite, has been studied in detailed investigations in Laxemar /25-107/ and Forsmark /25-108/ and in the ongoing studies of uranium series isotopes at the Äspö HRL. These studies show that reducing conditions have prevailed at repository depth for a long time and that oxygen is consumed in the upper 150 metres of the rock, if not already in the soil layer. Uranium series analyses have contributed to the information on redox conditions that has been obtained via studies of minerals and element distribution, partly because uranium is redox-sensitive, and partly because the analyses of the isotopes in the uranium series can distinguish processes that have been active during the past million years. A methodology study concerning uranium series analyses has resulted in a recommended analysis and sampling methodology. Special studies of Fe(III) oxides for the purpose of differentiating between young Fe(II) precipitations and older precipitations have been done and have yielded positive results. Fe isotopes and Fe(II)/Fe(III) have been determined, and detailed XRD has been combined with SEM studies /25-109/. With the aid of this method it is possible to identify artificial Fe(III) precipitates resulting from drilling.

The fracture-filling mineral and the country rock contribute available buffer capacity. It is therefore essential to determine the extent and quantity of, for example, minerals that can act as a buffer for redox or pH. Reddish coloration, i.e. oxidation, of the country rock is common along many fractures. Detailed studies have been made both in Forsmark /25-110/ and Laxemar-Simpevarp /25-111/ to describe the origin of the reddish coloration and to describe how much of the country rock's reduction capacity has been preserved. It was found that the alteration of the country rock, on both sites, is of hydrothermal origin (very old, in other words). It consists above all of a mineralogical transformation of biotite to chlorite and plagioclase to fine-grained mica (sericite), and partial oxidation of magnetite. The red colour is due to the fact that micrograins of hematite have impregnated above all the altered feldspar. The total reduction of Fe(II) is small and in many cases negligible.

A statistical processing of the quantities of fracture-filling mineral (based on mapping) has not previously been possible, but this has been successfully tested in both Forsmark and Laxemar /25-98, 25-86/.

The studies of weathering rates have resulted in a Ph.D. thesis /25-112/. The dissolution rates can be used to estimate the release of Fe(II) to groundwater.

Programme

Isotope studies of low-temperature precipitations of sulphide minerals in the fracture system (both naturally occurring and caused by disturbances due to drilling of the borehole) are planned in order to gain a better understanding of processes that can affect the sampling and analysis of sulphide in groundwater.

The lessons learned from the uranium series analyses along with determinations of the degree of oxidation of fracture-filling minerals will be applied to uranium-bearing minerals from fractures in Forsmark for the purpose of gaining a better understanding of which processes have caused the mineral precipitations and which processes dissolve them and cause the elevated uranium concentrations in the groundwater. Moreover, uranium series analyses together with mineralogy and trace element chemistry will be applied to study penetration of oxygenated water in glacial areas. The latter work is planned to be done within the framework of the Greenland Analogue Project, see Section 19.6.

Data will be gathered to study abiotic reactions between dissolved oxygen and different minerals kept in sterile flasks. The studies will examine the difference between different minerals and rock types, the effect of different mineral preparation methods (fragmentation) on the reaction surfaces, and the reaction rate. Preliminary results show that the presence of microbes has a strong accelerating effect on the loss of oxygen.

Studies of redox conditions and method development aimed at improving redox measurements will be pursued in the EU project ReCosy (Redox Phenomena Controlling Systems), where several of SKB's cooperation groups in Sweden and abroad are participants.

Studies of ion exchange processes are planned to gain a better understanding of how the salinity of the groundwater changes with time. These studies may result in a quantification of the exchange of Na and Ca when rain or sea water infiltrated the areas investigated by SKB. Sr isotope data obtained in Äspö could also contribute knowledge on these processes.

25.2.15 Reactions with the rock – sorption of radionuclides

Conclusions in RD&D 2007 and its review

Completion of a doctoral project concerning electrical methods for measurement of K_d values was planned in RD&D Programme 2007. Development of process-based modelling of sorption and reactive transport and execution of the analysis and reporting phases of LTDE-SD were also included in the programme.

In its review, SKI takes a positive view of the programme, but says that SKB should also link conceptual models of radionuclide sorption to the properties of the rock, for example the degree of mineral alterations, microfractures and the accessibility of the rock matrix. SKI also asks for greater clarity in the programme regarding the methodology for determination of K_d values.

In its review, the Swedish National Council for Nuclear Waste says that it is very urgent that the programme for radionuclide sorption be carried out as planned.

Newfound knowledge since RD&D 2007

Data from site investigations have been presented for Forsmark /25-96/ and Laxemar /25-97/. Sorption measurements (resulting in the distribution, or partition, coefficient K_d) have been done on crushed material and on whole pieces. Furthermore, CEC (Cation Exchange Capacity) and specific surface area

(BET) measurements have been performed. The sorption results indicate that rock material adjacent to fractures (different forms of altered rock) gives higher K_d values than unaltered rock. It can also be clearly observed that tracers that are expected to adsorb by cation exchange adsorb more strongly at lower ionic strength in the groundwater. Sorption measurements on a small number of samples from different deformation zones indicate varying sorption strength compared with intact rock, from weaker sorption for oxidized country rock to elevated sorption for material from shear zones.

The doctoral project concerning electrical methods for measurement of K_d values has been concluded /25-113/. The thesis describes a method for measuring sorption on whole pieces using electrical methods. The advantage of the method is that it is several orders of magnitude faster than using sorbing substances in a through-diffusion test where K_d can also be evaluated. A methodology for measurement of BET surface area on whole pieces has also been developed. The results indicate that both sorption and surface area measurements on crushed material can overestimate sorption properties and surface area, respectively. The methods have already been used to support experiments in the site modelling and are described in this context in /25-97/. The individual papers in the thesis are presented in /25-114, 25-115, 25-116, 25-117/.

A doctoral project on the subject of process-oriented sorption modelling has been initiated and is being pursued jointly at KTH and Chalmers. The background of the project is that sorption of radionuclides and other tracers to geological material in the bedrock has typically been quantified in detailed laboratory tests and interpreted in the form of empirical K_d models for use in safety assessment models. More recent investigations have, however, indicated conceptual uncertainties in such models, where above all specific surface area, mineralogy and particle size comprise dominant uncertainties that can potentially explain at least some of the large spread in K_d values that has been observed. The aim of the current project is to formulate and test process-quantifying models and systematically quantify the sorption (ion exchange and surface complexation) of radionuclides to geological material from Äspö via laboratory tests with a special focus on particle size dependence and localization (minerals and surface structures) of sorption.

During the initial period of the project, work has been done with the pure minerals labradorite, chlorite and magnetite. These minerals represent commonly occurring and especially sorptive minerals in Swedish granite. In the extrapolation of sorption results from laboratory tests to field conditions that is often done, sorption is as a rule assumed to be proportional to the specific surface area. However, experimental results show that the sorption of Ni^{2+} on chlorite is very similar between two different samples with very different specific surface areas /25-118/. This is a behaviour that can possibly be explained by the anisotropic properties of chlorite, which is a mixed-layer mineral where similarly reactive surfaces between the samples can arise in connection with differences in the morphology of the minerals. The result indicate that in order to obtain relevant K_d values, the sorption should be related to the reactive surface area, rather than the total specific surface area. The reactive surface area can be expected to play a crucial role for other important surface chemical reactions as well, such as mineral weathering and reduction of molecular oxygen at the surfaces of reducing minerals. However, there are no experimental methods today for direct and simple quantification of the reactive surface area of a mineral simple by experimental or theoretical methods.

The LTDE-SD experiment has been carried out in situ without interruption or other disturbances and was concluded according to plan after about six months. The results are described in Section 25.2.13.

Programme

A simpler setup of LTDE-SD, which will be tested in the Äspö HRL during the coming period (see Section 25.2.13), is expected to yield in situ K_d values (rather than diffusion values).

Continued development is planned for measurement of CEC and BET on whole pieces. Linked to this is development of a methodology for interpretation of different types of K_d measurements. This method development work is being done within the framework of SR-Site, but will continue even after the final reporting of SR-Site.

Development of mechanistic modelling of sorption (surface complexation and ion exchange) is continuing within the Äspö Transport Model project. During the period 2010–2012, specific surface area and porosity as a function of particle size will be studied in the project for a number of minerals and rock samples in an effort to improve our understanding of the relationship between sorption properties observed in laboratory tests and those observed in the field. Supportive laboratory tests of sorption of some radionuclides to the same mineral samples will also be carried out. In order to try to tie the sorption to the reactive surface area, autoradiography will be used to try experimentally to identify which specific minerals, in a complex geological material such as granite, sorb the radionuclide. Furthermore, the possibilities of using autoradiography to associate the sorption to special surface structures or grain boundaries in pure minerals or granite will be investigated. Autoradiography will also be used to try to investigate the sorption of radionuclides to different surfaces of an anisotropic mineral such as chlorite or biotite as a further step in investigating the importance of different surfaces for a mineral's resistivity. Based on results from these investigations and literature data, models of radionuclide sorption to granitic materials will be formulated according to the Component Additivity Method /25-119/. In order to investigate the reliability and usefulness of the models, model results will be compared with observations from laboratory experiments with sorption on granite.

25.2.16 Microbial processes

Microbes can affect groundwater chemistry by speeding up reactions that would proceed very slowly or not at all in their absence, for example sulphide formation. Redox reactions are particularly affected, but precipitation and weathering reactions and radionuclide transport processes can also be affected /25-120/.

Conclusions in RD&D 2007 and its review

RD&D Programme 2007 described results from the laboratory that had been set up in the Äspö HRL /25-121/. This work yielded greater knowledge of microbial processes under repository conditions with respect to microbiological diversity, chemical environment, dissolved gases, pressure and temperature. The effect of microbial biofilms on sorption of radionuclides and matrix diffusion was described in /25-122, 25-123, 25-124/. It was also shown that radionuclide transport can also be affected by complexing compounds excreted by microorganisms in groundwater /25-125/.

The programme included further research regarding the impact of microbes on sulphide formation, oxygen reduction and buffering of the redox potential, the impact of bacteriophages (viruses that attack bacteria) and the start of new modelling projects.

SKI judged that the research planned by SKB on microbial processes is urgent. The most important question, especially in cases with buffer erosion, must be regarded as understanding how active sulphate-reducing bacteria can be expected to be during the long-term evolution of the repository. The feasibility of predicting the extent of copper corrosion is in part dependent on how well limiting and controlling factors have been identified and investigated. SKI notes that SKB did not discuss microbial activity during periods of permafrost and glaciation in SR-Can.

Newfound knowledge since RD&D 2007

Research has continued on microbial processes that lead to sulphide formation and the ability of microbes to buffer the redox potential at a low level that is favourable for the repository /25-126/. The work has been carried out at repository depth in the Äspö HRL under in situ conditions with naturally occurring microbes and on the candidate sites Laxemar /25-127/ and Forsmark /25-128/.

New methods have been developed for determination of biomass in groundwater /25-129/. The occurrence of bacteriophages /25-130/ and their impact on sulphate-reducing bacteria /25-131/ have been investigated. Microbial mobilization and immobilization of radionuclides have been quantified under different conditions /25-132, 25-133, 25-134/.

A project has been initiated for the purpose of modelling microbial processes with results obtained from the site investigations and the Äspö HRL. As the research on microorganisms in the geosphere has generated new results, they have been incorporated in modelling tools /25-135, 25-136/.

Development of models has been focused particularly on microbial processes that can have an adverse effect on a barrier function, for example sulphate reduction to sulphide.

Programme

The research on the ability of microorganisms to maintain the chemical environment in the near- and far-field at a low and stable redox potential will continue. The focus will be on the capacity of the microbes for sulphide formation and oxygen reduction in the near- and far-field.

The relationships between content and distribution of gases and microbial processes in deep groundwaters will also be explored. Efforts are also planned for method development and for greater knowledge of gases and their origin in groundwater and rock matrix. Studies of the ability of microorganisms and viruses to influence the mobility of the radionuclides in a natural geosphere environment, where biofilms develop on fracture surfaces and fracture-filling minerals, will continue. The need for further development of modelling tools for microbial processes will be reviewed.

Microbial activity during periods of permafrost and glaciation is addressed within the framework of SR-Site.

25.2.17 Decomposition of inorganic engineering material

The process is of importance for the geosphere in an initial phase and close to the repository, when conditions are affected by the construction work and by steel and cement in the repository.

Conclusions in RD&D 2007 and its review

The research on this process in RD&D Programme 2007 focused on leaching of low-pH cementitious materials in pore water in an in situ project in Grimsel, Switzerland, and in a joint project with Posiva. Continued work within this project was included in the project.

In its review, SKI says that the planned experiments are urgent and that SKB should also shed light on the effects, if any, of cement leaching on the engineered barriers.

Newfound knowledge since RD&D 2007

The results from the experiment in Grimsel, Switzerland, aimed at studying how concrete pore water reacts with rock minerals have been published /25-137, 25-138/.

Development and testing of low-pH grouting materials, i.e. grouts that are supposed to have pore water with a pH lower than about 11, are described in Section 15.5.3. Leaching of low-pH cement was done in a joint study with Nagra, NUMO and Posiva for the purpose of investigating whether organic cement additives can be leached out and whether they can affect radionuclide sorption on rock /25-139/.

Programme

Studies of organic cement additives will continue. SKB will also continue the work of modelling the degradation of cementitious materials over long periods of time.

The other principal inorganic engineering material is steel. This field is judged today not to require any further research for KBS-3V, but it may have some bearing on KBS-3H /25-140/.

25.2.18 Colloid formation – colloids in groundwater

Colloids are particles on the nanometre to micrometre scale and therefore have a high specific surface area. Colloids can stay in solution for long periods if stability exists. In granitic groundwaters where mineral surfaces and groundwater have equilibrated over long periods of time, the salinity is relatively high and colloids are therefore not stable. The natural colloids consist largely of clay, silicon and iron hydroxide particles. Organic colloids may also be present, but at a depth of about 400 metres the concentrations are low.

The conditions for colloid generation from the bentonite buffer can change during a glacial cycle. Beneath an ice sheet, meltwater could penetrate down to repository depth, and in contact with the bentonite buffer there is potential for montmorillonite colloids to be released for further transport.

Release of montmorillonite colloids can cause mass losses from the buffer, whose functionality would thereby be reduced. In the event of a leaking canister, radionuclides sorbed to montmorillonite particles can also be transported out from the buffer.

Conclusions in RD&D 2007 and its review

Transport experiments with latex and bentonite colloids resulted in the conclusion that these two types of colloids behave similarly at high water flows. This confirmed the fact that latex colloids can be used as a model for bentonite colloids at high flows. Experiments with bentonite colloids in groundwater in the Äspö HRL showed that the bentonite colloids precipitated quickly and no transport occurred.

Studies of the stability of bentonite colloids at different temperatures and ionic strengths were included in the programme. Among other things, a joint project was planned with the Paul Scherrer Institute (PSI) and KTH for the purpose of testing the equilibrium concentration of bentonite colloids in different types of experiments.

SKI asked for a clearer picture of remaining questions and strategies regarding the connection between colloids and bentonite erosion.

Newfound knowledge since RD&D 2007

Measurements of natural colloids have been performed in Forsmark and Laxemar /25-127, 25-128/.

In contact with a water that is virtually free of salt, a concentration of montmorillonite colloids outside the buffer at equilibrium on the order of 100 milligrams per litre (mg/l) is expected /25-141/. To investigate the influence of flow rates, bentonite type and groundwater chemistry on gel propagation and further colloid release, tests were performed at AECL in Canada in an artificial Plexiglas fracture (results not yet unpublished). Bentonite was compacted to pellets and deionized water or synthetic groundwater was pumped past. Colloid concentrations outside the bentonite pellets in the tests with deionized water at equilibrium are in parity with the concentrations obtained in /25-141/. In the tests where synthetic Grimsel water was used, the equilibrium concentrations were lower, of the order of ten mg/l. These concentrations have been confirmed by static generation and sedimentation experiments /25-142/. When montmorillonite colloids are exposed to γ-radiation, their stability also increases /25-143/. Why this effect arises has not been completely clarified, but experimental results indicate that the Fe(II)/Fe(III) ratio in the montmorillonite increases, which results in a higher negative surface charge on the particles and a higher repulsion between the particles.

If montmorillonite colloids are generated and are stable in solution, they can be transported away from the buffer. Transport experiments on colloids in a clean fracture in a granite block have been carried out /25-144, 25-145/ to specially investigate the influence of flow, colloid size and colloid type. These experiments have been modelled and the retention in the system is ascribed to a reversible colloid sorption to the mineral surfaces and an irreversible physical filtration /25-146/. Transport experiments have also been conducted in columns packed with fracture-filling materials. Sorption of colloids to mineral surfaces under unfavourable conditions (repulsion between colloid and mineral) has been studied in parallel with the transport experiments /25-147/.

Dipole experiments have also been conducted in a fracture system on the TRUE-1 site with fluorescent latex colloids. Breakthrough curves show that the latex colloids were transported under these conditions, but the uncertainty in the data is too great to be able to calculate filtration constants. The results of the in situ experiments will be published in a final report for the Colloid Dipole and Transport projects at the end of 2010.

The competition between sorption of radionuclides to colloids and fracture-filling minerals in groundwater from Äspö (more saline groundwater) and from Grimsel (more dilute groundwater) has been studied at INE-FZK in Karlsruhe. In these experiments, U-233, Tc-99 and Np-237 do not associate with montmorillonite colloids, which, on the other hand, Th- 232, Pu-242 and Am-243 do. A transport experiment with radionuclides and montmorillonite colloids in a well characterized fracture from Äspö exhibits a slightly different trend, namely that Am, Pu and U, and to some extent Np, are transported with montmorillonite colloids. These experiments are in progress and will be reported in the final report for the Colloid Dipole and Transport project at the end of 2010.

Programme

Even if the necessary conditions exist for generating colloids at high concentrations, the consequences will be linked to whether there is significant transport from the buffer. Previous studies show that sorption and physical filtration are significant for colloid transport. There is also a possibility for colloids to clog small fractures and restrict flow paths. In order to be able to predict colloid transport in fractures with different properties, more experiments are needed. The purpose should be to eventually develop models for montmorillonite transport in fracture systems for the safety assessments. SKB is participating in the joint project CFM (Colloid Formation and Migration) in Grimsel that is planned to continue at least until 2013 and where in situ transport experiments with montmorillonite colloids are planned. A large-scale bentonite erosion experiment is also planned to be conducted within the framework of CFM.

When the bentonite swells and generates colloids, it is uncertain whether the released particles are in equilibrium. Images of Ca-montmorillonite colloids in one-year-old solutions show ellipsoidal colloids with regions of lower density inside the colloid and a shell of higher density /25-148/. Investigations are planned within the framework of the CFM project to find out whether the size and structure of bentonite colloids changes with time. The need for further studies of γ-radiation effects on montmorillonite colloids will be examined.

Sorption of radionuclides to colloids is an area where more data are needed, including kinetics data. The size distributions of the investigated colloids in existing studies are very wide and sorption values are normalized to mass, where they should reasonably be normalized to surface area. The sorption kinetics is usually very fast and desorption is very slow, which explains why so few rate constants have been determined. Experimental studies are therefore planned to examine the effect of colloid size and to determine the kinetics data that are needed for the models that are used in the safety assessments.

25.2.19 Colloid formation – radionuclide transport with colloids

Conclusions in RD&D 2007 and its review

In RD&D Programme 2007, SKB noted that questions remained concerning the impact of colloids on radionuclide retention. This was to be studied in the Colloid Project, where experiments were planned to be conducted in situ in a drill core.

SKI remarked that SKB's account did not provide a clear picture of what remaining problems are most important to study and what strategies are needed to tackle them.

Newfound knowledge since RD&D 2007

A doctoral project that has studied the formation, stability and transport of colloids has been concluded /25-147/. The results are discussed above (Colloid formation – colloids in groundwater), except for transport, which is dealt with here.

As mentioned in Section 25.2.18, column experiments have been performed with Na-montmorillonite and latex colloids that are transported through columns filled with fracture-filling materials. The results indicate that colloids sorb on the minerals and thereby retard colloid transport. Na-montmorillonite sorption is reversible, while sorption of latex colloids is irreversible during the course of the experiment. Modelling of colloid transport linked to the Colloid Project has been carried out in /25-146/.

Radionuclide transport with colloids is being modelled in SR-Site. Both reversible sorption on colloids and irreversible sorption on bentonite colloids are being studied within the framework of SR-Site.

Programme

Development of colloid modelling continues. This includes both development of modelling tools, where MARFA may be updated to incorporate radionuclide transport with colloids, and studies to gain a better understanding of the driving processes specifically linked to bentonite erosion and colloid formation, see Sections 24.2.20 and 25.2.18.

25.2.20 Gas formation/dissolution

The following gases usually occur dissolved in the groundwater, in order of decreasing concentration: nitrogen, methane, argon, helium, carbon dioxide, hydrogen and carbon monoxide. Traces of other hydrocarbons such as ethane, ethylene, acetylene, propane and propylene also occur. Nitrogen mainly originates from the formation of the Earth, when N_2 condensed with other materials when the planet was formed, and N_2 has been seeping through the geosphere into the atmosphere since then. Most other gases can have several sources.

Conclusions in RD&D 2007 and its review

The total quantity of dissolved gas in Swedish groundwaters varies between about 30 and 100 millilitres per litre (ml/l) of groundwater, at depths down to a kilometre below the ground surface. These quantities lie well below the solubility limits for the encountered gases at the pressure prevailing at the depth in question. Larger quantities of gas have been encountered in groundwaters sampled in Olkiluoto: up to 300–400 millilitres of gas per litre of groundwater /25-149, 25-150/. Here the gas also has a more varying composition.

In RD&D Programme 2007, SKB planned to develop methods and equipment for gas analysis in groundwater. The possibility of analyzing dissolved inert gases was to be explored. The origin and transport of methane and hydrogen were identified as important questions in the programme.

In its review, SKI said that SKB had identified important questions, and they support the plans for development of equipment and analyses for gas sampling.

Newfound knowledge since RD&D 2007

The database of dissolved gases in groundwater is being added to continuously /25-126/, both with data from groundwater around the Äspö HRL and with data from groundwater at the investigation sites.

Research is being conducted on the influence of the gases on the occurrence and activity of microorganisms, and on the influence of the microorganisms on the content and composition of the gases. The results of these investigations are reported in Section 25.2.16.

Programme

Studies aimed at understanding the origin and transport of methane and hydrogen are of importance for safety assessments of the final repository, since these gases can be used for microbial sulphate reduction to sulphide. A very useful parameter for understanding the diffusion of gases in the rock is helium isotope composition. The heaviest isotope, He-4, is formed by radioactive decay of uranium and thorium in the Earth's mantle and crust and diffuses up to the surface through the rock matrix and in fracture systems. The other isotope, He-3, is formed by decay of tritium, which normally originates in the atmosphere. Analysis of the helium isotopes can therefore be crucial in describing gas transport, which is why it is imperative to build further on existing data /25-151/. The equipment for sampling of gas and gas analysis needs to be developed further, and sampling and determination of the isotope composition needs to continue in both Äspö and Forsmark when detailed site characterization begins.

25.2.21 Methane ice formation

At low temperature and high pressure, water and methane form a solid phase called methane ice. Methane ice can form under permafrost, for example /25-152/.

Conclusions in RD&D 2007 and its review

Together with researchers from Finland and Canada, SKB completed studies of a gold mine in a permafrost area in Canada (the Lupin Mine) /25-153, 25-154/. No methane ice was found during the studies in the Lupin Mine. The programme included further studies in mines in Canada and further investigations of pore water and groundwater chemistry at the site investigations.

In its review, SKI says that continued efforts in these areas are warranted, but that they find it difficult to determine whether SKB's studies in mines in Canada are appropriate for their purpose.

Newfound knowledge since RD&D 2007

No methane hydrates were encountered in the Lupin Mine, but certain indicators of their earlier presence were noted in /25-155/. This included identifying possible areas beneath the permafrost layer where induced pressure fall due to mining activities may have caused melting of methane hydrates, which may have contributed to the dilution that has been observed in some subpermafrost groundwaters. The permafrost studies have continued in High Lake /25-156/.

The possibility of tracing earlier occurrences of methane hydrate has been borne in mind during site investigations /25-73, 25-74/, but no convincing evidence was found in the studies of matrix pore water.

Programme

In order to understand methane ice formation, knowledge is needed concerning the origin of the methane and transport possibilities. Simulations will be performed to assess the potential for formation of methane hydrate beneath future permafrost. Studies in permafrost areas will continue either in other Canadian mines or on Greenland. Continued vigilance will be maintained concerning methane hydrates in the hydrogeochemical studies at Äspö and Forsmark.

25.2.22 Salt exclusion

When saline water freezes slowly, most of the solutes (salts) are forced out into the solution that remains after the ice forms. This process can be of importance in a cold climate, for example during a period with permafrost.

Conclusions in RD&D 2007 and its review

In the SR-Can safety assessment, SKB conducted generic model studies aimed at investigating whether excluded salts beneath a permafrost layer could become so concentrated as to affect the swelling capacity of the backfill. The calculations show that the saline groundwater moves rapidly downward in transmissive fracture zones due to its higher density. The saline groundwater is relatively immobile in rock volumes with low hydraulic conductivity /25-157/.

SKB planned in RD&D Programme 2007 to continue to participate in the investigations of mines in permafrost areas, see Section 25.2.21.

Newfound knowledge since RD&D 2007

The studies in the Lupin Mine have now been concluded, and even if frost processes have taken place it has been difficult to quantify to what degree freezing has altered the groundwater chemistry in the area /25-155/. The difficulties are exacerbated by the type of site, where there is extensive contamination due to many years of mining as well as mixing of different types of groundwater beneath the permafrost layers.

Mixing of different types of groundwater has also complicated the interpretation of hydrochemical data from Forsmark and Laxemar, but an indication of freezing processes may have been preserved, particularly in Laxemar /25-73, 25-74/. The isotopes of oxygen, boron and chlorine may be sensitive to exclusion processes, whether they are enriched in the ice (O-18 and Cl-37) or in the remaining liquid (B-11). The interpretation of isotope data that has been applied at Laxemar (especially in brackish groundwater of glacial type that has usually been encountered at about 300–600 metres and that seems to have been relatively well preserved in the bedrock) shows a decrease in hydraulic conductivity with increasing depth. The groundwater analyses show an increase in B-11 in most of these groundwaters, which may be a result of freezing-out processes. Furthermore, the distribution of Cl-37 exhibits a weak correlation between the increase of Cl-37 and the depletion of B-11 in the same groundwater, which supports a freezing-out effect.

A number of pore water samples from depth intervals of 430–550 metres and 620–750 metres at Laxemar show high Na-Ca-SO_4 concentrations. The origin of this sulphate is unclear, but possible sources are dissolution of gypsum and/or changes brought about by freezing processes.

In summary, even though there is some support for freezing-out processes that occurred at Laxemar (less at Forsmark), it has not been possible to quantify the degree to which freezing has modified the chemistry in the groundwater, as was the case with the studies in the Lupin Mine.

Programme

Studies in permafrost areas will continue in the Greenland Analogue Project (GAP) and possibly in other mines in Canada (High Lake, Nunavut Territory). Continued vigilance will be maintained regarding indications of salt exclusion within the ongoing hydrogeochemical programmes on Äspö and Forsmark.

25.3 Modelling

25.3.1 DFN

Conclusions in RD&D 2007 and its review

The regulatory authorities state that SKB should continue to develop discrete fracture network models, take into account the interconnection of existing fractures, and develop methods for identification of fractures in tunnels, and that it is positive that SKB is continuing to develop discrete fracture network models for density-driven flow.

Newfound knowledge since RD&D 2007

Discrete fracture network (DFN) models have been developed for both Forsmark and Laxemar during the work with the site descriptions. Geological and hydrogeological fracture network models have been developed for both sites, where the geological model describes all fractures on the site, while the hydrogeological model only characterizes the open and interlinked fractures.

It is not a simple matter to predict the distribution of the fractures in terms of intensity, orientation and size based on data from surface boreholes and outcrops. The models exhibit differences depending on what assumptions are made. Both /25-158/ and /25-159/ present a number of geological fracture network models of different likelihood depending on which conceptual model they are based on. This, together with uncertainties, can generate a cumbersome number of models.

By taking into account more parameters when setting up the geological DFN model – such as data from an additional measurement scale, hydrogeological information, rock stresses or geophysical information – the range of outcomes can be narrowed and the number of model variants reduced. The range of outcomes could be further reduced by a better understanding of the connection between measurements on different scales, for example outcrop versus lineament, and of different dimensions, for example borehole versus tunnel wall.

A new way to incorporate uncertainty and variability in DFN models is described in /25-160/, which also presents a statistical method for linking rock units to fracture domains. An alternative method for interpreting data is also developed there, resulting in yet another model variant with fractures that are truncated against other fractures. Others who have studied alternatives for generating fracture networks are /25-161/.

A system for ranking credibility and tools for evaluating differences between different models at an early stage are needed so that all theoretically possible models are not propagated to the customers with the risk that the total range of outcomes is too great.

The geological model serves as a basis for construction of the hydrogeological fracture network models /25-18/ and /25-16/, but hydrogeological knowledge and data are also taken into account and other conceptual assumptions are made, for example the assumption of the smallest detected fracture in the borehole. A consequence of the method used in /25-18/and /25-16/ is that the intensity is overestimated and the size distribution of the fracture set with a dip close to 90 degrees is skewed. As a result, there are certain differences between e.g. the geological DFN model and the hydrogeological one.

An overestimation of intensity and size distribution, together with constant properties over the whole fracture plane, can lead to an overestimation of the flow capacity. Already in /25-88/ a network of a few fractures that were given heterogeneous properties for each fracture plane was studied by means of a pipe network model to see how heterogeneity within a fracture plane affects the total flow through the fracture network. The conclusion was that the internal heterogeneity was subordinate to the heterogeneity that arises between the discrete fractures. However, the whole fracture plane was assumed to be more or less open and thereby more or less flowing, which is not necessarily the case, see Figure 25-2.

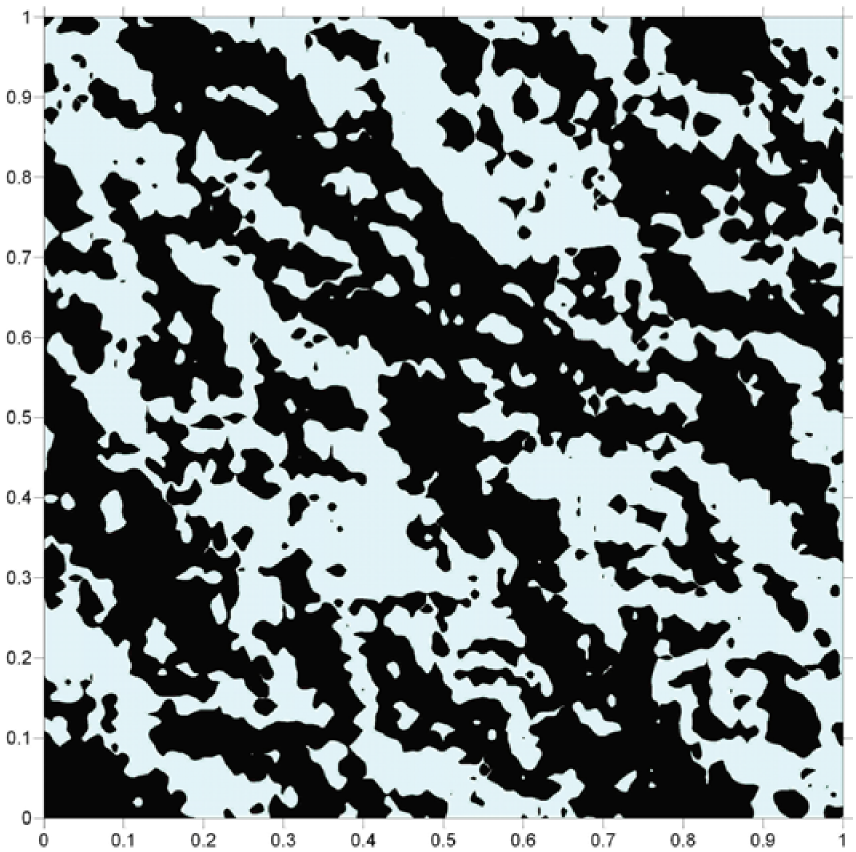


Figure 25-2. *Synthetic fracture plane (1×1 metres) showing areas with aperture (light blue) and without aperture (black).*

An increased understanding of the geometry of individual fracture planes can lead to a greater understanding of the flow through the fracture as well as its properties and contribution to flow in a discrete fracture network.

Programme

The work of gaining a better understanding of the fracture network in crystalline bedrock is continuing in order to get, if possible, a more integrated and thereby more realistic and less pessimistic picture that takes account of the geological, rock mechanical and hydrogeological aspects of the rock.

Based on the newfound knowledge, six points have been identified where time will be devoted to gaining a better understanding of discrete fracture networks in crystalline bedrock. SKB plans to:

- investigate how estimated intensity depends on the measurement method,
- study what extra data must be taken into account in order to limit possible values and combinations of values in the input data and thereby limit the range of outcomes for the models,
- develop a method for better evaluation of differences between different models,
- further explore the aperture distribution of the fractures over the fracture plane,
- study effects of interconnection of fractures, for example via channelling, truncation against other fractures, or alternative methods of generating fracture networks, and
- evaluate alternative concepts of fracture generation.

Depending on whether fractures are mapped along a line, for example a borehole, or on a surface, for example a tunnel wall, different measures of fracture intensity are obtained. Borehole mapping generally shows higher intensity than surface mapping, probably because fractures of smaller size are mapped in boreholes than on surfaces. This also has a bearing on depth trends, i.e. whether the properties of the fractures that are mapped on outcrops are representative of the fracture network at depth, which has not been studied by SKB to any appreciable extent since /25-162/. The relationship between data mapped along lines and on surfaces can be investigated more closely by a systematic review of currently available fracture data from the Äspö HRL and from site investigations. A joint project with Posiva (DEMO) has been initiated to demonstrate what resources and methods are needed to design deposition tunnels and holes for a KBS-3V concept. Data from this project can provide a better understanding of the relationship between fractures mapped in pilot holes and fractures mapped on a tunnel wall.

In the ongoing project to produce a new geological description of the rock mass in and around the Äspö HRL, the ambition is to incorporate data from several measurement fields in order to reduce the parameter span and thereby the range of outcomes of the discrete fracture network model.

Studying different fracture network models set up for different experiments at the Äspö HRL can provide guidance as to how different models can be compared to detect differences or similarities between two different fracture network models. Development of a tool that can compare two DFN models in a simple way would reduce the uncertainty regarding whether different models should be propagated as variants or whether they can be judged to be sufficiently equivalent to assume they cover each other.

Increased integration between models that describe all fractures and those that describe those that are flowing is desirable, i.e. the goal is to have **one** DFN model, with quantified variation and uncertainty. Properties can be associated with the fractures in this model in such a way that it is possible to distinguish those fractures that have a potential to carry flow from those that cannot flow, so that fractures that are not active in a given submodel can be extinguished.

Fracture geometries such as those in Figure 25-2, i.e. with flow paths that are made up of lobed channels in the fracture plane, result in the fact that the channels do not necessarily have contact with each other when two fractures intersect. The connection between two channels, within a fracture or between two intersecting fractures, can thus be non-existent or weak, which increases the flow resistivity and reduces the interlinking of the flow channels, which is not the case in today's models. By means of further studies of the properties of the fracture plane and the geometry of the aperture distribution, it should be possible to achieve realism for flow, and such a model should also be able to estimate stagnant volumes and flow-wetted surface area.

The numerical codes that are used today within SKB for generation of fracture network models are based on roughly the same concept, i.e. fractures are generated with location, orientation, size and properties instantaneously based on different distributions. Alternative methods generate only fracture seeds that then propagate with probability functions in the model volume according to set rules. A review of alternative concepts can reveal strengths and weaknesses in different approaches.

25.3.2 Integrated modelling – thermo-hydro-mechanical evolution

Conclusions in RD&D 2007 and its review

Different types of modelling and model development occur in RD&D Programme 2007, which are reported in different sections about rock strength and deformations. Often the modelling work has dealt with coupled issues concerned with thermal evolution, impact on hydraulic properties and mechanical consequences.

The regulatory authorities commented that in order to determine expected changes in transmissivity in fractures in a repository's near-field that could be caused by construction, thermal loading, swelling of the buffer or a glaciation scenario, SKB should take into account the possibility of formation of new fractures, fracture propagation and interconnection of existing fractures in the vicinity of the deposition holes. SKI said that new fracture patterns from each incremental type of loading can change the flow and the transmissivity in the near field of the deposition hole. According to SKI, this would mean that the 3DEC code needs to be modified to be able to create thermomechanical near-field models.

Newfound knowledge since RD&D 2007

The newfound knowledge concerning coupled modelling (HM, TM, THM couplings) is commented on in the above process-oriented sections on Gas flow/dissolution (Section 25.2.4), Thermal movement (Section 25.2.6), Reactivation – movement along existing fractures (Section 25.2.7), Fracturing (Section 25.2.8) and Time-dependent deformations (Section 25.2.9).

SKB has participated in the DECOVALEX 2011 project, which consists of three individual parts: i) ventilation experiment in a clay tunnel in Mt. Terri, ii) Äspö Pillar Stability Experiment (APSE) and iii) flow transport and hydrogeological modelling of a water tunnel in the Czech Republic.

The APSE experiment has been analyzed by seven different modelling teams, including one from SKB. The first part of the work has focused on developing models that can simulate uniaxial compressive strength tests on drill cores. Great success has been achieved by using models that simulate the mineral composition of the rock /25-163/. The current focus of the modelling of APSE is to fine-adjust coupled TM models to be able to calculate how the stress in the pillar changes as it is heated. The objective is to determine the spalling strength in the pillar. The models will, in a later phase with elasto-plastic modelling, try to simulate the geometry of the failure that occurred in the pillar. For this it must be possible to implement refined material models in the TM models.

Programme

An important point of departure in the design philosophy for the Spent Fuel Repository in Forsmark will be the application of the Observational Method, see Chapter 15. In order to make better predictions of the expected thermal, mechanical, and hydraulic behaviour, it is necessary to develop a model that can calculate parameters that will be measured as the construction of the repository progresses. This is not a trivial problem, since we can often measure what we cannot calculate and vice versa. This means that prediction and follow-up schedules must be worked out carefully, and with a good understanding of the nature of the problem. Against such a background, it is of interest to develop the application of proxy parameters in modelling.

In view of the premises of the Observational Method and in order to be able to handle different coupled rock mechanical issues, monitoring of computation programs that describe rock mechanical events on different scales is an important task for SKB, see Chapter 15. Of particular interest is the new generation of computation programs that have been developed in response to the need of the mining industry to understand the risk of failures in deep surface mines, or to instigate and control continuous failures in block caving (mass mining). This type of computation program tries to describe the composition of the rock mass from micro- to macro-scale, and failures can be computationally initiated in the weakest link, whether it be fractures or intact rock. That makes this type of computation program of long-range interest for analyses of failures due to earthquakes or thermal movement as well within the framework of the safety assessment's studies of coupled processes (THM).

SKB intends to execute a programme with new calculation codes and result verification based on unique databases established in different studies in Sweden or internationally, above all in cooperation with the different projects being conducted by Posiva in Onkalo. The goal of the programme is to ensure that the new modelling methods have essential advantages in relation to present-day approaches. Further, the programme should provide insights on the limitations of the respective model code. The programme should be regarded as a general development of modelling strategy so that the best possible modelling tools can be used when the underground works start in Forsmark.

SKB also intends to further develop the conceptual assumptions regarding stress-transmissivity relationships for fractures and deformation zones which are now used to evaluate results from mechanical and thermomechanical simulations of repository evolution.

The handling of the excess pore pressures during a glaciation scenario that is currently employed is based on continuum models with homogeneous hydraulic diffusivity. This probably overestimates the pore pressures. SKB intends to develop methods to take into account the discontinuous distribution of transmissivity and storage coefficient that exists in the bedrock.

SKB plans to further develop the DFN modelling, see Section 25.3.1. This further development will also take into account (T)HM-coupled applications.

Development of the calculation technique is planned for the earthquake simulations, see also Section 25.2.7. This should be done so that it is possible to use the results of the simulations to get an idea about what types of earthquakes are realistically possible given probable postglacial stress states, less conservative mechanical properties of potential earthquake zones, pore pressure conditions, etc. With an improved calculation technique it should be possible to distribute the strength (cohesion) over the area of the deformation zone instead of, as in the existing models, describing the reduction of strength to a predetermined residual strength. The failure can then be initiated by, for example, a pore pressure increase and thereafter be allowed to propagate spontaneously. The failure should then propagate as long as the gradually transmitted load is sufficient to exceed the strength in the failure zone.

With the goal of simulating typical conceptual fracture patterns that can occur in the Baltic Shield, SKB intends to carry out brittle-tectonic generic modelling in a relatively homogeneous crystalline rock type. Typical fracture mapping on outcrop surfaces of crystalline bedrock normally contain in the horizontal plane four more or less pronounced fracture systems that overlay possible foliation fractures or gneissification. The fracture systems are a result of a tectonic load history that may reflect varying rock stress situations ever since the bedrock was formed, i.e. including cooling deformations, the three-dimensional effect of various orogeneses, the effect of continental drift and the effect of glaciations in the observed bedrock region. In the brittle-tectonic modelling, boundary conditions and load cases are posited for the relevant bedrock based on an interpreted principal stress history. The project should be seen as an initial attempt at an alternative geometric description of possible flow patterns for bedrock water.

25.3.3 Integrated modelling – hydrogeochemical evolution

The simplest hydrochemistry model is a description of the spatial distribution of the concentrations of the most important solutes in the rock volume. The distributions of the concentrations of individual solutes can in some cases indicate specific ongoing chemical processes. More knowledge is obtained by statistical processing using multivariate analysis, which results in a subdivision into different classes. The different classes represent water which has undergone a certain evolution. By comparing the different classes, their different evolutionary pathways can be identified, regardless of where in the volume they occur. These classes then serve as a basis for further calculations of reactions and mixing ratios /25-82/. The calculated mixing proportions and the actual measured composition comprise the basis for calculating the scope of reactions. The code M3 (Multivariate Mixing and Mass balance calculations), which was developed with Matlab as a basis, is used for this hydrochemical modelling /25-82, 25-83, 25-84, 25-85/. The reasonableness of the results is checked by alternative modelling runs, for example geochemical simulations with the code PHREEQC.

Conclusions in RD&D 2007 and its review

No plans for further development of the method were presented in RD&D Programme 2007. The method was planned to be used in the site investigations and the safety assessments.

In its review, SKI expressed the opinion that some method development is needed in this area. SKI says that SKB should, in connection with an application for the KBS-3 system under the Environmental Code, pay greater attention to the risks of chemical-toxic effects on human health and the environment of e.g. grouts.

Newfound knowledge since RD&D 2007

The goals of SKB's hydrogeochemistry programme were 1) collect representative quality-assured data to be used as input data in the assessment of the repository's long-term safety, and 2) gain an understanding of current unaffected hydrogeochemical conditions and of how they may change in the future. For these purposes, different hydrogeochemical models were developed for Forsmark and Laxemar /25-73, 25-74/. The models included concepts based on qualification of field data, mixing models based on multivariate statistical analysis, models based on chemical equilibria, and models that attempt to couple groundwater flow with transport of solutes and chemical reactions. Furthermore, the hydrogeological models for Forsmark and Laxemar were calibrated by comparisons between calculated results and chemical field data /25-164, 25-15, 25-20/.

Models that couple hydrogeology with reactive transport have proved to be one of the most powerful tools for integrating hydrogeology with hydrochemistry. Large-scale coupled models of the Laxemar area have quantitatively integrated hydrogeological effects (i.e. the velocity of the groundwater) calculated with ConnectFlow with complex geochemical reaction models of the interaction between groundwater and minerals /25-165, 25-166/. Calculated results show that such an approach can reproduce the most important hydrochemical trends found in the Laxemar site investigation.

More in-depth knowledge of this area has been gained in the EU project FUNMIG (Fundamental Understanding of Radionuclide Migration). Of particular interest was the component called RTDC 4: "Processes and transport studies relevant for crystalline rock disposal concepts". Hydrogeochemical data from both Laxemar and Forsmark were used in these integrated projects, and several modelling methods were tested /25-167, 25-168, 25-169, 25-170, 25-171/.

Programme

A description of how groundwater composition is affected by sea water intrusion, permafrost and glaciation will be given in the SR-Site project. The coupling between hydrogeology and hydrogeochemistry will be done in a manner that resembles the method developed for SR-Can /25-80, 25-81, 25-79/.

The principal limitation for the models that couple groundwater flow with transport of solutes and chemical reactions is that the underlying constitutive equations are generally non-linear. This entails long calculation times and/or high demands on memory allocation, and thereby high costs. As a result, coupled model problems are normally solved by means of oversimplified geometries and simplified hydrogeological and chemical assumptions. New approaches are therefore required to go further. Possible ways may entail utilization of either effective calculation geometries or effective geochemical formulations that reduce the complexity and non-linearity that characterize the processes that describe the interaction between rock and groundwater.

Effective geometric solutions for models that integrate hydrogeology and hydrochemistry can be achieved by means of streamline simulation, where the geochemical processes are added along flow lines calculated with the hydrogeological models. Further research and development may be needed in the future in order to add mixing processes between different flow lines and equivalent dispersion. On the other hand, reactive transport experiments at the Äspö HRL may be useful for testing these models in the future.

25.3.4 Integrated modelling – radionuclide transport

Conclusions in RD&D 2007 and its review

In RD&D Programme 2007, SKB describes the plans to finish the TRUE project and expects to be able to use the results for evaluation of the SWIW tests in conjunction with the site investigations. The programme included testing and development of the MARFA and ConnectFlow codes, plus development of flows for reactive transport modelling. Transport modelling with MIKE SHE will continue in order to gain a better understanding of transport in the near-surface soil layers.

In its review, SKI considers that the programme presented by SKB is appropriate, but asks for clarification as to how results from retention experiments performed during a limited timespan are to be transferred to transport models for long-term safety.

Furthermore, SKI said that the programme for radionuclide transport in the geosphere should be more clearly linked to the corresponding programme for the biosphere.

Newfound knowledge since RD&D 2007

SKB has participated in the EU project FUNMIG (Fundamental Understanding of Radionuclide Migration; www.funmig.com). Of specific interest to SKB was RTDC number 4 (Processes and transport studies relevant for crystalline rock disposal concepts). A better understanding of processes linked to transport of radionuclides has been achieved in the project.

Within the Äspö Task Force on Modelling of Groundwater Flow and Transport of Solutes (TF GWFTS), Task 6 (Performance assessment modelling using site characterization data) has been concluded. The conclusions have been published in a number of contributing papers in a single scientific publication /25-172, 25-173/. A number of selected modelling papers were also published in the same issue; SKB's contribution is described in /25-174/. The most important conclusion is that it is not directly possible to use retention parameters evaluated from tracer tests on a field scale in safety assessment applications. The reason is that during the timespan for the tracer test, the properties of rock volumes nearest the fracture surfaces are mainly measured, while during the timespans for the safety assessment even deeper parts of the rock matrix are activated. A broader discussion of this aspect and its implications can be found in /25-175/. This report answers the regulatory authorities' questions regarding how results from retention experiments performed during a limited timespan (months–years) are to be transferred to transport models for long-term safety.

Development of the calculation code MARFA has continued and resulted in two separate versions: one for steady-state flow and one for transient flow /25-176/. MARFA is used within SR-Site to analyze specifically the effect of retention in tunnels on radionuclide transport, and possibly also to analyze radionuclide transport in transient flow fields (caused for example by shoreline displacement or glacial cycles). The methodology on which MARFA is based, i.e. a time-domain random walk, has also been published scientifically in /25-177/. In MARFA it is possible to use an upscaling algorithm based on transport simulations in small fracture domains, for parts of the domain where explicit fracture statistics are lacking. An alternative method for upscaling of transport based on data on fracture segments has also been developed and is presented in /25-178, 25-179/.

The SWIW tests performed within the framework of the site investigations and site modelling have been evaluated and reported in /25-180/. The tests confirm that retention takes place in the field, and further that the relative sorption between different tracers is consistent with laboratory tests. The TRUE experiments are concluded and are currently in a publishing phase. The tracer tests in a single fracture, TRUE-1, are summarized in a series of three articles /25-181, 25-182, 25-183/. The results indicate that the experimental TRUE-1 results can be explained with a retention model based on one-dimensional diffusion in an infinite matrix. The retention properties in the altered rock nearest the fracture are much stronger than indicated by laboratory measurements on unaltered material.

The tracer tests in a fracture network, True Block Scale, have also been summarized in a series of articles /25-184, 25-185/ in which the tests and the modelling are presented and fracture network simulations and retention models are described. The effect of heterogeneity in the superficial part of the matrix, known as the rim zone, has also been analyzed in /25-186/.

Models that couple hydrogeochemistry and radionuclide transport have been developed and tested, specifically the code Fastreact. This work is reported within the framework of SR-Site. The methodology is based on the use of streamtubes, which are obtained from hydrogeological modelling, see Section 25.2.3. A number of complex geochemical reactions, which in turn affect the retention of radionuclides, are solved along the streamtube. The advantage of this methodology is that it is fast, since only streamtubes of interest need to be analyzed (the hydrogeochemical reactions do not have to be solved in the whole domain) and it is easy to analyze heterogeneity when multiple streamtubes are introduced. The method can be used to analyze both transport from the repository and transport in the Quaternary deposits.

Programme

Further development of the MARFA code will continue. A judgement will be made as to whether additional retention processes should be included. Possible development includes transport with colloids, diffusion into stagnant water with accompanying matrix diffusion, and subdivision of the matrix into layers with different properties. In SR-Site, MARFA is used as a calculation tool for transport of radionuclides with colloids, but instead of being formally incorporated in MARFA, these colloid processes are handled via special versions of the code. MARFA will also be used in different applications and in future safety assessments.

A tracer test, SWIW with synthetic groundwater, is planned to be performed in the Äspö HRL. The synthetic groundwater is injected into the rock formation and then pumped back. The increase of the naturally occurring tracers in the returning water can then be analyzed and provide information about dominant diffusion processes in the fracture system. A preparatory modelling study has been carried out and is reported in /25-187/, showing that the experiment should be possible to perform. If this experiment is combined with injection of ordinary tracers, it should be possible to distinguish diffusion processes associated with exchange with stagnant water in the fracture plane and diffusion processes associated with exchange with the matrix.

Development of reactive transport modelling, i.e. coupled hydrogeochemical and radionuclide transport modelling, is continuing. The model Fastreact, based on the streamtube concept (see "Newfound knowledge" above), is an example of such a model. The effects of the spatial variability in fracture mineral composition are of particular interest. Data for this are available from the site investigations, and the results of this work will be reported within the framework of SR-Site. The knowledge that is obtained by the development of process-oriented sorption modelling (see Section 25.2.15) can eventually also be integrated in the reactive transport modelling.

In extension, the integrated transport modelling can handle the radionuclide transport taking into account the whole system, i.e. groundwater flow, geochemical reactions and radionuclide transport in both geosphere and biosphere. The distribution of streamtubes, from canister to biosphere with associated parameters (advective travel time, flow-related transport resistance), then comes from groundwater flow models that contain a detailed description of the rock's flow properties, see Section 25.2.3. In principle, the streamtubes can start in a model of the deep rock and then continue in a more detailed groundwater flow model of the surface system as described in MIKE SHE. Geochemical reactions and radionuclide transport are solved along the streamtubes.

26 Surface ecosystems

Surface ecosystems include loose deposits (soil layer and sediment), superficial groundwater, surface water, land surface and the lower part of the atmosphere. The consequences of a release from a final repository for spent nuclear fuel and other radioactive waste – in the form of a radiation dose to humans, animals and plants – occur when radionuclides are cycled in these ecosystems. Calculations of the accumulation and turnover of radionuclides in surface ecosystems, and of the risks associated with a release, are therefore an important part of the safety assessment. The calculated radiation risks are used to determine whether safety-related regulatory requirements (limit values) for human health and the environment are complied with, and as a yardstick for comparing different facilities, technical solutions or sites. Credible calculations require a realistic description of events and processes in the ecosystems with reasons why certain processes are important and why others can be ruled out. The states of the surface ecosystems also comprise chemical, hydrological and geological boundary conditions for the underlying rock.

The overall goal of the research and development programme for surface ecosystems is to describe, based on a scientific knowledge base, the most important processes and phenomena in the ecosystems from a radiological point of view and develop methodology and models that can be used to assess long-term radiological risks to humans and other organisms in conjunction with the final disposal of spent nuclear fuel and other radioactive waste.

SKB's basis for the planning of research and development regarding surface ecosystems during the coming programme period consists of the conclusions of RD&D Programme 2007, the regulatory authorities' comments on the programme, and insights gained from the site description and from the ongoing (2010) safety assessment SR-Site. The programme has been designed with a view to the requirements which future safety assessments, for both commissioned and planned disposal facilities, will make on the description of transport and accumulation of radionuclides in surface ecosystems.

26.1 Overview of the programme

The research programme for surface ecosystems has been divided into a number of research areas. They are: Terrestrial ecosystems (Section 26.3), Aquatic ecosystems (Section 26.4), Biogeochemistry (Section 26.5), Hydrology and transport (Section 26.6), Effects of long-term variations (Section 26.7), Landscape evolution and deposits (Section 26.8) and Radionuclide modelling (Section 26.9). These research areas are described in detail later in the chapter. A summary of the entire research programme follows below.

Conclusions in RD&D 2007 and its review

The emphasis in RD&D Programme 2007 as regards surface ecosystems was to increase process knowledge based on the site investigations at Forsmark and Laxemar-Simpevarp /26-1/. Continued gathering and analysis of site-specific data was planned, since this is a prerequisite for numerical descriptions of the sites. A review of the relevant scientific literature would, according to the plans, be carried out in parallel with this work. Furthermore, a need was identified to further develop the models used to assess radiological risk on the basis of transport, accumulation and biological uptake. The goal was to develop a methodology capable of describing exposure to all radionuclides of interest that includes effects on both man and the environment and that reflects relevant knowledge from the investigated sites with respect to landscape evolution and human use of natural resources. A comprehensive sensitivity analysis of the modelling results was planned, along with an improved description of permafrost conditions on the investigation sites. The programme also stressed the importance of continued national and international cooperation, as well as the importance of disseminating newfound knowledge via scientific publication and active participation in conferences and seminars.

In its review of RD&D Programme 2007, SSI was the expert authority for the biosphere area. SSI commented that SR-Can had shown how the knowledge gained from the biosphere programme had been applied in the safety assessment, and their review of the biosphere programme was therefore based to a great extent on the previous review of SR-Can /26-2/. SSI found that dose calculations based on the SR-Can methodology entailed clear progress in the development of the safety assessment, but noted a series of deficiencies that needed to be rectified prior to the licence application for the final repository. Among other things, the regulatory authorities pointed out that the methodology in SR-Can led to a dilution in the dose calculations, that certain transport processes were lacking, that the validation of underlying models against field data was deficient, and that an uncertainty analysis was lacking. SSI also called for clarification of how SKB planned to respond in the coming programme to the the regulatory authorities' comments on previous research programmes and safety assessments.

The Swedish National Council for Nuclear Waste also commented in its review that the biosphere programme had taken an important step forward with the development of site-specific models that describe transport and accumulation of radionuclides in relation to landscape changes in time and space. The Council welcomed the continued and intensified efforts in the field of the biosphere and said that the programme will provide important support to both the safety assessment and the EIA (Environmental Impact Assessment) process. However, the Council called for a clearer explanation of how the results of the work will be integrated in the safety assessment and the EIA, and what importance assessments of the biosphere will have for the siting of a final repository. Further, the Council felt that SKB should conduct sensitivity analyses of the modelling results and draw up a programme for modelling of the conditions associated with expected climate change due to the greenhouse effect.

Newfound knowledge since RD&D 2007

Since RD&D Programme 2007, knowledge of surface ecosystems has grown and the methodology for assessing radiological risks has been further developed. The biggest advances have been made in the work with site modelling of Forsmark and Laxemar-Simpevarp and the SR-Site safety assessment. Collection and analysis of site-specific data has generated new knowledge of surface ecosystems, which has been applied in numerical descriptions of the sites /26-3, 26-4/.

With this work, SKB has developed conceptual and numerical models that describe the ecosystems and transport of waterborne elements in surface ecosystems. Models of the extent of vegetation on land and in the sea and of surface-hydrological flows has been validated by comparisons with field data. The description of the sites has been supplemented with models of landscape evolution and ecosystem succession under different assumptions concerning future climate and land use. The methodology for calculating transport and accumulation of radionuclides in the landscape and assessing the risk to human health and the environment has also been developed. The calculations are largely based on existing knowledge of Forsmark and Laxemar-Simpevarp. The results obtained have been published in some 50 SKB reports and more than 30 scientific articles, and also presented at conferences, seminars and university courses.

Programme

Research and development during the coming programme period will build further on current activities and include studies to improve process understanding and development of calculation methodology. Sensitivity and uncertainty analyses of the calculations are being carried out. A great emphasis is placed on questions and uncertainties identified in current and future safety assessments. New efforts are planned to improve the description of aquatic and terrestrial ecosystems and the transition between them in an evolving landscape.

26.2 Basic principles for description and modelling of surface ecosystems

Since the end of the 1990s, a central part of SKB's research programme for surface ecosystems has been to identify, describe and quantify processes that are potentially important from a radiological perspective. A systems ecology approach is taken, where both biotic and abiotic processes in the ecosystems are taken into account. A safety assessment for a final repository is done for long time perspectives and varying environments, which often requires generalization of knowledge generated in the academic world.

Knowledge of the most important processes is updated continuously with current research findings and conclusions from SKB's own investigations. The results from the site investigations in Forsmark and Laxemar-Simpevarp have contributed greatly to a better understanding of important processes in surface ecosystems (see for example /26-3/ and /26-4/). Moreover, active knowledge feedback takes place from previous safety assessments, and the accumulated knowledge is then applied on the temporal and spatial scale and level of detail required in future safety assessments.

Each ecosystem is characterized by a large number of processes and complex interactions. Few processes play a quantitatively crucial role in a safety assessment, however. The numerical model used in the safety assessment can therefore be greatly simplified and adapted to a suitable temporal and spatial perspective. By using a systematic approach to identify the processes that are important for radiation dose to man and biota, the model can be simplified and the simplifications justified. One way to do this is to set up an interaction matrix /26-5/. In an interaction matrix, the system in question is divided into different elements, in the case of surface ecosystems both abiotic (e.g. Quaternary deposits and surface water) and biological components (e.g. primary producers and consumers). Interactions between the different elements are then identified, and these elements are assessed from the viewpoint of their potential importance for radioactive dose to man or the environment.

An interaction matrix for the biosphere was developed by SKB c. ten years ago /26-6/. This interaction matrix has comprised a tool in designing the site investigation programmes and ecosystem models. It is also used to ensure that all processes that have been identified as important from a dose perspective are included in the radionuclide transport models that are set up in the biosphere programme. These three components – site description, mechanistic ecosystem models and models of radionuclide transport – together comprise, along with the results of various research projects, the most important parts in the long-term iterative work of improving process understanding, see Figure 26-1.

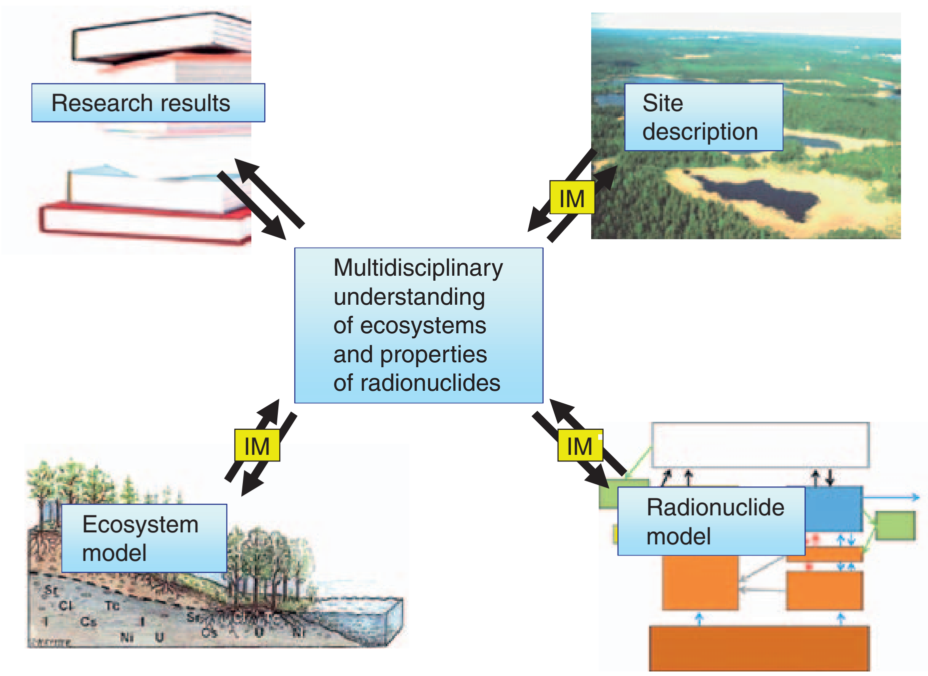


Figure 26-1. *Illustration of the long-term iterative work of identifying and improving the understanding of important processes in surface ecosystems. An initial version of the interaction matrix (IM) served as an important basis for the design of site investigation programmes, ecosystem models and models of radionuclide transport. Each of the different components generates new knowledge and new questions that are used to develop the other components.*

In its review of RD&D Programme 2007, SSI pointed out the lack of an integrated description of processes that are relevant for devising the models that are used to calculate doses in the safety assessment. In response, SKB has prepared ecosystem-specific interaction matrices for limnic, marine and terrestrial environments, based on the general interaction matrices for the biosphere. Preliminary versions of these matrices have been presented for the limnic and marine systems /26-7, 26-8/. In conjunction with the reporting prior to SR-Site, updated versions of ecosystem-specific interaction matrices will be presented, while definitions of the identified processes will be presented in a separate report.

In the comments on RD&D Programme 2007, the regulatory authorities also called for procedures for handling regulatory viewpoints as well as for version control of model codes and data. In connection with the execution of SR-Can, Subversion (http://svnbook.red-bean.com) was implemented, a version control system for report texts, input data, derived parameters, models and results. During the work with the SR-Site project, Subversion has been supplemented with Trac (http://trac.edgewall.org/), which is an issue and error tracking system. All regulatory viewpoints concerning the biosphere in conjunction with the reviews of SR-Can, RD&D Programme 2007 and SAR-08, along with SKB's handling of the matters, are documented in Trac in a traceable chain of measures.

26.3 Terrestrial ecosystems

SKB's description of the terrestrial ecosystems includes areas where the groundwater table is located beneath or close to the ground surface for a large part of the year. The terrestrial ecosystems thus span over many different biotopes, from well drained agricultural land, over drier and wetter forest types, to wetlands. Deep groundwater reaches the upper soil layers in the low-lying points of the landscape in particular. Radionuclides are mainly taken up in plants via roots and can, e.g. via primary production, be accumulated in biomass. For most elements, this biomass is the most important source of exposure of humans and herbivorous animals. Secondarily, there is also an accumulation of certain radionuclides in the upper part of the soil profile via root litter and litterfall. Accumulation of organic material, and of radionuclides associated with it, is greatest in wetlands where more or less anoxic conditions periodically exist, which inhibits decomposition and causes peat formation. Radionuclides can also be sorbed in wetlands when the groundwater runs through them.

Conclusions in RD&D 2007 and its review

RD&D Programme 2007 identified the need for a more thorough description of terrestrial biosphere processes, such as degradation and flux of organic carbon in the root zone, based on the site investigations. The programme also expressed an ambition regarding targeted studies of wetlands bordering on lakes, see Section 26.6. Key processes for transport and accumulation of radionuclides in terrestrial systems were to be identified by computer simulations (CoupModel, see below) and newfound knowledge was to be used to further develop methodology for calculations of activity concentration in terrestrial systems. SSI's review called for a clearer application of newfound knowledge, and a clear plan for the continued development of models that describe transport and accumulation of radionuclides in different types of wetlands. The Swedish National Council for Nuclear Waste said that more in-depth knowledge of agricultural land is needed, with particular reference to the cultivation of former accumulation bottoms in watercourses, lakes and seas, and use of wetlands.

Newfound knowledge since RD&D 2007

The investigations of recent years have led to a number of new insights regarding turnover and accumulation of organic material and distribution of elements in the terrestrial ecosystems in Forsmark and Laxemar-Simpevarp, see Figure 26-2. This has been summarized in /26-9/.

The quantity and turnover of live and dead biomass varies greatly between different types of forests and wetlands on the sites, and there can also be great variation from year to year /26-9/. It is above all variations in the water saturation of the soil that influence degradation and thereby the accumulation of organic matter in wetter environments. The calculations of carbon flux on the sites are based on field measurements of soil respiration /26-10, 26-11/, of production and degradation of tree litter /26-12/, and of the distribution and turnover of fine roots /26-13, 26-14, 26-15, 26-16/.

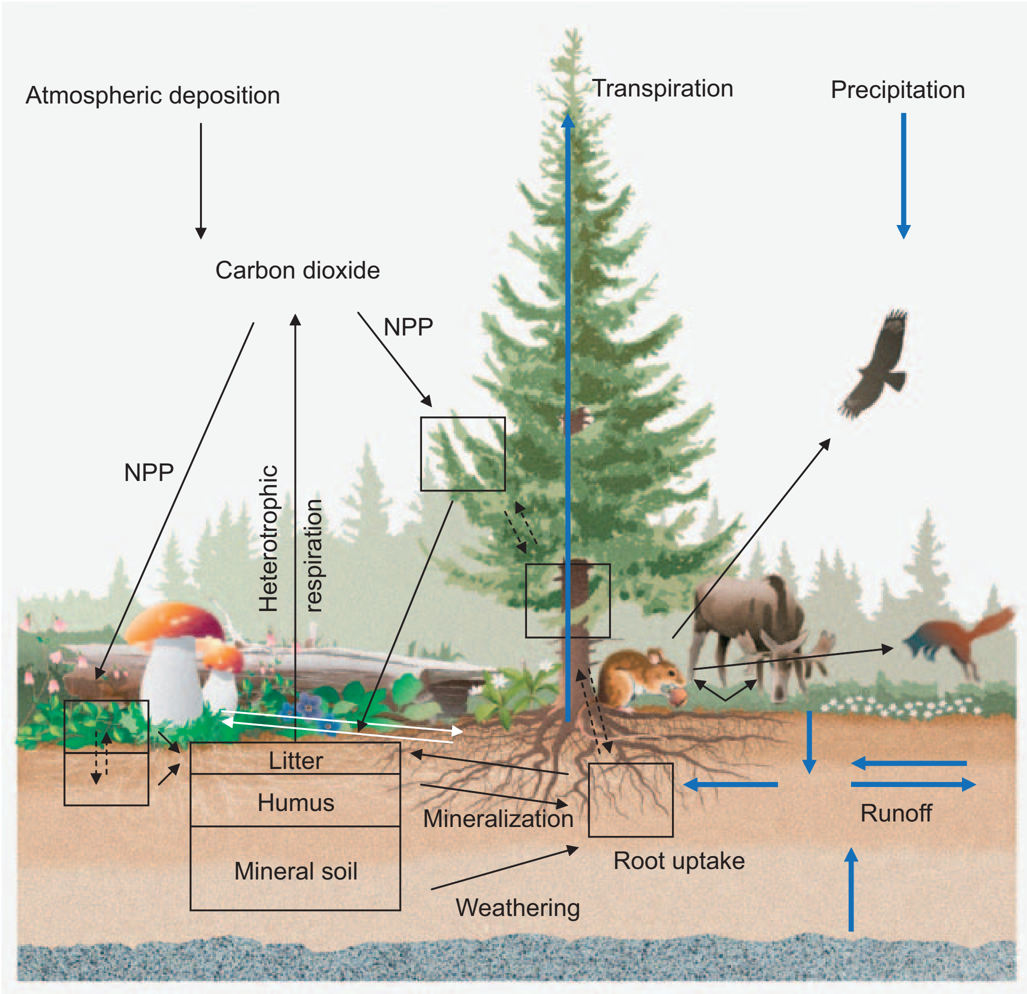


Figure 26-2. *Conceptual model of a terrestrial ecosystem. Black boxes symbolize pools of organic matter, while arrows symbolize element fluxes that are associated with carbon fluxes (black) or water fluxes (blue). The dashed arrows symbolize internal fluxes within plants. NPP = net primary production (figure modified from /26-9/).*

In a joint project with Lund University, a dynamic vegetation model (LPJ-GUESS) was used to describe pools and fluxes of carbon in different ecosystems (vegetation types) /26-9/. The modelling results agreed well with field data gathered during the site investigations. The carbon cycle was described for whole catchments by combining site data and results from computer simulations. The calculations showed that human use of natural resources affects both production and accumulation of organic carbon.

A large-scale survey of primary production in Forsmark and Laxemar-Simpevarp has been done as a part of the site investigations. In a separate joint project with Lund University, field data were combined with vegetation modelling and satellite information /26-17/ to prepare a geographic description of primary production /26-18/. The estimated primary production agreed on average well with field observations and has been used to estimate spatial variation.

Variations in vegetation development, biomass and primary production have been described for forest during free development (400 years) and for agricultural land (during 100 years) with the aid of computer simulations controlled with regional climate data from a 100-year period /26-9/. The forest achieved an equilibrium with respect to net primary production (NPP) after about 150 to 200 years. Production forests are generally kept at a high NPP by harvesting. The modelled variation in NPP for agricultural land was lower than that observed in the field. Similar computer simulations have been used to illustrate the vegetation in Forsmark under periglacial conditions and under conditions with a warmer climate than today /26-19/.

Populations of mammals and birds in Forsmark and Laxemar-Simpevarp have been monitored during the site investigations /26-20, 26-21, 26-22/, and annual hunting statistics have also been used for the moose populations /26-23, 26-24, 26-25/. Most populations have been stable between the survey occasions, but populations of wild boar, fox and squirrel have increased in all surveyed areas. In the site investigations, bioturbation by earthworms and ants was investigated at nineteen different sites in different vegetation types, from well drained pine forests to spruce swamps /26-26/. The sites exhibited great variation in population densities. According to the investigation, the calculated turnover rate of the humus layer exhibited a covariation with groundwater level and pH.

The distribution of a large number of elements in different forest ecosystems has been studied in a joint project with the University of Kalmar /26-27/. The distribution of a selection of elements in plants has been studied for the same ecosystems /26-9, 26-28, 26-29/, see Section 26.5.

Modelling studies have been used to identify key processes for accumulation of radionuclides in forest ecosystems. In a joint project between SLU and KTH, CoupModel, a mechanistic forest ecosystem model, has been supplemented with a module for following trace elements from a release to the groundwater /26-30/. According to the computer simulations, drier forests situated higher in the terrain exhibit a limited accumulation of released radionuclides. The same applies for swamp forests with a passive plant uptake of trace elements. Accumulation under these conditions takes place mainly in deeper soil layers and is controlled primarily by the soil's sorption capacity. Great accumulation in the humus layer in occurs in swamp forests and increases with factors that favour plant uptake (e.g. active uptake, deep root depth and low sorption).

Programme

Newfound knowledge in recent years confirms the importance of wetlands as recipients for radionuclides in the event of release from a final repository, see Section 26.6. In the continued programme we will strive for a deeper understanding of transport and accumulation of radionuclides in wetlands and in organogenic agricultural lands. Newfound knowledge will serve as a basis for updating SKB's methodology for calculating activity concentrations and exposure and for reducing uncertainties in the description of key factors in these calculations.

In the coming programme we plan to broaden the description of properties that characterize wetlands over a postulated succession gradient from coast to interior. The description will also be supplemented by modelling studies of uptake and accumulation processes in wetlands.

Ingestion of contaminated farm crops comprises an important source of human exposure to many radionuclides. The possibilities of utilizing wetlands for agricultural production will therefore be studied more closely. Important questions are how accumulated radionuclides are cycled when a wetland is drained and how long sustainable agriculture can be practiced on a drained wetland. Development of a mechanistic description of how radionuclides are taken up in crops is planned, see Section 26.5. In conjunction with this, SKB will conduct a literature review that will serve as a basis for planning and executing whatever supplementary field sampling and modelling work will be required.

26.4 Aquatic ecosystems

Aquatic ecosystems consist of running waters, lakes and seas. They are normally located in low-lying points in the landscape and thereby constitute potential discharge areas for deep groundwater that could be contaminated by radionuclides from a final repository. Furthermore, most transport of substances in the biosphere is mediated by water, which means that virtually all mobile substances in the terrestrial system will sooner or later end up in an aquatic system. Aquatic systems will therefore be central when it comes to assessing the effects of the potential release of radionuclides from a final repository. Radionuclides that reach an aquatic system can be taken up by organisms, bound in the sediments, emitted to the atmosphere or transported further downstream. Uptake of radionuclides in aquatic organisms comprises an exposure pathway for dose to man, since different kinds of aquatic organisms constitute food.

In many potential discharge areas, radionuclides from the final repository will pass a sediment layer. The permeability of the sediments affects the dispersion and dilution pattern, and due to adsorption processes different substances will be accumulated in the sediments. In the case of certain radionuclides, we can thus expect a higher concentration in sediments than in water, and sediments can constitute an important exposure pathway for aquatic organisms. In the short term, accumulation in sediments will probably reduce the outflow of radionuclides to the water mass and result in lower human exposure. In the long term, however, freeing of accumulated radionuclides due to resuspension, land uplift or land cultivation could result in elevated doses during a limited period.

Conclusions in RD&D 2007 and its review

In RD&D Programme 2007 it was noted that the state of knowledge regarding the dominant processes for accumulation and transport of different substances in aquatic environments is relatively good. It was said that the most important work remaining was to develop models and modelling tools that can handle the knowledge and to gather data from the sites. Prior to SR-Site, it was planned that the available knowledge from the site would be compiled in a report for lakes and streams and a report on the sea. Further development of dose models for lakes was planned, along with a special study of wetlands bordering on lakes with respect to interactions between lakes and surrounding wetlands due to water level variations in the lakes, see Section 26.6 for newfound knowledge.

In its review, SSI complained of the lack of concrete plans for further development of dose models for lakes and said this made it difficult to judge how well SKB's programme in this area responds to the regulatory authorities' comments on SR-Can.

Newfound knowledge since RD&D 2007

In order to provide a better understanding of important processes in the aquatic ecosystems, supplementary field investigations have been carried out within the site investigation programme since RD&D Programme 2007. Furthermore, models have been developed to describe important fluxes of organic carbon and a number of other substances. Knowledge of the aquatic systems in Forsmark and Laxemar-Simpevarp has been compiled in two reports, one for lakes and watercourses /26-82/ and one for the sea /26-92/.

The carbon flux models show that most of the Forsmark lakes are net autotrophic, i.e. primary production is greater than respiration, see Figure 26-3. In this way they differ from most other lakes in temperate areas, which are normally dominated by respiration and supplied with organic material from the surrounding catchment to a greater extent. Only a small fraction of the carbon that is fixed by primary producers in the lakes (7–10 percent) is carried further in the food webs to the top consumers. This means that most material that is fixed by primary producers circulates in the microbial food webs and is returned to the water mass or bound in organic sediments. The flow models show that a lot of material can be accumulated in the lake sediments. This material can later be returned to the food chain or be released when former lake bottoms are transformed into wetlands or agricultural land. The flow models also show the importance of adsorption/desorption to particles for radionuclide transport in aquatic systems. Substances that bind strongly to particles (such as thorium) are largely accumulated in sediments, while substances with low absorption to particles (such as iodine) are largely transported downstream with running water. Since sorption is an important process for the transport of certain substances, supplementary analyses have been done of the chemistry of suspended material /26-31/.

The spatial distribution of marine ecosystem components has been modelled in Forsmark and Laxemar-Simpevarp with the aid of geographic information systems (GIS) /26-92/. The dominant biomass on both sites consists of large benthic algae (macrophytes), which are mainly found in the more near-coastal basins. It is also in these basins that primary production is greatest. As in the case of lakes, only a small portion of the carbon that is fixed by marine primary producers is transported further in the food webs to the top consumers. In both Forsmark and Laxemar-Simpevarp, transport of different substances is dominated by the advective flows between sea basins, while accumulation in sediments is relatively low. The calculated biomass of the marine vegetation in the different sea basins in Laxemar-Simpevarp has been validated by field observations. The comparison shows relatively good agreement /26-32/.

Figure 26-3. The clear oligotrophic hardwater lakes that are typical of the Forsmark area today are relatively small and shallow, and the same applies to most of the lakes that will be formed in the future by land uplift. The main primary production in the lakes takes place on the bottoms, which often consists of a thick microbial mat (top right) or dense stands of stoneworts (bottom right).

Since sea water currents are of crucial importance for flows and accumulation of different substances during the marine period, a supplementary hydrodynamic model has been constructed for the SR-Site safety assessment to describe the water exchange in Öregrundsgrepen during the period 6500 BC to 9000 AD. The model is based on the oceanographic model that was previously developed by Engqvist and Andrejev /26-33/ for the Forsmark area, but uses MIKE 3 with a high flexible spatial resolution. The supplementary model describes how water exchange and flows in the coastal basins in Forsmark change over time as a function of the changing water depth as the shoreline is displaced in the model area. The modelling shows that water exchange and flows between basins decreases the shallower the model area becomes.

Within the framework of SR-Site, dispersion and uptake of elements in Öresundsgrepen has been simulated with a model with high spatial and temporal resolution. In the model, a detailed description of ecological processes is coupled with hydrodynamic flows (see above) in order to mechanically describe fluxes of carbon, nitrogen and phosphorus in three dimensions. The work is reported in conjunction with the reporting of SR-Site. The modelling tool that is used (ECOlab /DHI 2008/) has previously been used successfully to calculate the consequences of eutrophication in the Gulf of Finland /26-34/. By coupling a radionuclide module based on /26-35/ to the model, dispersion and uptake of six selected radionuclides has been calculated. In the simulations, concentration factors for phytoplankton were varied during the year and in space as a function of plankton biomass and radionuclide concentration in the water.

A literature compilation has also been made concerning a previously little-known ecosystem in deep groundwater /26-36/. It describes stygofauna, i.e. small animals such as nematodes and crustaceans that live in the deep groundwater present in the fractures in the bedrock. The report indicates that the probability of the occurrence of stygofauna in deeper groundwater under the conditions that prevail on the sites investigated by SKB is low, above all due to the low oxygen concentrations. The work has resulted in scientific articles /26-37/ as well as popular scientific articles /26-38, 26-39, 26-40/.

Programme

SKB has previously developed mechanistic ecosystem models with a detailed description of the food web in order to study transport and accumulation of radionuclides in an aquatic environment /26-35/. In these models, the flow of carbon is described as a function of primary production, consumption, respiration and degradation. Turnover and accumulation of other elements in the food web is described dynamically as a function of plant uptake, adsorption, consumption and excretion, with reference to the nutritional needs of the organisms and the properties of the elements. SKB will support the continued development of these models, with the goal of describing turnover and accumulation of radionuclides in organisms and dead organic matter in seas and lakes as well as in bordering wetlands. Model descriptions will be validated with existing data from Forsmark. The continued development of these models will take place in the form of a doctoral project at Stockholm University, and the work will mainly involve models for lakes.

SKB has previously developed a model that describes the succession when sea bays are isolated and form lakes, and the subsequent infilling of lakes to form wetlands /26-41/. The model, which is based on empirical relationships, has subsequently been further developed during the work with SR-Site, but it can be made more process-oriented by greater utilization of the knowledge of the evolution of individual objects in the Forsmark area. SKB therefore plans a development of the succession model.

26.5 Biogeochemistry

Studies of element concentrations in the superficial system, i.e. biogeochemistry, can provide basic information on the occurrence of the elements on different spatial scales, from large-scale distribution in the landscape to detailed descriptions of how the elements are distributed between organisms and the environment in an ecosystem. The stoichiometric relations between elements can provide good guidance on their origin and the processes that control the large-scale distribution patterns in nature. On a more detailed level, the same information can be used to describe how elements are taken up and enriched in terrestrial and aquatic ecosystems.

Element distributions in surface ecosystems can also be used to understand how radionuclides are transported and accumulated in the environment. Stable isotopes of many radionuclides occur naturally in the environment (e.g. C, I, Th, Ra, U, Ni). In the case of other radionuclides, the properties of a similar stable element (K as compared with Cs-137), or the properties of an entire group of elements (e.g. rare earth metals), can provide guidance on quantitatively important transport and accumulation processes.

Element ratios are used in most exposure models to calculate radionuclide concentrations in the environment and uptake in organisms. *Distribution, or partition, coefficients* (K_d) describe the distribution of elements between solid and aqueous phases (e.g. in deposits or on particles in surface water systems), while *concentration factors* or bioaccumulation factors reflect a biological uptake that is proportional to the concentration in the environment or the food. Element ratios summarize a variety of chemical, physical and biological processes and are assumed to reflect a state of equilibrium in the environment.

Conclusions in RD&D 2007 and its review

In its review of RD&D Programme 2007, SSI said that when element ratios for a specific site are to be estimated from general literature compilations, the uncertainty is often great, since it is difficult or impossible to relate literature data to site-specific biological, chemical and physical conditions. SSI also pointed out that it was unclear how SKB has taken into account the physical and chemical properties of radionuclides in their evaluation of flow paths and distribution in different ecosystems.

Newfound knowledge since RD&D 2007

During the site investigation programme, SKB has characterized element concentrations for deposits, pore water and surface water, as well as for a large number of terrestrial and aquatic organisms, see below. The resulting database is unique, both in the quantity of element information that has been collected from two distinct geographic areas and in the systematic and synchronized sampling that lies behind the site investigations. Newfound biochemical knowledge will be summarized and synthesized in the reporting in SR-Site. The goal in SKB's ongoing SR-Site safety assessment is to use site-specific estimates of element concentrations to calculate concentration factors.

The hydrochemistry at Forsmark and Laxemar-Simpevarp has been used to identify the original sources that affect the chemical composition of the water, as well as to distinguish different types of groundwater /26-42, 26-43/, see Figure 26-4. Data have also been used for hydrochemical mass balance calculations, see Section 26.6. Element concentrations in environmental samples and organisms from several different ecosystems in Forsmark and Laxemar-Simpevarp have been characterized /26-44, 26-45/ and the measurements have been supplemented with concentrations of a number of radionuclides (Pu-238, Pu-239, Pu-240, Pu-242, Ra-226, Th-229, Th-230, Th-232, U-233, U-234, U-235, U-236, U-238, Tc-99 and I-129) /26-46, 26-47/. Sediments, pore water, suspended material and filtered water have also been analyzed, and K_d values have been estimated for aquatic environments /26-31/.

The chemical composition of abiotic and biotic material from a sea bay in the Forsmark area has been studied in detail /26-48/. The distribution of 48 elements in large groups of organisms (phytoplankton, zooplankton, benthic microalgae, macroalgae, aquatic vascular plants and several different types of benthic organisms and fish), as well as in dissolved and particulate material in the water and in the sediment, was investigated. There is considerable variation in element concentration between different organisms, and between organisms and the environment. Differences in uptake and bioaccumulation were explained by the biological function of the elements, as well as the differences between the organisms in terms of habitat, ecosystem function, trophic level and morphology. It is interesting to note that when element concentrations are specified per unit of carbon, the results show that few elements are accumulated in the food chain.

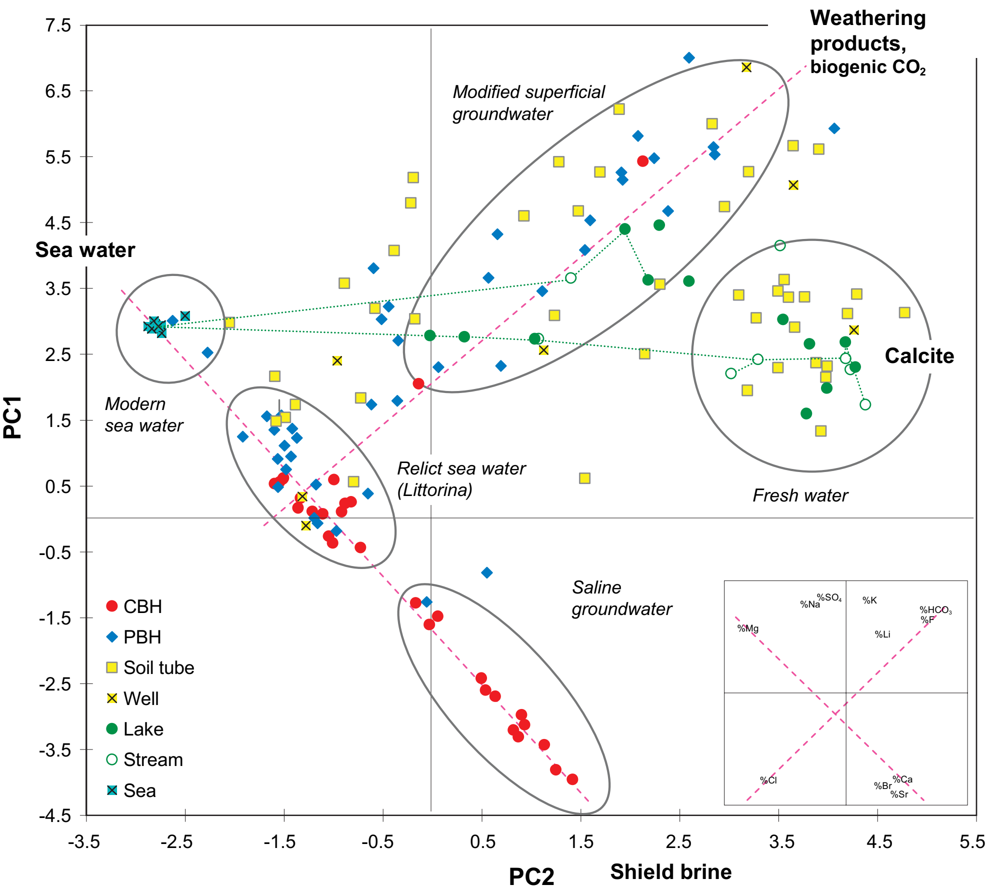


Figure 26-4. *Chemical composition of water samples from Forsmark. Five water types (ellipses) and four ion sources (bold text) have been identified by means of the covariation in water concentration for the main constituents. The axes in the figure (PC1 and PC2) are Principal Components which together summarize 80 percent of the total variation of water concentrations. CBH = Cored borehole, PBH = Percussion borehole. The figure is modified from /26-42/.*

The distribution of a large number of elements in different forest ecosystems has been studied in a joint project with the University of Kalmar (now Linnaeus University) /26-27/. By studying covariation in field samples, two main groups of elements were identified: 1) elements that were mainly found in the mineral soil (for example metals and actinides) and only reach low concentrations in organic matter and living tissue; 2) elements with relatively high concentrations in plants and humus, which mainly consist of plant nutrients. The distribution in plants of the stable forms of phosphorus, iodine, thorium and uranium has been studied for the same ecosystems /26-9, 26-28, 26-29/. As expected, the heavier elements (thorium and uranium) were mainly found in fine roots, while the plant nutrients iodine and phosphorus were also found in above-ground parts. No clear differences in the plants' allocation of the different elements could be distinguished between wetter and drier types of land.

The chemistry in soil and pore water has been analyzed in samples from a number of terrestrial ecosystems that are representative of present-day conditions in Forsmark and Laxemar-Simpevarp. Site-specific K_d values have been calculated and compared with a large body of data from Canada /26-49/. The estimates were normally within the same interval, with the exception of the K_d values for Cl, I and Se. K_d was also related to chemical and physical properties in the soils with the aid of regression analysis. The results showed that pH, clay content and organic matter content were the factors that best explained the variation in K_d between examined samples. Site-specific K_d values for sediments and suspended material in lakes and sea have also been calculated /26-49/ based on site data /26-31, 26-48/. For these samples, the type of solid phases (i.e. whether the sample contained sediment or suspended material) and the type of ecosystem (marine or limnic) affects the K_d values for many elements, while the sediment depth at which the sample was taken was only of importance for a few elements.

In SR-Site, site data have been used wherever possible to calculate concentration factors. SKB has also updated previous literature compilations /26-50/ with up-to-date data, in part from the IAEA's database /26-51/. Site and literature data have been combined with the aid of the statistical tool Babar, which was developed with support from SKB, Posiva, NRPA (Norwegian Radiation Protection Authority) and EDF (Electricité de France). This means that all available information is used to estimate concentration factors, but that data from the site has weighed more heavily than data from other places.

In addition to sampling at the sites, laboratory measurements have been made of the sorption properties of soil samples from the Laxemar area. Sorption was studied for seven elements (I, Cs, Sr, Ni, Eu, U and Np) in till, sand, clay, clay-gyttja, gyttja and peat /26-52. An evaluation of the results is currently in progress (spring 2010). An important step in the evaluation is to relate measured K_d values to chemical and physical properties of the soils in question /26-53, 26-54/.

In parallel with the development of how site-specific information is used to estimate concentration factors and K_d values, we are conducting long-term research to gain a better process understanding of retention and biological uptake. One example is the ongoing doctoral project related to the Krycklan catchment study, another is the process-based modelling of radionuclide retention that has been developed in a series of papers /26-55, 26-56, 26-57, 26-58/, see Section 26.6. SKB has also used mechanistic descriptions in modelling studies of the biological processes that control uptake and accumulation, among other things to study spatial and seasonal variation of element concentrations in a marine food chain, see Section 26.4. Research on surface system chemistry that is being conducted in cooperation with Linnaeus University, in association with Nova FoU, is described in Section 17.3.2.

Programme

The collection and compilation of information used to describe element concentrations on the investigation sites is unique. The continued work in the programme will mainly consist of processing collected data in order to obtain a better understanding and description of retention and biological uptake on different spatial scales.

SKB intends to improve the models of radionuclide uptake in plants, in both aquatic and terrestrial ecosystems. For this we will differentiate between active uptake of nutrient salts (and analogous radionuclides) and passive uptake of trace elements (transpiration, diffusion, adsorption), while also taking into account the bioavailability of substances that are taken up (see for example /26-59/).

In order to gain a better understanding of the causes of the variation of measured concentration factors, supplementary chemical analyses of samples from the Forsmark area will be done. The process-based modelling of radionuclide retention in the soil layers will also continue, see Section 26.6.

26.6 Hydrology and transport

This section deals with hydrology and solute transport in surface ecosystems, which means that the presentation is essentially limited to water flows and transport processes in the soil layers, on the ground surface and in various types of surface water. Even though many of the calculation models that are discussed include parts of the underlying rock, the problems that are taken up are primarily associated with processes in the soil layers and associated water systems. The research and development needs that are dealt with in the following thus concern the part of the system that lies on top of the rock surface and modelling that includes this volume. Equivalent descriptions of hydrology and transport in the rock are found in Chapter 25.

Transport modelling is a central component in both dose modelling and ecosystem modelling. These types of modelling are described separately in other parts of this chapter. As far as transport is concerned, we will concentrate in this section on modelling done to describe the investigated areas and to support the safety assessment with process understanding and parameter values. With regard to hydrological modelling, we discuss activities associated with site description as well as safety assessment and environmental impact assessment (EIA).

Conclusions in RD&D 2007 and its review

In the review of RD&D Programme 2007, viewpoints were offered regarding the use of site data in the safety assessment, the description of the transition between soil and rock, and the absence or representation of certain processes in our models. Specific problems that were mentioned concerned deficiencies in the description of the transport and retention in soil layers and sediments, the absence of studies of the size of the contaminated area, and the need to improve the description of the connection between soil layer/superficial rock and the deep rock.

Newfound knowledge since RD&D 2007

Most developments in the modelling of hydrology and different types of solute transport in surface ecosystems in recent years have taken place within the framework of site descriptive modelling and safety assessment. The site descriptive modelling has been concluded with the publication of the model version SDM-Site /26-3, 26-4/ and the results are being used in the ongoing safety assessment and EIA work. During the period a number of papers describing modelling methodology and results have been published in scientific journals and in conjunction with conferences /26-60, 26-61, 26-62/.

Great efforts have been made to develop surface hydrological models for description of surface waters and near-surface groundwaters in SKB's investigation areas in Forsmark and Laxemar-Simpevarp. These models have then been used for modelling of advective (water-borne) transport in the form of particle tracking and in advective-dispersive transport modelling (called AD modelling), where dispersion due to velocity variations and diffusion are also taken into consideration. The reporting of hydrology and transport modelling in SDM-Site Forsmark consists of an evaluating and summarizing main Report /26-63/ with appurtenant background reports describing data and data evaluation /26-64/, numerical flow modelling /26-65/ and transport modelling /26-66/. The reporting of SDM-Site Laxemar has a similar structure and includes a main report /26-67/, a data and data evaluation report /26-68/ and a report that describes the numerical modelling of flow and transport /26-69/.

The hydrological modelling makes use of the calculation tool MIKE SHE (see /26-65/), which can be used to model saturated and unsaturated flow in soil layers and rock, surface water flows on the ground surface and in the surface water system, water uptake and evapotranspiration in plants, and water exchanges between these hydrological subsystems. Important results from the site modelling include components of the water balance, distribution and changes in time and space of recharge and discharge areas, water exchanges between groundwaters and different types of surface waters, and site-specific values of hydrogeological parameters (mainly the hydraulic conductivities of the different soil materials). The calculation results are supported by a large body of data from hydraulic tests in groundwater monitoring wells, flow measurements in streams and water level measurements in groundwater and surface water, which have been used for comparisons between measurements and calculation results in model calibrations and sensitivity analyses.

The flow paths for outflowing groundwater from possible repository volumes in the rock are important for the safety assessment's biosphere modelling. Since discharge often takes place in lakes and wetlands, an understanding of the connection between surface water and groundwater in the saturated and unsaturated zones in the soil in such areas is of particular importance. The conditions around lakes and wetlands have therefore been specially studied in both data evaluations and numerical modelling (Figure 26-5), see summarizing descriptions of Forsmark and Laxemar-Simpevarp in /26-63/ and /26-67/.

The Forsmark area in particular contains many lakes and wetlands that are important both for the analysis of ecological consequences during the construction and operating phases and in the safety assessment of long-term radiation risks. Modelling has shown e.g. that the shore zones of the lakes play an important role in the water exchange between groundwater and surface water and that conditions beneath the lakes are periodically affected by evaporation processes in surrounding land areas. The lakes can occasionally switch from being discharge areas to being recharge areas for groundwater. Detailed studies of specific hydrological objects are being done in the ongoing safety assessment in order to develop and calculate parameter values for the models for the landscape objects in Forsmark, see further Section 26.9.

The coordinated evaluation of measured surface water levels and groundwater levels in soil and rock has served as a basis for an integrated surface-hydrological and hydrogeological model, see /26-63/. Together with other results from the site modelling, this model has served as a basis for continued surface-hydrological modelling in SR-Site, for example detailed studies of flow paths and water balances in current and future land areas, and modelling of the hydrology under possible future climatic conditions, including permafrost.

In order to support the EIA work, SKB has investigated the effects of an open repository on drawdown (lowering of the groundwater table) and surface water levels /26-70, 26-71/. Surface-hydrological modelling has been used to delimit the influence areas and to quantify ecological and other consequences of the drawdown. The most important development within this area is the application of the modelling results in the consequence analysis. The final results will be presented in the EIS with associated background reports.

Besides the aforementioned particle tracking and AD modelling, SKB has also carried out transport modelling with a focus on other processes. This modelling has revealed new knowledge about the turnover of a number of nuclides in groundwater and surface water and about the potential for retardation of radionuclides in the soil layers.

Element transport within catchments has been studied with hydrochemical mass-balance models /26-42, 26-43/. Mass balances were calculated for macroelements from water concentrations and flows on the sites. The calculations provided a good idea of the origin of elements, their main transport pathways and their accumulation in the landscape. We have also studied the hydrodynamics and marine transport of carbon, nutrients and radionuclides in dissolved and particulate phase, based on site data from Forsmark, see Section 26.4.

Process-based modelling of radionuclide retention in the soil layers has been investigated in a series of studies presented both in SKB reports /26-55, 26-56/ and in scientific papers /26-57, 26-58/. Potential retention processes and their effects on transport of selected radionuclides have been identified via a thorough literature review. Further, the retention of caesium, strontium, uranium and radium has been calculated. Retention modelling of other nuclides is being done within the framework of the ongoing safety assessment.

Radionuclide transport in near-surface systems is also being studied in ongoing SKB-funded research projects in connection with the Krycklan catchment study. This research began with analysis of data from a wetland in Laxemar-Simpevarp /26-72/ and is based on a large body of hydrological and chemical data (radionuclides and numerous other substances) collected in the Krycklan area in Västerbotten. The main purpose of the study is to describe transport and fixation of radionuclides around streams and in wetlands.

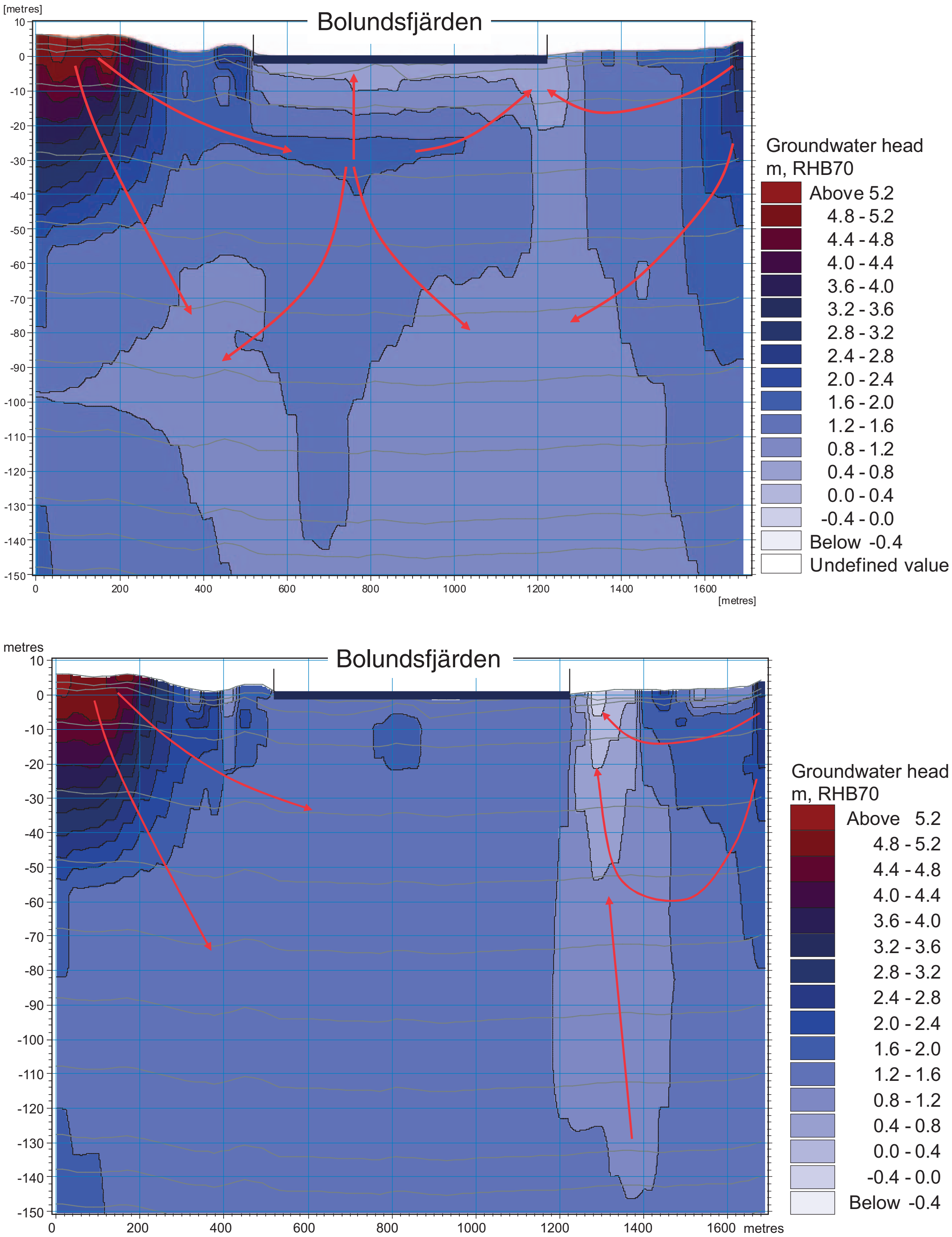


Figure 26-5. *Simulated groundwater heads in an East-West profile over Bolundsfjärden during a wet period (April 2006, top) and a dry period (August 2006, bottom) /26-65/.*

As an example of research work based on SKB's site data, it can be mentioned that measured time series from Forsmark were used to investigate calibration methodology and the value of additional data for reducing uncertainties /26-73/. Data from Forsmark and Laxemar-Simpevarp have also comprised a part of the basis for a series of studies of hydrology and transport processes where researchers at Stockholm University have investigated the distribution of the runoff among different outlets from near-coastal catchments, the importance of flow and degradation processes for the load on the downstream recipient, the contribution from catchments that are normally not monitored to the Baltic Sea as a whole, and method development regarding risk assessments /26-74, 26-75, 26-76, 26-77/.

Programme

Research and development during the coming programme period will largely build further on current activities. In addition, supplementary studies are planned of issues identified in the safety assessment. The plans for the coming period include both studies aimed at process understanding and a continued development of calculation tools.

The SKB-funded research in the Krycklan catchment study was recently extended. Important issues being studied are the link between organic matter and radionuclides in conjunction with transport to and within potential groundwater discharge areas (mainly streams and mires). The work within the ongoing doctoral project will be focused on completing and publishing the results of a number of subsidiary studies. An important part of the continued work is to link newfound knowledge from the Krycklan catchment study to SKB's investigation areas, i.e. to evaluate what results and conclusions might also be valid in Forsmark and Laxemar-Simpevarp.

Modelling is an essential part of such a generalization, and modelling resources have been made available to the Krycklan catchment study via a recently initiated post-doctoral project. Models with different degrees of complexity (K_d-based and mechanistic models) being developed in the Krycklan catchment study will be tested with independent data from Forsmark and Laxemar-Simpevarp.

The process-based modelling of advective-reactive radionuclide transport will continue and be developed to include more parts of the surface system. So far this modelling has only been carried out for the water-saturated part of the soil layers. The continued work will also include the soil's unsaturated zone and key processes such as uptake in plants. The intention is that this work should be pursued within an international cooperation project.

The work with mass balances as a way to describe and predict the distribution of elements in the landscape will continue. The number of elements will be increased and atmospheric deposition and sediment concentrations will be included in the calculations, which will be scaled up to landscape level. In order to improve the body of data, deposition data (precipitation chemistry) from the Forsmark area will be augmented during 2010.

SKB will continue its work of characterizing the size and location of discharge areas and the processes that drive transport there. We intend to study specific hydrological objects that represent all succession stages in landscape evolution, from marine basins via eutrophic lakes to wetlands and watercourses.

We plan a further development of the calculation tools used for modelling of hydrology (water flows) and advective transport of solutes (particle tracking and AD modelling). This further development includes improving the possibilities of describing the changeovers between frozen and unfrozen conditions in the uppermost part of the soil in permafrost environments. The plans also include supporting the development of tools with the capability to handle transport in different soil layers (saturated and unsaturated zone) and surface water, in order to facilitate the continued work of characterizing discharge areas and transport there. SKB currently uses a commercial tool (MIKE SHE) to model hydrology and advective transport and will support the development of the tool in cooperation with the software supplier.

During the coming programme period, SKB will continue to support university researchers who use data from SKB's investigation areas. A joint project has been under way for several years with Stockholm University, and new modelling activities were recently initiated at KTH.

26.7 Effects of long-term variations

Our understanding of how elements are transported and accumulated in surface ecosystems is based above all on descriptions of phenomena and processes that can be observed today in Forsmark and Laxemar-Simpevarp. Many processes are climate-dependent and have varied in intensity as the climate has varied. Other processes are secondarily related to the climate and changes in the climate. Land uplift, which greatly influences the evolution of the landscape in near-coastal areas such as Forsmark, is for example a return to a new state of equilibrium after depression of the bedrock by ice sheets. The salinity of the Baltic Sea is another example of how shoreline displacement interacts with runoff, both of which are affected by the climate. To assess the long-term safety of a repository, it is therefore necessary to take into account long-term variations caused by changes in the climate.

Climate-driven process domains /26-78/ (or simply climate domains) is a concept introduced by SKB to classify environmental conditions of importance for the performance and safety of a repository. Three climate domains have been identified as relevant for the SR-Site safety assessment, namely *periglacial* conditions with a cool climate and deep permafrost, *glacial* conditions when the site is covered by an ice sheet, and *temperate* conditions, which include today's climate as well as a considerably warmer climate. We have exemplified conditions during the different climate domains in order to judge how important processes and phenomena are affected by long-term variations in the climate. The three climate domains are also used in the work of devising possible future climate evolutions, including a so-called reference evolution based on a repetition of the conditions that prevailed during the last glacial cycle.

Conclusions in RD&D 2007 and its review

RD&D Programme 2007 described plans for a compilation of the function of the surface systems during a permafrost domain. The Swedish National Council for Nuclear Waste expressed a desire in its review for a better description of the evolution of the biosphere in climates that are warmer than today.

Newfound knowledge since RD&D 2007

Climatic conditions (e.g. temperature, precipitation, surface water balance and length of the vegetation period) during the different climate domains have been described within the framework of SR-Site /26-19/ (see Chapter 19). Based on this description and the relevant scientific literature, we have judged how the ecosystems are affected by variations in climate, and what effects this may have for factors that influence long-term radiological safety for humans and other organisms. Surface hydrology and transport have been modelled in the ongoing SR-Site safety assessment for possible future climates that differ from today's; both colder (permafrost) and wetter climates have been studied. The effect of a colder climate on eutrophication of lakes and formation of taliks (during permafrost, areas of unfrozen soil under e.g. lakes) has been modelled. The results have been used as a basis for describing surface ecosystems during possible future climate change in the ongoing safety assessment and will be reported in it.

A new model for shoreline displacement during the entire reference evolution has also been developed in conjunction with the SR-Site safety assessment /26-79/. The new model consists of a combination of the previous model, which described shoreline displacement from the last deglaciation until today /26-80/, and a new model for future shoreline displacement /26-81/.

SKB has initiated field studies of surface ecosystems under periglacial conditions on Greenland in the SKB-supported joint Greenland Analogue Project (GAP), see Section 19.6. The GAP is above all focused on glacial hydrology and geochemical conditions in and near an ice sheet. In the summer of 2008, SKB conducted initial field work for the purpose of exploring the feasibility of a broadened investigation programme that also includes ecology, surface hydrology and soil processes, see Figure 26-6.

Programme

During the coming programme period we will describe periglacial environments that exist today. Field investigations on Greenland and modelling of areas with potential groundwater discharge will be used to gain a better understanding of how processes and phenomena that drive transport and accumulation of radionuclides are affected by a colder climate. The results of the GAP project will also be used to describe how the connection between surface ecosystems and the rock is affected by periglacial conditions.

A trend towards a climate that is considerably warmer than today may entail changes for both the environment and human beings. Several of the processes that are important for transport and accumulation of radionuclides in surface ecosystems may be affected by e.g. water flux and ice formation time. SKB will therefore continue to keep track of new knowledge in the area, while conducting model studies to investigate how changes in e.g. temperature, hydrology and land use associated with a warmer climate can affect long-term radiological safety for humans and other organisms.

Figure 26-6. Conductivity is measured as a part of an attempt to characterize the origin of the surface water. The work was done during a field week with pilot studies in a periglacial environment near Kangerlussuaq, Greenland, in the summer of 2008.

26.8 Landscape evolution and deposits

As a basis for safety assessments, SKB has worked with a historical description of the phenomena and processes that have driven the evolution of the landscape in Forsmark and Laxemar-Simpevarp /26-82/. The historical description has been combined with an understanding of the present-day appearance and function of the landscape /26-83, 26-84/. Together, these descriptions constitute the basis for describing a probable evolution of the landscape under different assumptions regarding future climate and shoreline evolution.

During the site investigations and subsequent site modelling and in conjunction with the SR-Site safety assessment, we have described the geometries of the landscape, from the top surface of the bedrock via deeper deposits of till and clay to soils and superficial sediments. The physical and chemical properties of the deposits have been characterized, and the resulting site model is the point of departure for describing a changing landscape.

Climate variation and shoreline displacement are the principal phenomena that drive landscape evolution in Forsmark and Laxemar-Simpevarp, see Section 26.7 above. Natural climate change leads to a slowly repeated glacial cycle, with a succession from ice-age marine conditions to a temperate climate where the landscape is ultimately dominated by terrestrial ecosystems /26-82/. On top of these large-scale and slow changes, the landscape is affected by the deposition and reworking of sediments. Lakes get shallower and silt up. The chemical properties of soil and water change, along with the species composition of plants and animals as ecosystems succeeds one another.

Hydrological discharge areas are located in the topographical low-lying points in the landscape, for example around lakes, streams and wetlands, see Section 26.6. The potential for transport and accumulation of radionuclides in these areas is determined by geometries and properties in the local and regional catchments. The large-scale evolution of the landscape determines where and when groundwater can flow up to the surface. The natural succession of discharge areas and their potential use by future human communities for food production and habitat is a premise for calculating radiological risk to human beings and the environment, see Section 26.9.

Conclusions in RD&D 2007 and its review

The objective in RD&D Programme 2007 was to further develop the methodology for the dose calculations, which also takes into account the radiological impact on the environment so that it reflects current knowledge from the sites, for example with respect to landscape evolution and human use of natural resources. In its review of the biosphere programme, SSI said that SKB's development of an integrated landscape model that describes several ecosystems in succession is a step forward in the development of the safety assessment, but that the connection between the landscape model and process-based calculations of transport and accumulation was unclear. In its review of the safety analysis report for SFR-1 /26-85/, SSM has stressed the importance of a landscape description that takes into account the natural transition between ecosystems /26-86/. A review of issues related to the choice of size of biosphere objects was also considered urgent.

Newfound knowledge since RD&D 2007

During the programme period, SKB has developed conceptual and numerical models that describe the historic evolution of the landscape in the investigation areas up until today. The description includes geometric models of topography /26-79, 26-87/ and regolith depth /26-88, 26-89/, characterization of current Quaternary deposits and their physical and chemical properties /26-90, 26-91/, a conceptual model of the succession of the ecosystems in the regions, and a historical description of population and land use at the sites up to the present. The work has been summarized in site descriptions of surface ecosystems for Forsmark and Laxemar-Simpevarp /26-3, 26-4/.

As a part of the SR-Site safety assessment, the understanding gained from the historical evolution of the landscape has been used to formulate possible future descriptions of the landscapes in Forsmark and Laxemar-Simpevarp. These descriptions are based on a new model of shoreline evolution /26-79/ and cover the entire current interglacial (with periods of permafrost) and climate variants that occur in connection with global warming. A final report of the results of this work will be made within the framework of SR-Site. Process understanding from the site has been translated into numerical models that describe the hydrodynamics in the sea, see Section 26.4, and the future accumulation and erosion of sediments in the sea and on land /26-92/ for an entire interglacial, see Figure 26-7. The hydrology associated with future landscapes and climates has been linked to this, see Section 26.6. The result is a spatial description of how the landscape evolves continuously from sea to land, where sea bays are silted up and pinched off to form lakes, and lakes are filled in to form wetlands. Different variants of land use have then been applied on this landscape, from a pristine landscape used for hunting and fishing to a purely agricultural landscape where all suitable land is drained and tilled for cultivation.

Future areas where groundwater discharge from a repository is conceivable constitute the geographic unit used to make assessments of radiological risk: a biosphere object. The properties of the object (including the size of the catchment) and its evolution directly drive the modelling of transport and accumulation of radionuclides on which the safety assessment is based, see Section 26.9.

Programme

During the coming programme period, SKB will deepen its understanding of transport and accumulation of radionuclides on a landscape level and further develop the methodology for describing the landscape and its evolution. Research and development will mainly take place in cooperation with the universities and organizations that have previously participated in the work. The emphasis in the development work will be on two aspects: conceptual understanding, and improving the description of processes and parameters that are of crucial importance for safety assessments.

The development of the conceptual understanding of how radionuclides are transported and accumulated in a changing landscape will be based on the results of simulation studies and empirical results of planned studies of mass balances on a landscape level, see Section 26.6. The knowledge will be synthesized in a conceptual model of the landscape that will include a number of conceivable types of land use climate.

The sensitivity and uncertainty analysis within the SR-Site safety assessment will identify phenomena and processes that have a bearing on the description of landscape evolution and long-term safety. Preliminary analyses show that the rate of eutrophication, which is an important factor in landscape evolution, is subject to uncertainties. Knowledge from the planned biosphere programme will therefore constitute an important basis for refining models that describe the ecosystem succession from

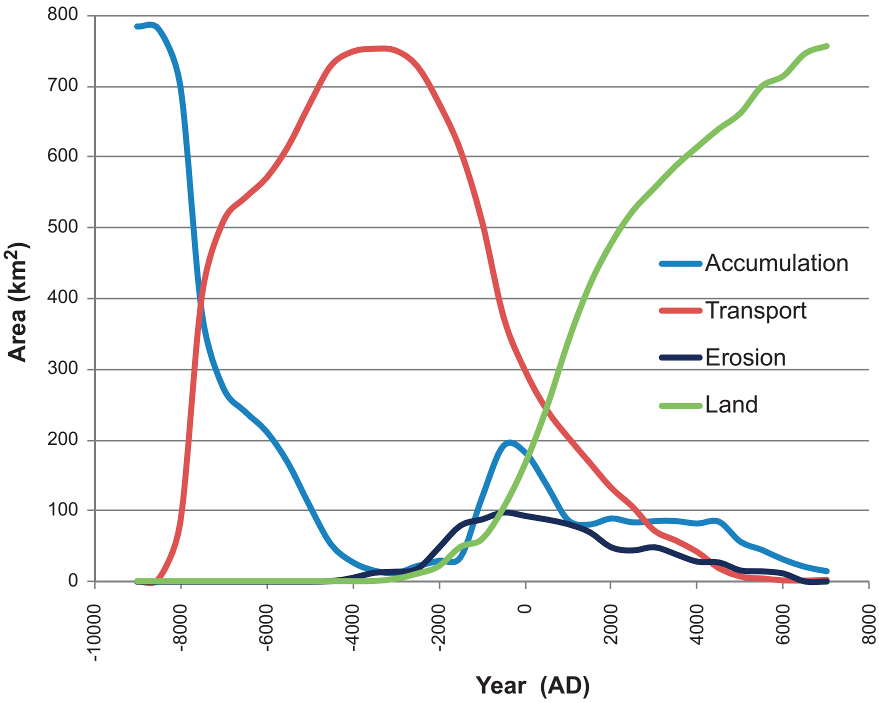


Figure 26-7. *Change in extent of sea bottom types and land in the Forsmark area. The change is based on today's description of topography, water depth and soil depth, plus models for sedimentation and shoreline displacement. The figure is taken from /26-92/.*

sea bay to wetland on the landscape level and how this succession is affected by climate change, see Sections 26.3, 26.4 and 26.7.

The calculated radiation dose to future inhabitants is based on assumptions and simplifications regarding how people can reside and use natural resources in the landscape. We will continue to work to gather knowledge about historical and contemporary self-sufficient cultures and human nutritional needs to support the description off future human habits. Realistic assumptions about people in the future will be based on a landscape perspective.

26.9 Radionuclide modelling

The research programme for surface ecosystems is aimed at furnishing data for a credible assessment of the radiological risks for humans and other organisms associated with the final disposal of spent nuclear fuel. A central part of the programme is therefore using our understanding of processes and phenomena to simplify reality so that transport and accumulation of radionuclides in the environment can be described with numerical models. The concentration of radionuclides in foodstuffs and drinking water is in turn estimated from the distribution of the radionuclides in the environment. In order to calculate exposure for a representative individual, concentrations of radionuclides in the environment and food are combined with knowledge of radiation effects on the body (dose conversion factors) and assumptions concerning the habits of the future inhabitants.

Over the years, the representation of the biosphere in SKB's safety assessments has evolved from general and simplified descriptions of individual ecosystems to models that describe a number of coupled recipients that evolve dynamically in the landscape as a function of land uplift and shoreline displacement, see Section 26.8. In today's dose model, radionuclide transport is determined by detailed knowledge of hydrological flows, and accumulation takes place by adsorption to suspended particulate material, sediments and soil particles. The model also describes how radionuclides are bound in organic matter via plant uptake and how they are subsequently cycled via consumption or degradation. An accumulation of radionuclides with long half-lives in organic matter is thus possible in environments with a long retention time, such as in the soil's humus layer or in wetlands.

Conclusions in RD&D 2007 and its review

The objective in RD&D Programme 2007 was to further develop the methodology for exposure calculations so that they can be used to assess risk to both human health and the environment. The goal was a methodology that can be applied to all important radionuclides and reflect current knowledge from the sites, for example with respect to landscape evolution, key factors for transport and accumulation and human use of natural resources. There was an ambition to harmonize the representation of processes in the various ecosystems and to replace empirical transfer factors between the environment and human food with a mechanistic description of plant uptake and bioaccumulation. Furthermore, a comprehensive sensitivity analysis of the modelling results was planned.

In its review of the biosphere programme, the Swedish National Council for Nuclear Waste felt that the trend towards a more mechanistic representation of processes (for example primary production, plant uptake and degradation) was an important complement to previously used models, and SSI took a positive view of Pandora and the ability of the tool to handle process understanding. Further, SSI felt that the integrated landscape model, which describes several ecosystems in succession, was a step forward in the development of the safety assessment, but contended at the same time that the connection between the landscape model and process-based calculations of transport and accumulation needs to be clarified. SSI further felt that the methodology used in SR-Can gave an effect of dilution in the calculations. Both SSI and the Swedish National Council for Nuclear Waste considered it necessary for SKB's assessment of the risks of a possible release to also include the consequences for the environment, and that calculations of activity concentrations should be provided with uncertainty estimates. The Council was also of the opinion that SKB should assess the chemical toxicity of a final repository by applying its models for transport and accumulation in the biosphere to stable substances such as heavy metals.

Newfound knowledge since RD&D 2007

In order to estimate future effects on the biosphere of a hypothetical release from SFR 1, the turnover and migration of radionuclides in surface ecosystems was modelled /26-85/. The modelling comprised a part of the safety analysis report for SFR 1 /26-93/.

Three snapshots of the landscape evolution in Forsmark were used for the calculations. Each landscape configuration consisted of a number of coupled objects (landscape objects) that reflected either an aquatic (coast of lake) or a terrestrial ecosystem (wetland and agricultural land). The representation of the landscape and radionuclide transport largely followed the methodology used in SR-Can /26-94/. With a few exceptions, the ecosystem models were the same as those used in SR-Can, as well as in the previous safety assessment of SFR 1 /26-95/. A separate model, based on equilibrium assumptions, was developed to assess the accumulation of, and future human exposure to, carbon-14 /26-96/. In order to judge the effects of a release on the environment, calculated concentrations of radionuclides in the environment were compared according to the Erica method /26-97, 26-98/.

A release from the repository to the landscape and to a well was considered in a 100,000-year perspective. Carbon-14 gave the highest calculated dose to human beings via lake food, as well as via drinking water from the well at high groundwater flows (associated with shoreline displacement). The uncertainty and sensitivity of the calculations were evaluated using probability-based methods. The estimates of the highest doses were associated with small uncertainties, since factors that determine transport and accumulation of carbon are well known and can be estimated with good certainty. The most important factors for the calculated exposure from ingestion of food were the quantity of dissolved inorganic carbon and the production of plant biomass in the lake, as well as (for the contribution from drinking water) the flow capacity of the well. For all investigated radionuclides, the concentrations in the environment were lower than the used limit values by five orders of magnitude or more /26-85/.

SSM recently reviewed SKB's biosphere methodology in the safety analysis report for SFR 1 /26-85/. SSM took a positive view of the fact that SKB analyzed the effects of a final repository on the environment, but identified a number of weaknesses in the methodology we have used for radionuclide modelling. The Authority recommended a revision of the assumptions on which the dose calculations are based, among other things to determine the degree of conservatism in the estimated risk /26-86/. It was considered particularly urgent to review the methodology for determining the size of the biosphere objects and uncertainties in the size of the catchment, as well as how to handle accumulation in connection with transitions between ecosystems.

For the safety assessment of the Spent Fuel Repository (SR-Site), SKB has further developed the methodology for exposure calculations in a number of respects. This development has been based on conclusions in RD&D Programme 2007, comments on this by the regulatory authorities and newfound knowledge. The radionuclide model has been implemented in the tool Pandora, and the tool Ecolego has been used to verify the calculations. The most important changes in the calculation methodology are presented below. Newfound knowledge in this area will be presented within the framework of SR-Site.

The most important methodology updates are in the calculations of the activity concentration in the environment. The current radionuclide model describes how hypothetically released radionuclides that reach the deepest deposits of a biosphere object are transported and accumulated in deposits, after which they are cycled in superficial soil layers, surface water systems and the atmosphere, Figure 26-8. For the calculations it is assumed that the entire release reaches one biosphere object. For each biosphere object the succession of ecosystems is described continuously as a function of land uplift, sedimentation and terrestrialization, see Section 26.8. The transition between sea and lake is described with an isolation time of 500 years, after which the size and depth of the lake diminish as a function of sedimentation and terrestrialization. Aquatic and terrestrial ecosystems thus occur simultaneously in one and the same biosphere object, but their size relationships change gradually over time. The model includes fluxes of organic and inorganic carbon in lake and soil water as well as in the atmosphere nearest the ground, which permits a dynamic description of activity concentrations in plants and the environment.

Existing knowledge of the sites has been used to a great extent in the modelling. The detailed description of the landscape and its evolution (see Section 26.8) has been translated to simple parameters such as areas, mean depths and sedimentation rates, which vary with time. Plant production and water exchange between sea basins change continuously with time as a function of landscape geometries and site data. Water balances from detailed hydrological calculations (see Section 26.6) have been translated to simple horizontal and vertical flows, which have been scaled to the surface runoff. Biological concentration factors (CR) and distribution (partition) coefficients (K_d) are based on field data from the sites.

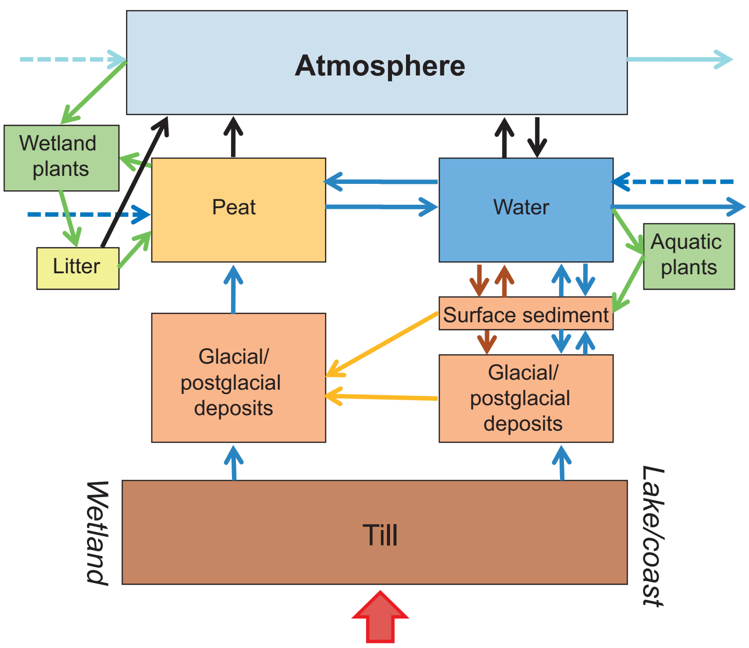


Figure 26-8. *Simplified schematic illustration of pools and fluxes of radionuclides used to calculate the activity concentration in the environment in a biosphere object. The red arrow represents a potential release, while other solid arrows represent fluxes of radionuclides that are driven by: blue = hydrological flows, black = gas exchange, brown = sedimentation/resuspension, yellow = terrestrialization, green = plant uptake and degradation. Dashed arrows represent uncontaminated flows of air and water.*

Further development of the methodology for calculating the future human exposure in a landscape object has also been carried out. During the time an object is just below or above sea level, humans in the object are assumed to support themselves as hunter-fisher-gatherers. When the object is located so far above sea level that it is possible to conduct sustainable agriculture, it is assumed that most of the wetland is drained and used as agricultural land, or for grazing. Exposure is calculated by taking several exposure pathways into account simultaneously (ingestion, drinking water, inhalation and external irradiation). The group of humans who get the highest exposure, over all times and landscape objects, comprises the most exposed group.

Dose conversion factors (i.e. the fraction of radiation from a release that contributes to harmful effects in a representative individual in the most exposed group) calculated with the current methodology (SR-Site) have been compared with the results of previous calculation methodologies (SR-97, SR-Can and SAR 08). The observed differences have been related to the development of the model structure, changes in parameter values and assumptions regarding the distribution of the release in the landscape.

The consequences of a release for the environment have been analyzed using the Erica method /26-97, 26-98/. Doses have primarily been calculated for organisms that live on the investigated sites. Calculated activity concentrations in the environment (soil, sediments and water) were combined with dose conversion factors and empirical transfer factors from the sites for the analysis.

A probability-based analysis has been used to quantify the uncertainty in the exposure calculations caused by uncertainty in model parameters. The sensitivity of the calculations to assumptions regarding future human use of food resources has also been quantified.

When it comes to the chemical toxicity of radionuclides in the spent fuel and the copper canister, SKB's considers it unwarranted to carry out further calculations with the methodology that has been used to determine the radiological risk in the safety assessment, since a rough pessimistic calculation has shown that all nuclides fall below the comparative concentration criteria. This calculation is being reported within the framework of SR-Site.

Programme

SKB will further develop the methodology for dose calculations based on newfound knowledge from the biosphere programme and applications of the methodology to safety assessments. The sensitivity and uncertainty analysis from SR-Site will serve as a guide for this development.

With current methodology for radionuclide modelling, concentration factors and distribution coefficients (K_d values) are key factors for the uncertainty in calculated activity concentrations in the environment and thereby the uncertainty in the calculated exposure. The ultimate ambition is to complement empirical transfer factors between the environment and human food with a mechanistic description of plant uptake and bioaccumulation, see Section 26.5. This development is particularly warranted for nuclides of commonly occurring elements in the environment.

Previous safety assessments of the final repository for radioactive operational waste (SFR 1) show that carbon-14 that dominates the calculated dose to future humans. In preparation for the safety assessment of an extended SFR, the representation of processes that drive the turnover of carbon, along with underlying assumptions, will therefore be re-examined.

26.10 National collaborations, international work and dissemination of information

SKB is participating actively in a number of international cooperation forums concerned with radiological safety for man and the environment, including IAEA (International Atomic Energy Agency) programmes and EU projects, as well as in organizations such as BIOPROTA and the International Union of Radioecology (IUR). SKB also supports research at a number of universities and institutes of higher learning. SKB believes it is urgent to disseminate newfound knowledge, as well as to make data and results available for national and international scrutiny, for example via scientific publishing and active participation in symposiums and seminars. In their review of RD&D Programme 2007, both SSI and the Swedish National Council for Nuclear Waste considered it positive that SKB has broadened national cooperation via the network SurfaceNet. The Council also takes a positive view of SKB's increasing publication in scientific journals.

26.10.1 Activities

SKB has followed the IAEA's programme for modelling of radiological risk in the environment (EMRAS). SKB's involvement in the programme will increase in EMRAS II, where SKB will assume leadership of Working Group 3 (WG3) from 2010. The working group's goal is to support the updating of the IAEA's recommendations for biosphere modelling, coupled to climate and environmental change, for assessing the safety of radioactive waste repositories.

SKB has acted as a reference group for two EU project: FUTURAE, where the formation of leading networks in radioecology has been assessed, and PROTECT, for determination of limit values for assessment of radiological risk to flora and fauna.

SKB has been active in BIOPROTA, a joint international project concerned with key issues for assessment of long-term radiological safety in the biosphere. Participants in the group include both organizations who perform assessments of radiological risk (e.g. SKB, Posiva, EDF and Nagra) and regulatory authorities (e.g. SSM and NDA). In its former role as chairman, SKB has encouraged an exchange of information and experience within the group, and has participated actively in the working groups for selenium and environmental exposure. Within the framework of BIOPROTA, SKB is supporting a study of iodine transport in a Canadian wetland. We have also participated in the group for radioactive waste within the IUR.

Cooperation with Posiva has continued, above all within BIOPROTA. SKB has also collaborated with Posiva in the Greenland Analogue Project (GAP, see Section 19.6), as well as with a number of other organizations: NMWO (Canada), NDA (UK) and GEUS (Denmark). The work has involved field sampling of surface ecosystems under permafrost conditions and SKB has arranged a number of workshops in connection with this.

SKB will continue to report newfound knowledge in scientific, popular scientific and professional journals, for example /26-38, 26-39, 26-40/. Furthermore, we have presented data and scientific results at a number of international and national symposiums. Among other things, SKB gave an introductory presentation on the integration of hydrology and ecology at the international symposium on hydrogeology (IAH) in Lisbon in 2007, which also resulted in a scientific publication /26-62/. SKB has further contributed with presentations and posters at a number of different symposiums: ECORAD in Bergen in 2008 /26-61, 26-99, 26-29, 26-100, 26-101/, ICEM in Brügge in 2007 /26-102, 26-60/, NKS-B FOREST in Helsinki in 2008 /26-28/, the 33rd International Geological Congress in Oslo in 2008 /26-103/, the DHI user conference in Dubrovnik in 2008 /26-104/ and the ESRI user conference in Stockholm in 2007 /26-105/, as well as the Swedish OIKOS meeting /26-106/. Members of the biosphere group are also regularly engaged as reviewers of scientific papers.

SKB lectures annually on the biosphere programme at several of the country's universities. For example, we have lectured in the post-secondary programme in ecotoxicology and biology at Stockholm University and in a doctoral programme for radiophysics arranged jointly by Gothenburg and Lund universities. SKB's has been consulted for its expertise on the biosphere by the National Research Center for Environment and Health in Germany (GSF) and the Svealand Coastal Water Management Association.

SKB has supported research projects at the Swedish University of Agricultural Sciences /26-15, 26-16/, Umeå University, Stockholm University /26-74, 26-75, 26-107, 26-108, 26-77, 26-76/, KTH /26-73/, Chalmers and Lund University /26-10, 26-17, 26-18/. The large body of data from the site investigations has been used by regulatory authorities and independent researchers at e.g. Linnaeus University in Kalmar /26-109, 26-27, 26-110, 26-111, 26-112, 26-113, 26-114/, at KTH and at Stockholm University, as well as at Universidade de Aveiro in Portugal /26-115/. SKB has further supported the seminar programme at the recently established Centre for Radiation Protection Research at Stockholm University.

Programme

SKB will continue to be active in BIOPROTA during the coming programme period, and its involvement in EMRAS II will also continue. Results and newfound knowledge will be disseminated via scientific publication and active participation at international symposiums as well as at seminars and lectures at Swedish universities and institutions of higher education. SKB will continue to support research groups at domestic and foreign universities. After the safety assessment in preparation for the licence applications for the Spent Fuel Repository has been concluded, it is our ambition to publish all newfound knowledge concerning transport and accumulation of radionuclides in an evolving landscape in, for example, a special theme issue of a scientific journal. We will also continue to monitor international development and evaluation of existing models for calculation of activity concentrations of e.g. C-14 and Ra-226 in BIOPROTA.

27 Other methods

There are two main approaches for managing the spent nuclear fuel. One entails regarding the fuel as a resource, the other as waste.

Both in Sweden and other countries with nuclear power reactors, certain principles and strategies for spent fuel management have been dismissed from further development because they are obviously unsuitable and/or not practically feasible. Other principles and strategies have been regarded as sufficiently promising to warrant further analyses. Such analyses have in turn resulted in the fact that certain systems have been found to be less promising than others. Further knowledge accumulation and development work on less promising systems has been put aside.

Figure 27-1 illustrates the principles and strategies for final disposal of spent nuclear fuel that have been considered on numerous occasions. The strategies that are of interest today are:

- Partitioning and transmutation (P&T).
- Geological disposal.

It is possible to considerably reduce the quantity of certain long-lived radionuclides in the spent nuclear fuel by means of P&T. Compared with spent nuclear fuel, the decay products from transmutation have shorter half-lives or are even stable. By unanimous judgement, P&T is a research area that requires several decades of development before demonstration plants can be built. The technology is thus not available today, which means that a commitment to P&T would shift the responsibility for disposing of the spent nuclear fuel to future generations. Even after P&T, radioactive waste still remains and must be disposed of, so this cannot be regarded as a strategy for final disposal. Using transmutation solely to reduce the quantity of high-level, long-lived waste is not efficient, in terms of either costs or resources. Partitioning and transmutation is instead a possible technology for more efficient utilization of the fuel in new types of reactors.

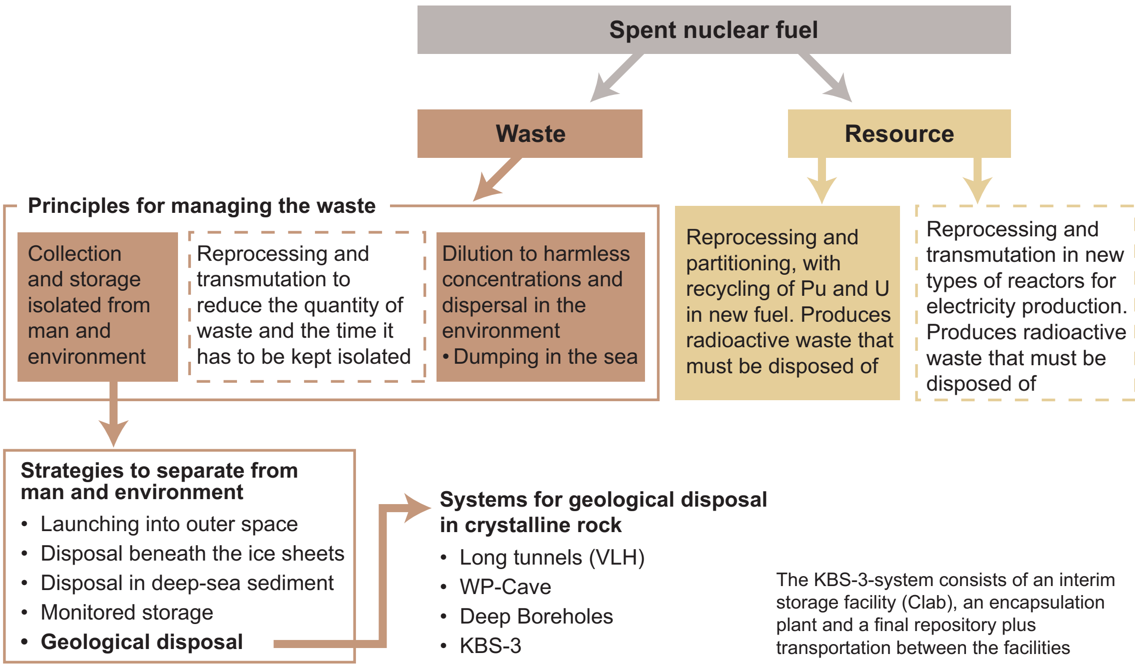


Figure 27-1. *Principles, strategies and systems for disposal of spent nuclear fuel. The principles in the dashed boxes are based on technology that is not available today.*

There is a broad international consensus regarding the principles for disposal of spent nuclear fuel and high-level waste, and in most countries with nuclear power these disposal systems are under development. The methods are based on systems with multiple barriers located at great depth in geological formations. The system that is still held forward by some as being of interest for Sweden besides the KBS-3 method is disposal in deep boreholes.

P&T and disposal in deep boreholes are described in Sections 27.1 and 27.2, respectively.

27.1 Partitioning and transmutation (P&T)

The research on partitioning and transmutation (P&T) in Sweden and other countries is searching for feasible methods for eliminating, or at least considerably reducing the quantity of, long-lived radionuclides that have to be deposited in a final repository. The most important long-lived radionuclides are the so-called transuranics, or transuranium elements, which are elements heavier than uranium. Transuranics are formed in nuclear reactors by the capture of one or more neutrons by uranium atoms, which are then transformed by radioactive decay to neptunium, plutonium, americium or curium. Plutonium comprises about 90 percent of the total quantity of transuranics in spent nuclear fuel from today's light water reactors. Small quantities of even heavier elements than curium may also be formed, but they are of minor importance in this context. A few fission products, such as technetium-99 and iodine-129, may also be of interest for transmutation, since they have long half-lives.

The long-lived radionuclides can be transformed (transmuted) to more short-lived or stable nuclides by nuclear-physical processes. In theory and on the laboratory scale, a number of such processes are possible. But in practice, the only process that has been used thus far for transmutation on a large scale is irradiation with neutrons. Neutrons can split the nuclei in transuranic atoms so that they are transformed into other nuclides. Large-scale transmutation of transuranic elements from spent nuclear fuel must thus take place in plants that resemble a nuclear reactors in many respects. The nuclear fission process releases large quantities of energy that can be used for power production.

A prerequisite for transmutation by neutron irradiation is that the nuclides to be transmuted are separated from other nuclides in the spent fuel. In particular, residual uranium must be removed in order to avoid the formation of more plutonium and other transuranics. Separation, or partitioning, of the different elements can in principle be achieved by mechanical and chemical processes. Today this is done in large reprocessing plants that separate uranium and plutonium from each other and from other elements in spent nuclear fuel. However, these plants cannot separate the other transuranics – neptunium, americium and curium – from the high-level waste that must be disposed of.

The goal of the current research on partitioning is to find and develop processes that are suitable for industrial-scale separation of heavier transuranics and possibly certain fission products.

The goal of the current research on transmutation is to define, investigate and develop plants that are suitable for transmutation of the aforementioned long-lived radionuclides on an industrial scale.

The processes and plants that may be the result of this development must meet very tough requirements on safety, radiation protection and environmental protection. They must be economically defensible and provide good security against the diversion of fissionable material. The large quantities of energy that are released in the transmutation process should be put to use. The processes and the plants must be accepted by society.

Research on P&T started back in the 1950s when the expansion of nuclear power began and was mainly associated with the development of breeder reactors during the following decades. When this expansion slowed down to a very low level in the early 1980s, interest in P&T more or less died. This interest was reawakened in the 1990s, with a focus on system accelerator-driven systems (ADS). In recent years, however, interest in fast reactors for transmutation has been reawakened due to a renewed interest in nuclear power.

Conclusions in RD&D 2007 and its review

SKB's conclusion was and still is that P&T presumes continued nuclear energy production on a time scale of 100 years or more, i.e. that the energy supply system will include nuclear power and that new types of nuclear reactors will succeed the old ones. We believe that this conclusion has been verified by global developments.

Wet separation was and still is at the centre of this development, even though pyrochemical processes that are being developed internationally may be useful for certain types of fuels with high americium and curium contents. Swedish efforts have been concentrated, as before, on wet separation while development of other methods has been monitored.

Through its work on P&T, SKB mainly intended to keep abreast of international developments, while also maintaining the necessary nuclear competence in the country. The know-how that has been acquired via research on advanced nuclear fuel cycles and nuclear waste management has also proved to be valuable for development of safety, fuel supply and waste management for existing light water reactors.

The research on transmutation was focused on safety, feasibility and reliability. Acquiring knowledge of materials and fuel has been an important interim goal. The importance of analyzing future advanced fuel cycles was emphasized. The intention was to foresee what different types of waste may arise.

SKB also announced its intention to increase its spending in this area from SEK five million per year to between and SEK six and seven million to better be able to participate in the EU's projects in the area. This spending increase was then implemented.

In the review of RD&D Programme 2007 led by SKI, SKB was encouraged to continue conducting or participating in studies of systems for partitioning and transmutation, and SKI had no objections to the announced budget increase.

SKI also cited reasons for investing in partitioning and transmutation:

- In order for SKB's programme to be as comprehensive as is required by the Nuclear Activities Act.
- Sweden must actively follow developments in order to be able to judge the potential of the technology.
- Participation in the international work provides knowledge for both the nuclear waste programme and other areas of nuclear technology.
- Maintaining the level of research on nuclear technology in this way will benefit the management of existing waste.

SKB concludes that this is completely in line with the chosen strategy.

The Swedish National Council for Nuclear Waste also found the level of efforts in the P&T field outlined by SKB to be appropriate and concurred with the assessment that transmutation presumes a very long-term commitment to nuclear energy.

The Council concludes that P&T also leads to long-lived waste, although in smaller volumes. It is quite correct that both high-level and long-lived wastes are unavoidable residual products in all solutions that have been proposed thus far. It is not even certain that the volumes will decrease. If long-lived low- and intermediate-level waste is included, the volumes will probably increase, since processing of the waste also generates a secondary waste fraction. Special efforts are needed to keep down the volume of secondary waste. An example of this is the development of "ash-free" separation chemicals that only consist of the elements carbon, hydrogen, oxygen and nitrogen.

Newfound knowledge since RD&D 2007

At the request of SKB, a reference group of Swedish researchers published a report on the status of P&T research in early 2010 /27-1/. The information for the following brief summary of the current state of knowledge on P&T is taken from this report. The report summarizes developments since 2007, when a similar report was produced on behalf of SKB /27-2/. For a more detailed account of the state-of-the-art, the reader is referred to these reports and the annual reports /27-3 to 27-10/.

In the past three years, interest has swung from ADS back to using fast critical reactors for transmutation. There is agreement today that a spectrum of fast neutrons is needed to efficiently transmute transuranics. However, these conditions can be achieved with both fast breeder reactors and ADS. Accelerator-driven systems are usually regarded nowadays as a method of treating remaining smaller actinide fractions, such as americium, after the main fraction of the transuranics, and in particular plutonium, have been transmuted by recycling in light water reactors as well as fast reactors. The joint European interest organization SNETP (Sustainable Nuclear Energy Technology Platform) has adopted a strategic research plan in which development and construction of a sodium-cooled fast reactor for partitioning and transmutation has been given highest priority. Design of such a reactor is in progress.

Interest in Europe is focused on EU-funded research programmes. The EU's framework programmes are strongly linked to the national programmes in the member states and a few other European states. Other large programmes are being pursued in e.g. Japan, Russia and the USA.

EUROTRANS was the biggest European project directly focused on transmutation. In addition to EUROTRANS, a number of smaller research projects are under way where different specific questions are being studied. Each one is much smaller than EUROTRANS, but together they represent a comparable research volume. EUROTRANS started in April 2005 lasted four years with a budget of 43 million euros, half of which has been paid by the European Commission (EC). The project gathered 47 organizations from 14 countries, 10 of which represented the industry, 19 were research centres and 17 were European universities. The universities were represented collectively in the project organization of ENEN (European Nuclear Education Network). Sweden was represented by KTH in Stockholm and by the neutron research group at Uppsala University. The project was focused on accelerator-driven transmutation and dealt with questions such as design, coupling of accelerators with subcritical cores, development of transuranic fuel, material properties in connection with cooling with liquid lead-bismuth and irradiation with fast neutrons as well as basic nuclear data.

The international research on partitioning is following two main routes: hydrochemical processes and pyrochemical processes. European efforts are coordinated within the EU project ACSEPT (Actinide reCycling by SEParation and Transmutation), which started in 2008 and will be concluded in 2012. This is a collaborative project in the EU's seventh framework programme and is a continuation of the project EUROPART in the preceding framework programme. The project has a total budget of EUR 23.8 million, of which the EC is contributing EUR 9.0 million. ACSEPT deals with questions relating to hydrochemistry, pyrochemistry, process development and training. Sweden is represented by the nuclear chemistry group at Chalmers in Gothenburg.

Internationally, P&T still occupies a prominent position within research and development of future nuclear power and nuclear fuel. It is attracting considerable interest among students in nuclear technology. Interest in the nuclear energy industry has increased with the increasing attention being given to fast reactors. P&T is regarded as an interesting possibility for future energy systems based on advanced nuclear reactors, advanced nuclear fuel and advanced nuclear fuel cycles. A successful development of P&T within the framework of advanced fuel cycles will, however, not eliminate the need of final repositories for high-level and long-lived waste. Furthermore, the separation processes generate waste flows that tend to increase the volumes of high-level short-lived waste (mainly Cs and Sr) and long-lived low- and intermediate-level waste. All of this must be taken into account in a complete system.

In connection with the research on transmutation, we have also commissioned a small study of fusion as a source of fast neutrons /27-11/. If the development of fusion is successful, it could be an interesting possibility for transmutation.

Programme

The goal of SKB's research on partitioning and transmutation of long-lived radionuclides is unchanged and includes to:

- Examine how this technology is developing and how it will influence waste streams from nuclear installations and their nuclide content.
- Judge whether, and if so how, this can be utilized to simplify, improve or develop a system for final disposal of the nuclear fuel waste from the Swedish nuclear power plants.

Research is being pursued in accordance with annual activity plans. Overall assessments are made prior to important decisions in the nuclear waste programme. For the period from 2011 to 2013, SKB's annual budget for these activities is estimated to be unchanged at between SEK six and seven million. SKB was the principal sponsor of research on P&T in Sweden from 1995 up to the end of 2009. In October 2009, however, the Swedish Research Council granted SEK 36 million for the research project Genius (Generation IV research), which involves research on the next generation of critical reactors. Such reactors have a potential for efficient transmutation. The Swedish Research Council's initiative has thus entailed to an increase in Swedish activity in the transmutation field.

Our research activities mainly serve as support for ongoing research at universities and institutes of technology. This research is being pursued in broad international cooperation, above all within the EU through active participation in projects funded within the EU's periodic framework programmes. It is unclear what support will be available in future programmes. It should be pointed out that EU funding is also dependent on the political situation in countries where public opinion is strongly against nuclear power. As long as P&T was regarded as synonymous with ADS, even countries that were sceptical about nuclear power were willing to support it to some extent, since they saw it as a "clean-up" rather than an excuse to prolong the use of nuclear power. Within the Commission, an unchanged level of funding of ADS research seems to be expected for the foreseeable future, whereas funding of research on fast reactors could increase.

The international trend is expected to be towards the use of fast reactors for transmutation, and interest in ADS is waning. Much of this is dependent on the introduction of the fourth generation of reactor systems, which includes fast reactors. For example, France announced that a decision may be made in 2012 to build the sodium-cooled fast reactor ASTRID (/27-12/, sid 10). Questions regarding material and fuel are still of central importance for safety, feasibility and reliability. For Sweden, it is also important to know which processes may be included in future advanced nuclear fuel cycles so that the different types of radioactive waste that may arise can be identified and described.

Hydrochemical separation is expected to attract most of the interest internationally. Swedish efforts will be concentrated on this, while development of other methods such as pyrochemical separation is being monitored. Liquid extraction, which is used in the hydrochemical methods, is an efficient and well known method capable of achieving high purity of the separated phases and small losses. Pyrochemical separation, on the other hand, can be advantageous for fuel with high burnup by offering a lower risk of criticality.

27.2 Deep Boreholes

The concept of disposal in deep boreholes, where canisters of spent nuclear fuel are emplaced in boreholes at a depth of 2–4 kilometres, is described in /27-13/. The upper two kilometres of the boreholes are sealed. In the deep boreholes concept, the canisters have a diameter of 500 millimetres and hold four BWR assemblies or one PWR assembly, which means that the diameter of the borehole at repository depth must be about 800 millimetres. The required area depends on the distance between the boreholes, the quantity of fuel to be deposited and the size of the canisters. It is uncertain how closely spaced the holes can be. In previous studies, a spacing of 500 metres has been assumed to be necessary with a view to the risk of "collision" between boreholes that deviate from the vertical and the heat output of the deposited fuel. With this spacing and assuming that the diameter of the boreholes at repository depth is 800 millimetres, a final repository with a capacity for the waste from the operation of the Swedish NPPs would require a total area of more than 13 square kilometres (60 holes). If the distance between the boreholes is halved, the area requirement is reduced to just over three square kilometres.

In disposal in deep boreholes, the rock is the most important barrier for isolating the waste and preventing radionuclides from spreading to the biosphere. The concept is based on the assumption that groundwater conditions at great depths are stagnant. The reason for the stagnant conditions is that the groundwater has high salinity (and thereby also high density) and therefore tends not to mix with the lighter fresh water above. Figure 27-2 illustrates schematically how properties such as water flux, salinity, temperature and rock stresses change with depth. Figure 27-3 shows a proposal for a conceptual model of the uppermost five kilometres of the bedrock in Sweden /27-14/. Any groundwater movements that do occur at great depth are not believed to have any contact with the ground surface. This means that radionuclides from the deposited spent nuclear fuel could not be carried up to the surface by the groundwater.

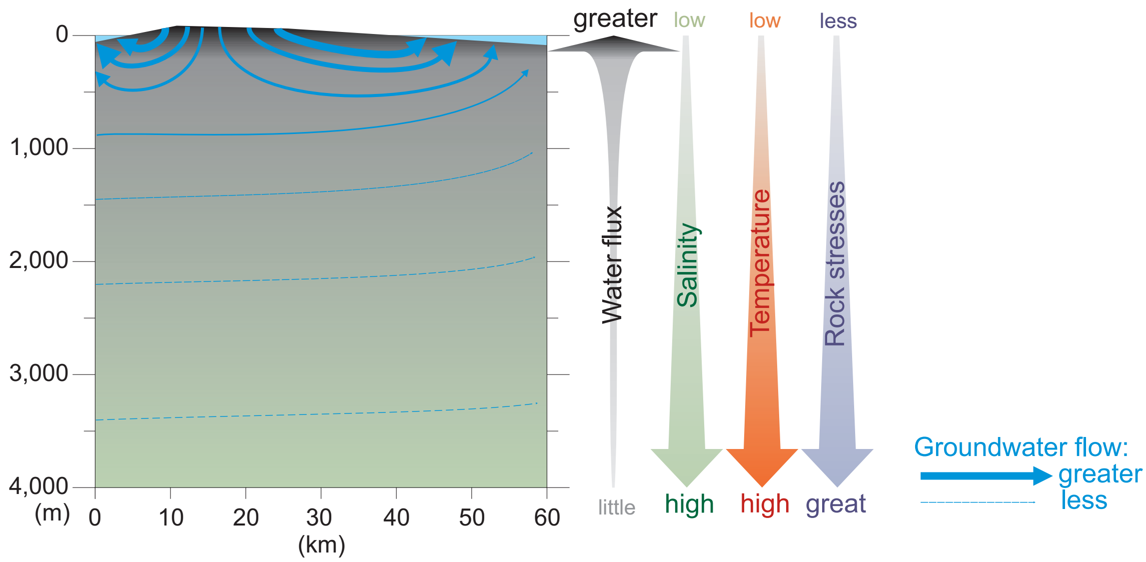


Figure 27-2. Change in the properties of the Swedish bedrock with depth.

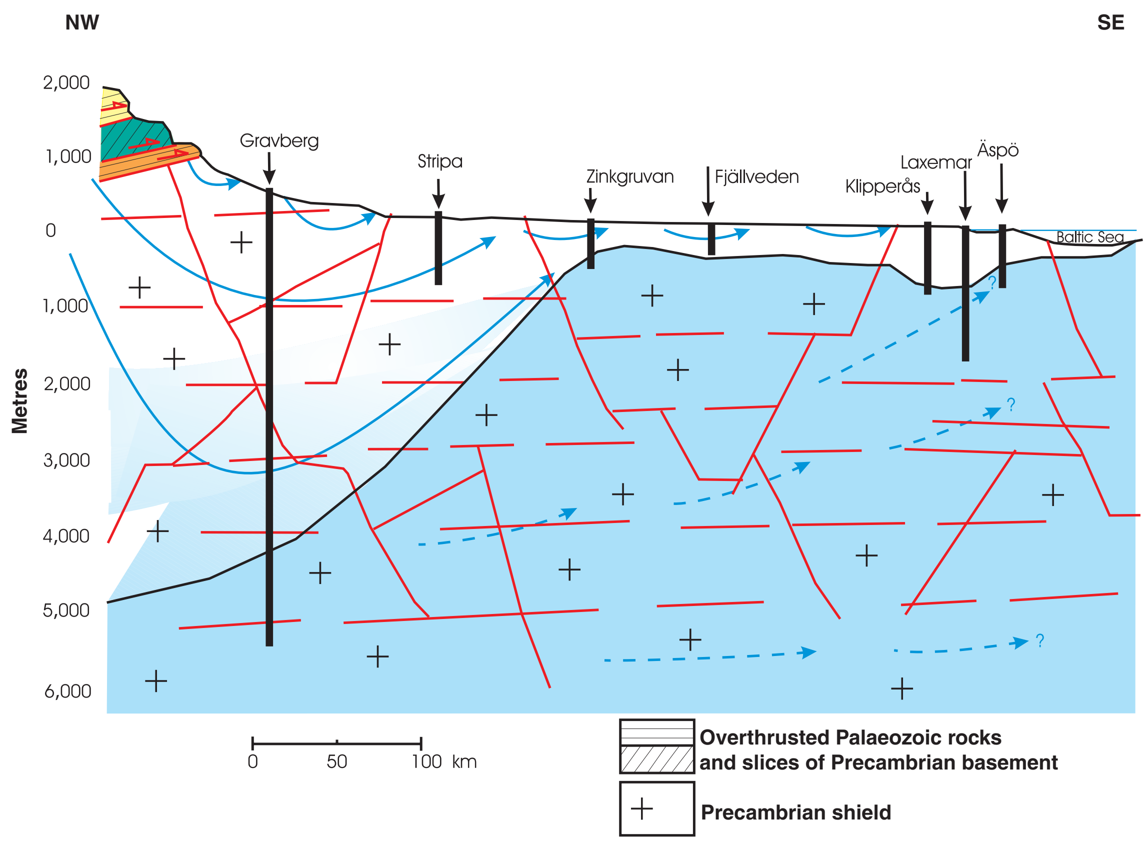


Figure 27-3. Water circulation and variations in salinity along a profile from northern Dalarna to eastern Småland /27-14/.

Conclusions in RD&D 2007 and the supplement to RD&D 2007 and their review

SKB made the same judgement in RD&D Programme 2007 as in RD&D programmes 2001 and 2004. There is no reason to believe that disposal in deep boreholes would increase the safety or reduce the costs of the final disposal of the spent nuclear fuel. Fundamental weaknesses remain, such as the fact that the concept is based on a difficult-to-check deposition procedure, a single barrier after a short time and great uncertainties regarding the evolution of the repository, particularly during a future ice age.

In RD&D Programme 2007, SKB said that they would present a comparison between the KBS-3 method and disposal in deep boreholes by the time of the licence application for the Spent Fuel Repository.

As far as continued work is concerned, SKB said that they will continue to follow developments within the Deep Boreholes field. There is, however, no justification for pursuing an independent research programme in the field. Available resources will instead be concentrated on realizing a final repository according to the KBS-3 method.

SSI said that SKB should study the Deep Boreholes concept further to determine whether it has potential for development and compare it with the long-term protective capability of the KBS-3 method.

SKI supports SSI's viewpoint that the basis for comparison with disposal in deep holes needs to be strengthened for a licence application and that this can be followed up within the framework of the consultations. Furthermore, SKI notes that a repository concept based from the start solely on the rock as a barrier violates SKI's regulations.

The Government decision of 20 November 2008 regarding SKB's RD&D Programme 2007 states that SKB must "give an account of the state of knowledge regarding alternative final disposal methods such as deep boreholes". This was done in a supplement to RD&D Programme 2007 which SKB submitted to the Government in March 2009.

RD&D Programme 2007 included a summary of SKB's efforts to monitor international research and development work on the two concepts of Partitioning and Transmutation and disposal in Deep Boreholes. The account of the state of knowledge concerning alternative final disposal methods given in the supplement to RD&D Programme 2007 is primarily focused on such systems within the strategy of geological disposal that SKB has found reason to either study further or follow the development of, i.e. the KBS-3 method and disposal in deep boreholes /27-15/. In addition, a brief account is given of the state of knowledge for P&T as well as for monitored storage with a focus on Dry Rock Deposit.

In its commentary on the supplement to RD&D Programme 2007, SSM offers viewpoints on the account of alternative methods. Like other reviewing bodies, SSM feels that SKB should gather more background data regarding disposal in deep boreholes. SSM calls for a) an in-depth expert assessment of feasibility (drilling technology and deposition), and b) a more thorough analysis of the uncertainties surrounding the stability of the groundwater at great depths. SSM deems these additional studies to be necessary to permit a systematic comparison to be made between disposal in deep boreholes and the KBS-3 method.

In SKB's view, a more detailed analysis of the stability of the groundwater at great depths can only be made in qualitative terms. As a point of departure we can observe that the outcome of such an analysis is wholly dependent on both the conditions around a deep borehole and the conditions in the borehole itself. It is also dependent on the specific situation in a larger geographical area as well as future climatic conditions. The qualitative analysis can take as its starting point currently existing conditions and from there consider possible future changes. It is generally known today that the chemistry of the groundwater changes with increasing depth and that the groundwater at a depth of several kilometres is often, but not always, saline. We know from SKB's own investigations in the 1,700 metre deep borehole in Laxemar that the salinity increases sharply between 900 and 1,700 metres. We also see that the increase in salinity with depth is strongly correlated with distance to the sea. In other areas, such as Klipperås, which is only about 100 kilometres to the south, no increase in the salinity of the groundwater can be seen at a depth of 700 metres. The difference between Äspö-Laxemar and Klipperås is that Klipperås was not covered by the sea following the latest glaciation. At present we can state that this is a possible explanation for the observation of different salinities in the groundwater, but we don't know if other factors are involved. It is also not known today how the stability of the groundwater at great depths will be affected by a future glaciation or earthquake.

SSM also said that SKB should obtain an in-depth expert assessment of the feasibility (drilling technology and deposition) of disposal in deep boreholes. Such an assessment will be included in the general comparison between the KBS-3 method and disposal in deep boreholes which SKB will present in conjunction with the licence application for the Spent Fuel Repository.

Newfound knowledge since RD&D 2007

New studies of Deep Boreholes

No country recommends disposal in deep boreholes as a preferred alternative for disposing of spent nuclear fuel. In 2008 and 2009, reports on work in this field were published in the UK, Canada and the USA. They mainly discuss disposal in deep boreholes of special waste types with small volumes, for example plutonium from scrapping of nuclear weapons.

UK

A study commissioned by the Nuclear Decommissioning Authority (NDA) was published in the UK in 2008 /27-16/. The study focuses on drilling-related questions. Questions concerning encapsulation and deposition technology require further study before it is possible to determine whether the concept may be viable. The overall conclusion in the study is that the Deep Boreholes concept may, under certain circumstances, be a credible option for the UK, but that a great deal of development work would be required on both drilling technology and deposition technology.

The technology for drilling at great depths has been developed strongly in the petroleum industry during the past 25 years. However, practical experience is with holes with smaller dimensions and mainly in sedimentary bedrock. No experience exists from drilling of deep holes with large diameters and in crystalline bedrock.

Of particular interest for Sweden are the assessments of the problems involved in:

- installing casing in the holes,
- cementing the gap between casing and rock,
- maintaining control at depth during lowering of the canisters, so that they are not damaged.

It was further stated that only completely vertical holes should be considered. The possibilities of drilling such deep holes were evaluated for four different diameters: 300, 500, 750 and 1,000 millimetres free inside diameter in the casings. That means that the diameter of the drilled hole needs to be 20–50 percent larger. With today's experience and existing equipment, it is judged to be possible to drill holes with a free inside diameter of 300 millimetres down to a depth of 4,000 metres. It is also deemed possible to drill holes with a free inside diameter of 500 millimetres down to 4,000 metres with today's technology and further developed equipment, but no experience of this is available as yet.

Holes with a free inside diameter of 750 and 1,000 millimetres are not judged to be possible to drill today to a depth of 4,000 metres. It is deemed possible to reach down to 3,000 metres with a hole diameter of 750 millimetres under favourable conditions.

Another report that was commissioned by the NDA and published in 2008 describes alternatives for geological disposal of long-lived high-level waste and spent nuclear fuel that might be implemented in the UK /27-17/. Twelve different concepts for geological disposal are described in the report, including disposal in deep boreholes. The description includes design, origin, maturity, constructional, operational and environmental aspects, and which countries have the concept in their programmes. The geological environments judged to be suitable for each method are also described.

It is observed in the report that the evaluations that have been made of the Deep Boreholes concept have focused on the feasibility of drilling deep holes. However, not much has been done regarding disposal technology or management of the spent nuclear fuel. According to the study, much is unclear with regard to safety in the handling of the canisters, including the risk that the canisters will be damaged when they are placed on top of one another. The biggest disadvantage of the Deep Boreholes concept is deemed to be the lack of both a detailed design and a thorough evaluation of safety.

The conclusion in the report is that Deep Boreholes is better suited for disposal of small quantities of high-level waste and fissionable material than for disposal of spent nuclear fuel.

Canada

The Nuclear Waste Management Organization (NWMO) in Canada has made a compilation of methods for disposing of spent nuclear fuel /27-18/. Regarding the Deep Boreholes concept, the NWMO observes that no practical demonstration of this concept has taken place and that bringing it to the same level of understanding as the KBS-3 concept would require considerable additional R&D. They further observe that monitoring and retrieval would be much more difficult for the Deep Boreholes disposal concept.

The report also describes an alternative version of Deep Boreholes that was developed for storage of carbon dioxide. The concept entails deviating the hole at a depth of 3,000 metres so that it is sub-horizontal. According to the report, this approach would reduce stress on the stored canisters, facilitate retrieval and permit monitoring and control for a very long time.

The concept is based on the fact that the deeper hydrogeological system is isolated from the system near the surface. According to the report, such geological areas can be found in western Canada.

USA

Sandia, one of the US Department of Energy's energy laboratories, recently published a report on disposal of radioactive waste in deep boreholes /27-19/.

The report proposes that the holes be five kilometres deep with a diameter of 445 millimetres at repository depth. According to the report, such holes can be drilled with existing drilling equipment of the type used to drill geothermal wells. The waste is deposited at a depth of between three and five kilometres. After deposition, the upper part of the holes is sealed. The proposed distance between the holes is 200 metres. The holes are lined with casing with an inside diameter of 381 millimetres. The canisters are made of steel casing with an outside diameter of 340 millimetres, with one fuel package (PWR or BWR) in each canister. The canisters must be strong enough to withstand the waste emplacement (deposition) procedure, but do not need to possess other intrinsic isolating characteristics for the radioactive waste.

The canisters are deposited in crystalline rock. Overlying strata can consist of sedimentary formations. Such formations are found at many places in the USA, which means that final disposal can take place at multiple locations near local storage facilities and nuclear reactors. This reduces the transportation need.

The preliminary evaluation presented in the report is that disposal in deep boreholes may have good potential for excellent long-term safety performance. Features, events and processes requiring further research and development are identified in the report. However, there is no analysis of how the canisters containing the radioactive waste could be deposited in the borehole in a safe manner.

Programme

SKB's stands by its assessment from previous RD&D programmes: that disposal in deep boreholes is not a realistic method for final disposal of spent nuclear fuel.

Moreover, in accordance with the promises made in RD&D Programme 2007, work is under way on a report comparing the KBS-3 method with disposal in deep boreholes. The report, together with documentation justifying SKB's choice of the KBS-3 method, will be presented in conjunction with the licence application for the final repository system.

SKB nevertheless intends to continue to monitor developments in the areas of drilling and disposal in deep boreholes.

Part V

Social science research

28 Overview – social science research

SKB has been conducting and funding research in the social sciences since 2004. The research results have contributed to a deeper understanding of historical, financial and public opinion aspects. The social science research has contributed to increasing the general knowledge base and has also proved useful in SKB's practical work.

When SKB has submitted licence applications under the Nuclear Activities Act for final disposal of spent nuclear fuel and under the Environmental Code for the KBS-3 system, they will be processed within the framework of the democratic system, locally and nationally. SKB would therefore also like to be able to present to decision-makers and the general public material independent of the applications that can shed light on important societal aspects.

In the eight feasibility studies SKB carried out between 1993 and 2000, considerable interest was devoted to the societal aspects. The feasibility study reports contain descriptions and analyses of population development, the business sector, psychosocial aspects, labour market, municipal activities and economy, transport and communications, tourism, property values, etc. The reports contain forecasts and evaluations of the development of the municipality and the region, both with and without the establishment of a final repository. Other reports dealing with the social sciences were also published during the time of the feasibility studies /28-1 to 28-3/.

The Swedish National Council for Nuclear Waste identified the need for high-quality social science research in the nuclear waste field early on, for example in their review of RD&D Programme 2001. The need for further studies of the societal and democratic aspects of the nuclear waste issue has also been noted by the concerned municipalities.

The Council has held a series of seminars for researchers and decision-makers to address issues of democracy, decision-making in complex issues and, not least, ethical aspects. A number of publications dealing with these central issues have been issued by the Council and contributed to a better state of knowledge /28-4 to 28-9/. With its broad expertise, the Council is in a good position to ensure that societal issues continue to be given due consideration and proper treatment in the research.

The purpose of the social science research supported by SKB is to:

- Broaden the perspective on the societal aspects of the nuclear fuel programme. This will facilitate evaluation and assessment of the programme in a larger context.
- Provide deeper knowledge and a better body of data as a basis for site- and project-related studies and analyses. The results of the social science research will thereby provide a sounder basis for various decisions.
- Contribute with data and analyses to research on the societal aspects of large industrial and infrastructure projects. In this way, experience gained from the nuclear fuel programme can benefit other similar projects.

The main emphasis in the research areas we support is on applied research, but there may also be interfaces with basic research. The different research areas and the research projects that have been completed since RD&D Programme 2007 are presented in greater detail in Chapters 29 to 32.

In 2003, in preparation for a social science research programme, SKB surveyed most Swedish research relating to the waste issue, as well as the most important international research. The results of the survey have been compiled in a searchable and publicly accessible database of 400 records. Two preparatory research seminars were held in 2002 and 2003 where invited researchers and representatives from municipalities, regulatory authorities and the Swedish National Council for Nuclear Waste contributed with valuable viewpoints on the form and content of the programme.

The participants at the preparatory seminars contributed greatly to identifying four general research areas as being relevant for the waste issue and the municipalities:

- Socioeconomic impact – macroeconomic effects.
- Decision processes.

- Public opinion and attitudes – psychosocial effects.
- Global changes.

It is important to note that the research projects in many cases span several of the research areas and that certain projects could just as naturally fall under the area of, for example, "Public opinion and attitudes" as the area of "Decision processes" or "Global changes".

Conclusions in RD&D 2007 and its review

The Swedish Nuclear Power Inspectorate (SKI) concluded that SKB's social science research programme had contributed to an overall picture of the work of final disposal of the spent nuclear fuel. SKI further considers that the research funded through the programme embraces a wide field of social science and provides additional knowledge and insights on previous issues. Oskarshamn Municipality finds the programme to be ambitious and of laudable breadth. SKI wishes SKB to further clarify the relationship between social science research, the licence applications, the EIS and social science studies. They decry the lack of a clear link to the knowledge base resulting from the work done by SKB during the feasibility study phase and the studies and research conducted by other actors, for example the Swedish National Council for Nuclear Waste. Further, SKI asks how the research that has been conducted under SKB's previous social science research programmes relates to the work just completed.

The Swedish National Council for Nuclear Waste, Östhammar Municipality and the environmental movement call for a research programme that is independent of SKB. The Swedish National Council for Nuclear Waste believes that the programme should be supplemented by studies of future economic consequences of how the nuclear waste issues are handled, mainly cost-benefit analyses. Another research field deemed by the Council to be particularly urgent is global changes and safety culture. The Council also emphasizes the importance of continuing the social science research programme even after an application has been submitted.

The relationship between research and other documents

SKI wants SKB to further clarify what the relationship between social science research, the licence applications, the EIS and social science studies looks like.

The EIS should describe the consequences of the applied-for activity for man and environment. The purpose of the social science research programme is to shed light on various societal aspects and provide a broadened knowledge base on political and social aspects in preparation for the licensing of the Spent Fuel Repository. The main target group is decision-makers, local and national. The programme is independent of both the EIS and the licence applications.

Research and studies work from different premises. Compared with the social science studies conducted within the framework of the feasibility studies and the site investigations, the social science research is not primarily municipality-specific but is aimed at gathering new and general knowledge. A study is governed by a clear order where the client formulates the issues, while research is characterized by an open-minded testing of theories and a high degree of independence when it comes to formulating research issues, choosing a methodology and drawing conclusions from the results obtained. Research and studies also differ when it comes to the review procedure. The research undergoes the customary academic review and is also reviewed via the RD&D programme by the scientific committee (see paragraph "Scientific committee") and at open seminars. In the case of studies, the review is carried out by the client, SKB, and concerned municipalities.

Since both the social science research and the social science studies often treat the same topics, a mutual exchange of knowledge and experience is sometimes fruitful, but this is not an end in itself. There is no direct connection between the studies and the research conducted by SKB during the feasibility studies and the site investigations, or by other actors, such as the Swedish National Council for Nuclear Waste.

The results of the research are communicated to the municipalities by regular presentation at open seminars, by publication of articles in the yearbook for the research programme and the final reports for the projects, and via postings on SKB's website.

SKI asks how research conducted within SKB's previous research programmes relates to the current programme. The social science research programme started in 2004 after two years of preparatory work in which both researchers and representatives of concerned municipalities participated to identify interesting research areas. SKB has thus not pursued any programme since then.

The Swedish National Council for Nuclear Waste recommends that the research programme be supplemented by studies of the future economic consequences of how the waste issue is handled and research projects that shed light on global changes and safety culture. The Council also believes that the programme should continue even after an application has been submitted.

SKB intends to publish a report in 2011 that provides a summarizing evaluation of the activities of the social science research programme so far and the needs that may exist of future research. An English-language version will also be published. An important opportunity for obtaining international validation of the programme is at the international conference that will be held in the spring of 2011. Only after that will an assessment be made of future research needs based on the new premises that apply during and after licensing and decision.

Work forms

The social science research programme issued the first call for proposals for funded research in 2004, and the programme has since been built up in a dialogue with the concerned municipalities. The programme has remained open to new issues that have arisen over the years. Certain reviewing bodies, such as the Swedish National Council for Nuclear Waste, want the social science research to be funded by other actors than SKB. It would benefit the entire research field if other actors than SKB also took the initiative to fund social science research. How an industry works with a research programme in practical terms is important. If the research community and the rest of the global community do not perceive the scientific results as being independent in relation to the funder, they can lose much of their value. Regardless of whether the project is funded by a research council, a university, an interest group or an industry, the research results should be the same, other factors being equal.

The researchers that have been engaged for different research tasks formulate their own research topics and take responsibility for methodology, results and conclusions. A clear division of roles and tasks in combination with good research communication lays the foundation for independent research. The work forms that have been established for the social science research programme are aimed at achieving this.

Calls for proposals

An initial call for proposals in the four general research areas was sent out to a number of universities and institutes of technology in the spring of 2004. Eight research projects were prioritized in this first application round. Additional calls were then issued annually between 2005 and 2009. Invitations were then sent to all universities in Sweden. The invitations have resulted in 18 research projects concerned with 15 different scientific disciplines and engaging 23 researchers from ten universities and other institutions of higher education.

Two main criteria have to be met for a research project to be considered for funding. First, the application must be of intradisciplinary relevance and good quality, and secondly it must be relevant to SKB's mission. Unless both of these criteria are met, the application is rejected. The application procedure takes place in two steps. The call for proposals stipulates relatively briefly sketched proposals, which are screened by the scientific committee, SKB and concerned municipalities. The applicants whose ideas are deemed to be particularly interesting are urged to come in with complete project descriptions. The approved projects proceed further to contract signing.

The research programme has so far engaged researchers from the universities in for instance Lund, Gothenburg, Linköping, Stockholm, Uppsala and Umeå and has a good geographic spread, Figure 28-1. The programme also has good breadth in terms of research topics, including disciplines such as economic history, cultural geography, jurisprudence, psychology, sociology, media science, and ethics, see Figure 28-2. Most of the projects in the programme are two-year projects.

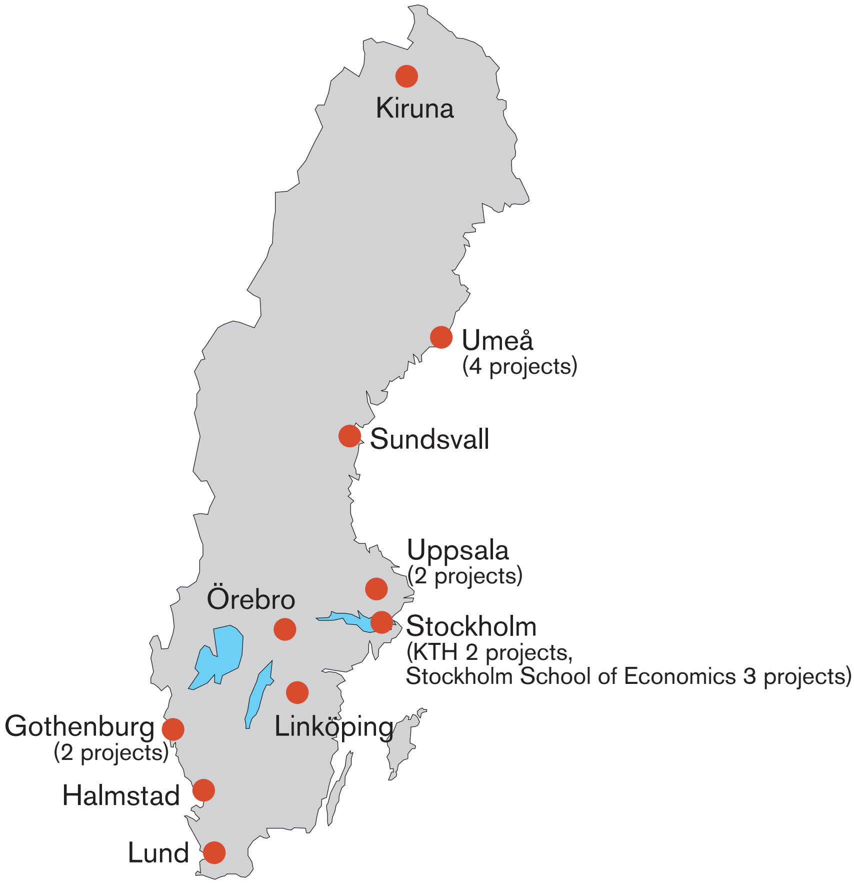


Figure 28-1. *Geographic breadth of social science research programme.*

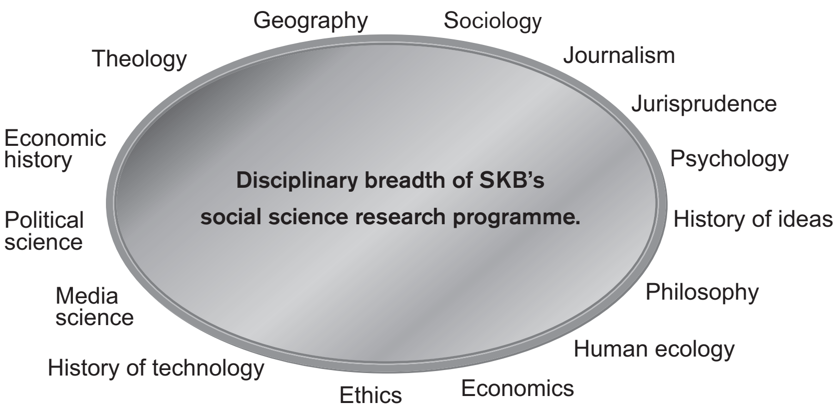


Figure 28-2. *Disciplinary breadth of social science research programme.*

Communication of findings

Decision-makers and other stakeholders should be given an opportunity to share the findings of the research programme. The research task therefore includes communicating interim findings and final results to various target groups via seminars and publishing the research results both in scientific journals and in more popular form, such as in SKB's yearbooks on social science research. Information on and results of the social science research programme can also be found on our website. Via these channels it is possible for municipal politicians and officials, private citizens, regulatory authorities, environmental organizations and scientists to follow the progress of the projects.

Research seminars were held in 2007, 2008 and 2009 for the purpose of presenting the results of the ongoing research. A total of six two-day seminars and two half-day seminars have been held since the start of the programme. The seminars are an important forum where municipal representatives, interest groups and regulatory authorities have direct contact with the researchers who are active in the programme. SKB will hold an international seminar in Stockholm in the spring of 2011.

Annual status reviews of the programme have been carried out since 2006. Interviews have been held with representatives of municipalities, regulatory authorities, scientists and the environmental movement. The interviewees were asked to offer viewpoints on the programme to date and to propose new ideas for research.

Scientific committee

A special scientific committee has been appointed to support the contents and development of the programme. The committee is responsible for ensuring that the research projects have the necessary scientific quality and relevance and that the tasks are assigned to suitably qualified researchers and research groups.

The scientific committee consists of researchers in the social and behavioural sciences: Professor Britt-Marie Drotz Sjöberg, Norwegian University of Science and Technology in Trondheim, Professor Boel Berner, Theme Technology and Social Change, Linköping University, and Professor Einar Holm, Dept. of Social and Economic Geography, Umeå University. The scientific committee also oversees the progress of the work. The projects submit semiannual reports, on which the scientific committee comments.

An important task of the scientific committee is to review the articles for SKB's yearbooks and the final reports that are published in the programme. The work of the scientific committee does not entail any scientific control of the research; rather, it is aimed at achieving greater clarity and relevance in articles and reports.

Newfound knowledge since RD&D 2007

The fourth and fifth calls for proposals were issued in the spring of 2008 and 2009, guided by the viewpoints on the programme that had been received from municipalities, regulatory authorities and other reviewing bodies. The following projects were granted research funds:

- The industrial organization of the final repository – Pitfall or consistency?, Uppsala University.
- Democratic core issues – A study of opinion and external changes affecting the political decision about the final disposal of nuclear waste, Mid Sweden University, Sundsvall.
- The time frame of nuclear waste in comparison. Royale Institute of Technology, Stockholm

New and complementary questions may arise during the examination of SKB's licence applications for the Spent Fuel Repository. It is, however, SKB's ambition that the contents of the research programme during the period 2004–2011 should meet the needs to have various societal aspects illuminated. An overview of the programme period is shown in Figure 28-3 and a list of the projects with starting and finishing times is shown in Table 28-1. The ongoing dialogue that has been held during the course of the programme has provided premises for this.

Evaluation of the programme

A summarizing report has been published on the research conducted during the period 2004–2009 /28-10/. It contains a presentation of all projects so far and the most important results. Furthermore, a discussion is being held on a number of central issues of importance for an understanding of the social and societal consequences of the nuclear waste issue. These are issues that have been addressed and examined in the different projects. For example, patterns of societal change, different pictures of the local and national benefits and risks of the Spent Fuel Repository, opinions and attitudes at different levels, and the long-term importance of the Spent Fuel Repository are also being discussed.

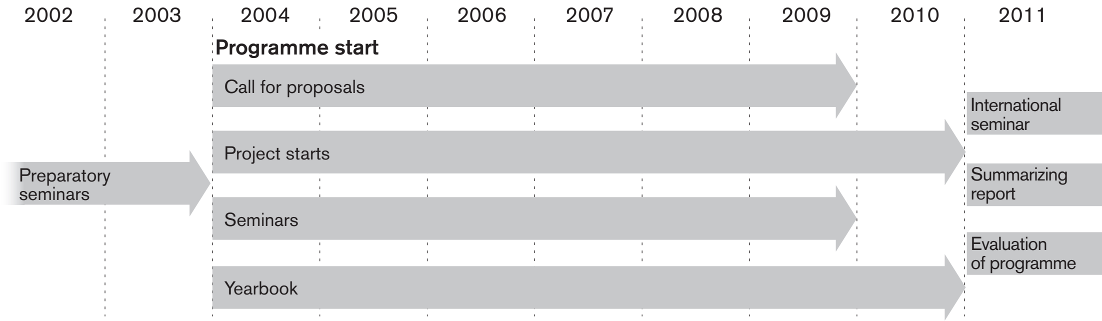


Figure 28-3. *Overview of SKB's social science programme.*

Table 28-1. List of projects funded during the programme period.

Project	Institution of higher education	Period
Attitudes to a final repository for spent nuclear fuel	Stockholm School of Economics	2004–2006
Nuclear waste – from energy resource to disposal problem	Linköping University	2004–2006
Public, experts and deliberation	Örebro University	2005–2006
Identity and security in time and space – cultural theory perspectives on the existential dimensions of the nuclear waste issue	Lund University	2004–2006
Local development and regional mobilization around large-scale engineering projects	Umeå University	2004–2006
Long-term socioeconomic effects of large investments on small and medium-sized communities	Umeå University	2004–2006
National nuclear fuel policy in a European Union	University of Gothenburg	2005–2006
Like night and day despite the same nuclear origin?	Umeå University	2006
Resource or waste? International decision-making processes relating to spent nuclear fuel	Royal Institute of Technology, (KTH) Stockholm	2005–2007
Ethical argumentation in the final repository issue	Stockholm School of Economics	2006–2007
Towards activism or indifference? (2 projects)	Uppsala University	2006–2007
How young people view democracy and technology issues	Stockholm School of Economics	2006–2008
Assumption of responsibility in the back end of the nuclear fuel cycle – a legal perspective	University of Gothenburg	2008–2009
Ethical and philosophical perspectives on the nuclear waste issue	Royal Institute of Technology, (KTH) Stockholm	2007–2009
Participatory democracy and legitimacy of decision-making in multi-level governance systems	Umeå University	2007–2009
The industrial organization of the final repository – Pitfall or consistency?	Uppsala University	2009–2010
Democratic core issues – A study of opinion and external changes affecting the political decision about the final disposal of nuclear waste	Mid Sweden University, Sundsvall	2009–2010
The time frame of nuclear waste in comparison	Royal Institute of Technology, (KTH) Stockholm	2010–2011

The report also provides a summarizing evaluation of the activities of the social science research programme so far and the need for future research, and can also serve as a source of inspiration for other researchers. It will be supplemented by the projects completed during 2010–2011. An English-language version will also be published. After this supplement, which will serve as a final report for the entire social science research programme, a final assessment will be made of the results of the programme. An important opportunity for getting an international validation of the programme as well is at the international conference we will hold in the spring of 2011. Only after that will an assessment be made of future research needs based on the new premises that apply during and after licensing and decision.

29 Socioeconomic impact – macroeconomic effects

The purpose of the research in the area of socioeconomic impact is to gain better knowledge and understanding of how the local community's economy and population structure are affected by the establishment of a large new facility. This knowledge can in turn make valuable contributions to SKB's, the concerned municipalities' and other stakeholders' assessments of how the establishment of the the Spent Fuel Repository will affect the community's economy and demographic development.

Socioeconomic impact includes both narrow economic aspects such as employment, industrial establishment, entrepreneurial spirit, property values, municipal finances and tourism, and broader socioeconomic effects such as travel to and from the town, in- and out-migration to or from the town, and the town's reputation and attractiveness.

Conclusions in RD&D 2007 and its review

SKI says that an integrated account of the results in the context of earlier research findings would have permitted a more complete appraisal of the socioeconomic consequences for a municipality. SKI further believes that the results of the research projects could have been presented in greater detail.

According to division of roles that has been established between SKB and the researchers engaged in the programme, it is the researchers who decide what other research of intradisciplinary relevance is to be integrated in the different projects. The results are presented in the final reports published for each project and in the yearbooks published each year. The space available within the framework of the RD&D programmes is limited, but a slightly fuller project account will be provided in RD&D Programme 2010. SKI also calls for socioeconomic research in a longitudinal time perspective. This is a question for the coming evaluation of the current programme.

Newfound knowledge since RD&D 2007

The calls for proposals issued in 2008 and 2009 have not resulted in any more project applications in the area of "Socioeconomic impact – macroeconomic effects", which would have been of interest. The work forms of the social science programme allow the researchers great freedom to formulate their own research topics within the overall research areas and to work independently in relation to the funder. It follows from this that SKB as a funder must accept the priorities of the research community in this case.

30 Decision processes

The siting of a final repository for spent nuclear fuel is a controversial issue, in part because the time perspective is difficult to grasp. The issue has repercussions for local community planning, national energy policy and international nuclear waste management. By focusing on political issues of this special character, the research can lay the foundation for general knowledge and new perspectives for decision-makers and others to weigh into their decision-making. The actual nature of the decision process for a final repository establishment is one thing; how it is perceived is another. Lessons can be learned from both Swedish and foreign decision-making processes, for example to what extent decisions are perceived to be legitimate, fair and efficient.

Conclusions in RD&D 2007 and its review

SKI says that the studies of decision processes make an important contribution in this field and is particularly appreciative of the research on work in other countries. SKI says that the studies do not fully reflect the complexity of, for instance, decision processes at various administrative levels in a long time-perspective. Stronger consensus-building and feedback to the main parties in the process, particularly the municipalities, is desirable.

Newfound knowledge since RD&D 2007

A number of initiatives have been taken to promote feedback of knowledge and results to concerned parties. The initiatives have enabled politicians and officials in concerned municipalities and other stakeholders to benefit from the results of the research programme. The research task includes communicating results to various target groups at annual two-day seminars and publishing the research results both in scientific journals and in more popular form, such as in SKB's yearbooks on social science research. Information on and results from the social science research programme are also posted on SKB's website. At the annual seminars, municipal representatives have an opportunity to meet the researchers in joint discussions, and the municipalities can extend invitations to the researchers they deem interesting. This permits both consensus-building and feedback to other parties during the course of the work.

The projects that have been carried out meet SKI's requirement of shedding further light on issues related to decision processes at different political and administrative levels and decision processes outside Sweden.

Resource or waste? International decision processes relating to spent nuclear fuel, KTH

Disposing of high-level nuclear waste in the bedrock is not the only alternative that has been discussed internationally. Different countries have come with different viewpoints and proposals. Other alternatives that are and have been considered are reprocessing of the spent nuclear fuel and ultimate removal by exportation. The purpose of the project "Resource or waste? International decision-making processes relating to spent nuclear fuel" is to clarify and analyze decision processes in an international and historical perspective /30-1/. The focus is on four countries: Finland, Germany, Russia/USSR and Japan. Understanding what forces and factors have determined the choice of strategies provides a better understanding of international cooperation in the field of nuclear power and nuclear waste.

The history of the issue has been characterized by sudden policy reversals, uncertainty and continuity. To analyze why different countries have chosen one or another alternative and how the strategy has changed over time, the project focuses on eight central dimensions.

- Does the country produce nuclear weapons?
- Does the country have an expansive or a stagnant nuclear power sector?
- Does the country have strong or weak nuclear expertise?
- Does the country have a strong or a weak anti-nuclear movement?
- Is the country a democracy or a dictatorship?
- Does the country have strong or weak local political power?

- Does the country have good geological conditions for a final repository?
- To what extent does the country have access to domestic uranium resources?

These factors all influence how different countries have managed spent nuclear fuel, what conflicts have existed and what choices have been made. They provide a picture of very dynamic processes, where geological and technological premises, as well as the availability of domestic uranium resources, have been important but where politics has also played a central role. Examples of the influence of politics are the absence of a democratic public discussion in Russia, the existence of political consensus in Finland, and strong political conflicts and local challenges in Germany and Japan. The anti-nuclear movement has questioned the suitability of the salt formations that have been considered for a final repository and has demanded that other alternatives be explored. Geology has thereby become another factor that has influenced the political processes in Germany.

Domestic availability of uranium has historically played a very big role for the strategic choices of different countries when it comes to the management of spent nuclear fuel. In Japan, which has no significant domestic uranium resources, fear of a shortage of uranium has influenced political decisions and contributed to a commitment to reprocessing technology and breeder reactors. A similar way of thinking can be discerned in the Soviet Union's early nuclear power history, even though it has since become clear that the country has good uranium resources. In Finland and Germany, the current strategy of direct disposal of the spent nuclear fuel imply confidence that the supply of uranium will not be a problem in the future. Scandals, political controversies and strong anti-nuclear movements have, particularly in Germany and Japan, more or less led the countries into a dead end.

Due to the absence of opposition in Russia – and previously in the Soviet Union – reprocessing has continued to be the preferred option, albeit with great economic and technical uncertainties, while the democratic consensus that has emerged in Finland has led to a decision on the selection of a site for a final repository for spent nuclear fuel.

The basis for decisions in the nuclear waste issue, experiences of the legislative basis and the EIA process, Umeå University

The aim of the project "The basis for decisions in the nuclear waste issue, experiences of the legislative basis and the EIA process" is to analyze the multi-level governance process in connection with the siting and design of a final repository for spent nuclear fuel /30-2/. Since no such facilities have been built in Sweden, there is no established practice for how laws should be coordinated and interpreted. The study sheds light on three general questions:

1. What is the formal decision-making mandate and what are the decision-making bodies at different levels – local, regional and national – according to the legislation? What questions of interpretation are experienced by the different actors?
2. What private citizens and what organizations, besides those with a formal decision-making mandate, have participated in the EIA consultations, and what viewpoints have they expressed regarding the process?
3. How have judgements of, and reactions to, risk related to a final repository been handled in the process?

The study is mainly based on two different sources of material. A literature review with a focus on nuclear fuel management has been carried out within the social sciences field. Special interest is devoted to the content of the legislation in relation to the EIA process, its background and design. The EIA process is of special interest, since it involves both formal decision-making bodies and interest groups. In addition, the literature review deals with theoretical perspectives regarding perceptions and communicating of risk assessments.

The literature review also include minutes from SKB's consultations during the period 2001 to 2007 plus two interview studies with somewhat different aims. The aim of the first interview study is to investigate how participants with a formal role in the consultations perceive legislation, the EIA process and licensing processes. Fourteen interviews were conducted with representatives of municipalities (politicians and officials with legal expertise in the nuclear waste issue), county administrative boards, SKB, SSI, SKI, regional environmental courts and the Ministry of the Environment. The second interview study concerns the consultations held during the EIA process. A total of 20 interviews were held with the different parties that participated in consultations.

The interviews show that the municipal level and local environmental organizations have an active role, while the regional level via the county administrative boards is not as active. On the national level, particular attention is paid to the state's influence on the EIA process via e.g. decisions on financial support for the participation of different groups via the Nuclear Waste Fund. A universal perception is that most of the actors perceive their own role as clear, while the interaction between sectoral laws and the Environmental Code is unclear in some respects.

The environmental organizations and other parties have to a great extent had different perspectives on the role of the consultations. More established parties, including regulatory authorities, represent a planning paradigm where the project itself and the political decisions are in focus. The environmental organizations traditionally represent an environmental paradigm that focuses on the precautionary principle for potentially environmentally hazardous activities. These differences have characterized the whole consultation process.

The results of the interviews also show that the environmental organizations cannot participate in the consultation process on the same terms as government authorities and the nuclear power industry. The resources of the industry cannot be matched by any other party when it comes to expertise and information. Some believe that the County Administrative Board should play a more central role, given its responsibility for coordination. With the exception of the environmental organizations, there is wide agreement that the industry (as is prescribed by Swedish law) should be responsible for the EIA process.

When it comes to differences in risk perception between different actors, the study shows that actors with an expert role tend to emphasize low risks and good opportunities for risk control, while laymen often emphasize the risks to man and the environment. Furthermore, the study suggests that laymen cannot be regarded as a homogeneous group. The interviews shed light on clear differences between, for example, nearby residents and the environmental organizations. Nearby residents often speak of short-term risks associated with consequences for their immediate environment, while the environmental organizations focus more on long-term risks for the environment.

Industrial organization of the final repository – pitfall or consistency?, Uppsala University

SKB is owned by the nuclear power plant owners and is responsible for disposal of the nuclear waste arising in Sweden, including a final repository for spent nuclear fuel. Questions concerning ownership, governance, and relationship to stakeholders and the local community are of paramount interest. Regardless of which site is chosen, the Spent Fuel Repository will be a major industrial project. The project will entail the same kind of industrial and organizational problems that are always encountered in such enterprises. At the same time, the final repository project includes unique elements that make it a special case. For example, its organization extends over a long time horizon and is subject to very strict legal requirements.

Due to the fact that the discussion has mainly had to do with safety and choice of site, little attention has been given to the construction of the final repository and its industrial dimensions. As a result, the questions that are usually put to business leaders concerning organization, governance principles and social responsibility have not been asked. This study takes up these questions by examining the Spent Fuel Repository Project as an industrial project and examining what specific organizational and corporate governance problems that project will entail.

The research questions the study intends to answer are:

- How will SKB, in view of its legal responsibility for the nuclear waste, organize the industrial side of the project – construction, operation and closure – over time?
- What are the reasons for and the consequences of these organizational choices?
- What problems and opportunities do the organizational choices create in relation to the expectations of important stakeholders on the final repository project?

The research project will not primarily study the company SKB, but rather focus on the organization of the industrial units within the actual final repository system. Nor is the project concerned with the socioeconomic conditions in the concerned municipalities. Its relevance lies rather in showing how organization choices have consequences for various stakeholders in the final repository project and how this can clarify possible roles and responsibilities in the process. The question of what is required to organize a construction project with a very long time horizon is also studied. The final report on the project will be submitted in the autumn of 2010.

31 Public opinion and attitudes – psychosocial effects

Opinions and attitudes are highly changeable phenomena and are influenced by various forces, as well as by personal characteristics. As phenomena they are therefore complex research topics. The establishment of a Spent Fuel Repository is moreover a drawn-out process, involving different actors during different phases. The purpose of this research is to study how opinions and attitudes are formed and change. This knowledge can make important contributions to an understanding of the decisions made by the different actors and to the conduct of the consultations. Opinions and attitudes are not just a reflection of decision-making, actual events and communicated messages. Individual characteristics and perceptions of reality also play a role. Deep-seated values and norms, group identification, perceived fears, anxiety about risks, and self-interest are some examples of factors that are also of importance. It is therefore also important to shed light on the "symbolism" surrounding the Spent Fuel Repository and its activities.

Conclusions in RD&D 2007 and its review

SKI considers the results that have emerged to be of the utmost relevance and that they justify further research activities.

Newfound knowledge since RD&D 2007

Three new projects have been carried out in the research area. One of the projects consists of two studies that have been reported separately.

Towards activism or indifference? How Swedish young people view democracy and the environment, science and technology in an international and longitudinal perspective, Uppsala University and Halmstad University

The project "Towards activism or indifference? How Swedish young people view democracy and the environment, science and technology in an international and longitudinal perspective" includes two studies: an international cross-sectional study /31-1/ and a Swedish longitudinal study /31-2/. A point of departure for the project is the idea that a person's attitude now and in the future to a final repository for spent nuclear, to decisions regarding its realization and siting as well as its ultimate legitimacy cannot be regarded as a separate, isolated attitude here and now. Rather, a person's perception of such a specific issue forms part of a context of attitudes to other phenomena, which are in turn formed by more fundamental notions of values, risks and decision systems. For example, attitudes to a Spent Fuel Repository may be related to the broader issue of attitudes to nuclear power. The project focuses on young people, who will have to live for a long time with the results, and who will carry something of their early attitudes throughout their life.

The project analyzes how certain elements in this tapestry of attitudes and motive forces are interrelated and evolve. The longitudinal study poses the question of how young people's attitudes to nuclear power, other technologies, technology and science in general and democracy have changed over the past few decades in Sweden. What role does age play compared with year of birth for these changes in attitudes to technology, especially the use of nuclear power, and democracy since the 1980s? What processes have altered young people's views of science, technology, the environment and democracy during this period?

The Swedish longitudinal study suggests that young people's criticism of the system has diminished since the 1960s and 1970s when it comes to both nuclear power and the workings of the political system. It is above all the baby boomers (those born in the 1940 and 1950s) who have changed their attitude and become increasingly less critical to nuclear power and the political system as such. Attitudes to nuclear power are better explained by age than by year of birth, while the opposite is true of attitudes to democracy.

The conclusion of the Swedish longitudinal study is thus that young people do not differ very much from older people with regard to their view of democracy. The international cross-sectional study gives the same results. On the other hand, the Swedish material suggests that young people differ from older people by immediately adopting new technologies. These new technologies, which have led to drastically changed behaviour and systems for communication in recent decades, also appear to have contributed to an increased acceptance of new technologies for energy production. The international cross-sectional study combines four perspectives on how young people view democracy and politics by (1) regarding them in a dynamic perspective of change, (2) regarding them as being formed by the Swedish political culture, (3) comparing them with their older countrymen and (4) analyzing their view of democracy and politics as an expression of basic values.

The analyses are mainly focused on those defined in the project as "nascent adults". Nascent adults are persons between the ages of 18 and 27 years who satisfy no more than one of the criteria of being married, having children and having a full-time job. This is a relatively new developmental age, falling between youth proper and adulthood. The analysis shows two central value dimensions: "emancipative and self-transcending values" and "openness to change and secular rational values".

The self-transcending trait manifests itself in a positive view of the common good, active participation in civil society, tolerance for social minorities and the belief that individual autonomy is more important than economic development and law and order. Openness to change manifests itself in putting a high value on creativity, freedom and excitement in preference to the status quo in the form of traditional authorities such as religion and family. Sweden turned out to be a special case with high values in these two dimensions. In all countries, including Sweden, nascent adults score higher when it comes to both openness to change and secular values.

But when it comes to emancipative and self-transcending values, adults in many countries exhibit higher values than nascent adults. Sweden is an exception in this respect, however. A universal finding for the 24 countries studied is that the differences between nascent adults and other adults are less than the differences between countries, see Figure 31-1. By comparison, Swedes have a very positive view of democracy and politics, and the gap between Swedish nascent adults and other adults is smaller than in other countries. The nascent adults emulate their adult compatriots with respect to perceptions of democracy and politics. There is therefore no reason for pessimism regarding the possibility that Swedish young people might be moving away from the Swedish political tradition.

The analysis also shows that general views of democracy and citizenship have changed in the direction of greater individualism – from a society-oriented concept of democracy to a more individual-oriented one, from a patriotic and/or collectivistic view of citizenship to a more activist approach. This change is most evident in Sweden.

A concluding portion of the project analyzes the question of whether the different attitude of young Swedes to democracy on the one hand and science/technology on the other is specifically Swedish. Does it differ from young people's attitude in other countries? The analysis shows that Sweden has the

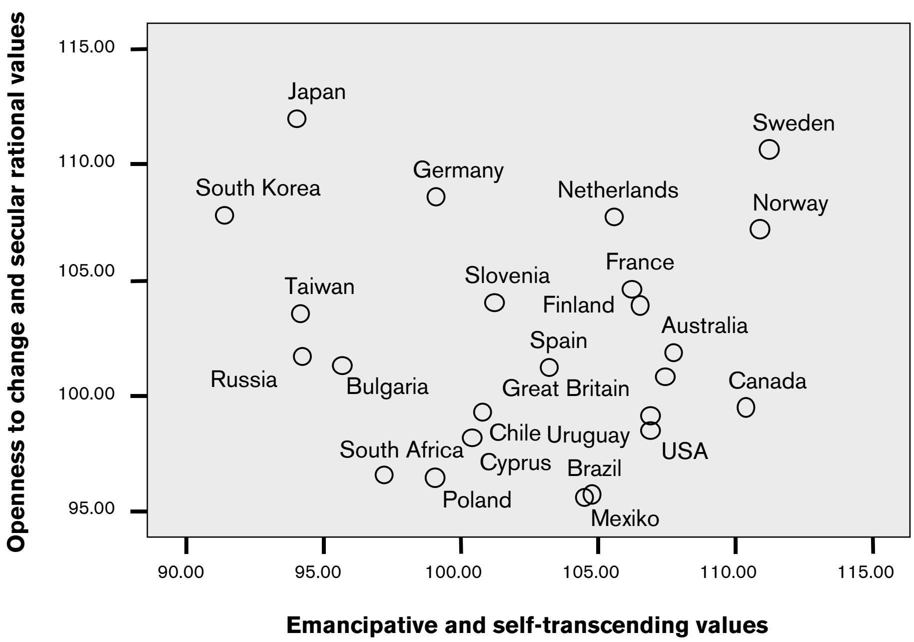


Figure 31-1. *Mean values for the two fundamental value dimensions in 24 countries /31-1/.*

highest score of all surveyed countries for a pro-democratic attitude, both among adults and nascent adults, after Norway and Germany. Bulgaria and Brazil are the countries that exhibit the lowest values.

When it comes to having a positive view of science and technology, Bulgaria, Poland and South Africa have the highest scores and Japan, Slovenia and Uruguay the lowest. Sweden shows medium-high scores and ranks sixth, right after the USA but before Canada. Hopes with regard to science and technology appear to be highest in the countries that come later in adopting this positive view.

Swedish young people are just as positive to a democratic form of government as their older countrymen. However, when it comes to attitudes to science and technology, both younger and older Swedes exhibited average and expected scores. The answer to the main question in this concluding analysis is that Swedish nascent adults are similar to adults in other countries in having a similar restrictive view of science and technology. However, they are much more pro-democratic and thus in this matter as well more similar to their adult compatriots. This differentiates Swedish young people from young people in other countries.

Ethical argumentation in the final repository issue, Stockholm School of Economics

The project "Ethical argumentation in the final repository issue" analyzes the ethical value differences that exist regarding the issue of a final repository for spent nuclear fuel /31-3/. The idea is that ethical values are manifested in the argumentation and can be identified by studying the arguments used by different actors in debates and discussions. Ethical values can in turn be assumed to have an influence on what decisions are made. Opposing values may lie behind some intractable ethical differences of opinion. By the same token, where similarities in values exist between different actors, important issues, on which there is an ethical consensus, may never come up for discussion. The purpose is to contribute to a better understanding of the ethical value differences that exist in the final repository issue. The analysis is premised on a number of questions. Who participates in the public debate about nuclear waste disposal? What topics are discussed? What arguments are used and on what ethical values are they based?

Given the study's selection principles, the answer to the question of who participates in the public debate about nuclear waste disposal is primarily a number of environmental organizations, chiefly MKG (Swedish NGO Office for Nuclear Waste Review), Milkas (Swedish Environmental Movement's Nuclear Waste Secretariat) and Oss (Opinion Group for Safe Final Disposal). Other participants such as the Swedish National Council for Nuclear Waste, SKB, municipalities, regional councils, county administrative boards and SSM are all more sparing in their written argumentation, compared with the environmental organizations.

Three topics in particular are discussed. Which site should be selected? Which method is sufficiently safe? What should the decision process look like? When it comes to site selection, the focus on the municipalities of Östhammar and Oskarshamn as primary siting alternatives for a Swedish final repository has met with criticism, particularly from a number of environmental organizations. Their thesis is that alternative sitings of a Spent Fuel Repository should be given greater consideration. Milkas, for example, says that SKB "isn't selecting a municipality based on where the best technical prospects exist, but based on where nuclear waste disposal is accepted politically by the municipal leadership and the public". SKB, along with the Swedish National Council for Nuclear Waste, says that the issue is multidimensional and that a concept such as "the best possible site" has no meaning without a definition of in what respect the site is "best" and under what conditions the concept is to be applied.

Using local political and public acceptance as a criterion for site selection also creates conflicts. SKB contends that political and public opinion support is a prerequisite no matter where the Spent Fuel Repository is built, while certain environmental organizations say that it should not be considered at all as a site selection criterion. Opinions differ with regard to method selection as well, and the Government statement in 2001 regarding the KBS-3 method as a planning premise for SKB's site investigations has given this method a unique position. The environmental organizations in particular are sceptical towards the method and believe that alternatives such as disposal in deep boreholes should be given greater consideration. Opinions also differ widely regarding the question of retrievability. What the actors disagree on is whether retrievability is compatible with long-term safety and whether retrievability is a prerequisite for the freedom of choice of future generations. The decision process and the division of responsibilities between different actors in the process is also criticized, especially by the environmental organizations. They believe that the decision process leaves much to be desired from a democratic point of view.

Based on the arguments offered, the actors are grouped in four categories.

1. Process drivers (SKB).
2. Observers (regulatory authorities, municipalities, county administrative boards, Swedish National Council for Nuclear Waste).
3. Process critics (MKG, Oss, the Waste Network).
4. Opponents of nuclear power (Milkas, Greenpeace).

Differences of opinion are great between these different actors. The project concludes that despite the fact that the different actors stand far apart from each other in the discussion, they share important ethical values. These are above all the principle of avoiding harm, the idea of equity between the generations, the principle of producer responsibility and the principle of co-determination. The project also identifies a major underlying point of dispute regarding the relevance of functional values in relation to ethical values. This refers to a fundamental difference of opinion regarding the relevance of process efficiency in terms of time, costs and general socioeconomic effects in the final repository issue.

In conclusion, there is a discussion of whether there may be a risk that the final repository issue and the nuclear power debate creates a much too narrow and symbol-charged framework for public argumentation. The nuclear waste discussion is lively and at times infected, in relative isolation from other important environmental ethical issues. Blame can be assigned to both the nuclear power industry, which focuses on the waste as a solvable problem, and the environmental organizations, which have made the waste their main issue. Can the question of nuclear power and the waste problems it entails be discussed at all without reference to the pros and cons of other energy sources and environmental problems, resource utilization and welfare priorities in general?

Attitudes to a final repository for spent nuclear fuel – structure and causes, Stockholm School of Economics

The overall purpose of the project "Attitudes to a final repository for spent nuclear fuel – structure and causes" is to study the structure and causes of the attitudes to a final repository for spent nuclear fuel /31-4/. Previous research results show differences between young and older people, between men and women and between the site investigation municipalities Oskarshamn and Östhammar, as well as between these municipalities and the rest of the country. The project is aimed at explaining the differences in attitude to a final repository between different ages and genders, but also at delving deeper into the question of geographical differences. A secondary purpose is to study more closely the attitude to nuclear power, since it has previously been found to be of great importance in understanding the attitude to a final repository. There is a more positive attitude to a final repository in the site selection municipalities, and the risk entailed by nuclear waste is deemed to be low there. In Oskarshamn and Östhammar there is a clear majority among the men for a final repository, while there is some doubt among the women, see Figure 31-2.

Previous data have shown that the attitude to a final repository became more positive in Oskarshamn and Östhammar during the period 2001–2005, see Figure 31-3. These differences are only partially attributable to the fact that many people work in the nuclear industry. In an analysis of the site selection municipalities and the rest of the nation, the most important explanation for the differences is the benefits such a facility is believed to entail for the municipality. Additional important explanatory factors are confidence in science (epistemic confidence), emotional reactions, attitude to nuclear power and assessment of risks.

There are small differences between younger and older people, where older people are more positive in both the site investigation municipalities and the country as a whole. The age effect is greater among women than among men, and men are generally more positive than women to a final repository. There is an interaction between gender and place of residence entailing that the gender difference is less in Oskarshamn and Östhammar than in the rest of the country, see Figures 31-4 and 31-5.

When it comes to SKB's proposed method, KBS-3, people in Oskarshamn and Östhammar are the most satisfied, while the respondents in the rest of the country more often want to see a development of other methods. Young people in Oskarshamn are more positive to SKB's method than young people in Östhammar. Furthermore, women are less positive to the method than men. There is a correlation between party political preferences and attitude to a final repository. A fairly positive attitude exists to the Swedish nuclear power programme. Perceived benefit is the most important component here, see Figure 31-6. There is, however, some concern, and attitudes have become less positive in Oskarshamn and Östhammar since 2005.

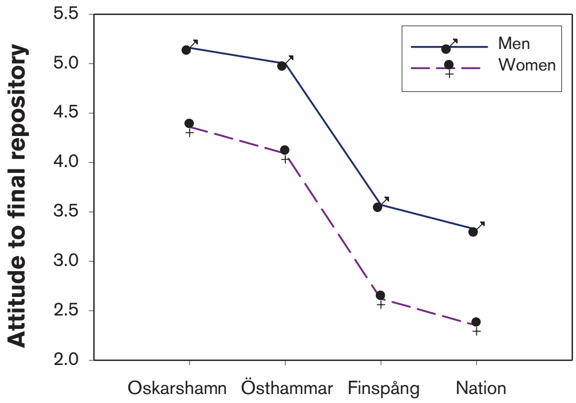


Figure 31-2. *Attitude to a final repository in one's own municipality.*

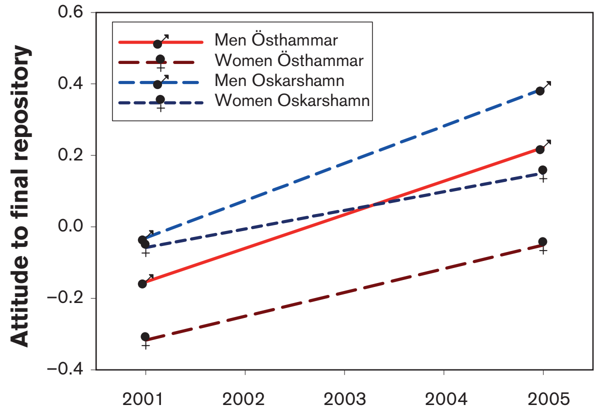


Figure 31-3. *Attitude to final repository 2001–2005. Standardized scale.*

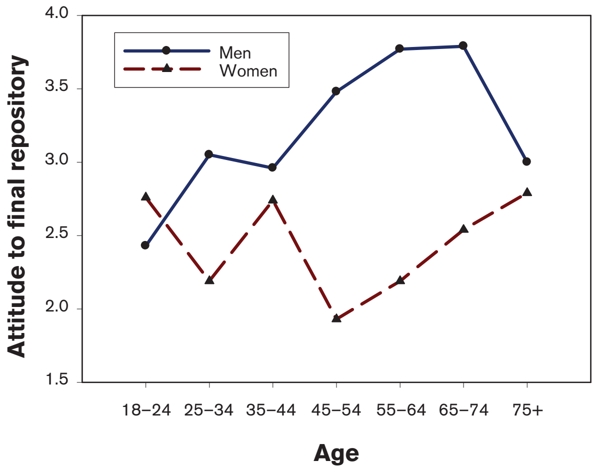


Figure 31-4. *Attitude as a function of age, data from the whole country.*

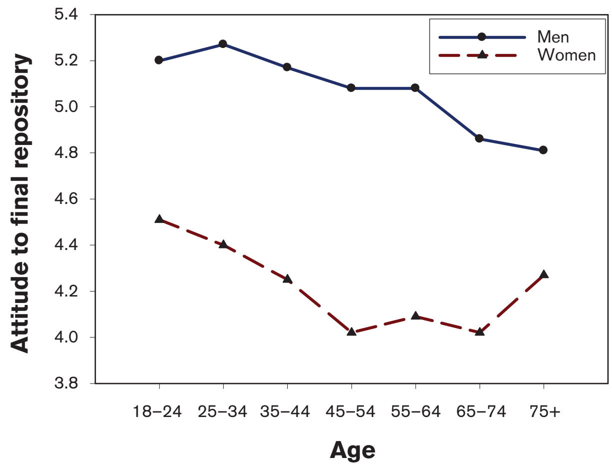


Figure 31-5. *Attitude as a function of age, aggregate data from Oskarshamn and Östhammar.*

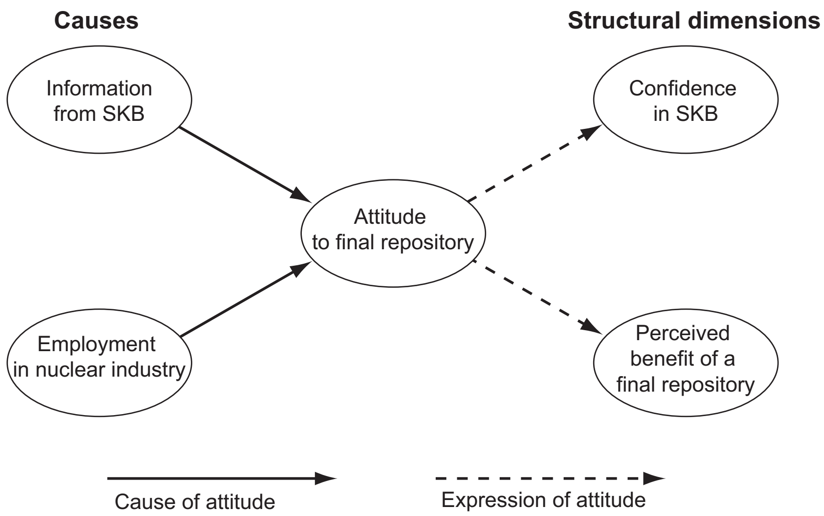


Figure 31-6. *Examples of causal and structural dimensions in relation to attitude to a final repository.*

Twenty different types of actions are judged in the survey, and the respondents were asked to state to what degree they "have participated in information get-togethers arranged by SKB" and "spoken with relatives, friends or acquaintances about this issue". The responses were combined to an index to measure action proneness, and the study shows that the majority of respondents have not been very active, and that those who have reported more extreme attitudes also reported a greater action proneness. This means that those who had the strongest positive or negative attitudes are also those who are most active. The study shows that if a survey targets people who are particularly committed to an issue, such as the final repository issue, it will also come into contact with individuals with unusually strong attitudes.

The most interesting results in the survey are based on two series of questions having to do with information. 1. Where did you get your information? 2. Did this information cover what was important? The questions refer to information from SKB, regulatory authorities and opinion groups and information from the person's own municipality. The results show that information from SKB in particular seems to have had a great positive effect on the attitudes of all groups, and that residents in the site investigation municipalities in particular have received, or solicited, such information. Similar effects, but weaker, are found for information from regulatory authorities and from the municipalities. Information from opinion groups appears to have reached few people, but has had some effect on the attitude to a final repository in Östhammar in particular.

When it came to judgements of the quality of the information, i.e. whether the information covers what is important, the quality of the information from SKB is perceived to be high, while that from regulatory authorities and the person's own municipality is slightly lower. Information from opinion groups is perceived less positively in Oskarshamn and Östhammar, but more positively in the rest of the country. The study finds that the information has reached older people to a slightly greater extent than younger ones, and men slightly more than women.

The attitude to a final repository for spent nuclear fuel can be explained on the basis of perceptions regarding the risks and benefits of the final repository, epistemic and social confidence, and what the respondents believe other people think. In a model where these variables are regarded as structural factors, i.e. as consequences of the attitude and not its causes, a high explanatory percentage has been achieved (86 percent of the variation in attitude). The perceived benefit is above all a strong explanatory factor. Separate model fittings for men and women, young and old, and for residents at different places give similar results.

In summary, the study reports that there is a tendency for older people to have a more positive attitude to a final repository than younger ones. This tendency can be explained by the fact that younger people are more negative to nuclear power and less interested in the waste issue. When it comes to the attitude to nuclear power, it goes in another direction than the attitude to technology in general. Young people are more positive to technology in most cases, but not when it comes to nuclear power. Men and women differ considerably in their attitude to a final repository. In this case as well, the person's attitude to nuclear power is an important explanatory factor, with anxiety coming in second place and not interest, as in the case of the age difference. Women feel greater anxiety in most respects, but the difference is particularly great when it comes to nuclear power. Younger women in particular have a negative (or less positive) attitude to nuclear power. This could contribute to making younger people as a group less positive to a final repository than older people.

Democratic core issues – A study of opinion and external changes affecting the political decision about the final disposal of nuclear waste, Mid Sweden University, Sundsvall

The project "Democratic core issues – A study of opinion and external changes affecting the political decision about the final disposal of nuclear waste", which has been started within the research area "Opinions and attitudes", also has clear links to the areas "Decision processes" and "Global changes". The issue of final disposal of the nuclear waste from the Swedish nuclear power plants is interesting to analyze from a democracy point of view for a number of reasons. In the first place, the whole nuclear power issue exhibits ideological, but also to some extent cross-bloc, party political conflict patterns, which emerged clearly in connection with the Swedish nuclear power referendum in 1980. There the Centre and Left parties were for a phaseout, while other parties advocated two different lines with continued nuclear power production. In the second place, the nuclear waste issue touches upon fundamental standpoints relating to the demands of the ecological society on consideration for the environment and sustainable development on the one hand and the demands of the industrial society on reliable energy supply and economic development to guarantee welfare on the other hand. In the third place, the issue of the geographic location of the Spent Fuel Repository also involves possible tensions between central and local power centres and different stakeholders at these levels.

In all of the conflict dimensions discussed above – the ideological-political, the ecological-economic and the central-local – there are at present uncertainties regarding the future development of nuclear power and the disposal of the nuclear waste. With regard to the latter question, several uncertainties can be noted: How might the political decision-making and the political "wiggle room" before a decision is made on final disposal be affected by changes in the prevailing public opinion climate? How might the political decision-making be affected by changes in global issues such as environmental threats, sustainable development, technological development, legal issues and economic growth? And finally, how might changes in the public opinion climate and global issues affect the trade-offs between national and local perspectives in the nuclear waste issue?

The purpose of the project is to shed light on the Swedish political decision-making process relating to a final repository for spent nuclear fuel, taking into account changes in the public opinion climate and global events. In contrast to studies of public opinion trends and the influence on public opinion exercised by e.g. political parties and the media, the purpose of this study is to analyze how the political agenda is set in a complex interaction between political parties, interest organizations, regulatory authorities, industry and media. It can be assumed that all these actors have their own interests and rational motives, but are also receptive to changes in public opinion and the surrounding world. The result of this setting of a political agenda can also be assumed to serve as a basis for what decisions can be made in a political issue.

The following questions are of central interest to the study:

- What characterizes the parliamentary and the party political debate in the nuclear waste issue, and in what way have actors, standpoints and arguments changed over time?
- What characterizes opinion formation and news reporting in the media in the nuclear waste issue and in what way have actors, standpoints and arguments changed over time?
- In what way are the political agenda and the media agenda nuclear related to each other when it comes to the issue of the Swedish nuclear waste?

A final report on the project will be published in the autumn of 2010.

32 Global changes

The establishment of a final repository for spent nuclear fuel is a unique project with unique features. In the end, only one location in Sweden can be chosen. In 2009, SKB selected Forsmark in Östhammar Municipality as the site of the Spent Fuel Repository. At the same time it is a question that is very clearly related to what is happening in the rest of the world. The purpose of the research is to gain greater knowledge about relevant global factors and global changes. This knowledge can make a very valuable contribution to planning, studies, consultations and decision-making before and after the permit applications. The knowledge may also be important for the future operation of the final repository. The economic situation and trend in the local community is dependent on a variety of circumstances in the surrounding world. What will the Swedish state, which bears ultimate responsibility for the final repository, look like in the future? Legislation, regulation and financing, as well as the country's economic situation, are factors of importance. Another important global change is Sweden's participation in the development of the European Union. What kind of relationship will Sweden have with the EU in 30 years? What will the EU look like? What impact will a deeper European integration in the future have in nuclear waste management, and to what extent will this affect Sweden's own waste management programme?

Conclusions in RD&D 2007 and its review

SKI is positive to the research that has been conducted on Sweden's national responsibility viewed in the perspective of the country's membership in the EU, but wants SKB to monitor international events that may affect the process of managing spent nuclear fuel.

Newfound knowledge since RD&D 2007

Two projects have been carried out during the period. "Ethical and philosophical perspectives on the nuclear waste issue," Royal Institute of Technology, Stockholm, and "Assumption of responsibility in the back end of the nuclear fuel cycle – a legal perspective," School of Economics and Commercial Law at Gothenburg University.

Ethical and philosophical perspectives on the nuclear waste issue, Royal Institute of Technology, Stockholm

The project "Ethical and philosophical perspectives on the nuclear waste issue" is presented in eight different essays with the aim of putting the nuclear waste issue in a broader perspective /32-1/. The texts argue for a more rational approach and emphasize the possibilities offered by science and technology. They do not take a stand in concrete issues relating to management of nuclear waste or formulation of energy policy. However, they do attempt to provide a basis for taking such stands.

Risk and uncertainty

The word "risk" is constantly used in discussions of nuclear waste, as well as in other discussions of the hazards to which we are exposed in society. On closer examination, it turns out that "risk" is used in different senses in different contexts. In some technical contexts a careful distinction is made between risk and uncertainty. Sometimes it seems as if the difficulties different interest groups have in understanding each other has to do with the fact that they do not understand (or perhaps do not respect) each others' use of these terms. In order to be able to have a fruitful discussion, it is important to clarify what is meant by these words, something which this essay attempts to do.

Radiation as an ethical problem

Issues concerning ionizing radiation arouse strong feelings. Are radiation risks unique, or should we think of them in the same way as other risks we encounter in our daily lives? This essay presents the new research area of radiation ethics and points out the connections between the basic issues in radiation protection and moral philosophy. Among the questions discussed are: Is the fact that radiation from a given source is less than the naturally occurring background radiation reason to accept it? Do we have to take into account (probable) effects that cannot be proven because the statistical relationships are too weak? Should radiation protection mainly be adjusted to the exposure of an individual or the total exposure of the whole population?

The time perspective of the nuclear waste

The aspect of the nuclear waste which has received the most attention is that it is hazardous very far into the future. Nuclear waste management has long been the only societal issue that has been discussed broadly in this long time perspective. In recent years, the climate issue has also come to be treated in very long time perspectives. Furthermore, we are engaged in a public discussion of sustainable development, which does not set any actual time limits. The two main perspectives in which the very long-term effects of what we do today have been discussed are presented here: Economic discounting and sustainable development. The problems with both of these perspectives are discussed and alternative approaches are presented. The nuclear waste issue proves to be one of many issues deserving of thorough discussion in a long time perspective.

Science and its limitations

The controversies surrounding nuclear waste management have been concerned to a great extent with how much we can know about what will happen in the future. Expert judgements have been questioned, and sometimes confidence in the ability of science to answer the crucial questions has been low. The critics are right to the extent that science is not infallible. It is often rational of a decision-maker to take into account the possibility that the scientific experts could be wrong. This is one of the reasons why extra safety margins are usually built into complex technical systems. But it is also an important insight to realize that in real life we constantly have to take action and make decisions without being entirely certain. This essay arrives at the conclusion that the best we can do is to act at every given time on the basis of the best available science, while at the same time trying to estimate the level of uncertainty in this science and make allowances for the greatest uncertainties.

What does the precautionary principle say?

It is important to distinguish between caution and the precautionary principle. Caution as a general concept means avoiding actions that can lead to very negative consequences, even if their probability is low. The precautionary principle has to do with handling scientific uncertainty. This essay is about how the precautionary principle should be interpreted and how it fits into the way we accumulate and apply scientific knowledge. An important conclusion is that the precautionary principle is really not a special principle, but concerns how to use all available information when making practical decisions.

How much is a life worth?

Cost-benefit analysis is the traditional economic method for analyzing risk decisions. But the method is controversial, not least because many people are hesitant about the “costs” that are assigned to human life in the analyses. This essay discusses the advantages and disadvantages of the method. The conclusion is that while cost-benefit analysis can be useful, it is important to be aware of its limitations. While the method can be used to support societal decisions, it cannot be the final word but must be combined with other information.

Engineering certainty

Engineers have been developing practical work methods and rules to avoid accidents for a very long time. This area, called “safety engineering”, involves risk management practices that are valuable because they deal with a part of the problem that is difficult to cover in a risk analysis that is based on probabilities. This essay presents three of the area’s most important principles: inherent safety, safety factors and multiple safety barriers. Comparisons with the “engineering philosophy” that is applied in other technical areas can be useful in discussions of nuclear waste disposal.

Risk management as a political issue

Like many other risk issues, the nuclear waste issue is very complicated from a political point of view. This essay examines the risk issues in its sociopolitical context. What division of roles should be striven for between experts and elected representatives in the crucial decisions? Is there any justification for the criticism of local influence that is usually referred to by the abbreviation NIMBY (Not In My Back Yard)? Why is the concept of "acceptance" accorded such importance in risk contexts? What demands can be made on a democratic decision process in a controversial risk management issue? The purpose of this essay, as well as the others, is to show how the nuclear waste issues are related to larger ethical and philosophical issues. It is then up to the reader to arrive at his or her own standpoint.

Assumption of responsibility in the back end of the nuclear fuel cycle – a legal perspective, School of Economics and Commercial Law at Gothenburg University

One overall purpose of the project is to study and analyze how different types of responsibility are regulated in current legislation on nuclear activities /32-2/. The project has investigated whether the division of responsibilities that is established in Swedish legislation is appropriate in the light of the objectives formulated in the legislation's travaux preparatoires (legislative history). Can these objectives be realized with the current body of legislation, or do the changes that have occurred necessitate legal reforms?

The project consists of three studies:

1. Responsibility for safe management of spent nuclear fuel.
2. Responsibility and parallel regulation.
3. Responsibility for non-proliferation of nuclear weapons.

Responsibility for safe management of spent nuclear fuel

This study analyzes, based on Section 10 of the Swedish Nuclear Activities Act, the legal structures surrounding the issue of responsibility for safe management and final disposal of spent nuclear fuel. The purpose is to shed light on the legal aspects that must be considered in the coming licensing process and thereby contribute to a better understanding of the importance of the legal structures for the decisions concerning final disposal that lie ahead. What interpretation will be given to the requirements of the Nuclear Activities Act on "safe management and final disposal" of the spent nuclear fuel in the coming licensing process?

One conclusion that can be drawn from this study is that the Swedish regulation of nuclear activities creates a legal basis for exacting far-reaching industrial responsibility from the reactor owners, but also for an extensive state influence over the activities. The RD&D programmes play a central role in the model for division of responsibility established by the Nuclear Activities Act. They reflect a political will to meet the requirement of "safe management" through research. The legal forms for organization of the programme also reflect an ambition to place great responsibility for execution and financing on the industry. There is also an ambition for the state to maintain and possibly increase its control and influence. However, it is difficult to judge whether the hopes of the 1980s regarding public influence over the process have been fulfilled. The realistic answer to the question of whether the requirements stipulated in Section 10 of the Nuclear Activities Act are met is nevertheless that the interpretation lies in the hands of the political majority that owns the question when licensing takes place. In other words, the judgement of what is safe according to the Nuclear Activities Act is primarily a political one. If at the time of the decision there is a political majority that opposes granting a licence, a legal argumentation may occur.

Responsibility and parallel regulation

Responsibility for management and disposal of spent nuclear fuel is regulated in Sweden by several different laws, which means that the regulatory frameworks overlap each other to some extent. Different reports have paid particular attention to the fact that licences for a final repository must be applied for under both the Nuclear Activities Act and the Environmental Code. This study describes in general terms the parallel regulation of environmental, nuclear safety and radiation protection matters on the national and European levels. It begins with a discussion of the Swedish regulation of environmental responsibility for ionizing radiation, where the Environmental Code is not generally applicable today. This part also deals with the regulation of environmental responsibility at a European level. It shows that the division of responsibility embodied in the relationship between the Swedish regulations is only partially reflected at the European level. The study also describes the relationship between the EU Treaty and Euratom and gives examples of regulation of responsibility for waste management and for furnishing of information in connection with licence applications for a final repository. Finally, the study discusses the consequences of this type of parallel regulation for the Swedish licensing process.

A conclusion that can be drawn from the study is that, due to the parallel regulation at a national level, there are certain risks of overlaps and contradictions. This applies in particular to the conditions for exacting environmental responsibility for nuclear safety and radiation protection. There still seems to a lack of clarity about the extent to which the environmental courts can or should issue conditions on the basis of nuclear safety and radiation protection considerations. This lack of clarity stems from the introduction of the Environmental Code.

The Swedish parallel regulation is only partially reflected at the European level. The international regulatory framework largely follows an established international regulation tradition entailing that the issue of ionizing radiation is of such a special and hazardous nature that it should be regulated separately.

Conclusions from the parallel licensing processes that have taken place in Sweden, for example in connection with licences for power increases at nuclear power plants, show that it works satisfactorily in purely practical terms. However, the study claims that any lack of clarity that exists may be regrettable from the perspective of environmental responsibility. There is also a potential risk that parallel licensing and parallel regulation could create an ambiguity that affects the legitimacy of the licensing decision.

Responsibility for non-proliferation of nuclear weapons

Perhaps the greatest challenge for the development of civilian nuclear energy production is the need to create a regulatory framework that effectively prevents civilian nuclear activities from leading to an increased proliferation of nuclear weapons. The purpose of this study is to describe and analyze how responsibility for fulfilling international commitments regarding non-proliferation of nuclear weapons is concretized in connection with the planning of a final repository for spent nuclear fuel in Sweden. The study gives an account of the development of the multilateral regulatory framework for preventing proliferation of nuclear weapons and analyzes how this regulatory framework has been implemented at the European and national Swedish levels. What responsibility is borne by the holder of a licence to build a final repository for spent nuclear fuel? What is the scope and temporal extent of this responsibility, and what potential problems can exist in the application of the current regulations?

One conclusion that can be drawn from the study is that the complex regulatory system that governs the implementation of international commitments to prevent the proliferation of nuclear weapons has an effective administrative application in Sweden. At the same time there is a risk that the parallelism between national Swedish regulations and applicable rules in Euratom could lead to difficulties in identifying commitments and lines for exacting responsibility. These problems could grow larger if Sweden tries to defend a national regulatory autonomy concerning non-proliferation at the same time as the Community regulatory framework in the area becomes increasingly fine-meshed.

The most obvious problem identified by the study with regard to responsibility for Sweden's international commitments to non-proliferation in connection with a final repository concerns the temporal extent of the responsibility after the operating phase is over and closure has taken place. Under the current Swedish regulatory framework, SKB's responsibility will probably not end when the obligations under the Nuclear Activities Act have been fulfilled. The endpoint will instead be defined by a political decision to discharge SKB from responsibility, after which the state will assume responsibility. At the same time as SKB's responsibility ceases, however, the state's obligation to fulfil international commitments regarding non-proliferation and illicit trafficking of nuclear material remains. In concrete terms, this means that the state also assumes responsibility for fulfilment of the obligations regarding nuclear safeguards and physical protection of the closed final repository. In order to provide greater predictability and clarity, there is a need to formulate criteria for when a decision on discharge from responsibility can and should be made. In connection with the transition, it should also be clarified whether the state assumes ownership of the final repository and the spent nuclear fuel kept there.

Programme

A new research project has been launched and will be completed at the end of 2011.

The time frame of nuclear waste in comparison, Royal Institute of Technology, Stockholm

The project is a study of how the time perspective in the nuclear waste issue is related to the time perspectives that are applied in other societal areas. It touches upon several aspects that have not previously been addressed in the research.

The differences between the time perspectives that are applied in different societal areas are very great. For example, the nuclear waste issue can be compared with the climate issue, where a hundred-year perspective is still the rule. The project will investigate the background and consequences of these differences, and also discuss whether, and if so how, they should be reduced. These kinds of comparisons with other areas can help to create greater clarity regarding the premises of a decision on a final repository for spent nuclear fuel. The study also has relevance for other societal areas. A systematization of the time perspectives in important societal decisions is desirable, in part because it can contribute to better coordination between long- and short-term goals.

Each decision actually has three different time perspectives: validity time, evaluation time and effect time. The validity time is the expected period of time during which the decision will be valid, for example until the decision is to be re-examined. The evaluation time is the period of time that has been taken into consideration in evaluating the effects of the decision. The evaluation time is often longer than the validity time. The effect time is the period of time during which the decision will have practical effects. The effect time often follows from natural and other circumstances, while the validity and evaluation times are under the decision-maker's control. The relationship between effect time and evaluation time is different for different decisions. They coincide for the crucial decisions regarding management and disposal of the spent nuclear fuel. In climate and infrastructure decisions, the evaluation time is generally much shorter than the effect time.

Choosing a time perspective, i.e. validity and evaluation times, for a decision is a part of choosing the framework for this decision. The choice of decision framework also has other components besides the time perspective and can affect the time perspective.

- *Actor perspective.* One and the same decision or decision area can be discussed and described from the perspective of different actors. For example, it makes a big difference where the nuclear waste issue is discussed from a corporate perspective, where both legislation and national energy policy are among the given premises, or from a political perspective, where these are factors that can be influenced by the decisions that are made.
- *Feasibility perspective.* When a decision is described, only the options that are perceived as being possible to choose or realize are normally mentioned. Here it is possible to choose between two kinds of feasibility criteria. It is obvious that physically impossible options should be omitted, but it is not as obvious that options considered to be politically impossible should be omitted.
- *Creating an agenda.* There may be many ways to subdivide a decision area into different partial decisions. This subdivision can have great importance for what decisions will be taken in practice. The two main strategies for creating an agenda have advantages and disadvantages. One strategy is to subdivide a complex decision into as many independent partial decisions as possible. The other is to gather the decisions into a smaller number of big and coordinated partial decisions.

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SKBdoc id	Version	Title	Issued by, year
1049861	1.0	Preliminary decommissioning plan for SFR.	SKB, 2007
1056406	5.0	Organization, leadership and management – Construction and commissioning.	SKB, 2009
1175162	3.0	Welding in connection with fabrication and closure.	SKB, 2010
1175208	4.0	Fabrication of canister components.	SKB, 2010
1175236	2.0	Reliability in friction stir welding of canister.	SKB, 2010
1179633	3.0	Nondestructive testing of canister components and welds.	SKB, 2010
1193244	3.0	Criticality safety calculations of disposal canisters.	SKB, 2010

Appendix

Abbreviations

Ab initio	Latin for "from the beginning". Ab initio calculations are quantum chemistry calculations of electron structures.
ABAQUS	Finite-element computer code used for THM model calculations.
ABM	Alternative Buffer Materials. Experiment at Äspö HRL.
AD	Anno Domini, the year of our Lord. After the year zero in our calendar.
AD modelling	Modelling of advective (waterborne) transport in the form of particle tracking in advective-dispersive transport modelling.
ADS	Accelerator-Driven System for partitioning and transmutation of transuranics.
AECL	Atomic Energy of Canada Ltd, Canada.
ALARA	As Low As Reasonably Achievable. All radiation doses should be kept as low as reasonably possible, one of the ICRP's and SSM's three main principles in activities involving ionizing radiation.
ANDRA	Agence National Pour la Gestion des Dechets Radioactifs, France.
APSE	Äspö pillar stability experiment. Experiment at Äspö HRL.
Asha 230	Possible material for backfilling of final repository for spent nuclear fuel.
ATB	Radiation-shielded transport cask for intermediate-level waste.
ATB 1T	A new cask for transport of long-lived low- and intermediate-level waste in BFA tanks.
Baclo	Development, testing and demonstration of backfill and closure in a final repository. Cooperation project between SKB and Posiva.
BB	Big Bertha. Full-scale test for KBS-3H, swelling of bentonite out of supercontainer.
BC	Before Christ. Before the year zero in our calendar.
BeFo	Stiftelsen Svensk Bergteknisk Forskning (Swedish Rock Engineering Research Foundation), Stockholm.
BET	Specific surface area.
BFA	Rock cavern on Simpevarp Peninsula for dry interim storage of operational waste.
BFA tank	Tank for storage of long-lived low- and intermediate-level waste.
BIOPROTA	International collaboration on biosphere aspects of assessment of the long-term safety of the deep repository.
BKAB	Barsebäck Kraft AB.
BLA	Rock cavern for low-level waste in SFR 1.
BMA	Rock cavern for intermediate-level waste in SFR 1.
BP	Before Present.
BTF	Concrete tank repository in SFR 1, mainly intended for dewatered ion exchange resins.
BWR	Boiling water reactor.
CAD	Computer Aided Design.
CBI	Cement och BetongInstitutet (Swedish Cement and Concrete Research Institute).
CEA	Commissariat à l'Energie Atomique, France.
CEC	Cation Exchange Capacity.
Clab	Central interim storage facility for spent nuclear fuel.
Clink	Clab and the encapsulation plant as an integrated unit.
Code Bright	Computer code for thermal-hydraulic-mechanical calculations.
ConnectFlow	Computer code for groundwater flow calculations.
CRT	Canister Retrieval Test. Experiment at Äspö HRL.
Dawe	Drainage, artificial watering and air evacuation. Technology for drainage, artificial watering and air evacuation of a KBS-3H repository.
DarcyTools	Computer code for groundwater flow calculations.
DECOVALEX	DEvelopment of COupled Models and their VALidation against EXperiments in nuclear waste isolation. International project.
DFN	Discrete Fracture Network.
Ecolego	Calculation code that is intended to be used for probabilistic radionuclide transport calculations in Project SFR Extension.

EDZ	Excavation Damaged Zone. The rock around a rock excavation where irreversible changes have taken place.
EdZ	Engineering disturbed Zone. The rock beyond the damaged zone (EDZ) where the changes due to rock excavation are reversible.
EMRAS	Environmental Modelling for Radiation Safety. IAEA project.
ENRESA	Empresa Nacional de Residuos Radiactivos, Spain.
EPRI	Electric Power Research Institute, USA.
EQUIP	Evidence from QUaternary Infillings for Palaeohydrology. EU project.
Erica	Environmental risk from ionizing contaminants. EU project.
ESDRED	Engineering Studies and Demonstration of REpository Designs. Euratom's sixth framework programme for nuclear research and training (2002–2006).
FEM	Finite Element Method.
FEP	Feature, Events and Processes.
FHA	Future Human Activities.
FKA	Forsmarks Kraftgrupp AB. Company that operates three nuclear power reactors at Forsmark.
Forge	EU project that runs during the period 2009–2012 in which Lasgit is included.
R&D	Research and Development.
FSW	Friction Stir Welding.
RD&D	Research, Development and Demonstration.
FUTURAE	Assessment and management of the impact of radionuclides on man and the environment. EU project.
GAP	Greenland Analogue Project. A research project on western Greenland for studying hydrological processes at an existing ice sheet.
GEUS	De Nationale Geologiske Undersøgelser for Danmark og Grønland (Geological Survey of Denmark and Greenland).
GIA	Global Isostatic Adjustment.
GPS	Global Positioning System.
Holocene	Current interglacial that began around 11,500 years ago.
IAEA	International Atomic Energy Agency. Agency of UN.
IBECO RWC-BF	Possible material for backfill and pellet fill.
ICRP	International Commission on Radiological Protection. Independent not-for-profit organization that serves as an international advisory body for radiation protection.
INE-FZK	Institut für Nukleare Entsorgungstechnik im Forschungszentrum Karlsruhe, Germany.
IPCC	Intergovernmental Panel on Climate Change.
ISO containers	Containers of sizes standardized by the International Organization for Standardization (ISO) that can be loaded onto railway cars, trucks and ships.
ITT	Isothermal Test. Field test conducted at URL, Canada.
ITU	Institute for Transuranium Elements, Karlsruhe.
IUR	International Union of Radioecology.
JAEA	Japan Atomic Energy Center.
KASAM	Former Swedish name of Swedish National Council for Nuclear Waste.
K_D	Element-specific distribution (partition) coefficient that describes the distribution of elements between solid and aqueous phases.
k_{eff}	Effective neutron multiplication factor. The design-basis requirement with a view to criticality is that k_{eff}, including uncertainties, shall be less than 0.95 during handling and disposal of canisters.
KTB	New type of cask for transport of spent nuclear fuel from Clink to the Spent Fuel Repository.
Lasgit	Large Scale Gas Injection Test. Experiment at Äspö HRL.
LILW	Low- and intermediate-level waste.
LOT	Long Term Test of Buffer Material. Experiment at Äspö HRL.
LTDE-SD	Long Term Diffusion Experiment – Sorption-Diffusion. Experiment at Äspö HRL.
MARFA	Migration Analysis for Radionuclides in the Far Field. Computer code for calculation of radionuclide transport in the far field.
MATLAB	Commercial computer code for mathematical calculations.
MICADO	Model uncertainty for the mechanism of dissolution of spent fuel in a nuclear waste repository. EU project.

MOX	Mixed Oxide Fuel.
MX-80	Sodium bentonite from Wyoming. Possible backfill material.
Nagra	Nationale Genossenschaft für die Lagerung von Radioaktiver Abfälle, Switzerland.
NDA	Nuclear Decommission Authority, UK.
NDT	Nondestructive testing.
NEA	Nuclear Energy Agency. A cooperation organization for nuclear energy matters within the OECD.
NUMO	Waste management organization of Japan.
NWMO	Nuclear Waste Management Organization, Canada.
OECD	Organization for Economic Cooperation and Development.
OKG	OKG Aktiebolag (Oskarshamns kraftgrupp). Company that operates three nuclear power reactors at Oskarshamn.
PADAMOT	Palaeohydrogeological Data Analysis and Model Testing. EU project.
Pandora	SKB's and Posiva's modelling tool for dose calculations in the biosphere.
Posiva	Posiva Oy, Finland.
PROTECT	An evaluation of the practicability and relative merits of different approaches to protection of the environment from radiation. EU project.
PSAR	Preliminary Safety Assessment Report.
PSI	Paul Scherrer Institute, Switzerland.
PSU	Project SFR Extension.
PWR	Pressurized Water Reactor.
RAWRA	Radioactive Waste Repository Authority, Czech Republic.
RH	Relative Humidity.
RNR	Radionuclide Retention Experiment. Experiment at Äspö HRL.
P&T	Partitioning and Transmutation.
SAR	Safety Analysis Report.
SAR-08	Safety analysis report for SFR, delivered by SKB in April 2008.
SEM	Scanning Electron Microscope.
SFL	Final repository for long-lived waste.
SFR	Final repository for short-lived radioactive waste.
SFR 1	Final repository for radioactive operational waste.
SIMFUEL	Uranium dioxide containing non-radioactive fission product elements and metal particles similar to those in spent fuel.
Silo	Repository part for intermediate-level waste in SFR 1.
SR-Can	Report on the long-term safety of the final repository, published by SKB in November 2006.
SR-Site	Assessment of the long-term safety of the final repository.
STF	Safety-related technical specifications.
SSM	Swedish Radiation Safety Authority.
SVAFO	AB SVAFO, which is owned by Ringhals AB, Forsmarks Kraftgrupp AB, OKG AB and Barsebäck Kraft AB, is a member of the Vattenfall Group. Its activities include management and treatment of waste from previous research and development activities at e.g. Studsvik for the purpose of final disposal.
SWIW	Single Well Injection Withdrawal tracer test. Experiment at Äspö HRL.
TBM	Tunnel Boring Machine.
TBT	Temperature Buffer Test. Experiment at Äspö HRL.
TF EBS	Task Force on Engineered Barrier Systems. Experiment at Äspö HRL.
TF GWFTS	Task Force on Modelling of Groundwater Flow and Transport of Solutes. Experiment at Äspö HRL.
THM	Thermal-Hydraulic-Mechanical.
TRUE	Tracer Retention Understanding Experiments. Experiment at Äspö HRL.
URL	Underground Rock Laboratory, Canada.
XANES	X-ray Absorption Near Edge Structures.
XRD	X-Ray Diffraction.
3DEC	Computer code for rock mechanical analyses.