Technical Report

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RD&D Programme 2013

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

September 2013

Svensk Kärnbränslehantering AB
Swedish Nuclear Fuel
and Waste Management Co

Box 250, SE-101 24 Stockholm
Phone +46 8 459 84 00

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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. Under the Nuclear Activities Act, the licensees for the Swedish reactors shall, in consultation, 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 the nuclear power plants. The licensees have delegated responsibility to SKB to act on their behalf and submit the programme to Swedish Radiation Safety Authority (SSM). SKB is now presenting RD&D programme 2013, which has been produced in collaboration with the nuclear power companies to comply with these requirements.

After more than three decades of research, technology development and investigations, SKB applied to the regulatory authorities in 2011 for licences to build and operate a final repository for spent nuclear fuel according to the KBS-3 method in Forsmark and a facility part for encapsulation of spent nuclear fuel adjacent to the interim storage facility for spent nuclear fuel in Oskarshamn. The licensing process is under way and is expected to take several more years. In parallel with this, SKB is conducting the research and technology development needed to design, build and operate the KBS-3 system in a rational manner while complying with the requirements on long-term safety, low radiation dose and safety in the operation of the facilities as well as a good external environment. This means that the schematic solutions presented in the applications must be translated into an industrialized production system that meets the stipulated quality requirements. The research is focused on processes of importance for long-term safety, particularly those factors which the SR-Site safety assessment has shown contribute to the risk of harmful effects and where further research can improve the premises for the assessment.

SKB will soon also submit applications for extension of the final repository for short-lived radioactive waste in Forsmark, SFR, to provide capacity for disposing of additional waste from operation and decommissioning of the nuclear facilities. The applications have been preceded by consultations with concerned stakeholders, a siting study, a site investigation, site modelling and design. An important supporting document is the preparatory preliminary safety analysis report that is now being prepared and will be included in the application under the Nuclear Activities Act. The focus of the continued research regarding the waste and the barriers in SFR will largely be determined by the results of the safety assessment.

The final repository for long-lived low- and intermediate-level waste, SFL, is the repository that is planned to be put into operation last. Work is under way on how to manage and dispose of the long-lived waste, and the programme presents the alternative courses of action that are available as well as the development and research that is required for this work to progress.

Stockholm in September 2013
Svensk Kärnbränslehantering AB

Christopher Eckerberg President	Olle Olsson Vice President

Summary

The Swedish system for management and disposal of radioactive waste is divided into two main parts: one for the low- and intermediate-level waste and one for the spent nuclear fuel. The interim storage facility for spent nuclear fuel, Clab, and the final repository for short-lived radioactive waste, SFR, have been in operation since the 1980s, along with the transportation system. Other facilities or facility parts remain to be built.

In RD&D programme 2013, SKB presents its plans for research, development and demonstration during the period 2014–2019. The programme consists of five parts:

Part I SKB's activities and plan of action.

Part II Low- and intermediate-level waste.

Part III Spent nuclear fuel.

Part IV Research for assessment of long-term safety.

Part V Social science research.

In this RD&D programme, SKB presents the research and technology development that is needed to be able to design, build and operate the planned facilities and facility parts as well as to maintain safe operation of the existing facilities. This work is largely being done under SKB's own auspices, but also in cooperation with universities and institutes of technology all over the world. SKB is also participating in the work of a number of international organizations to resolve the issue of how spent nuclear fuel and radioactive waste can best be managed and disposed of. SKB is also collaborating with sister organizations in other countries, and in 2013, planning was initiated to deepen SKB's collaboration with Posiva in Finland. The goal of the collaboration is to develop common solutions for the final repository system for spent nuclear fuel.

In addition to RD&D programmes, SKB submits other accounts of plans and activities to SSM, for example the applications for the Spent Fuel Repository and the encapsulation plant in 2011, upcoming applications for the extension of SFR, studies and plans for decommissioning of nuclear power reactors and other nuclear facilities, and periodic overall assessments of facilities in operation and cost estimates in the form of Plan reports. In order to avoid redundancy, the content of these accounts is not repeated in this programme, but rather referred to as needed.

The focus in RD&D programme 2010 was on the plans to build and commission the remaining facilities for low- and intermediate-level waste, which includes an extension of SFR and a final repository for long-lived low- and intermediate-level waste. The planning and method development being done in preparation for the decommissioning of the nuclear facilities was also described. The technology development in preparation for the final disposal of the spent nuclear fuel was presented, along with SKB's research in support of the assessments of long-term safety for the Spent Fuel Repository and the extension of SFR.

In its review of RD&D programme 2010, SSM expressed that SKB should present plans and strategies regarding decommissioning of the nuclear power plants, including Ågesta. Further, SSM wanted SKB to present a more in-depth programme for long-lived low- and intermediate-level waste including different alternatives for management and final disposal of this waste. The Government's decision regarding the programme also mentioned the need for such a programme. This decision also contained a condition that the reactor owners should consult with SSM prior to RD&D programme 2013 in matters pertaining to final disposal of long-lived low- and intermediate-level waste as well as plans and studies for decommissioning of nuclear facilities. Consultations were held during 2012 and 2013.

Part I SKB's activities and plan of action

Part I presents the overall plans for operating SKB's existing facilities and realizing the remaining parts of the waste system.

Activities and facilities

The Swedish power industry has been generating electricity by means of nuclear power for more than 40 years. A large part of the system for management and disposal of the radioactive waste and the spent nuclear fuel from the operation of the nuclear power reactors has been built up during this time.

What remains to be done for disposal of the spent nuclear fuel is to build and commission the facilities for encapsulation and final disposal. This work includes building a facility part for encapsulation of the spent nuclear fuel at Clab, developing transport casks for shipping canisters of spent nuclear fuel, and building a final repository where the canisters will be deposited. For disposal of the low- and intermediate-level waste, SFR will be extended, containers will be developed for transportation of long-lived waste, and eventually a final repository for long-lived waste, SFL, will also be built.

Existing systems and facilities must be continuously maintained and modernized. This is necessary in part because the planned operating times for the Swedish reactors has been prolonged, which means that the operating time for the waste system must be prolonged accordingly.

Much of the research and development for final disposal of nuclear waste and encapsulation 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 Hard Rock Laboratory, the Canister Laboratory and the Bentonite Laboratory. These laboratories also provide an opportunity to demonstrate that the final repositories' barriers can be fabricated and installed with the quality required to meet the requirements on long-term safety.

Plan of action

The reactors in Forsmark and Oskarshamn are planned to be operated for 60 years, as are the Ringhals 3 and 4 reactors. The planned operating time for Ringhals 1 and 2 is 50 years. These times are important premises in SKB's planning. Assuming that all the reactors have been taken out of service by 2045 and decommissioned by around 2052, SKB's three final repositories (the Spent Fuel Repository, SFR and SFL) can be closed in about 60 years.

The plan of action describes in general terms the measures needed to meet SKB's obligations and when applications and other legally mandated reports for the facilities are planned to be submitted.

The process of construct and commission a new facility or facility part undergoes different phases. In March 2011, SKB submitted 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. An application under the Nuclear Activities Act for the encapsulation plant was submitted in 2006. It was amended in 2009 regarding integration of the encapsulation plant and Clab to a single facility, Clink, and in 2011 regarding parts dealing with the KBS-3 system. The licensing is under way, and SKB regularly answers questions and requests for supplementary information from SSM. The estimated start of construction for the Spent Fuel Repository is 2019 and for the encapsulation part of Clink 2021. These facilities will then be able to be commissioned simultaneously in 2029.

Today SKB has a licence to store 8,000 tonnes of fuel in Clab. This storage capacity can be increased by the use of compact storage canisters. SKB plans to apply for a licence for this purpose in 2018. Clab has been in operation for about 30 years, and certain system upgrades and component replacements will be necessary in the future. A system inventory has been initiated to investigate the need for improvements.

In the spring of 2014, SKB plans to apply for licences under the Nuclear Activities Act and the Environmental Code to extend SFR. The extended repository is expected to be ready for operation in 2023 to meet the needs of the nuclear power industry to dispose of nuclear waste from operation and decommissioning of the nuclear power reactors.

SKB plans to apply for a licence to build the next final repository, SFL, in around 2030. According to current plans, the facility will be put into routine operation in about 2045.

The relatively long time horizon covered by SKB's planning means that the planning premises may change in the meantime. SKB's planning must be able to handle such changes. In many cases, this is expected to require only minor modifications, without any major changes in the long-term plan.

Part II Low- and intermediate-level waste

Part II covers both current and planned management and disposal of low- and intermediate-level waste, i.e. waste arising in connection with operation and decommissioning of the nuclear facilities in Sweden and other radioactive waste from industry, research and medical care. In this work, experience from the operation of SFR constitutes an important knowledge base for the development and construction of new repositories.

Management of low- and intermediate-level waste

SFR is a hard rock facility beneath the sea in Forsmark with rock cover of about 60 metres. The facilityis reached via access tunnels from the ground surface. SFR has been in operation since 1988 and has been operated by SKB since 2009. When SFR was built, the intention was that the facility would be in operation up until 2010, when the nuclear power plants were to be shut down according to plans at that time. Since then the planned operating times for the nuclear power plants have been extended and SKB is pursuing a programme for remedial and preventive maintenance in order to ensure reliable operation, personal safety and facility availability.

Before the short-lived low- and intermediate-level waste is transported to SFR, it undergoes treatment by the waste producers. Besides packaging the waste, other purposes of this treatment can be to reduce its volume or modify its physical and chemical properties. The waste producers work constantly to reduce waste volumes by reducing the inflow of material on the controlled side and by clearance and source-separation of materials. Improved technology for cleaning air and process water helps to reduce radionuclide releases to the environs.

Long-lived low- and intermediate-level waste is interim-stored today either wet in pools at Clab or at the NPPs, or dry in interim storage facilities at the NPPs. In addition, long-lived waste is stored in an interim storage facility on the Studsvik industrial site. This facility contains legacy waste that is managed by AB SVAFO and waste from Studsvik Nuclear AB's activities. The prospects for establishing a central interim storage facility for long-lived waste will be studied.

Extension of final repository for short-lived radioactive waste

SKB plans to extend SFR to obtain sufficient capacity to dispose of future short-lived low- and intermediate-level waste from operation and decommissioning of the nuclear facilities. The planned extension will increase the capacity of the facility and provide room for nine BWR-reactor pressure vessels. The estimated quantity of waste to be deposited in SFR is surrounded by numerous uncertainties. Some are difficult to foresee, such as changes in laws and political decisions, changed operating conditions or changed operating times for the nuclear power plants. The SFR extension will also be designed to enable interim storage of long-lived waste, such as core components from the NPPs.

In preparation for the applications for the extension of SFR, a siting study has been conducted. It shows that an extension of the existing SFR will allow the purpose to be achieved with minimum damage and detriment to human health and the environment. The site has been investigated, and a site descriptive model and a layout have been developed. The extension will be built at a greater depth than the existing facility. The main reason for this is to avoid water-conducting structures near the existing part of SFR.

The design of the engineered barriers in the extension of SFR will differ from the design of the barriers in the existing facility.

Repository for long-lived waste

SFL will be the smallest of SKB's final repositories and is planned to be the last to be commissioned. Several important milestones must be passed before SFL can be put into regular operation, such as choice of repository concept and site, investigations of the site, evaluation of long-term safety, design, preparation of applications with associated licensing, and construction of the facility.

The work of selecting a final repository concept for SFL has begun. The work is essentially following the methodology SKB has used earlier for system analysis of different solutions for disposal of spent nuclear fuel. The study, which will be published at the end of 2013, shows that SFL will be based on geological disposal. In this connection, different alternatives for handling and conditioning of the waste, suitable waste packages and engineered barriers will be studied. These studies are also linked to the technology development planned for SFL. Since RD&D programme 2010, SKB has held consultations with SSM concerning long-lived low- and intermediate-level waste.

Near-surface repositories

Today there are near-surface repositories for very low-level operational waste in the industrial areas at the Forsmark, Oskarshamn and Ringhals nuclear power plants. There is also a near-surface repository at Studsvik. Approximately half of the short-lived very low-level waste from decommissioning of the NPPs could be disposed of in near-surface repositories instead of in SFR. SKB is therefore studying the possibility of building a central near-surface repository.

Technology development for final disposal of low- and intermediate-level waste

The technology development SKB is conducting in preparation for the extension of SFR is largely focused on selection of barriers and materials as well as development of methods for fabricating and building them. Experience from SFR, which has been in operation for about 25 years, constitutes an important point of departure for this. A review of the status of the barriers in the rock vault for intermediate-level waste (BMA) shows that there is damage to the concrete barriers that has been caused by corrosion of reinforcement bars or stems from cooling after casting of the barriers. SKB will develop recipes and methods for casting of concrete barriers without reinforcement. The damage to BMA that has been identified will be repaired.

Technology development for SFL includes methods and solutions for conditioning of waste, waste containers, and engineered barriers for the final repository. Concepts for a number of waste containers for SFL waste have been developed to permit efficient handling. In the concept study for SFL, SKB has developed four different barrier solutions. Three are individual engineered barriers of concrete, bentonite and crushed rock/gravel, respectively, and the fourth is a combination of the three. In the study, which will be presented at the end of 2013, SKB will specify which concept or concepts will be included in a planned safety evaluation. The technology development for SFL will focus on the selected concept or concepts.

Decommissioning of nuclear facilities

According to the Nuclear Activities Act (1984:3) the holder of a licence for nuclear activities is responsible for safely decommissioning the licensed facilities when the nuclear activity is discontinued. Decommissioning shall be described in plans where the degree of detail in the account increases as the facility in question approaches the time for decommissioning.

Decommissioning includes defuelling, shutdown operation and dismantling and demolition. Activities needed for clearing the facility are pursued during dismantling and demolition. The first commercial reactors in Sweden that will be decommissioned are the reactors in Barsebäck and Ågesta.

SKB has been given the assignment of planning and executing the decommissioning of the nuclear power plants. SKB coordinates the planning so that general methods and procedures are used for the decommissioning work, which also includes calculations of waste volumes, nuclide inventory and costs.

The studies for the decommissioning of all Swedish nuclear power plants, including Ågesta, have now been finished. They give estimates of the quantity of waste and the costs for decommissioning of each plant. Since RD&D programme 2010, the licensees for the reactors have, together with SKB, held consultations with SSM on the subject of decommissioning.

Part III Spent nuclear fuel

An important task for SKB is safe management and disposal of the spent nuclear fuel. The fuel is transported from the nuclear power plants by ship to Clab in Oskarshamn, where it is interim-stored awaiting final disposal. To permit final disposal of the spent nuclear fuel, an encapsulation part must be built adjacent to the interim storage facility (Clab) and a final repository for spent nuclear fuel must be built in Forsmark. Part III of this RD&D programme focuses on the technology development that is needed to design, construct and operate the final repository system for spent nuclear fuel.

Technology development

As is evident from the applications submitted for Clink and the Spent Fuel Repository, a reference design has been adopted for the repository barriers for long-term safety that fulfils the design premises for the KBS-3 system. At the same time, a feasible way towards production and an inspection programme has been presented.

The structure presented by SKB in previous RD&D programmes, where technology development is divided into a number of production lines, has been retained and refined in RD&D programme 2013. This means that the development work for the barriers for long-term safety is being pursued in production lines for fuel, canister, buffer, backfill, closure and underground openings. The production lines for buffer, backfill and closure share a number of common issues, so the development work for these production lines is integrated. Furthermore, technical systems are being developed for e.g. logistics and machines that are unique for the final disposal facility.

Continued technology development is being pursued to proceed from schematic solutions to solutions that are tailored to an industrialized process with stipulated requirements on quality, cost and time. A large part of the remaining development work consists of building up a production system with quality control.

The delivery control model presented by SKB in RD&D programme 2010 has been applied and developed to control the research, development and demonstration work that is needed to commission the facilities for encapsulation and final disposal of spent nuclear fuel. According to the delivery control model, technology development is subdivided into four phases: concept phase, design phase, implementation phase and operation and maintenance phase. For each phase there is a specification of what must have been achieved and what is needed as a basis for a decision to proceed to the next phase.

Fuel handling

Research activities are being pursued to progressively improve the state of knowledge regarding the spent nuclear fuel and its properties. This is being done for the needs of the entire KBS-3 system. The work includes both calculations of source term and radiation characteristics. It also includes reporting radiological properties for fuel with different burnups and decay times. A programme is being carried out for the interim storage facility for spent nuclear to measure decay heat, which influences several parts of the KBS-3 system. In the continued work, criticality analyses for spent fuel in SKB's present-day and future facilities will be performed and the chosen method for drying the fuel before it is encapsulated will be evaluated.

In addition, a plan will be presented for how fuel that requires special handling is to be managed. This includes damaged fuel and odd types of fuel.

Canister

Reference methods have been selected for fabrication of the canister's components and welding of lids and bottoms. The methods enable canisters to be produced that conform to requirements on reproducibility and industrial production.

Development of the canister is now in the design phase. Together with Posiva, SKB is planning efforts aimed at determining the canister's reference design. The work of analyzing the canister's mechanical strength is continuing with calculations of different load cases the canister may be exposed to in the final repository and during handling in the facilities.

SKB's production system for canisters consists of a number of external suppliers of the canister's various components. SKB intends to build a canister factory where the components are assembled and the properties of the canisters are inspected. SKB is collaborating with Posiva in this matter, since a common canister factory may permit synergy gains.

Buffer, backfill and closure

When the applications were submitted in 2011, development of buffer and backfill had passed the concept phase. Since then, technology development has been further pursued in preparation for system design for buffer, backfill and plugs in deposition tunnels. Development of the closure has not come as far, but efforts are being made to finish the work in the concept phase.

Detailed design for buffer, backfill and plugs will commence during the coming RD&D period. Furthermore, studies are planned to finalize the production system, with its inspections, for the manufacturing of buffer and backfill blocks and pellets. In addition, full-scale installation tests of both buffer and backfill are planned.

The outer section of the Prototype Repository in the Äspö HRL has been dismantled since RD&D programme 2010. The backfill as well as the buffer and the canisters in two deposition holes has been retrieved. The results of field work and laboratory programmes are currently being compiled and evaluated, and the preliminary conclusion is that the barriers have evolved more or less as expected.

Rock

Technology development for rock includes detailed characterization, design, construction and maintenance of the Spent Fuel Repository's underground openings. The development work also includes methods for investigations and modelling. It also includes rock construction, including sealing and rock support measures, as well as development of special equipment with a focus on the rock conditions prevailing in Forsmark.

Since RD&D programme 2010, SKB has developed a framework programme for detailed characterization. The Äspö HRL has been extended during the period with new tunnels, which has made it possible to integrate detailed characterization and design. Here it has also been possible to apply the Observational Method to the rock excavation work. Methods for detailed characterization will also be tested in the new tunnels. Follow-up inspections will determine how well the tunnelling requirements have managed to minimize the excavation-damaged zone, especially in the floor.

The methodology for underground design and the application of the Observational Method need to be further developed. This applies mainly to strategies for detailed adaptation of deposition areas and coordination of detailed characterization and building production.

Methods and equipment for detailed characterization with associated modelling will also be developed. This development work will be carried out in a first step prior to construction of the Spent Fuel Repository's access tunnels and in a second step prior to construction of the deposition areas.

Technical systems

Many of the machines, vehicles and technical systems that will be used in the Spent Fuel Repository and Clink for handling and transport of canisters, buffer and backfill material, etc are specific for SKB's activities and cannot be bought on the open market. They must have high operational availability and be able to function together with other machines and technical systems and be user-friendly for the personnel.

Technology development of SKB's specific systems proceeds incrementally and takes many years, which means it needs to start long before the equipment has to be in place. Prototypes are needed in some cases for verification and validation of the systems. Parts of today's technology development are therefore concerned with development and design of prototypes. Examples of this are the prototype of the deposition machine that was further developed and tested in the Äspö HRL and the stacking robot that was developed to install backfill blocks in the deposition tunnels.

Horizontal deposition – KBS-3H

Together with Posiva, SKB is studying whether horizontal deposition of copper canisters (KBS-3H) could constitute an alternative to vertical deposition, which is SKB's reference design. Current project will continue until 2016, after which an evaluation of KBS-3H will be performed. Decisions and planning for continued development of KBS-3H will follow this evaluation.

Part IV Research for assessment of long-term safety

Part IV describes SKB's research programme in natural science, which is being pursued based on the need to assess safety in connection with the final disposal of radioactive waste and spent nuclear fuel. The research covers both general issues that are shared by the different repository systems and issues related to a specific repository system. This means that the research programme is designed to cover the needs of the Spent Fuel Repository, SFR and eventually SFL as well.

The current level of scientific understanding of issues relevant to long-term safety is quite advanced. This is due to decades of research in the Swedish programme, but also in other countries' national programmes and joint international projects.

The research on the evolution and safety of the Spent Fuel Repository has led to an understanding of important processes such as copper corrosion, shearing of canisters and other potential causes of canister failure, as well as of processes related to radionuclide retardation. In the SR-Site safety assessment, SKB showed that it is possible to build a final repository for spent nuclear fuel that meets SSM's demands on long-term safety. Research on processes crucial to safety continues, and in particular research on factors which SR-Site has shown contribute to risk and where further research can clarify the assumptions for the assessment.

As far as short-lived low- and intermediate-level waste is concerned, SKB is in the midst of an evaluation phase for SFR. Continued research for this waste will largely be guided by the results of the ongoing safety assessment for SFR, SR-PSU. The work for the long-lived low- and intermediate-level waste is in the concept stage. Efforts that specifically concern the long-lived waste will be progressively concretized, even though quite a few processes will probably resemble those that occur in SFR and can therefore be co-studied.

Safety assessment

The safety assessment is the instrument that is used by both SKB and SSM to determine whether a repository for radioactive waste conforms to the regulatory requirements on long-term safety. The assessment methodology has been developed over a long time, in parallel with the development of the KBS-3 system. In recent years, this development has largely been pursued and reported in conjunction with the execution of the safety assessments, most recently SR-Site.

The development needs during the coming RD&D period pertain to supplementary development of methods for sensitivity analyses and radionuclide transport, and further development of quality assurance of SKB's safety assessments. In addition, preparedness exists to handle methodology-related viewpoints that may be expressed in conjunction with SSM's ongoing review of the SR-Site safety assessment.

Climate evolution

SKB's climate research touches upon and affects several other research areas. Climate-related processes such as the growth of ice sheets and permafrost can affect sea levels, groundwater flows, stresses in the Earth's crust and, not least, living conditions at the ground surface.

The focus is not only on global warming or determining exactly when the next ice age will occur but also on describing a number of possible climate evolutions. Together they are aimed at reducing uncertainty regarding climate change in a very long time perspective. The climate cases constitute examples of possible climate evolutions and of extreme limiting cases for the analysis of different functions of the repository. The needs of the different repositories differ somewhat due to the fact that they are located at different depths, have barriers of different materials and have analysis periods of different lengths.

For the development of the Spent Fuel Repository it is especially important to study processes that could affect the buffer and the canister up to a million years in the future. Future efforts will mainly be aimed at obtaining more information on historic variations of the ice sheets. The purpose is to be able to set limits on how thick future ice sheets could become. Important questions also have to do with the evolution of the repository during an ice age and what the hydrogeological conditions look like beneath an ice sheet. To improve knowledge concerning these matters, SKB, together with the sister organizations in Finland and Canada, is pursuing a major research project on Greenland called the Greenland Analogue Project.

The expected climate evolution during the next ten thousand years is of greater importance for the assessment of the long-term safety of SFR. Here it is important to get an idea of, for example, when permafrost might form at the earliest and how deep it could reach during the assessment period.

Short-lived low- and intermediate-level waste

The short-lived low- and intermediate-level waste is highly varied and complex in nature. The properties of the waste at closure are dependent not only on the waste materials, but also on the waste containers in which the waste is placed and how it has been conditioned. The properties at the time of closure – the initial state – serve as a starting point for the safety assessment. To ensure the initial states for the different types of waste in the safety assessments for SFR, their content is regulated by acceptance criteria and type descriptions.

When the repository is closed and becomes water-saturated, the waste will interact with the penetrating groundwater and various processes will be initiated. A report dealing with these processes has been compiled since RD&D programme 2010. Furthermore, an updated estimate of the total material and radionuclide inventory at closure of SFR, which, besides operational waste, also includes waste from dismantling and demolition of the Swedish nuclear facilities. Continued research on short-lived low- and intermediate-level waste will largely be guided by the results of the safety assessment currently being done in SR-PSU.

Long-lived low- and intermediate-level waste

SFL is in a concept stage, and SKB is planning for a repository with a retarding safety function. In this way, many of the components and materials used in SFR will probably be found in SFL as well. Groundwater will seep in after closure and interact with the waste. This means that the processes that are expected to occur in SFL will for the most part resemble those that occur in SFR, which in turn means that much of the research can be coordinated with equivalent programmes for the short-lived waste. Specific research needs for long-lived low- and intermediate-level waste will be identified as the design of SFL and its inventory is concretized.

Concrete barriers

Concrete barriers with the main function of retarding releases of radionuclides are installed in SFR and will probably also be found in SFL. The retardation is achieved because the radionuclides sorb on the concrete and the concrete structures reduce the water flow through the rock vaults and the waste. These functions are best achieved by a fracture-free concrete barrier. The research being done on concrete barriers is therefore aimed at developing recipes and methods for concrete casting and increasing our understanding of degradation of concrete.

Fuel

As long as the copper canister containing the fuel in the Spent Fuel Repository is intact, no radioactive substances can escape from the repository. If a hole should occur in a canister and water should enter the canister, the properties of the fuel are crucial for determining if and when radionuclides can escape from the canister into the rock. The fuel, which generally dissolves very slowly in the water present in the repository environment, can retard any dispersion of radionuclides. Research on fuel dissolution is therefore an area for further research by SKB. For example, research activities are planned to gather data on fuel dissolution under repository-like conditions and to clarify the mechanisms of the different processes that contribute to fuel dissolution.

During their time in the reactor, certain radionuclides segregate to the surface of the fuel pellets and are thereby less tightly bound to the fuel. In the event of a canister breach, these substances dissolve faster than the fuel matrix and determine the initial pulse of radionuclides that is released and possibly transported to the ground surface. There are indications that high-burnup fuel contains a larger fraction of loosely bound radionuclides, which has occasioned further research into the properties of high-burnup fuels. Research is also being conducted to investigate new types of fuels, for example fuels to which chromium has been added to optimize the fuel's performance in the reactor. Investigations of these new fuel types are largely being undertaken in international cooperation.

Canister

The copper canister is the barrier in the KBS-3 repository with a containment function. In the reference scenario in SR-Site, the canister remains intact throughout the analysis period. There are two factors that could cause penetrating breaches in the canister: mechanical loads and copper corrosion. Central questions related to mechanical loads have to do with how large isostatic loads the canister can tolerate and how large shear movements in the rock the canister can withstand. The creep properties of copper are an important topic of research in this context.

The main research efforts are being made in the area of copper corrosion. There are, for example, phenomena whose underlying processes need to be better understood in order to determine if and how these processes can affect the canister's durability. Experiments are, for example, being conducted on copper in oxygen-free water where hydrogen gas is generated in quantities much greater than predicted by thermodynamic data.

At the same time, efforts to reduce uncertainties surrounding processes that can affect the long-term durability of the canister are continuing. An example is work on sulphide corrosion, which in SR-Site was the corrosion process that made the greatest contribution to risk and is thereby also included as an important subject of continued research. Stress corrosion cracking is another process on which further research is needed.

Buffer and backfill

The canister in the Spent Fuel Repository is surrounded by a protective buffer of bentonite clay. The function of the clay is to limit the groundwater flow into the canister. If a canister is breached, the buffer will also retard the transport of radioactive particles out into the rock.

In conjunction with the safety assessment work, it has emerged that the buffer might under certain conditions dissolve and be carried out into fractures in the rock, a phenomenon known as buffer erosion. This could happen if large quantities of meltwater with low calcium content from an ice sheet penetrate down to the repository. The biggest research efforts therefore have to do with clarifying under what conditions the clay is stable, which is being done in an extensive experimental programme both in SKB's own projects and in large joint projects within the EU.

Important efforts are also being made to develop a programme for characterization of clay. Important progress has been made here as a result of the experiments with alternative buffer materials that are being conducted in the Äspö HRL. An example is an experiment aimed at understanding the mechanisms behind changes in the clay, i.e. homogenization. New models have been developed to better describe this process in the clay, and development efforts in the area are continuing.

Bentonite clay will also be used to backfill the deposition tunnels in the Spent Fuel Repository. The research being conducted on the buffer also covers the needs that arise in connection with the development work for the backfill.

The research on buffer-related issues in the Spent Fuel Repository is largely applicable to conditions in SFR and will also be applicable to SFL. Some efforts are also aimed at gaining a better understanding of how bentonite clay and concrete interact.

Geosphere

Research on the geosphere embraces four disciplines: geology, hydrogeology, hydrogeochemistry and transport properties of the rock. Future research in these areas will be aimed at broadening the knowledge base concerning rock conditions of great importance for the outcome of the safety assessments. SR-Site clearly shows which properties and processes are of the greatest importance for the Spent Fuel Repository. The aim is to obtain greater knowledge of how the properties vary in the rock volume and as a function of the rock types that are present. SKB is also improving the modelling tools used to describe and predict the processes that take place in the rock over a very long time.

Research efforts in the discipline of geology will focus on gaining a better understanding of spalling caused by stresses in the rock and by high temperature, methodology for identification of large fractures, further studies of glacially induced faults, seismic measurements to support earthquake modelling and general increased knowledge regarding seismicity in Swedish bedrock. Discrete fracture network models, which serve as a basis for analysis of rock movements and solute transport in fractures, is a major field of research with a bearing on hydrogeology, geology and rock mechanics.

Data on the hydrogeology of surface systems is mainly being gathered in the investigations on Greenland. The goal is to be able to describe how the flow changes and varies during a period dominated by permafrost. The results are being used, for example, in the safety assessment for the extension of SFR. Hydrogeological research on deep groundwater systems involves integrating data and models with other disciplines (geochemistry, rock mechanics and transport) and maintaining and improving the codes that are used for flow and transport calculations. Flow conditions during a glaciation with extensive permafrost have consequences for the situation at repository depth. Investigation data from the project on Greenland are being used both to understand the flow processes and optimize the modelling tools.

Research in the field of geochemistry continues to focus on reactions between water and rock and effects of movements of the water in the fracture system. Efforts are being focused on models where geochemical conditions are integrated with hydrogeological conditions and the transport properties of the rock. Microbial processes are also an increasingly important topic. Examples are the importance of acetogens (acetate-generating microorganisms), the interaction between microbes and viruses in the rock, biofilms on fracture surfaces and microbial processes in the presence of hydrogen, methane and sulphide.

The programme examining how solutes are transported in groundwater includes studies of flow-related transport resistance, channelling (width and frequency) and diffusion in stagnant zones. There are also plans to improve the K_d concept for calculating distribution coefficients where the hydrogeochemical conditions in fractures are taken into account. The purpose is to be able to use the newfound knowledge in future safety assessments. International research is being pursued within the SKB Task Force for Groundwater Flow and Transport of Solutes. Modelling tasks are being carried out by several groups from different organizations using data from experiments and investigations at the Äspö HRL or other hard rock laboratories.

Surface ecosystems

Research and development in the area of surface ecosystems is mainly aimed at gaining a better understanding of those processes that affect transport and accumulation of radionuclides in the surface systems and at developing the methodology for calculating and assessing the radiological risk to human health and the environment. There will be an emphasis on questions and uncertainties that have been identified in the work with the safety assessments for the Spent Fuel Repository and the extension of SFR. Questions identified during preparations for the assessment of the future repository for long-lived radioactive waste, SFL, will also be investigated.

Studies conducted so far show that if radionuclides from a repository should get out into the rock and be transported up to the ground surface, they will eventually end up in low points in the landscape. If this occurs to terrestrial environments, the receiving area will in all probability be a wetland or agricultural land. An important area for future research efforts in the field of surface ecosystems is therefore processes that control transport and accumulation of radionuclides in wetlands and agricultural lands. It is also important to gain a better understanding of processes that control the transport of radionuclides from terrestrial areas to lakes and watercourses.

The coastal area in Forsmark is changing over time as a function of climate variation and shoreline displacement, which affects the possible exposure of man and the environment to radioactive substances. In order to assess the consequences if radionuclides should escape from a repository, knowledge is needed of both landscape evolution and how important processes change when the climate changes. Activities aimed at refining the description of the landscape and its evolution are therefore planned within the programme for surface ecosystems. In order to gain a better understanding of how processes will change when the climate in Forsmark gets colder, SKB will continue its work to describe periglacial environments on Greenland.

When it comes to refinement of calculation methodology, efforts are planned to replace or supplement the use of concentration factors with mechanistic models of retention and biological uptake in order to reduce the uncertainties. SKB will also continue its development of the methodology for dose calculations based on newfound knowledge from the biosphere programme and from applications of the methodology to completed and ongoing safety assessments.

Other methods

SKB is following the development of other methods besides the KBS-3 method for management and disposal of spent nuclear fuel. Such alternative methods include disposal at great depths in boreholes, but also utilizing the energy in the fuel in new reactors by means of partitioning and transmutation (P&T).

Part V Social science research

Part V describes social science research in two chapters:

- SKB's programme for social science research.
- Information preservation across generations.

SKB's programme for social science research

SKB carried out a programme for social science research during the period 2004–2011. The purpose has been to broaden the perspective on the societal aspects of the nuclear fuel programme and to provide deeper knowledge and a better body of data as a basis for site- and project-related studies and analyses. A total of 18 projects have been carried out within the four research areas included in the social science research programme: Socioeconomic impact, decision processes, public opinion and attitudes – psychosocial effects, and global changes.

SKB believes that the research results have contributed to a deeper understanding of historical, economic and public opinion aspects of issues related to the final disposal of nuclear waste and have thereby contributed to broadening the general knowledge base for SKB's work. The results have also been put to use in SKB's practical work, and they are also relevant to SKB's work with final disposal of the low- and intermediate-level waste.

In SKB's view, further research in this area should primarily be funded in the manner that is customary in the academic world, i.e. by having researchers apply for funds to various research funding bodies, for example state and private research councils or the like. SKB therefore does not intend at present to initiate a new research programme of the type that has now been concluded.

Information preservation across generations

There are two fundamental principles for how information can be passed on to future generations: successive information transfer, and information transfer aimed directly at a distant future. All the countries that are working with long-term information preservation are focusing on successive transfer. This entails transferral of information between generations via, for example, archives. Some countries are also working with information transfer directly to a distant future, involving the use of markers.

Regardless of who the future recipients are, it is important that the information be accessible and comprehensible.

The important questions regarding information preservation do not need to be resolved until the final repository for spent nuclear fuel is to be closed. At the central level, the question has to do with how information and knowledge of a final repository for radioactive waste can be preserved for future generations. The overall goal of SKB's current work is therefore to find ways and means to continue to keep the question alive and updated. SKB's current efforts include its own research projects, participation in an OECD-NEA project and a joint project with Andra, SKB's French counterpart. The goal of SKB's own research is to learn more about how information and knowledge can be preserved for a long time. This is being done by studies of how present-day knowledge of historical and ancient phenomena has been acquired as well as how language develops and changes over time.

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Part IV Research for assessment of long-term safety

Part I

SKB’s activities and plan of action

1 Management of radioactive waste and spent nuclear fuel

The Swedish power industry has been generating electricity by means of nuclear power for more than 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 Sigrid and containers 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 building a facility part for encapsulation of the spent nuclear fuel at Clab, procuring transport casks for canisters of spent nuclear fuel, and building a final repository where the canisters will be deposited. In addition to these facilities, SKB plans to build a canister factory.

For the low- and intermediate-level waste it will be necessary to add an extension to SFR, a repository for long-lived radioactive waste, (SFL) and casks for shipments of long- lived waste.

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

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 – now and in the future.

1.1 Premises

1.1.1 Relevant regulatory framework and SKB's mission

Under the Act (1984:3) on Nuclear Activities (the Nuclear Activities Act), the holder of a licence for nuclear activities is responsible for ensuring the safe management and final disposal of radioactive waste and spent nuclear fuel arising in the activities. The licensees are also responsible for safely decommissioning the licensed facilities when the nuclear activity is discontinued. The licensees for the nuclear power plants in Forsmark, Oskarshamn, Ringhals and Barsebäck are Forsmarks Kraftgrupp AB, OKG Aktiebolag, Ringhals AB and Barsebäck Kraft AB. These companies are referred to below as the nuclear power companies.

On behalf of its owners, Svensk Kärnbränslehantering AB[1], SKB, is responsible for management of the radioactive waste and the spent fuel from the Swedish nuclear power plants (NPPs). SKB's responsibility begins when the waste leaves the NPPs. For this purpose, SKB owns and operates a transportation system and facilities for waste management, see Section 1.5. SKB is owned by Vattenfall AB, OKG Aktiebolag, Forsmarks Kraftgrupp AB and E.ON Kärnkraft Sverige AB.

The nuclear power companies are responsible for decommissioning and demolition their nuclear power plants. In this context, SKB has been contracted by the nuclear power companies to participate in the planning and execution of the future dismantling and demolition. SKB's participation primarily entails coordination of general methods and procedures including calculation of volumes, radionuclide inventory and costs for dismantling and demolition.

Under the Nuclear Activities Act, the nuclear power companies, working in consultation, shall 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 the nuclear power plants. Such a programme (RD&D programme) shall be submitted every three years to the Swedish Radiation Safety Authority SSM). The programmes are reviewed and evaluated

[1] Swedish Nuclear Fuel and Waste Management Co.

by SSM after extensive referral for consideration and comment. They are also reviewed by Swedish National Council for Nuclear Waste. SSM and the Council submit viewpoints to the Government, which stipulates conditions for the continued research and development programme.

It is SKB who, on behalf of and in cooperation with the nuclear power companies, prepares the RD&D programmes and submits them to SSM, see Section 1.2.

Under the Nuclear Activities Act, 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. According to the Act (2006:647) on Financial Measures for the Management of Waste Products from Nuclear Activities (the Financing Act), the nuclear power companies are obligated to pay a fee for future waste management and decommissioning. On behalf of the nuclear power companies and pursuant to the Financing Act, SKB prepares a cost calculation every three years, see Section 1.4. Paid-in fees are managed by the state Nuclear Waste Fund.

In addition to the radioactive waste SKB receives from the nuclear power companies, SKB also receives some radioactive waste from other companies. This is regulated by commercial agreements between SKB and the respective companies.

1.1.2 Fundamental principles

The management of radioactive substances is regulated by legislation. The orientation 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 reactors shall be disposed of within Sweden's borders with the consent of the concerned municipalities.
- Sweden shall not dispose of waste from other countries.
- The spent nuclear fuel shall not be reprocessed.
- The final repository shall be established by the generations that have derived benefit from Swedish nuclear power.

SKB plans for geological disposal of the 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 spent nuclear fuel into space, disposing of it beneath the seabed or burying it in the continental ice sheet. Most countries and organizations, such as the IAEA and the OECD-NEA, are agreed today that geological disposal is a solution that satisfies all requirements on safe final disposal and feasibility. Moreover, geological disposal is supported by the EU's waste directive.

The following principles underlie the design of SKB's final repository:

- 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 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 requirements on the barriers in the final repository for short-lived radioactive waste are less stringent than those on the repositories for spent nuclear fuel and for long-lived radioactive waste.

The above principles, along with a number of other considerations, such as that construction of the repository must be technically feasible, have led SKB to choose the KBS-3 method for final disposal of spent nuclear fuel. In an evaluation of different strategies and systems for disposal of spent nuclear fuel (SKB 2010a), SKB explains the background and reasons for the choice of the KBS-3 method. The evaluation was done against stipulated requirements, both overall societal requirements and environmental, safety and radiation protection requirements.

One of the prerequisites for building new facilities is that they conform to the design premises and comply with the requirements stipulated by SKB, the regulatory authorities and other stakeholders. To ensure this, SKB employs systematic requirements management (Morén and Wikström 2007).

1.1.3 The KBS-3 method

SKB's method for final disposal of spent nuclear fuel is called the KBS-3 method, since it is based on the third report in the KärnBränsleSäkerhet (Nuclear Fuel Safety) Project. Development of the method began in the late 1970s and has come so far today that the work is mainly focused on adapting the technical solutions to an industrialized process. The KBS-3 method is characterized by the following:

- the spent nuclear is encapsulated in leaktight, load-bearing canisters that are resistant to corrosion,
- the canisters are deposited in crystalline rock at a depth of 400–700 metres,
- the canisters are surrounded by a buffer that protects them and prevents water flow,
- the openings in the rock that are required for deposition are backfilled and sealed.

Internationally, the KBS-3 method is the method for final disposal of spent nuclear fuel that is the most developed.

In March 2011, SKB submitted an application under the Nuclear Activities Act (SKB 2011a) for final disposal of spent nuclear fuel and an application under the Environmental Code for the KBS-3 system (SKB 2011b). An application under the Nuclear Activities Act for the encapsulation plant was submitted in 2006 and supplemented first in 2009 and thereafter in conjunction with the applications in March 2011 (SKB 2011c).

Finland has also chosen to build its final repository according to the KBS-3 method. In 2000, the Finnish parliament passed a resolution determining the method and site for the final repository. At the end of 2012, SKB's Finnish sister organization Posiva Oy submitted an application for a licence for construction of an encapsulation and final disposal facility for spent nuclear fuel according to the KBS-3 method. The facility is planned to be built at Olkiluoto in Eurajoki. In 2013, SKB and Posiva began planning an in-depth collaboration with the goal of developing common technical solutions for the final repository system.

The KBS-3 method is being considered as a final disposal method in numerous other countries as well, including Canada, South Korea, the UK and the Czech Republic.

1.1.4 The radioactive waste and the spent nuclear fuel

The plans for disposal of the radioactive waste and the spent nuclear fuel are determined to a great extent by the properties of the waste. The waste is divided into categories according to its level of radioactivity (low-, intermediate- or 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 high-level spent nuclear fuel require radiation-shielded handling, while 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 planning is based on the waste from the current nuclear power programme. The waste quantities are dependent on the reactors' operating time as well as availability and other operating conditions. Estimated quantities of radioactive waste and spent nuclear fuel are based on the forecasts of the nuclear power companies. Assumptions regarding the future operation of the reactors directly influence the capacity and operating time of SKB's different facilities. The long-term planning for the waste system is based on the nuclear power companies' current planning premises. Current plans call for the reactors in Forsmark and Oskarshamn to be operated for 60 years. The plan for Ringhals is that reactors 1 and 2 will be operated for 50 years, while reactors 3 and 4 will be operated for 60 years. A decision was made in 2013 to extend the planned operating time for the reactors in Forsmark and for Ringhals 3 and 4 from 50 to 60 years' operation. The quantities to which this scenario gives rise are reported below. The influence of other assumptions for the nuclear power programme is discussed in Chapter 3.

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[2] 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.

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

Short-lived waste is deposited today in SFR or near-surface repositories. The near-surface repositories, where waste with very low-level activity is deposited, are operated by the waste producers, while SFR is operated by SKB. According to current forecasts, SKB will dispose of a total of about 170,000 cubic metres of short-lived waste and provide room for nine reactor pressure vessels. Most of the short-lived waste comes from the nuclear power plants. Other waste comes from Clab and Clink (central facility for handling, interim storage and encapsulation of spent nuclear fuel) and from facilities belonging to Studsvik Nuclear AB and AB SVAFO.

The long-lived waste from the NPPs consists of used core components, reactor pressure vessels (RPVs) from pressurized water reactors (PWRs) and control rods from boiling water reactors (BWRs). The long-lived radionuclides are formed from stable elements in e.g. steel when they are exposed to strong neutron radiation from the reactor core. The total quantity of long-lived waste is estimated at about 16,000 cubic metres, about one-third of which comes from the NPPs. The rest comes from facilities belonging to Studsvik Nuclear AB and SVAFO. SKB plans to dispose of the long-lived waste in SFL.

Specified volumes include an allowance for uncertainties in the waste forecasts.

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 how long the fuel has decayed (cooled) and its 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 achieve as efficient utilization of the fuel as possible. 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 6,300 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.

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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.

In addition to all spent fuel from today's Swedish nuclear power plants, the 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. Approximately 20 tonnes of spent nuclear fuel from Ågesta and approximately two tonnes of spent nuclear fuel from Studsvik Nuclear AB'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. During reprocessing, uranium, plutonium and waste products are separated. Sweden has also sent spent fuel for reprocessing to Sellafield in England. According to current plans, the uranium and plutonium obtained from this reprocessing will be used to fabricate eighty MOX fuel assemblies (about 14 tonnes) for the Oskarshamn Nuclear Power Plant.

1.2 Programme for research, development and demonstration

The Nuclear Activities Act regulates the periodicity and scope of the RD&D programme. The programme shall provide an overview of all measures that are needed to manage the radioactive waste and describe in greater detail the measures intended to be adopted within a timespan of at least six years. The programme shall be submitted every three years to SSM, who conducts a review and evaluation of planned research and development activities, reported research results, alternative management and disposal methods and planned measures. After extensive referral for consideration and comment, SSM hands over the matter to the Government, which rules on whether the programme meets the requirements in the Nuclear Activities Act and any guidelines for the continued activities. The Swedish National Council for Nuclear Waste, which is an independent interdisciplinary committee that advises the Government, will also submit its own assessment of the programme.

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. When the new Nuclear Activities Act had entered into force in February 1984, the applications were supplemented with SKB's first programme, RD&D Programme 84, which thereby became a supporting document. In June 1984, the Government granted the nuclear power companies fuelling permits 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 published by SKB up to and including 2007 was given in the previous programme, RD&D programme 2010. All programmes have been circulated for consideration and comment and then approved by the Government, in some cases with demands for supplementary information or directives on how SKB should respond to viewpoints expressed by the reviewing bodies. Figure 1-1 shows an overview of research and development programmes and other milestones in the development work.

RD&D Programme 2010

The RD&D programme 2010 was divided into five parts: overall plan of action, the LILW programme, the Nuclear Fuel Programme, research for assessment of long-term safety and social science research. The first part gave a general picture of SKB's planning to construct and commission new facilities and facility parts at existing facilities. The overall plan of action also described the flexibility (and limitations on this flexibility) of both programmes in response to a number of specific changes in the premises.

RD&D programme 2010 elaborated above all on the plans for implementing the LILW programme (the programme for disposal of low- and intermediate-level waste). An account was given of the plans for a stepwise extension of SFR. In addition to final disposal of short-lived waste, the extension is also planned to be used for interim storage of long-lived waste. The plans for SFL were presented with a focus on research and development work in the immediate future.

Technology development in the Nuclear Fuel Programme is now proceeding from schematic solutions to solutions that are tailored to an industrialized process. The planning for this was described for the production lines that were defined in RD&D Programme 2007 and describe the production flow for the parts of the repository.

SKB's research in support of the assessments of long-term safety for the Spent Fuel Repository and the extended SFR was described. Much of this research will also be relevant for the safety assessment for SFL.

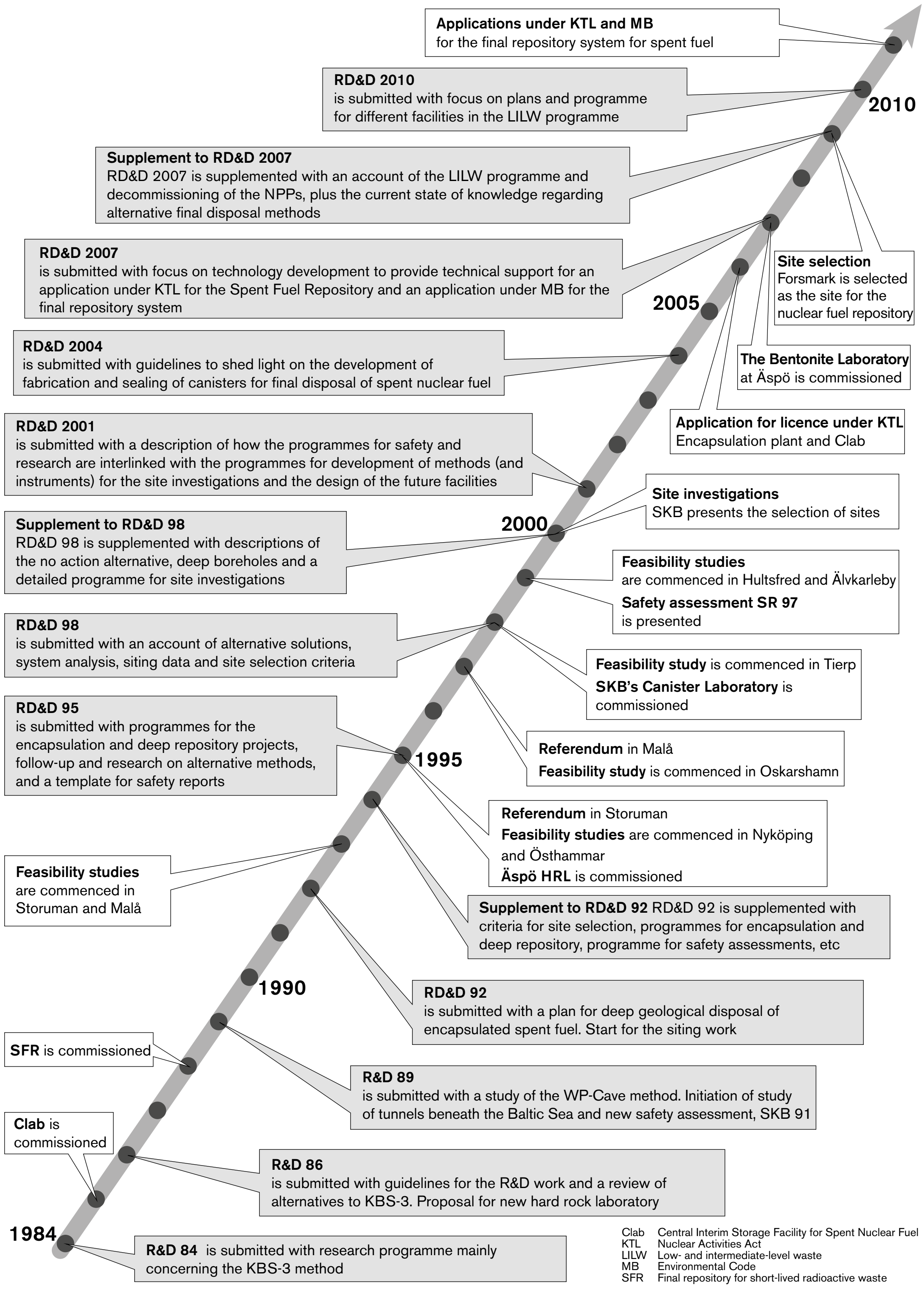


***Figure 1-1.** Milestones in SKB's development programme.*

SSM said above in its review that SKB should present more detailed plans and strategies regarding decommissioning of the nuclear power plants and the Ågesta combined heat and power plant (CHP). Further, SSM says that SKB should present a more in-depth account of the programme for long-lived low- and intermediate-level waste including different alternatives for management and final disposal of the waste.

1.3 The RD&D programmes in relation to other accounts submitted to SSM

In addition to the RD&D programmes, SKB and the nuclear power companies also submit other accounts to SSM where they explain their plans for the management of radioactive waste and spent nuclear fuel. In order to avoid redundancy, references are made to these accounts in the present RD&D programme. The most important of these accounts are:

- the applications SKB has submitted for the Spent Fuel Repository and the encapsulation plant, and requested supplements,
- future applications for extension of SFR,
- studies and plans for decommissioning of the nuclear power reactors and other nuclear facilities,
- periodic overall assessments including safety analysis reports (SAR) for SKB's commissioned facilities Clab and SFR,
- the regular Plan reports.

The applications under the Nuclear Activities Act for final disposal of spent nuclear fuel (SKB 2011a) and under the Environmental Code for the KBS-3 system were submitted in March 2011 (SKB 2011b). The activities that will lead to construction, operation and final deposition are described here, along with an account of pre- and post-closure safety. An application under the Nuclear Activities Act for the encapsulation plant was submitted in 2006. It was supplemented in 2009 with regard to integration of the encapsulation plant with Clab to a single facility, Clink. In March 2011 it was supplemented again with regard to those parts dealing with the KBS-3 system (SKB 2011c). In the course of reviewing these applications, SSM requests supplementary information from SKB.

In order to dispose of the waste from dismantling and demolition of the nuclear power reactors, in this case the shutdown reactors in Barsebäck and Ågesta and the research reactors in Studsvik, SFR needs to be extended. SKB plans to submit applications for an extension of SFR in the spring of 2014.

Regarding facilities in operation, Clab and SFR, SKB is obligated as licensee to submit various accounts to SSM that are not discussed in this RD&D programme. According to the Nuclear Activities Act, an overall evaluation of safety and radiation protection must be done at least every ten years. In this context, an account is also given of what changes can be foreseen ahead. The assessment shall take into account advances in technology and science. It shall also include analyses and descriptions of the prospects that regulations and conditions will be able to be met by the next safety evaluation. In the case of the commissioned facilities, there is also in this context a safety analysis report (SAR) that must be continuously updated. A decommissioning plan shall be submitted at the same time as the account of the periodic overall evaluation. The plan shall be kept up-to-date until the facility is decommissioned, and significant changes in the plan shall be reported to SSM.

Following the nuclear accident in Fukushima, Japan, in March 2011, the EU Council of Ministers agreed that all EU Member States should conduct stress tests at their nuclear power plants. The purpose of these tests was to assess how well the plant withstands highly improbable events. It was the nuclear power industry that conducted the stress tests, while SSM reviewed its assessments and compiled a national report that was then reviewed by international experts. The national reports have thereafter been followed by national action plans. The Government decided that Clab should also undergo a stress test. The results of the stress test and action plans based on these results were presented in a report submitted to SSM in 2011.

Another account linked to the RD&D programme is the Plan report, see Section 1.4. Here the future cost for disposing of the radioactive waste and the spent nuclear fuel and decommissioning of the nuclear power reactors is calculated. The cost calculation is based on the plans presented in the RD&D programme.

1.4 Financing

The costs for disposing of the operational waste are paid as they arise, but financing of the rest of the nuclear waste programme is based on the payment of fees to a special fund, the Nuclear Waste Fund. This is regulated in the Financing Act and the Financing Ordinance.

Every three years, SKB prepares a cost calculation, a Plan report, on behalf of the nuclear power companies. The report is submitted to SSM, who 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 pledged for Barsebäck, which is determined by SSM). The nuclear power companies pay the fees to the Nuclear Waste Fund, whose assets are deposited in an interest-bearing account with the National Debt Office or in debt instruments issued by the state and covered mortgage bonds, in accordance with Government instructions.

At the end of 2012 there was about SEK 49 billion in the nuclear power companies' shares of the Nuclear Waste Fund (market value). In addition, some SEK 34 billion (current price level) has been spent building and operating today's system and for the research and development work. During the period 2012 to 2014, the average fee is 2.2 öre (100 öre = 1 Swedish krona) per kilowatt-hour of electricity produced for the nuclear power plants that are in operation. Barsebäck Kraft AB pays an annual fee of SEK 842 million.

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.

1.5 Description of the waste system

Figure 1-2 provides an overview of the system for management and disposal of Sweden's radioactive waste and spent nuclear fuel. 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 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 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. All facilities in the KBS-3 system will be operated by SKB.

SKB is responsible for the transportation system, which is the same for both low- and intermediate-level waste and spent nuclear fuel. The shipments go by sea, since all nuclear power plants and nuclear waste facilities are situated on the coast. The transportation system consists of a specially built ship, different types of transport containers for different waste types, and special vehicles for loading and unloading.

A new ship, m/s Sigrid, has been launched. She replaces m/s Sigyn, which was built in 1982. Like the old ship, the new ship has a double bottom and a double hull. This design protects the cargo in the event of grounding or collision. Sigrid is more fuel-efficient and has a lower environmental impact than her predecessor. She can carry twelve fuel and waste containers instead of ten as before. 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.

The facilities in both systems, as well as SKB's facilities for research, development and demonstration, are described in the following sections.

1.5.1 Facilities in the system for low- and intermediate-level waste

In order to be able to manage and dispose of 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.

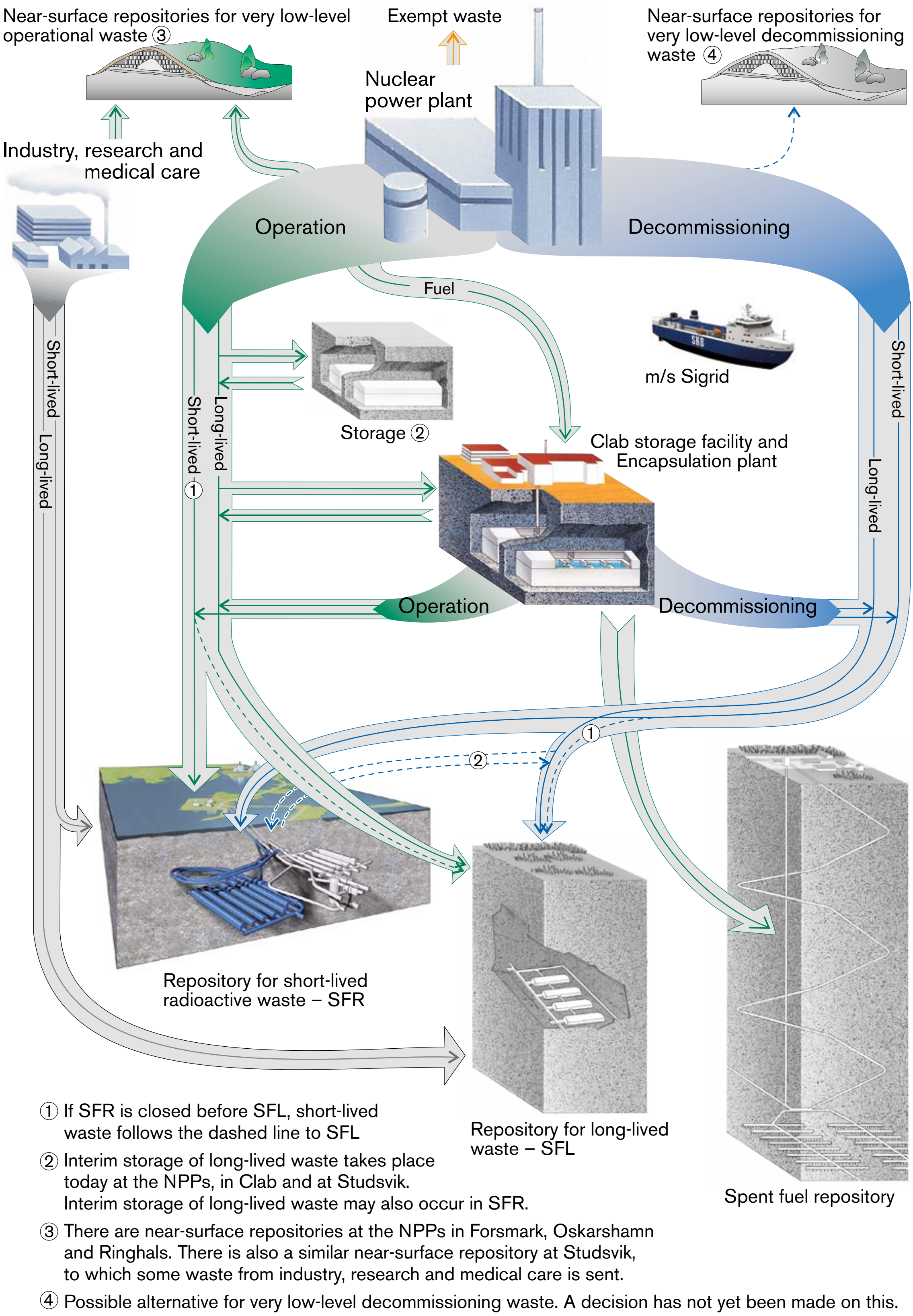


Figure 1-2. *System for management and disposal of Sweden's radioactive waste and spent nuclear fuel. Solid lines represent transport flows to existing or planned facilities. Dashed lines represent alternative handling pathways.*

SKB's final repository for short-lived radioactive waste, SFR, has been in operation since 1988. The nuclear power companies, SVAFO and Studsvik Nuclear AB, operate local treatment plants, interim storage facilities and near-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 low- and intermediate-level waste in SFL (final repository for long-lived waste).

Facilities for treatment of waste

There are treatment plants for 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 near-surface repositories. The purpose of the treatment may be to clear the material for free release, reduce its volume, concentrate its activity or solidify it.

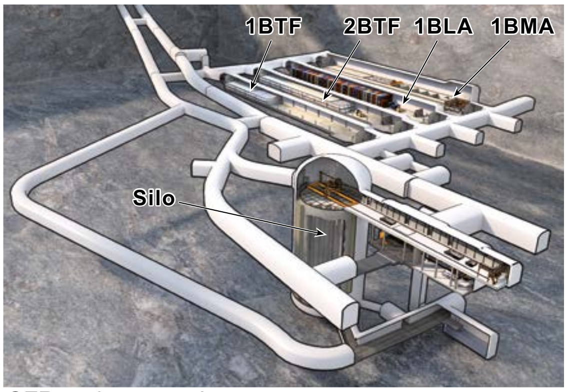
Near-surface repositories

Near-surface repositories is used for very low-level waste. There are near-surface repositories today on the industrial sites at the nuclear power plants in Forsmark, Oskarshamn and Ringhals as well as in Studsvik. SKB is exploring the possibility of building a near-surface repository for very low-level waste from dismantling and demolition of the NPPs. After about 50 years, the radioactivity in this waste has fallen to such low levels that it can be cleared from the viewpoint of radiation protection.

Final repository for short-lived radioactive waste, SFR

SFR is located at the Forsmark nuclear power plant, see Figure 1-3. 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.

Aerial view of the surface part



SFR under ground

Rock vault for ILW

View over top of silo

Figure 1-3. Final repository for short-lived radioactive waste, SFR.

The facility consist today of four 160-metre-long rock vaults 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 cubic metres. Low-level waste is kept in one of the four rock vaults (1BLA). Intermediate-level waste with lower activity levels is kept in two of the rock vaults (1-2BTF). The intermediate-level waste with the highest activity levels is emplaced in the fourth rock vault (1BMA). The silo will contain most of the radioactive substances in SFR. The waste in BLA and 1-2BTF is handled without radiation shielding.

The waste in SFR comes from the nuclear power plants, Clab, Studsvik and Ågesta. At the end of 2012, 34,822 cubic metres of waste had been deposited. Today only operational waste is disposed of in SFR. In order to be able to dispose of all additional short-lived operational waste from dismantling and demolition, SKB plans to extend SFR.

Interim storage of long-lived waste

Today most of the long-lived waste, mainly used core components, 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 and Ringhals AB also have facilities for dry interim storage on their power plant sites.

SKB plans to be able to store long-lived waste in the extended SFR or on another site and is therefore investigating this possibility.

Barsebäck Kraft AB is planning to construct a new building on its site for dry storage of segmented irradiated RPV internals. The reactor internals that are classified as short-lived waste to be disposed of in SFR will be moved there when the extension is commissioned. The internals that comprise long-lived waste will be moved from Barsebäck for interim storage in SFR or another prepared place.

Besides waste from the NPPs, there is long-lived waste that derives from research. This waste belongs to SVAFO and Studsvik Nuclear AB and is interim-stored today in Studsvik.

Final repository for long-lived waste, SFL

SKB is planning to dispose of the long-lived waste at a relatively great depth in a facility with rock vaults adapted for the different waste types. This final repository (SFL) will be the last facility to be put into operation. Work is currently under way to propose possible repository concepts, and siting of the repository is 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 16,000 cubic metres.

Transportation system for low- and intermediate-level waste

The transportation system for low- and intermediate-level waste consists of the ship m/s Sigrid, special vehicles and different types of transport containers. The ship and the vehicles are also used for shipments of spent nuclear fuel in the KBS-3 system.

Short-lived waste is shipped today from the nuclear power plants and Clab to SFR. Low-level waste does not need any radiation shielding and can therefore be transported in 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 shipped in transport containers (ATB) with 7–20 centimetre thick walls of steel, depending on how radioactive it is, see Figure 1-4.

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. A new waste transport container (ATB 1T) is being developed for shipping long-lived waste in steel tanks intended for dry interim storage.

Figure 1-4. *M/S Sigrid plus transport container for short-lived radioactive waste (ATB) and transport cask for core components (TK).*

1.5.2 Facilities in the KBS-3 system

SKB's interim storage facility for spent nuclear fuel, Clab, has been in operation since 1985. SKB is planning to build a facility part for encapsulation of the nuclear spent fuel adjacent to Clab and a final repository, the Spent Fuel Repository, in Forsmark. In addition to these facilities, SKB plans to build a canister factory in Oskarshamn where finish machining, assembly and quality assurance of the copper canisters will be done. Furthermore, a new type of transport cask for canisters with spent nuclear fuel is needed for the transportation system.

Central interim storage facility for spent nuclear fuel, Clab

Clab is located at the nuclear power plant 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. Besides spent nuclear fuel, control rods from boiling water reactors and core components are interim-stored today.

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 top edge of the fuel is eight metres below the water surface. The water in the pools serves both as a radiation shield and a cooling medium. The radiation level at the edge of the pool is so low that the personnel can stand there without restrictions.

At year-end 2012 there were 5,577 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. According to current forecasts, this amount is projected to be reached in 2023. The pools can hold a total of about 11,000 tonnes of fuel. A new licence is required to increase this storage capacity. SKB's general plans to increase the storage capacity in Clab are presented in Section 2.3.2.

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

Before the spent nuclear fuel is disposed of it will be encapsulated in copper canisters. SKB plans to encapsulate the spent fuel in a new facility part adjacent to Clab, see Figure 1-6 and Figure 1-7. When this encapsulation part has been connected with Clab, the two facility parts will be operated as an integrated facility, Clink.

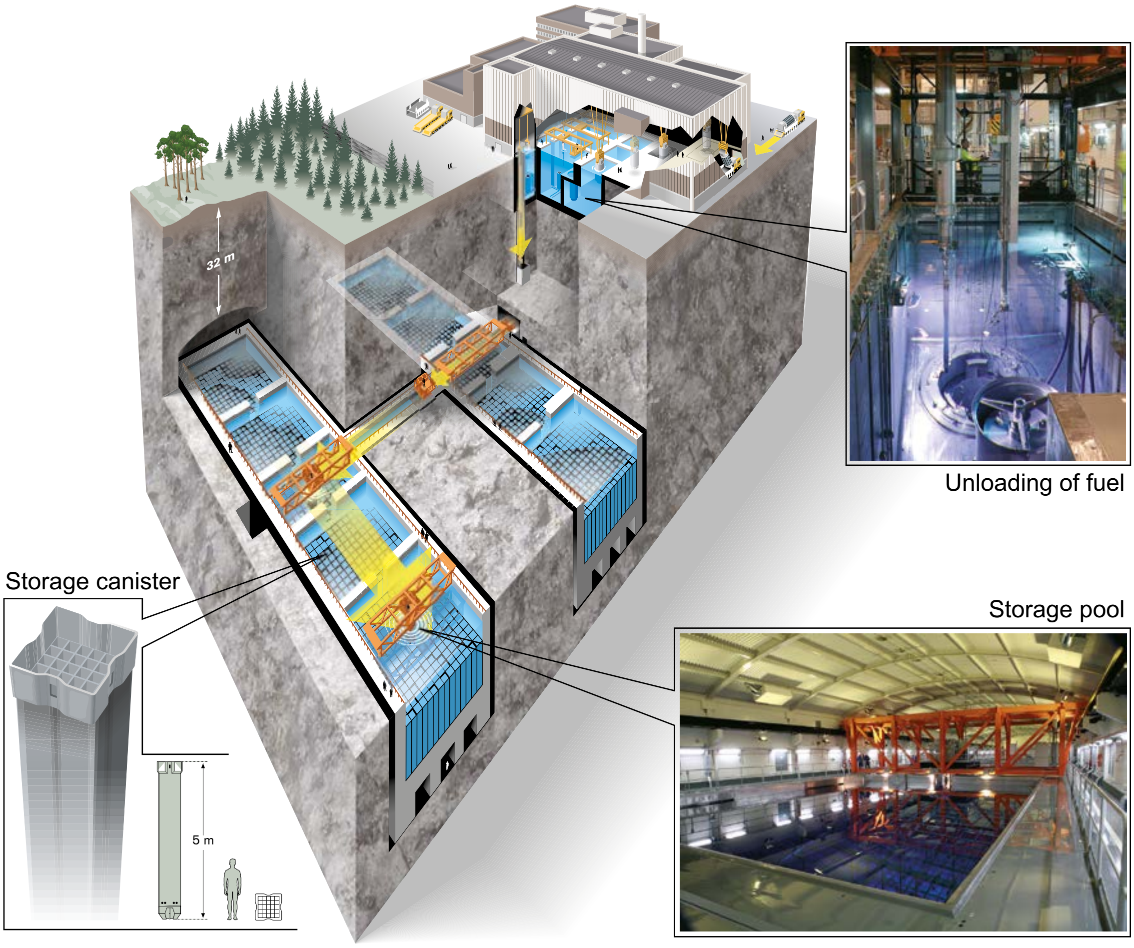


Figure 1-5. *The central interim storage facility for spent nuclear fuel, Clab.*

Figure 1-6. *Photomontage showing the integrated facility for interim storage and encapsulation, Clink. The buildings outlined in red are the planned building part for encapsulation and a smaller terminal building.*

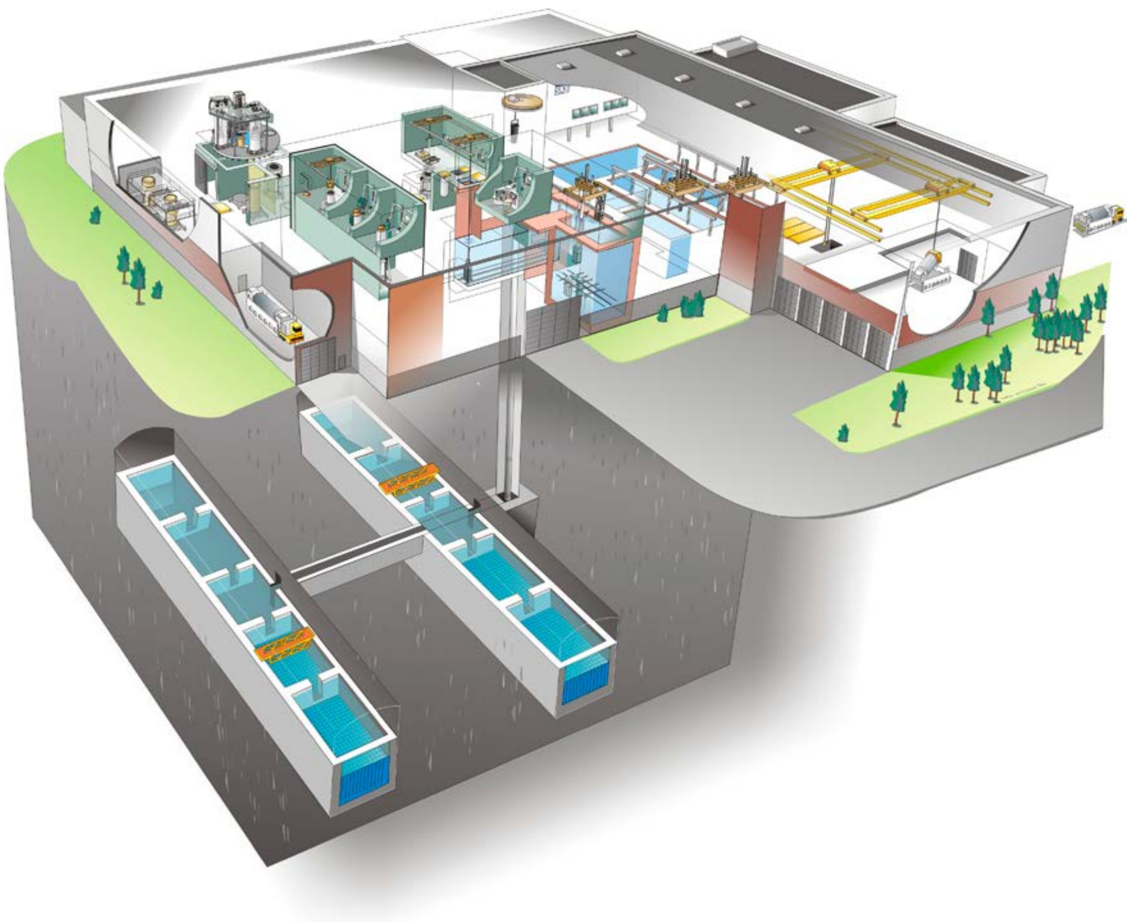


Figure 1-7. In Clink, the fuel will be taken from the interim storage pools underground to the encapsulation part with its different work stations.

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 1-8. There are two types of inserts: one that holds twelve fuel 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 1.1.4. They can be placed in one of the two insert types. According to current plans, SKB will build a canister factory in Oskarshamn where the canister's different components will be finish-machined, assembled and quality-assured. The canister factory will not be a nuclear facility.

There will be a number of stations for different work operations in Clink. 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. The copper canister is sealed by means of friction stir welding. The quality of the weld is inspected, and if it is approved the canister is taken to the machining station, where excess material is removed. 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. Clink is designed for encapsulation of 200 canisters per year.

Spent Fuel Repository

The work of finding a suitable site for a final repository for spent nuclear fuel has spanned 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. A decisive factor in the selection of Forsmark was that the prospects of achieving long-term safe disposal were judged to be better there.

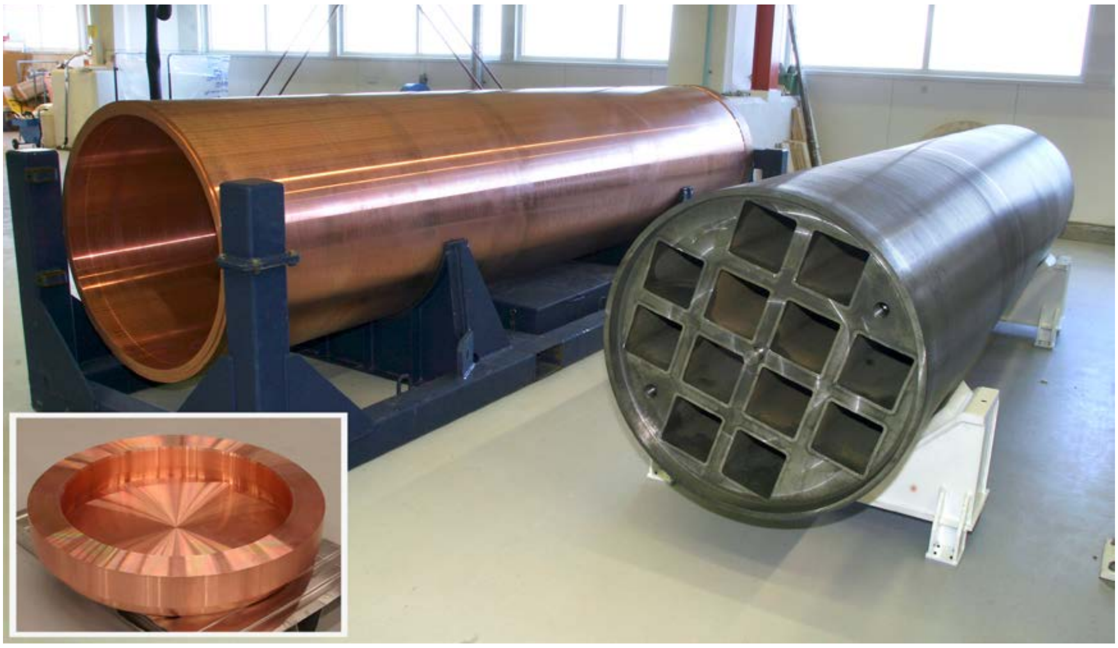


Figure 1-8. *Copper canister with nodular iron insert (the inset photo shows the copper lid).*

The final repository will consist of a surface part and an underground part, see Figure 1-9. 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 the location of large deformation zones, the occurrence of large or highly water-conducting fractures, and the thermal conductivity of the rock. The above-ground facility consists of operations area, rock heap, ventilation stations and storehouse. The facility is designed for a maximum deposition capacity of 200 canisters per year.

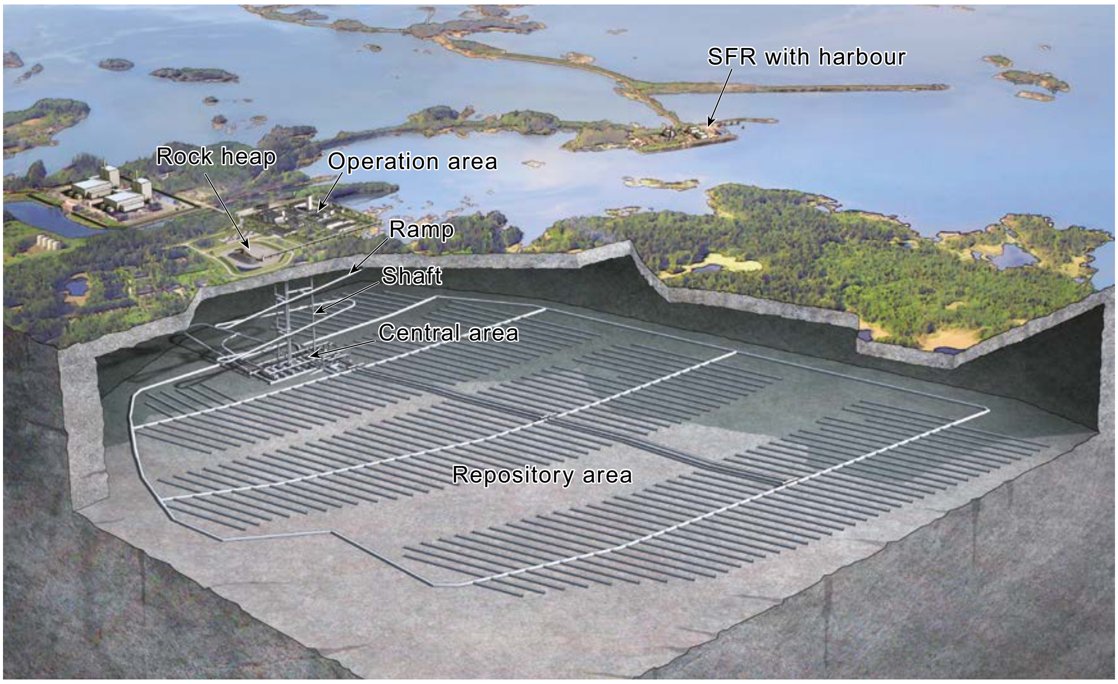


Figure 1-9. *Spent Fuel Repository in Forsmark.*

When the transport casks with 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 carried 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 and sealed with a concrete plug. When all fuel has been deposited, other openings are also backfilled and the above-ground facilities are decommissioned.

Transportation system for spent nuclear fuel

The transportation system consists of the ship m/s Sigyn, special vehicles and different types of transport casks and containers. The ship and the vehicles are also used for transportation of low- and intermediate-level waste.

Today the spent nuclear fuel is shipped from the nuclear power plants to Clab in transport casks (TB) with roughly 30-centimetre-thick steel walls, see Figure 1-10. These casks have cooling fins to remove the decay heat generated by the fuel.

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

1.5.3 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 SKB conducts research and development projects, mainly for the barriers in the Spent Fuel Repository. SKB also conducts research and development in close cooperation with Posiva in Onkalo. The results of the experiments and the projects in the 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.

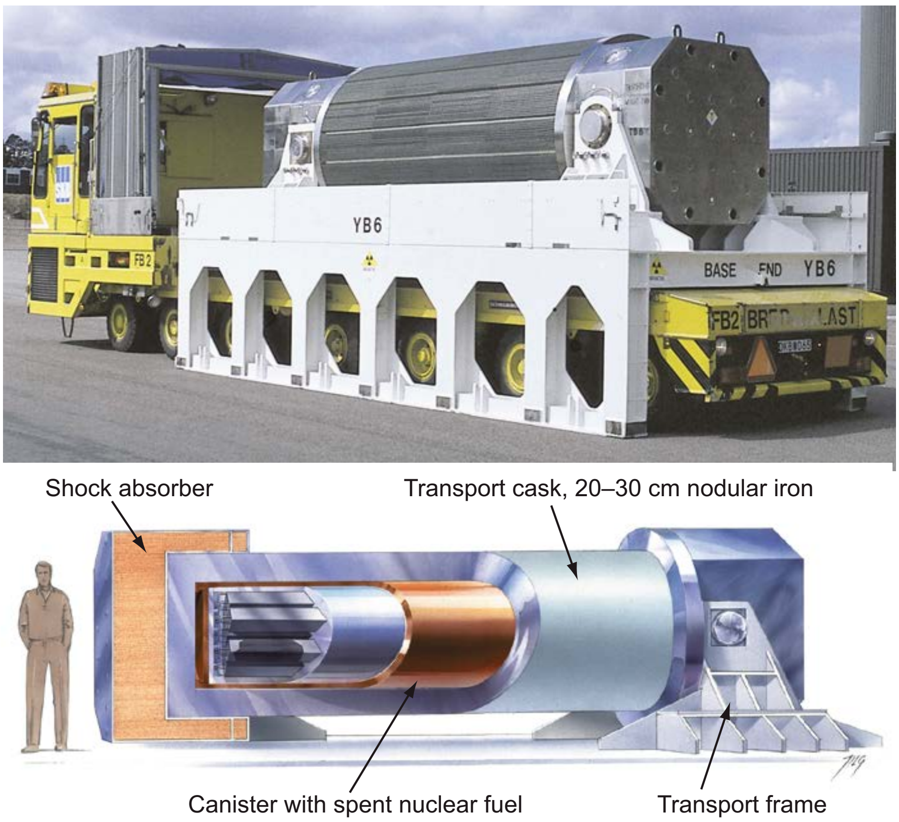


Figure 1-10. *Transport cask for spent nuclear fuel (TB, top) and canisters with spent nuclear fuel (KTB, bottom).*

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 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 activities at the Äspö HRL, which was built during the period 1990–1995, are a continuation of the work that was previously pursued in the Stripa Mine in Bergslagen. The laboratory is situated on the island of Äspö north of the Oskarshamn Nuclear Power Plant, see Figure 1-11. 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 and demonstration tests are conducted in niches and short tunnels that branch out from the main tunnel. An illustration of the HRL is shown in Figure 1-12.

The Äspö HRL has played a central role in the development, testing and verification of technology and methods for the site investigations that have been carried out in Laxemar and Forsmark. It is playing the same role for the coming detailed characterization in the Spent Fuel Repository in Forsmark.

The laboratory is also 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 spent fuel from reaching the ground surface.

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. The laboratory will therefore be in operation roughly until the Spent Fuel Repository is commissioned.

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. The parts of the facility near the surface are suitable for experiments for SFR, while the deeper parts are suitable for studying processes and testing technology to support the realization of SFL. SFR- and SFL-related experiments are already being conducted today for such purposes as studying how cylindrical concrete containers with different (non-radioactive) waste materials age in a real repository environment.

Figure 1-11. *The Äspö HRL situated on the island of Äspö north of the Oskarshamn Nuclear Power Plant.*

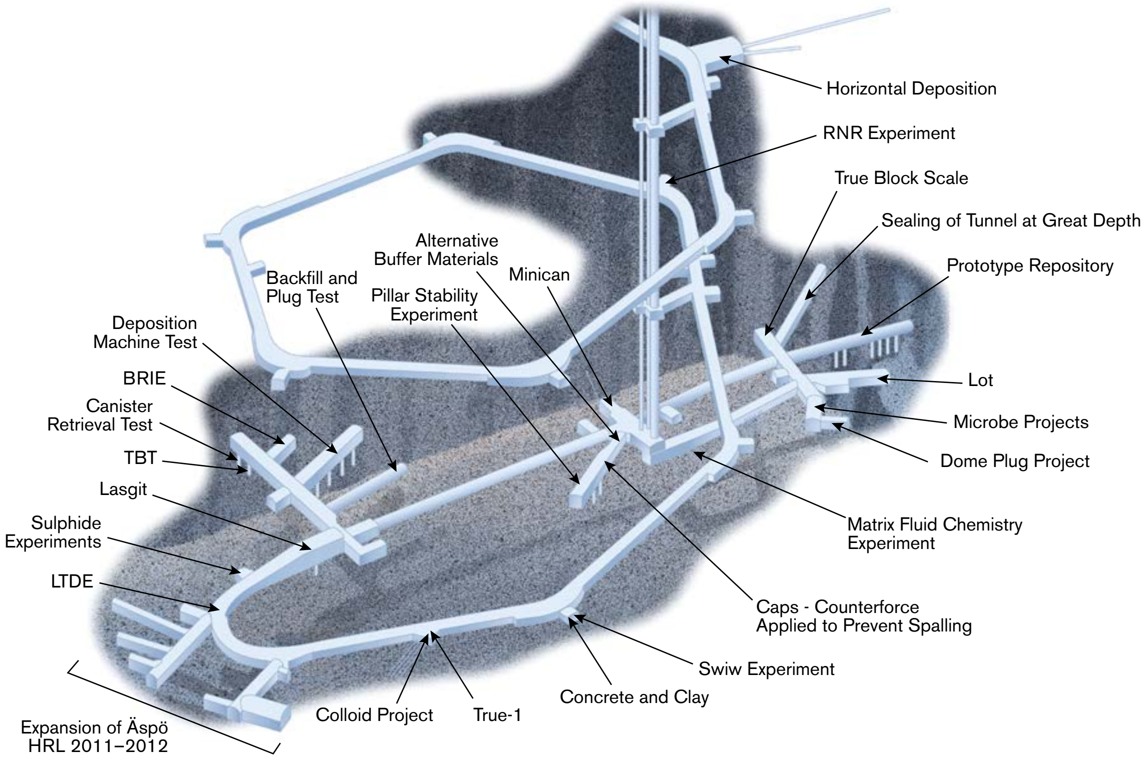


Figure 1-12. *The Äspö HRL.*

Many different countries and organizations are participating in the experiments being conducted at the Äspö HRL. In different forms and project groups, SKB is working with sister organizations, research institutes and universities in e.g. 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. The technology for welding the bottom and sealing the lid on the canister is being tested and developed in the Canister Laboratory. The methods that SKB will use to inspect the canister's components and welds are also being developed and demonstrated here. Development of the methods that will be used to fabricate the parts of the canister is being led from the Canister Laboratory. Investigation and evaluation are done to a great extent at the Canister Laboratory, while the fabrication trials are conducted at external suppliers. The goal is to develop methods for fabrication and inspection that meet stipulated quality requirements and have sufficiently high reliability to be used in future canister production and in Clink. Important equipment in the laboratory includes a system for friction stir welding with a rotating tool, equipment for nondestructive testing and a handling system for full-size canisters. Figure 1-13 shows the equipment for friction stir welding.

SKB also plans to use the facility for training purposes in preparation for future production, for example commissioning of the encapsulation part in Clink. The Canister Laboratory will therefore be in use until encapsulation of the spent nuclear fuel commences.

Bentonite Laboratory

SKB has been conducting research and development in the Bentonite Laboratory in Oskarshamn since 2007, see Figure 1-14. 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 will also be used for backfilling of the tunnels in the repository. In the Bentonite Laboratory, SKB is testing 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.

The tests performed in the Äspö HRL are often preparatory tests on various scales in preparation for full-scale tests at repository depth in the Äspö HRL. The laboratory also has equipment and space for reception of bentonite deliveries and mixing of bentonite to the desired water content.

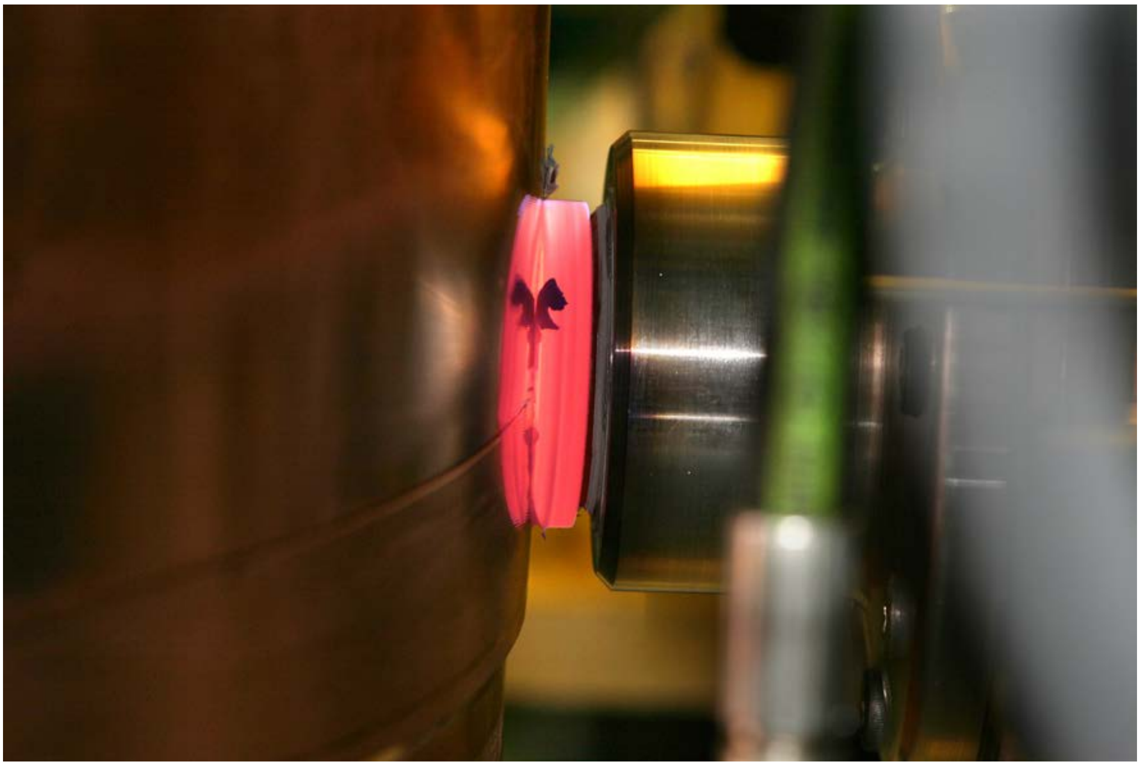


Figure 1-13. *The Canister Laboratory's equipment for development of friction stir welding.*

Figure 1-14. *Bentonite Laboratory.*

2 Plan of action

This chapter gives a general picture of SKB's planning for the construction and commissioning of new facilities and facility parts at existing facilities. The chapter provides a description of the current situation and SKB's plans for execution of the nuclear waste programme.

A fundamental principle for SKB's development is that the day-to-day activities shall be conducted safely and efficiently at the same time as new activities are prepared and built up. An important part of this work is continuously assuring and developing SKB's competence and organization, both in the day-to-day work and for the future.

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 accounts, as well as periodic safety analysis reports.

The process of building and deploying a new facility or facility part undergoes different phases. Before construction can begin, applications under the Nuclear Activities Act and the Environmental Code to build, own and operate the facility must be submitted and approved. During the licensing process, the applications are reviewed by SSM and the Land and Environment Court after they have received comments from different reviewing bodies. After the consent of the municipality has been solicited received in accordance with the veto provision in the Environmental Code, the Government can grant permissibility and a licence. Then conditions under the Nuclear Activities Act are issued by SSM and under the Environmental Code by the Land and Environment Court. Before a nuclear facility may be built, a preliminary safety analysis report (PSAR) must be compiled and submitted to SSM for approval, along with a description of how construction of the facility affects safety during operation and, in the case of a final repository, after closure as well. This procedure also applies to major conversions or modifications of an existing facility. In the case of new facilities, a building permit is also required from the municipality.

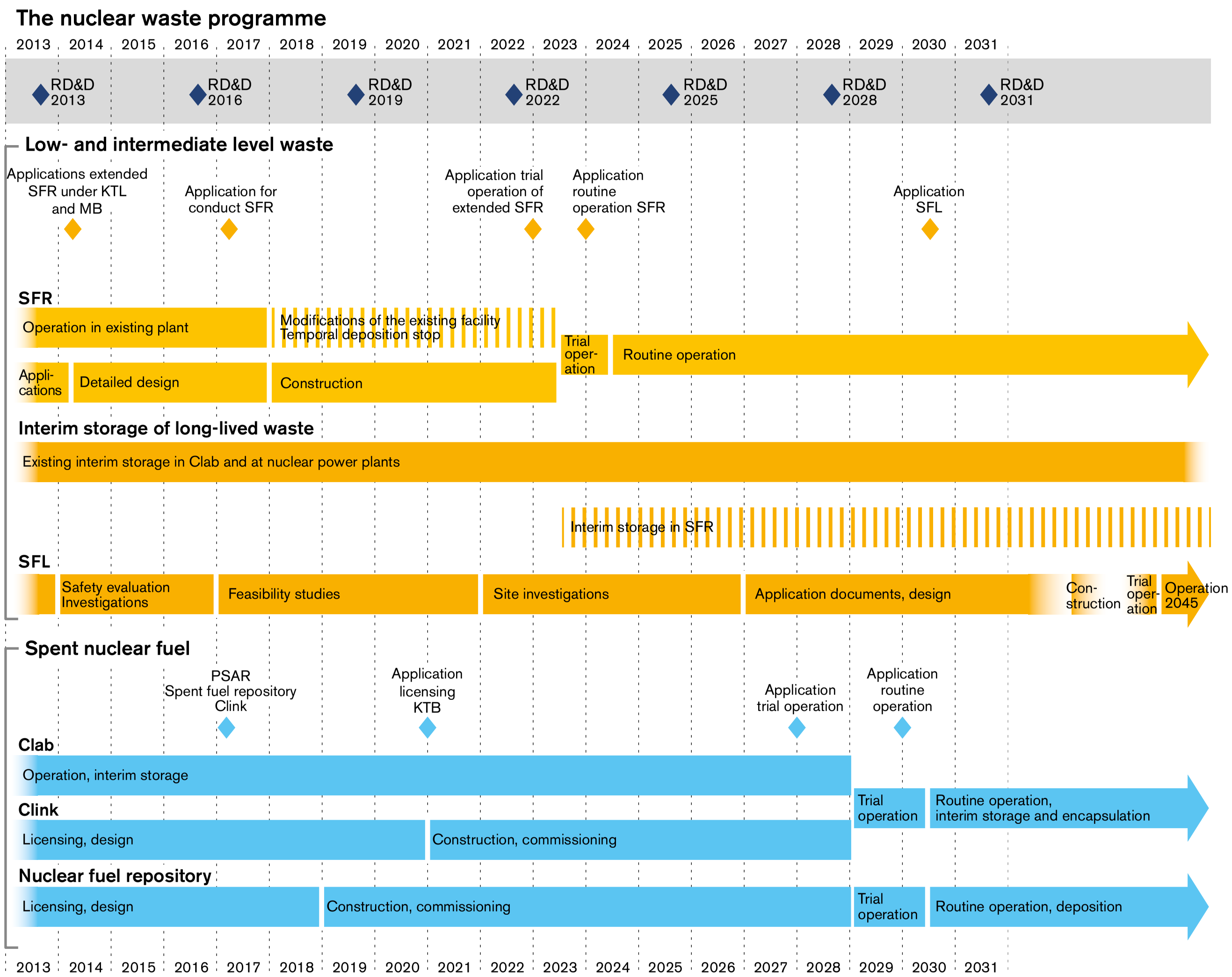
SKB will submit applications to SSM for a licence to commence trial operation when systems and processes have been tested and work as intended. Trial operation entails that nuclear material or radioactive waste is brought into the facility and handled there. Before trial operation of a facility may commence, the safety analysis report (SAR) must be updated so that it reflects the as-built facility. Before the facilities may be put into routine operation, the safety analysis reports must be supplemented taking into account experience gained from trial operation.

The plans of action for low- and intermediate-level waste and spent nuclear fuel are presented in the following sections.

2.2 Plan of action for low- and intermediate-level waste

SKB's programme for low- and intermediate-level waste includes both management of existing waste and planning and construction of the system that is needed to safely dispose of future low- and intermediate-level waste. The final repositories SKB plans to establish for low- and intermediate-level waste include an extension of SFR (final repository for short-lived radioactive waste) and construction of SFL (final repository for long-lived waste). SKB is also investigating the option of near-surface repository for very low-level waste.

In this section, the current situation for this programme is described in brief, after which the overall plan for coming activities is presented for short-lived and long-lived low- and intermediate-level waste respectively. Figure 2-2 illustrates a general timetable together with important milestones, where hatched bars in the timetable mark both uncertainties and flexibility in the planning. In order to clarify the connection between the programme for low- and intermediate-level waste and the decommissioning of the NPPs, this planning is also illustrated in the figure, where the starting year for dismantling and demolition is indicated for each reactor.



Clab	Central Interim Storage Facility for Spent Nuclear Fuel
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
SFL	Final repository for long-lived waste
SFR	Final repository for short-lived radioactive waste

Figure 2-1. *Main timetable for SKB's nuclear waste programme. Hatched bars mark uncertainties and flexibility in the planning.*

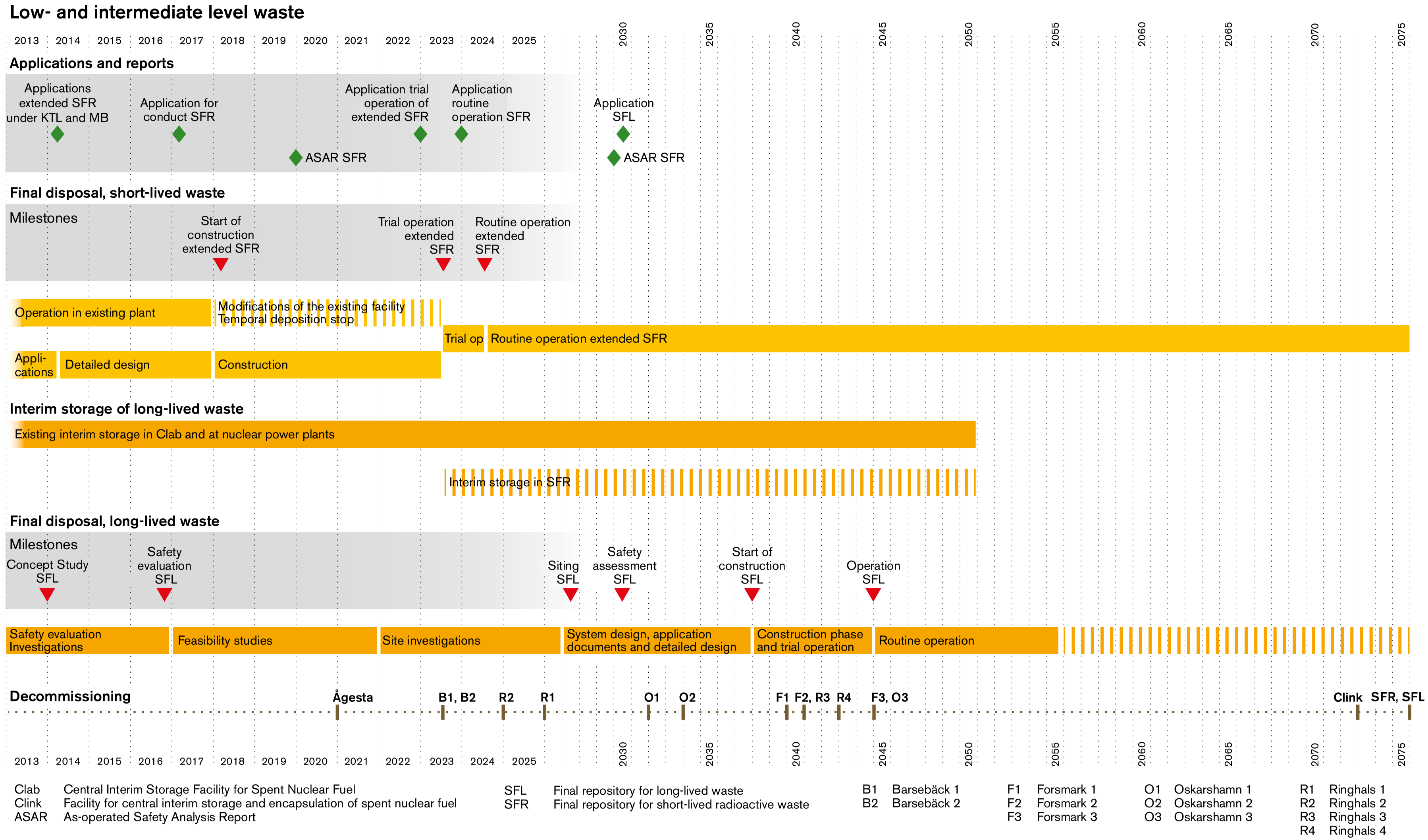


Figure 2-2. *Timetable for work with low- and intermediate-level waste and decommissioning of the NPPs. Hatched bars mark uncertainties and flexibility in the planning.*

The point of departure is that the reactors in Forsmark and Oskarshamn, as well as Ringhals 3 and 4, are operated for 60 years, while Ringhals 1 and 2 are operated for 50 years. The first reactors in Sweden that will be dismantled and demolished are the reactors in Barsebäck and Ågesta and the research reactors on the Studsvik industrial site. Dismantling and demolition is estimated to take five to seven years, depending on such factors as the size of the reactor.

The current status of the work with low- and intermediate-level waste can be summarized in the following points:

- The applications under the Nuclear Activities Act and the Environmental Code to extend SFR will be submitted in the spring of 2014.
- SKB wants to have the option of interim-storing long-lived waste in SFR. In the applications for extending SFR, interim storage of long-lived waste is therefore also treated as a part of the programme.
- Site-specific studies for decommissioning of the nuclear power plants in Forsmark, Oskarshamn and Ringhals that describe waste inventory, technology and costs have been completed in the summer of 2013.
- Based on the results of the site-specific studies for decommissioning, SKB has arrived at the conclusion that approximately half of the short-lived low-level waste from dismantling and demolition of the nuclear power plants in Forsmark, Oskarshamn and Ringhals could be disposed of in near-surface repositories instead of in SFR.
- A study of different repository concepts for SFL is under way and the results will be presented in the end of 2013. The goal of the study is to choose one or two repository concepts to proceed with.
- A new container for transport of steel tanks with long-lived low- and intermediate-level waste (ATB 1T) is under development.
- SKB is investigating the option of dry interim storage as an alternative to interim storage of BWR control rods in pools in Clab. Methods for segmentation of control rods have therefore been studied, and suitable waste containers for interim storage and future deposition in SFL have been evaluated, see Section 8.2.

2.2.1 Short-lived low- and intermediate-level waste

An overall timetable for the extension of SFR is presented in Figure 2-3. The different phases in the timetable are discussed in greater detail below.

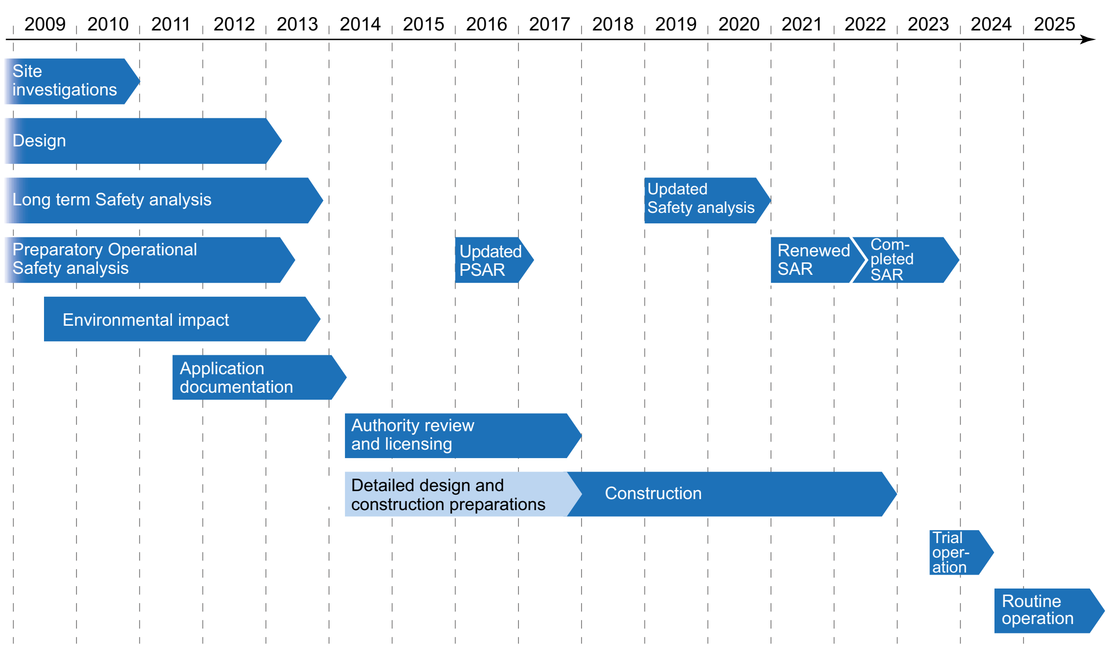


Figure 2-3. *Overall timetable for extension of SFR.*

Compared with the plans presented in RD&D programme 2010, commissioning of the extended SFR has been delayed by three years until 2023. The main reason for the delay is that SKB now estimates that the construction of the facility will take a longer time, but also that more time has been set aside for licensing, as requested by SSM in its review of RD&D programme 2010 and in consultations with SKB. The main reason why construction will take longer is that the rock vaults are being located at a greater depth.

Preparation of application documents

The SFR Extension Project started in 2008. In an initial phase, its task is to prepare application documents under the Nuclear Activities Act and the Environmental Code and a facility description for an extended SFR. The application documents are planned to be ready in the spring of 2014 to be submitted to SSM and the Land and Environment Court, respectively, for preparation for the Government. The project includes site investigations, overall design of the facility, assessment of long-term (post-closure) safety, assessment of operational (pre-closure) safety, consultations on and preparation of an environmental impact statement, compilation of application documents and certain building preparations. The latter includes e.g. preparation of system documents.

In the applications, the activity inventory will include not only operational waste, but also waste from dismantling and demolition. The estimated quantities are based on waste forecasts from the recently finished site-specific studies for decommissioning and on updated waste forecasts for operational waste. In conjunction with the applications to extend SFR, SKB is planning to apply for a licence to deposit both operational waste and waste from dismantling and demolition in the entire facility. This will provide better conditions for optimal control of the waste streams to SFR.

In the applications for the extension, SKB will also apply for a licence to interim-store long-lived waste from the nuclear power plants in SFR. In the safety analysis report included in the applications, interim storage of long-lived waste will be treated as a part of the activity. The safety risk entailed by storing SFL waste in SFR will be evaluated in a safety assessment. When SFL is put into operation, the interim-stored long-lived waste will be transferred to SFL.

Licensing and detailed design

Licensing starts when the applications under the Environmental Code and the Nuclear Activities Act have been submitted in the spring of 2014. The applications will then be processed and reviewed by the regulatory authorities, the Land and Environment 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 consideration 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. Before a licence can be issued for construction, there is a formal requirement that a preliminary safety analysis report (PSAR) must be prepared. A PSAR will therefore be prepared at this point for the construction application.

Detailed design and building preparations are proceeding in parallel with the licensing process. Provided the licensing process is expeditious, the necessary licences are expected to be obtained so that construction can commence at the beginning of 2018. During this phase, other requisite licences will also be obtained, such as a building permit for extension of the above-ground facility and notification of the facility changes that will be made.

Extension and handover to operating organization

The phase for construction and handover to operating organization includes the activities construction, trial operation and handover to routine operation. During the construction phase, occasional suspensions of deposition may be necessary, at least when rock work is in progress. At the same time as SFR is being extended, the existing facility will be upgraded, partly due to the fact that the operating time has been prolonged in relation to that originally planned.

Before trial operation a SAR will be prepared. This will later be supplemented with experience from trial operation before routine operation ensues. SKB plans to submit an application for trial operation at the end of 2022. Trial operation with deposition of waste in the extended part of SFR is assumed to start about six months later. SKB plans to submit an application for routine operation at the end of 2023. It is assumed that a licence for routine operation will be obtained by mid-year 2024.

The extension of SFR may not adversely affect safety in the existing SFR. To guarantee this, SKB will employ up-to-date 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 facility change matter will undergo safety review in accordance with the procedures that apply at SKB for primary and independent safety review.

2.2.2 Long-lived low- and intermediate-level waste

An overall timetable for the work with SFL is presented in Figure 2-4. Several important milestones must be passed, such as selection of repository concept and site, evaluation of long-term safety, preparation of applications, etc. SKB plans to apply for a licence to build the repository in about 2030. The timetable is discussed in greater detail in the following section. The time from applications to commissioning of the repository is rather uncertain, since it involves activities in a relatively distant future. The current assessment is that SFL can start routine operation about 2045.

SKB's line for the development of SFL is a stepwise and iterative process where assessments of long-term safety determine the choices of direction and the choice of site. The process is planned to ensure the shortest possible development time. When all necessary steps in the process are weighed together, routine operation during the first half of 2040s is a reasonable estimate. This is in line with SKB's current planning for routine operation in 2045. Other alternatives for commissioning of SFL are described in greater detail in Section 6.3.

Within the framework of the SFR extension project, SKB has studied the prospects for interim storage of long-lived waste from the NPPs in the extended SFR. The details will be presented in the coming applications for extending SFR. SKB is also investigating the possibility of interim storage of long-lived waste from the NPPs at another location than in SFR.

Closure of SFL is intended to take place when all interim-stored long-lived waste and the long-lived waste from dismantling and demolition of the last nuclear power plant has been deposited there, see Figure 2-2. Before closure, SKB must ensure that the waste from dismantling and demolition of Clink is suitable for SFR and does not have to be disposed of in SFL. In the figure, this uncertainty in the timetable has been illustrated with hatched bars.

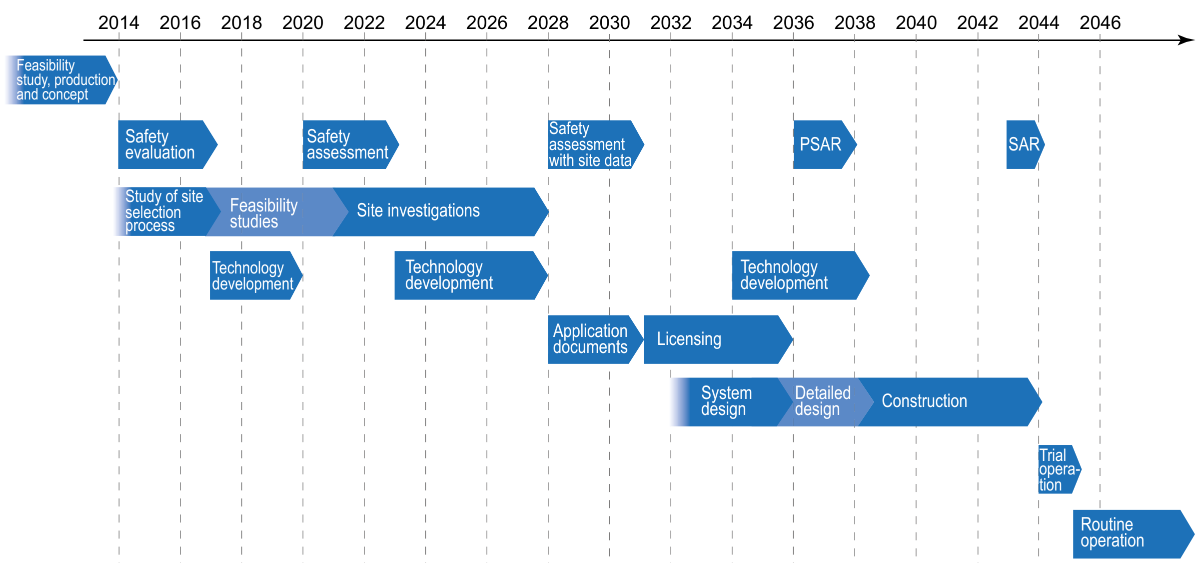


Figure 2-4. Estimated timetable for the work leading up to the commissioning of SFL.

A new container for transport of steel tanks with long-lived low- and intermediate-level waste (ATB 1T) is under development. The project is being pursued together with a French supplier, and licensing is being carried out by French regulatory authorities. The supplier has had considerable difficulties delivering a concept that meets all the requirements. In view of this, SKB has been compelled to explore alternative ways to develop ATB 1T. A new timetable for the project is expected to be finalized in early 2014.

Coming RD&D-period

During the period 2014–2016, the main focus of the work with SFL will be the evaluation of long-term safety that is to be presented in 2016. The purpose of the safety evaluation is to choose a main alternative among proposed repository designs. The safety evaluation will moreover develop the set of requirements for properties of the waste, the engineered barriers and the rock.

It will not be possible to adopt acceptance criteria for conditioned waste destined for SFL until a decision has been made on the repository concept, i.e. not before the safety evaluation is concluded. The nuclear power plants should not commence final conditioning of waste until a verified repository concept exists.

A study will be initiated during the period to plan the site selection process for SFL. Based on the results of the evaluation of long-term safety that is planned for 2016, preliminary requirements on the repository site can be formulated and siting factors identified. In the light of the identified siting factors, contacts can later be made with municipalities where the prospects for a siting are judged to be favourable. Feasibility studies will then be commenced in interested municipalities.

Technology development of the chosen repository concept will commence after safety evaluation is concluded.

The period up to the applications

A safety assessment without specific site data is planned. The assessment is based both on the safety evaluation that will be presented in 2016 and the technical solutions developed for SFL. These solutions may emerge from the research programme that is initiated based on the results of the safety evaluation.

The safety assessment will be followed by the next step in the technology development work. This work is aimed at developing detailed engineered barrier solutions. Technical solutions for the operating phase and for backfilling and closure will also be developed.

The feasibility study phase in the siting work will be concluded and site investigations will commence on one or two sites in the early 2020s. The site investigations are expected to lead to a final choice of site around 2027.

After site selection, the work of compiling supporting material for applications will commence, including a site-specific safety assessment. The goal is to submit applications under the Nuclear Activities Act and the Environmental Code around 2030.

The period between the applications and routine operation

After applications have been submitted, work will continue on e.g. system design and equipment development. Detailed design will commence when a licence to build SFL has been obtained. Construction and trial operation will be followed by routine operation around 2045.

2.3 Plan of action for spent nuclear fuel

SKB's programme for the future management of spent nuclear fuel consists mainly of the following parts:

- Work on SKB's part in the licensing of the KBS-3 system.
- Technology development of the KBS-3 system to enable it to be put into industrial operation.
- The Spent Fuel Repository Project, which is responsible for planning, design, construction and commissioning of the final repository in Forsmark.
- The Clink Project, which is responsible for planning, design, construction and commissioning of the integrated facility for interim storage and encapsulation in Oskarshamn.
- Planning for increasing the interim storage capacity in Clab to more than 8,000 tonnes fuel.
- Planning, design, construction and commissioning of the production system for canisters.
- Safety analysis reports for the KBS-3 system.

The two construction projects and the work with safety analysis reports for the KBS-3 system are primary beneficiaries of the technology development that is being carried out for the KBS-3 system.

A prerequisite for the planning is that operation of the system will start as soon as possible, but with realistic timetables for licensing, construction and commissioning.

The current status of the work for management and final disposal of spent nuclear fuel can be summarized in the following points:

- Based on the 2011 applications, SKB has planned and structured the work that remains up until the start of trial operation of the facilities included in the KBS-3 system.
- Licensing is under way and SKB is responding to questions and requests for supplementary material from both SSM and the Land and Environment Court.
- The Spent Fuel Repository Project is established in Forsmark, and the final phase of system design of the final repository's facility parts and technical systems is currently under way. A long-range project is under way to build up the organization and competence needed to implement technology development and, in extension, safe operation of the Spent Fuel Repository. Compensatory environmental measures have been adopted on the site, and the day-to-day work of monitoring the site and managing buildings, land and drilling sites is under way.
- Work is under way with safety analysis reports for the facilities in the KBS-3 system. Work is under way on the structure and content of the Spent Fuel Repository's PSAR, which has to be submitted prior to the start of construction. This work is based on experience from the preparation of the safety analysis reports SR-Site and SR-Drift.
- The technology development project is structured in accordance with the production lines that have been established for the Spent Fuel Repository's barriers for long-term safety. The projects are governed by a strategic technology development plan that links the deliveries from technology development with construction projects and future safety analysis reports.

2.3.1 Overview

Establishment of the facilities in the KBS-3 system is divided into four main phases: licensing (design), 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 each facility in Sections 2.3.2 to 2.3.5. The milestones shown in Figure 2-5 refer to times for delivery of 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 10.

Over the next few years, SKB will gradually prepare the organization for the start of construction of the Spent Fuel Repository and Clink. Start of construction for the Spent Fuel Repository is in 2019 and for Clink in 2021 so the facilities can be commissioned simultaneously in 2029. Each major step in the work is preceded by a decision gate aimed at evaluating the programme from a holistic perspective and establishing a more detailed plan for the period up to the next decision gate.

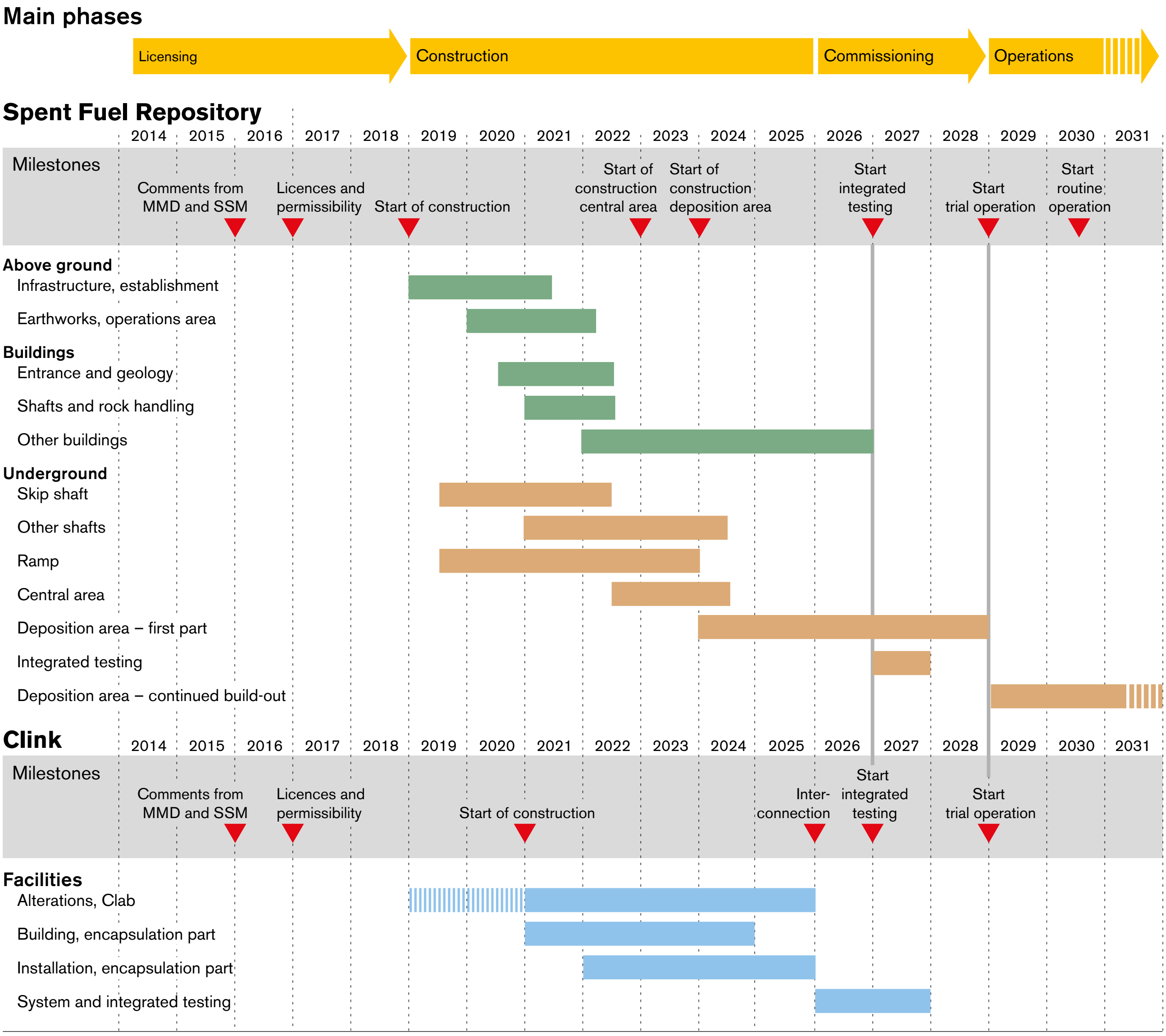


Figure 2-5. Timetable for establishment of the Spent Fuel Repository and Clink. Technology development needed by the specified milestones is discussed in Chapter 10. Note that the timetable only shows activities related to establishment. Extensive detailed design efforts will be initiated after the Government's decisions on licences and permissibility.

During the licensing process, the work pace will be adapted to the supplementary information requested from SKB and new information in the form of comments by regulatory authorities. The most important milestones to be considered are:

- the Land and Environment Court's and SSM's review statements to the Government,
- the Government's decisions on licences and permissibility.

As soon as the above milestones are passed, SKB will increase the pace of the preparatory work. For example, extensive efforts relating to detailed design of facility parts and technical systems will be initiated after the Government's decisions on licences and permissibility.

Before construction of the facilities can be commenced, two documents must be submitted and approved by SSM:

- the preliminary safety analysis report (PSAR), which also includes a description of the build-out of the repository's deposition areas and how this affects safety,
- a special document (Suus) that describes how the construction of the facility affects safety during the operating phase and after closure.

The documents submitted to SSM shall take into account results of e.g. technology development and design activities since the applications were compiled. The construction projects shall at this time be finished with detailed design and tendering specifications for the first facility parts to be built. The organization shall be ready to conclude procurements and begin works on the site.

2.3.2 Interim storage

Increase of storage capacity

SKB's current licence for Clab covers interim storage of 8,000 tonnes of fuel. According to current forecasts, this amount will be reached about 2023. Since the fuel from the storage pools cannot begin unloadning before 2029, SKB will apply for a licence to increase Clab's permitted storage capacity. SKB plans to submit such an application to SSM in around 2018. Decisions are then required by both SSM and the Government. Before an application is submitted, SKB will study what an increase in storage capacity would entail for the facility. The storage pools hold about 11,000 tonnes of fuel. It has not yet been decided what capacity SKB will ultimately decide to apply for.

In order to receive fuel for interim storage up to unloading begins, measures need to be adopted to permit the existing storage volume to be utilized more efficiently. There are two options: the fuel and control rods can be stored more densely, and core components and control rods can be interim-stored dry, see also Section 3.3. A study of possible measures is planned during the upcoming RD&D period.

Upgrade of existing facility

Clab is approaching 30 years of operating time, and system upgrades and component replacements will be necessary in the future.

A number of projects are already being pursued today. An upgrade of the cooling chain is being carried out in order to obtain increased cooling capacity and redundancy, galvanized fire-water pipe is being replaced with stainless steel pipe, and a membrane filtration system is being installed for improved treatment of effluent. New compact storage canisters have been ordered and are now being fabricated.

An inventory of the systems in Clab has been commenced to investigate the long-term need for improvements. A number of improvement projects are scheduled today. They include modernization and upgrading of the fuel elevator and facility modifications for handling of new fuel transport casks. Further changes being considered are new electricity supply pathways and modernization of overhead cranes. Planning and execution of the alterations in Clab will be integrated with planning and construction of the new facility part for encapsulation.

2.3.3 Encapsulation

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. After SSM's initial assessment, the authority announced that the application needed to be supplemented and that Clab and the encapsulation plant should be regarded as a single facility, Clink. SKB submitted the requested supplementary information in October 2009. In March 2011, SKB supplemented its application under Nuclear Activities Act regarding Clink while at the same time applying for a licence under Nuclear Activities Act for the Spent Fuel Repository and under the Environmental Code for the entire final repository system, including Clink and the Spent Fuel Repository.

Current and planned technology development mainly concerns the processes around handling of the spent nuclear fuel, fabrication of components for the canister, seal welding and nondestructive testing of components and seal welds. This is described in Chapters 10, 11 and 12.

Licensing

Design, construction and commissioning of Clink will be pursued by SKB in the Clink project.

The preparatory preliminary safety analysis report (P-PSAR) for the facility that was submitted together with the application describes how nuclear radiation safety in Clink will be maintained. SKB will also submit supplementary material to the Authority during the licensing process. The design of the facility and how the requirements are met will gradually be clarified and detailed as technology development progresses and viewpoints are received from the Authority during the licensing process.

During licensing and up to the start of construction of the encapsulation part, SKB will design Clink and procure its construction. This phase is called the design and procurement phase of the project and will be divided into four partially overlapping subphases: facility configuration, system design, detailed design and procurement of construction.

During the facility configuration phase, SKB intends to update the set of requirements for Clink based on requirements identified during technology development and viewpoints expressed by the Authority during licensing, and to modify the facility's preliminary layout as needed.

SKB then intends to proceed with system design of Clink. The further detailing of the design and the activity that occurs in conjunction with this will be described in a PSAR for the integrated Clink facility. SKB will then prepare a project plan for construction of the encapsulation part and changes in the interim storage part.

When SSM has approved the PSAR for Clink, SKB will continue detailed design and carry out the procurements required to commence construction of the encapsulation part and implementation of the changes that need to be made in the interim storage part. SKB will also announce the changes that need to be made in the interim storage part in the form of "change matters" in accordance with the requirements in SSMFS 2008:1.

The work during detailed design is supposed to provide a basis for procurement of the facility and system and also 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.

Construction

The construction phase will begin when SKB has obtained all licences and conditions needed to start construction of the encapsulation part. This means, among other things, that a PSAR must have been submitted to and approved by SSM.

Construction will be divided into two parallel parts, one for the changes in the existing Clab and one for construction of the new encapsulation part.

In Clab, the facility changes that are needed in the form of installations and building works will be executed. In parallel, rock, building and installation works will be carried out for the encapsulation part.

Operation of Clab will continue throughout the construction of the encapsulation part. This entails that the contract works have to be adapted so that safety can be maintained in Clab and operational disturbances can be minimized. The extension of Clab, stage 2, was carried out in a similar manner, and experience from that project is important in designing and building the encapsulation part.

The construction phase will be concluded when the two facility parts are connected, physically and process-wise. This will take place by opening of the walls between the interim storage part and the encapsulation part and interconnection and co-testing of the installations in both facility parts.

Commissioning

When the facility parts have been connected, initial tests of operation will be conducted. Spent nuclear fuel will be received for interim storage throughout the commissioning of Clink. The commissioning of Clink therefore has to be adapted so that safety can be maintained in interim storage part and operational disturbances can be minimized.

The initial tests of operation will consist of three phases: testing and commissioning of individual systems, integrated testing of Clink and integrated testing of the whole KBS-3 system.

Integrated testing of Clink entails testing of the function of the facility by test-running the whole encapsulation process without spent nuclear fuel, i.e. with dummy fuel assemblies and copper canisters. Integrated testing of the KBS-3 system entails testing of Clink together with the transportation system and the Spent Fuel Repository.

SKB plans to apply for a licence to commence trial operation of Clink at the end of 2027. It is assumed that trial operation will start one year later. An application for a licence for routine operation is planned to be submitted at the end of 2029, and a licence is expected to be obtained by mid-year 2030.

Production system for canisters

SKB's production system for canisters should secure a long-term supply of canisters to Clink in preparation for the start of operation of the Spent Fuel Repository. The production system for canisters will include a number of external suppliers who fabricate nodular iron inserts and copper components for the canisters according to SKB's specifications. SKB plans to build and operate a canister factory where final machining, assembly and quality assurance of canister components will take place. The result will be a canister that is delivered to Clink. The supply of canisters must be reliable and cost-effective and capable of meeting SKB's need for canisters during the expected operating time.

The canister factory will not be a nuclear facility, since spent nuclear fuel will not be handled in the plant. Activities will include machining of canister components, testing against established criteria, and assembly of canister components. The work with the canister factory is in an early stage compared with the Spent Fuel Repository and Clink. This is because the timetable for construction and commissioning of the canister factory is substantially shorter than for the other facilities in the KBS-3 system. A feasibility study describing the factory's functions, layout and machinery has recently been published. One of the purposes of the feasibility study was to update previous studies and clarify strategic decisions regarding e.g. boundaries with external suppliers and the scope of the quality work in canister production.

During 2013, Posiva and SKB have deepened their cooperation regarding technology development. One of the matters that has been discussed is the prospect of operating the canister factory together, which could create synergies in the future.

2.3.4 Transportation

The transportation system will be supplemented with a new type of transport cask for shipping encapsulated spent nuclear fuel from Clink to the Spent Fuel Repository. This transport cask is called the canister transport cask (KTB). Due to its activity content, it will be certified according to the IAEA's transport recommendations (IAEA 2012) as a Type B package.

Feasibility studies of possible transport casks for copper canisters were conducted in 2005 with international suppliers. An update/supplement of the earlier studies that takes into account new regulatory requirements and changed premises in SKB's system is planned during 2013. The new feasibility study should serve as a basis for system design of the encapsulation part of Clink and the Spent Fuel Repository as well as for the future procurement of KTBs.

The transport cask will be designed in cooperation with the chosen supplier. Design will be done in an iterative process with a focus on satisfying the regulatory requirements and SKB's own premises, specific requirements and expressed preferences. The design and safety-related properties of the cask are described in a safety report as a basis for obtaining a licence from the competent authority in the country where it is fabricated. Before the cask may be used in Sweden, a validation of the licence must be obtained from SSM.

The time required for design and licensing is estimated to be 3.5 years and for fabrication 1.5 years. The canister transport cask will be tested together with Clink and the Spent Fuel Repository before the start of operation.

The first cask will be delivered to Clink and the Spent Fuel Repository prior to the testing of individual systems. With today's timetable, this testing can be started in around 2025. The initial system-specific tests will be conducted one year before integrated testing of the whole KBS-3 system is done. The remaining casks will be fabricated and delivered as they are produced during the period 2027–2030, in parallel with integrated testing and trial operation.

2.3.5 The Spent Fuel Repository

Licensing

A principal task for the Spent Fuel 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 todays. 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 in Forsmark will be to continue to build up an organization that is suited to this.

Efforts centred on the final repository's environmental impact will also continue during the licensing process. This includes current efforts to create a habitat for the pool frog, which lives today in a pond that will be filled in during construction of the facility. Four new ponds have been created to which the frogs will be relocated. SKB has applied to the County Administrative Board in Uppsala for a species protection exemption in this matter. The exemption granted by the County Administrative Board has been appealed.

The impact, if any, of the facility on the water levels in surrounding wetlands will also be studied and possible measures to maintain the water levels will be evaluated based on practical tests.

The final repository is being designed in parallel with the licensing process. System design of the final repository's facility parts and technical systems is currently in its final phase. The intention is that this design step will serve as a basis for continued facility development, followed by detailed design. This will in turn result in construction documents and procurement 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. During licensing, detailed design is therefore carried out primarily 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 in some cases as part of the actual construction process.

Construction-related and engineering geological 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 final repository.

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 well-established work flows.

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. 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. Up 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 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 drawn 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. Such a methodology was described in Appendix VU in the application under the Nuclear Activities Act (SKBdoc 1199888).

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 driven in which deposition holes are bored. One purpose of finishing 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. Results and documentation from technology development are also needed for execution of detailed design, see Chapter 10.

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.

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 built.

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 Spent Fuel Repository under realistic conditions. All operational activities will then be carried out, including deposition of a number of canisters, but without any spent nuclear fuel. Deposition takes place in the first deposition area built during the latter part of the construction phase. Finally, integrated testing of the KBS-3 system is carried out, which entails that the Spent Fuel Repository is tested together with Clink and the transportation system.

The commissioning phase is concluded when SKB obtains a licence for trial operation of the final repository system. All functions and resources, as well as rooms for deposition, should then be available so that trial operation can be commenced.

SKB plans to apply for a licence to commence trial operation of the Spent Fuel Repository at the end of 2027. It is assumed that trial operation will start one year later. Application for a licence for routine operation is planned to be submitted at the end of 2029, and a licence is expected to be obtained by mid-year 2030.

3 Flexibility in the face of changed premises

SKB's planning is based on the assumptions that apply today for the current nuclear power programme, see Section 1.1.4. Planning of the nuclear waste programme is based on the strategic assumptions that are judged to be most realistic today. The current time horizon is about 70 years, so SKB has to assume that changes will occur in the planning premises and that the current assumptions for the planning may be re-evaluated.

This means that great flexibility will be required to adapt to changes in the premises for the programmes. It is often possible to adapt 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 measures, 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 in adapting 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 the assumption that the reactors in Forsmark and Oskarshamn, as well as Ringhals 3 and 4, will be operated for 60 years, while Ringhals 1 and 2 will be operated for 50 years. This means that the last reactors will be taken out of service in about 2045. Any changes in these assumed operating times entail changes in the planning premises for the nuclear waste programme.

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 programme for spent nuclear fuel, current planning premises entail final deposition of about 6,300 canisters. Extending the operating time of a reactor by one year means that another 10–30 tonnes of spent nuclear fuel need to be interim-stored and another 5–15 canisters need to be disposed of. The volume depends on the size of the reactor and its operating time during the year. When the additional quantity of fuel needs to be stored, unloading of fuel from Clink is expected to have started so that there will be room for this fuel. According to current plans, unloading will begin in 2029. 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 an extension of SFR is currently under way. The design capacity is based on current operating plans for the nuclear power reactors, but with an uncertainty allowance 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 dismantling and demolition can begin. If the operating time for an individual reactor is extended, this means that the need for deposition of waste from dismantling and demolition is deferred for this particular reactor, without affecting the planning for the other reactors. If all the reactors' operating times are extended, there will be a general deferral of decommissioning and of the need for disposal volume for the waste from dismantling and demolition. The increased quantities of nuclear waste and spent nuclear fuel would require postponement of the closures of the final repositories. This would 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 take changed premises into account 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 systems. All existing and planned facilities for disposal of nuclear waste and spent nuclear fuel will nevertheless be needed. 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.

If all the reactors' operating times are shortened, decommissioning and disposal of waste from dismantling and demolition will have to commence earlier. Reduced quantities of nuclear waste and spent nuclear fuel would also means that the nuclear waste programme could be concluded sooner. The scope of these reschedulings depends on how many reactors are affected and how much the operating times are shortened.

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 Commissioning of the extended SFR

According to current plans, the extension of SFR will be finished so that deposition of waste from dismantling and demolition of the Barsebäck plant, Ågesta and the plants on the Studsvik industrial site can begin in 2023. During the spring of 2014, SKB plans to submit applications for the extension in accordance with the Environmental Code and the Nuclear Activities Act. Construction is planned to start at the beginning of 2018.

The disposal volume available today in SFR is not large enough to accommodate the short-lived waste from the decommissioning of Barsebäck. Furthermore, the licence that exists today is only for deposition of operational waste in SFR. Planning for the dismantling and demolition of Barsebäck must therefore be coordinated with the extension of SFR. Barsebäck Kraft AB plans to begin decommissioning with removal of the reactor internals. These components are long-lived and will be interim-stored on the power plant site awaiting transport for interim storage in the extended SFR. Due to the delay of the extension of SFR, Barsebäck Kraft AB is exploring different options for interim storage of the radioactive waste so that dismantling and demolition can begin before SFR is extended. In the event interim storage is not possible, a delay of SFR will entail increased costs as a consequence of extended shutdown operation. A delay would not have any negative radiological consequences.

A delay of the extension of SFR could also have consequences for the timetable for dismantling and demolition of Ågesta. The possibility of interim storage to become independent of SFR is currently being explored. A delay here will also entail increased costs due to increased transport and interim storage needs.

Of the reactors that are in operation, the first reactors, Ringhals 1 and Ringhals 2, will, according to the nuclear power companies' planning, be shut down in around 2025–2026. In the event the SFR extension has not been put into operation by this time, dismantling and demolition of these reactors will be delayed. Alternatively, Ringhals AB can apply for a licence to store waste from dismantling and demolition temporarily on the power plant site, similar to what Barsebäck Kraft AB is planning. Ringhals AB has a special storehouse for interim storage of short-lived and long-lived operational waste. The storehouse has a capacity of about 10,000 cubic metres. The company estimates that the storehouse has capacity for interim storage of the RPV from Ringhals 1 and large components from dismantling and demolition of the Ringhals plants. A licence for interim storage of waste from dismantling and demolition will then be needed. Ringhals AB also has a building where end-of-life steam generators have been stored. The RPVs from PWRs are planned to be interim-stored in this building awaiting commissioning of SFL.

Deposition of operational waste could also be affected by a delay in the extension of SFR. The repository part that is expected to be fully utilized first is the rock vault for low-level waste (BLA), which could be filled up in only a couple of years. Before the extended SFR is commissioned, the low-level operational waste will be interim-stored, for example at the power plant sites. Opportunities for interim storage are currently good at all power plants, and capacity is judged to be sufficient even in the event of a couple of year' delay. The margins in other repository parts in SFR are relatively large and are not judged to be affected by such a delay.

3.3 Commissioning of the Spent Fuel Repository and Clink

SKB plans to commence trial operation of the Spent Fuel Repository and Clink in 2029. This means that unloading of fuel from the interim storage part will begin then. The storage capacity in Clab has two limitations: the permissible quantity of spent fuel in the facility, and the number of physical storage positions in the pools. SKB currently has a licence to store 8,000 tonnes of fuel in Clab. According to today's forecasts, this amount will be reached around 2023. SKB is therefore planning to apply for a licence to increase Clab's storage capacity around 2018. Besides spent nuclear fuel, control rods and core components are also stored in Clab today. If the facility were only used for interim storage of fuel, the storage pools would be able to hold about 11,000 tonnes of fuel.

According to current forecasts, the number of storage positions will be filled by about 2026 unless specific measures are adopted. There are, however, a number of possible measures whereby the existing storage volume could be used more efficiently. It is possible to transfer the fuel stored today in normal storage canisters to compact storage canisters. The control rods could be stored more densely, or core components and control rods could be interim-stored elsewhere.

Transferring fuel to compact storage canisters is a relatively simple measure and has been done previously when SKB got a licence in 1992 to increase Clab's storage capacity from 3,000 to 5,000 tonnes of fuel. (Since then Clab has been extended with a second rock cavern.) Switching to only using compact storage canisters would free up volume so that the storage capacity would suffice until 2029.

Different types of core components are stored in Clab today. They include boron plates, fuel boxes and different types of components that have been replaced at the NPPs. One possibility is to transfer the core components to steel tanks (BFA tanks) for dry interim storage. Steel tanks are already used today for interim storage of this type of waste. This would free up storage volume equivalent to about 200 tonnes of fuel.

Control rods from BWRs are stored in Clab in storage canisters that hold nine control rods. The control rods from PWRs are integrated in the fuel and require no extra room in Clab. SKB is investigating the possibility of segmenting the BWR control rods, see Section 8.2.5. The purpose is to achieve more compact interim storage, either for continued wet interim storage in Clab or for dry interim storage in e.g. steel tanks. In the event the control rods continue to be stored in Clab, SKB estimates that segmentation could reduce their volume requirement by half. A compaction of the control rods could free up room for about 600 tonnes of fuel. The alternative of interim-storing the control rods dry would free up room for about 1,200 tonnes of fuel.

This provides flexibility in the event of a delay in the commissioning of the Spent Fuel Repository and Clink as well. Based on today's forecasts, Clab's storage positions will be filled by 2036. This is assuming core components and BWR control rods are stored dry and that all fuel is stored in compact storage canisters.

If it should prove necessary, it is also possible to extend Clab with a third rock cavern with storage pools. This would only be necessary in the event of a long delay in the programme for spent nuclear fuel. An extension would probably be done in a similar manner as Clab stage 2, which was put into operation in 2008. If Clab is extended with a third rock cavern, the facility would be able to receive another 5,000 tonnes of fuel. Before SKB makes a decision on a third rock cavern, the option of dry interim storage of fuel will also be explored. Among other things, aspects related to long-term safety must be analyzed. Dry interim storage is used today by a number of countries, including Spain and Germany.

3.4 New nuclear power reactors

Given the possibility of generational changes in the nuclear power stock, existing reactors may have to be replaced with new ones. Even though SKB's mission is restricted to the present-day nuclear power programme, it could be broadened so that SKB's competence can be utilized for waste from new reactors as well.

3.4.1 Replacement of existing reactors of the third generation

In 2012, Vattenfall AB submitted an application to SSM for a licence to replace one or two existing reactors with new ones. Licensing to build and commission one new reactor is a stepwise process that will take many years. A licence will be required from the Government. At this point, a plan for waste management will also be needed.

New reactors would entail that the total volume of waste could be much greater than has been assumed. It is not possible today to predict the size of the waste volumes that would result from generational changes of the reactors. These volumes depend above all on how future economic considerations affect the willingness to invest in new nuclear power.

Current plans call for today's ten reactors to be taken out of service during the period 2025–2045. If new reactors are phased in at that time instead of the old one, the new scenario entails a total remaining operating time of nuclear power of up to 90 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 power reactors. They are in many respects 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 from todays. 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 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 today's 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 nuclear power companies have over the years improved the treatment and compaction of the waste for the purpose of minimizing its volume.

The spent nuclear fuel from the new reactors will have a higher average burnup than the nuclear fuel from today's reactors, considered over the entire life of the reactors. The higher burnup means that the nuclear fuel will need a longer period of interim storage before it can be encapsulated and disposed of. An interim storage period of about 60 years will be required. This, together with the new reactors' longer operating times, means that interim storage and encapsulation would need to continue well into the 22nd century. The fuel assemblies in the new reactors will probably be longer than those used in today's reactors. Clink will probably not be able to be used for fuels from new reactors. New facilities will then be needed for both interim storage and encapsulation. Well-developed technology is available today for both wet and dry interim storage of spent nuclear fuel, and both techniques are thereby conceivable.

It is difficult to know at present whether the capacity of the Spent Fuel Repository will be sufficient for the new nuclear fuel as well. This requires new investigations. But tunnels and shafts could be kept open for the time required. The need for alterations in the Spent Fuel Repository is deemed small. The need for a new repository will thereby probably be determined by the available disposal capacity.

It is assumed below that all of today's ten reactors are replaced with new ones, but another conceivable scenario is that only some of the existing reactors are replaced. The net power 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 should then be most advantageous in terms of both safety and economy to site the new repository areas adjacent to the Spent Fuel Repository. 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 disposal requirement, SKB can investigate the possibility of building an additional level in the Spent Fuel Repository. This would enable the upper level to be closed after this generation of nuclear power plants, at which point operation of the lower level would commence.

Furthermore, a scenario with replacement reactors entails an additional need of disposal capacity for operational waste of roughly the same scope as SFR with the currently planned extension. Additional disposal rooms can be built adjacent to SFR, provided the bedrock is considered suitable, or as a separate facility on another site.

There will also be a need for additional disposal capacity in the final repository for long-lived waste, SFL. Construction of SFL would have to be commenced roughly as planned today in order to dispose of waste from today's reactors, but the facility needs to be kept in operation for a longer time.

3.4.2 Future fast reactors of the fourth generation

The trend in various parts of the world is towards fast reactors. These reactors have the potential to utilize uranium much more efficiently (50–100 times) than light water reactors. A great deal of development work remains to be done on both the reactor technology and the associated fuel cycle, and fast reactors are not expected to be available for commercial operation on a large scale until the second half of this century.

Fast reactors have another type of fuel cycle that could affect the Swedish nuclear waste programme in several ways:

- Some of the spent nuclear fuel is saved to start a fast reactor programme[3], while the Spent Fuel Repository continues to be operated for most of the spent fuel.
- The Spent Fuel Repository is designed so that waste from reprocessing of the existing fuel and new fuel from fast reactors can be disposed of there.
- Since fast reactors require reprocessing, new waste streams will be generated in the Swedish system. Reprocessing gives rise to high-level vitrified waste and new types of low- and intermediate-level waste. In addition, considerable quantities of long-lived neutron-induced waste are obtained from the reactor core (SFL waste).

SKB's assessment is that the development of fast reactors does not affect the work with the Spent Fuel Repository. The main reason for this is that even if fast reactors should come into commercial operation on a large scale in around 2050, so much plutonium will be available in spent fuel around the world at this time that most of the spent fuel will still need to be disposed of.

[3] Plutonium from light water reactor fuel will only be needed for the first few years' operation of a fast reactor. The reactor then generates its own plutonium from depleted uranium, which is used to fuel the reactor.

Part II

Low- and intermediate-level waste

4 Management of low- and intermediate-level waste

The Swedish nuclear facilities generate low- and intermediate-level waste in the form of operational waste and, in the future, waste from dismantling and demolition. SKB works together with the waste producers to dispose of the low- and intermediate-level waste in a satisfactory manner. In order to be able to manage and store radioactive waste, it is necessary to categorize it based on the half-life and activity content of the radionuclides, as described in Section 1.1.4 "The radioactive waste and the spent nuclear fuel."

Part II of the RD&D programme deals with both current and planned management and disposal of the low- and intermediate-level waste that arises during operation and decommissioning of the Swedish nuclear facilities. Experience from the operation of the existing final repository for short-lived radioactive waste (SFR) provides vital knowledge in the work of developing and establishing new final repositories for low- and intermediate-level waste. The overall planning of this work is presented as a part of the company's planning in Section 2.2.

This chapter gives an account of the management of the existing short-lived and long-lived low- and intermediate-level waste and programmes for the development work being pursued by SKB in this area. The operation of SFR and the current and planned development of the facility, as well as the technology and method development being pursued by the waste producers, is also described.

What is included in the planned management and disposal of low- and intermediate-level waste is described in the following Chapters 5 to 9. The work of extending SFR to hold all short-lived waste arising during operation and decommissioning of the nuclear facilities is described in Chapter 5. The current situation and planned activities for the future final repository long-lived waste (SFL) are described in Chapter 6, which also includes an account of the waste quantities and when they are expected to arise. Waste with very low activity is currently managed by the nuclear power companies themselves. SKB is investigating whether near-surface repositories can offer an alternative for fractions of the waste from decommissioning, which is dealt with in Chapter 7. The technology development currently being pursued to develop new final disposal solutions for low- and intermediate-level waste, but also the development needed to maintain and guarantee the functions of the existing SFR, is described in Chapter 8. Chapter 9 describes the division of responsibilities, planning and technology for decommissioning of nuclear facilities. The scientific research pertaining to low- and intermediate-level waste is described in Part IV of the RD&D programme.

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

In recent years, the Swedish waste producers have worked on various ways to reduce releases of radionuclides to the environment, such as by improved technology for treatment of air and water and by reducing discharges of contaminated water. This can affect both the quantity and the nuclide content of the short-lived low- and intermediate-level waste. Before the waste is transported to the final repository, it undergoes treatment by the waste producers. Besides packaging the waste, other purposes of this treatment can be to reduce its volume or modify its physical and chemical properties. In order for the waste to be deposited in the repository, a type description approved by SSM (Swedish Radiation Safety Authority) is required. Deposition rules for waste deposited in SFR are presented in Section 4.3.1.

Conclusions in RD&D 2010 and its review

RD&D programme 2010 gave a general account of established technologies and programmes for treating water and conditioning waste.

SSM pointed out in its comments on RD&D programme 2010 that a more detailed account should be given of methods that are not so well established. SSM further noted that it is urgent that the work of reducing radioactive releases should continue and that methods for managing and disposing of the waste should be developed.

SSM wanted SKB and the waste producers to show what steps are being taken to reduce the waste quantities as far as is reasonably practicable.

4.1.1 Sources of short-lived low- and intermediate-level waste

Technologies that are utilized to a varying extent today by the different waste producers to treat effluent are filtration, ion exchange, evaporation, membrane filtration and centrifugation. In some cases, columns of activated charcoal are used to reduce atmospheric emissions. The treatment of water and air gives rise to operational waste in such forms as filters, ion exchange resins and evaporator concentrates.

Programme

Ringhals AB (RAB) has been working for some time to put an evaporator into operation. The evaporator has been test-run, and after the storage space for evaporator concentrates has been increased it will be put into continuous operation. In order to achieve a higher dry matter content of the evaporator concentrates, a vacuum evaporator will be used as a secondary treatment step. The boron concentration in the evaporator concentrates must not be too high, which is why RAB is investigating the possibility of pretreating boron-containing drainage water from the pressurized water reactors with a pressure-driven membrane filter. Testing of a suitable membrane for boron removal has begun.

There are also plans to use pressure-driven membrane filtration at Clab (the central interim storage facility for spent nuclear fuel) and OKG Aktiebolag (OKG). Design of a membrane system for treatment of wastewater containing a mixture of radioactive silver nuclides and organic matter is under way at Clab. A test rig was installed at OKG in the spring of 2013 where an ultrafilter-type membrane is being tested for treatment of water from floor drainage. Testing is expected to take about 1.5 years, after which a decision can be made on whether the technology will be adopted or not.

Forsmarks Kraftgrupp AB (FKA) has used evaporators to treat effluent for many years. Due to problems with water-soluble radionuclides in the vapour phase, they are currently looking at the possibility of augmenting the evaporator with a cesium-selective ion exchanger and increasing the capacity and efficiency of the evaporator. Evaluation of the cesium-selective ion exchanger is expected to lead to a decision on whether to install one or not. The evaporator is undergoing a feasibility study and testing programme where different operating parameters will be tested, after which a decision is expected to be made during the coming RD&D period.

Finally, SVAFO is planning to develop the treatment technology for effluent by building a vacuum evaporator. Today, low-level liquid waste from Studsvik Nuclear AB and SVAFO are mixed. Treatment involves the addition of iron chloride followed by sedimentation, where the sludge from sedimentation is solidified in campaigns. Besides reducing the activity level in the effluent, the introduction of a vacuum evaporator will also mean that low-level liquid waste from SVAFO does not have to be treated together with low-level wastewater from Studsvik Nuclear AB.

4.1.2 Reduction of short-lived low- and intermediate-level waste

The waste producers work constantly to reduce waste volumes by reducing the flow of material into the controlled area and by clearance and source-separation of materials. Each facility's waste plan includes clearance, and the waste producers generally have their own internal procedures to ensure that as much material as possible is cleared from regulatory control, taking into account costs and environmental aspects. Hazardous substances are usually cleared to permit suitable management and disposal.

If the waste has a very low radionuclide content but is not "clearable", it may be eligible for deposition in a near-surface repository. This is a way to reduce the waste quantities in SFR and can in principle be regarded as a "deferred clearance". In terms of radioactivity, waste deposited in a near-surface repository generally lies between "clearable" waste and the waste deposited in BLA (rock vault for low-level waste) in SFR.

Each near-surface repository has a specified year when the quantity of radioactivity in the repository shall be below the limit for clearance, expressed in the form of radionuclide-specific limit values. Furthermore, for each near-surface repository there is a limit on the maximum surface dose rate on the waste packages of 0.5 millisievert per hour (mSv/h), and a limit on the maximum permissible total activity content in the repository. Deposition takes place solely in the form of deposition campaigns where the waste producer applies for a permit to SSM and gives notification of deposition to the County Administrative Board.

Programme

In order to meet today's requirements on low background radiation in connection with clearance measurements, both RAB and OKG intend to house the clearance activity in a separate building or part of a building and purchase new monitoring equipment. At OKG a building section has already been modified for this purpose and monitoring equipment purchased. RAB has plans to introduce decontamination for clearance of materials with washing and sand blasting in a separate building. FKA is also investigating the possibility of a separate building with lower background radiation for clearance, which would mean shorter measurement times. Barsebäck Kraft AB (BKAB) is currently carrying out its own clearance of tools and equipment as well as waste oil and has signed an agreement with Studsvik Nuclear AB for other materials. BKAB is planning to eventually acquire its own expertise for in-house clearance of more materials. SVAFO is designing a mobile monitoring station that can be moved between different facilities for clearance measurements.

In order to raise the level of knowledge in the industry in the area of decontamination, a project is under way to compile national and international experiences. This work will result in a decontamination manual, and plans also exist for a training course.

4.1.3 Characterization of short-lived low- and intermediate-level waste

Both direct and indirect methods of measurement, as well as correlations and calculations, are used to determine the radionuclide-specific content of waste. RD&D programme 2010 described a development project to establish methods for measurement of nickel-63. As a result of this work, FKA and OKG analyze nickel-63 by means of liquid scintillation after sample preparation by double solid-phase extraction. Yield is determined by analyzing stable nickel, and the fraction of interfering radionuclides is determined by gamma spectrometry. Another method involving sample preparation in the form of separation by ion chromatography followed by analysis of nickel-63 by liquid scintillation is mentioned in RD&D programme 2010. This method is used by RAB, but is being abandoned in preference to the method now being used by FKA and OKG. RAB has purchased the method for analysis of nickel-63 used by FKA and OKG. The Swedish Defence Research Agency (FOI) is working under contract to RAB to certify and deploy the method so that it can also be used on boron-containing water.

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

Methods that are used to a varying extent by the different waste producers to condition the waste include solidification with cement and bitumen and grouting in concrete. These methods are sometimes preceded by incineration, segmentation and treatment with UV light to break down organic matter. In order to reduce the waste volumes and better utilize the capacity of the ion exchange resins, RAB and OKG reuse ion exchange resins from condensate cleanup to treat floor drainage water.

Programme

The waste producers are working in various development projects to find ways to manage future waste streams effectively.

RAB is conducting experiments with electrochemical degradation of evaporator concentrates to reduce the fraction of complexing agents. Efforts are being made to optimize the method. RAB has also worked on reducing the content of complexing agents in the waste by replacing some soap and cleaning agents with products with a lower content of complexing agents. This has reduced the need for electrochemical degradation of floor drainage water, but it is still necessary for certain decontamination solutions. The plan is to finally embed the evaporator concentrates in cement in concrete moulds, and a type description for the waste is currently being developed.

FKA's trials with plasma arc incineration of evaporator concentrates (described in RD&D programme 2010) have been discontinued.

In order to better utilize the volume in moulds, OKG is introducing steel moulds instead of concrete moulds. Changing from concrete to steel moulds increases volume utilization by 70 percent.

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

4.2.1 Long-lived low- and intermediate-level waste

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

- Strongly neutron-irradiated core components. This waste arises in connection with both maintenance and dismantling and demolition of reactors.
- Control rods from BWRs. Control rods are consumed during reactor operation.
- Pressure vessels from PWRs. The waste arises during dismantling and demolition of reactors. It may be handled with core components and reactor internals left in the pressure vessel.
- Long-lived waste from Studsvik Nuclear's activities and from medical care, research and industry. This waste arises continuously and is not associated with the operation or decommissioning of the nucleas power plants (NPPs).
- Legacy waste from research and development within the Swedish nuclear power research programmes. This waste is managed and interim-stored by SVAFO.

Conclusions in RD&D 2010 and its review

In RD&D programme 2010, SKB presented updated predictions of when the long-lived low- and intermediate-level waste will arise. The total volume was given at 10,000 cubic metres of conditioned waste. The interim-stored quantity of waste, which was given at about 6,000 cubic metres, consists primarily of legacy waste which is managed and interim-stored by SVAFO. SKB showed how the quantity increases with time based on the current planning for replacement of core components and dismantling of reactors.

SSM said that the forecast corresponded to the account they had requested as supplementary information to RD&D programme 2007. SSM observes that around 60 percent of the waste has already been produced, while the rest of the waste is expected to arise when the nuclear power reactors are dismantled in the 2030s and 2040s, according to current plans.

Current situation

The work with the updated reference inventory for SFL is presented in Section 6.4.

4.2.2 Interim storage of 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 (mainly steel tanks) in different at-reactor interim storage facilities.

Legacy waste is managed by SVAFO and interim-stored – along with waste from Studsvik Nuclear's activities and from medical care, research and industry – in interim storage facilities on SVAFO's and Studsvik Nuclear's sites.

Conclusions in RD&D 2010 and its review

In RD&D programme 2010, SKB specified the capacity of the existing interim storage facilities (not counting storage pools at the NPPs and Clab). Their combined volume clearly exceeds the planned total volume of all long-lived low- and intermediate-level waste that will be disposed of in SFL. SKB also presented the plans for applying for permission to interim-store long-lived waste in SFR. By making it possible to interim-store long-lived waste in SFR, SKB ensures 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.

SSM had no essential objections to SKB's analysis and account of the interim storage of long-lived waste in SFR. SSM will not take a final stand on this option until the licence application for extension of SFR is considered.

Current situation

Within the framework of Project SFR Extension, SKB has studied the prospects for an interim storage facility for long-lived waste from the NPPs in the extended SFR. As a result of the study, SKB is planning to initiate interim storage of long-lived waste in SFR when the extended facility is put into operation. The details will be presented in the coming applications for extending SFR. SKB is also investigating the possibility of interim storage of long-lived waste from the NPPs at another location than in SFR.

4.3 SFR – Final repository for short-lived radioactive waste

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 respect to the waste to be emplaced in them, see Figure 4-1. SFR has been in operation since 1988 and has been operated by SKB since 1 July 2009.

Conclusions in RD&D 2010 and its review

In RD&D programme 2010, SKB presented the general principles for distribution of the waste between the different parts of SFR and the work that is under way or planned to extend the operating time of SFR. When the applications for an extension of SFR are submitted, a plan will also be submitted for how the facility with its planned extension can be utilized optimally.

SSM was also positive to the overhaul of SFR that was described in RD&D programme 2010. The comments revealed that they also thought that a review of the rules for directing waste to different parts of SFR should be broadened to include other types of final repositories such as near-surface repositories and SFL. This work should also include a review of the possibilities of minimizing the waste quantities as far as is reasonably practicable.

Finally, SSM discussed the need of improving the traceability of individual waste packages and different versions of type descriptions, as well as the distribution of actinides in SFR.

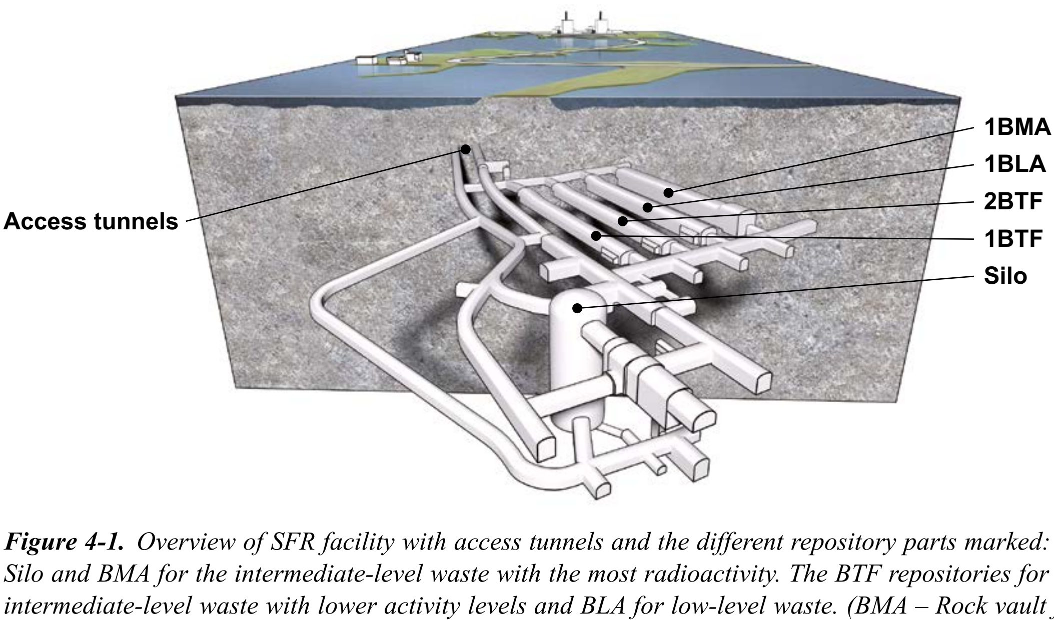


Figure 4-1. *Overview of SFR facility with access tunnels and the different repository parts marked: Silo and BMA for the intermediate-level waste with the most radioactivity. The BTF repositories for intermediate-level waste with lower activity levels and BLA for low-level waste. (BMA – Rock vault for intermediate-level waste, BTF – Rock vault for concrete tanks, BLA – Rock vault for low-level waste.)*

4.3.1 Operation of the facility

In 2012, SKB developed new acceptance criteria for waste in SFR. The acceptance criteria are a part of the safety analysis report for SFR and are intended to provide an overall picture of the requirements on the waste and ensure that the waste has the right properties at deposition.

The verifications prescribed in the type descriptions ensure that the acceptance criteria for the waste are fulfilled. Moreover, verifications are to be done in connection with transport, including inspection of the waste and checking of the waste package's waste data. At SFR, a visual check is made that the identity of the incoming waste agrees with what is specified in the shipping documents. In addition, SKB conducts waste audits at the waste producers.

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

The overall requirements on waste to be deposited in SFR are:

- The waste packages shall not give rise to unacceptable dispersion of radionuclides either pre- or post-closure.
- The waste packages shall be able to be handled without unacceptable impact of ionizing radiation on man and the environment.

It is above all the properties and origin of the waste that determine 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 repository destined for different parts of the repository are given in SFR's safety analysis report and SKB's waste manual for low- and intermediate-level waste. Table 4-1 presents the general principles.

In addition to the radionuclide content and the material in itself, the properties of the waste at closure are also dependent on the environment in SFR and the grout in which it is embedded. The operating and deposition procedures used in SFR, for example procedures for grouting, thereby determine the properties of the waste properties at closure.

There is an established forum for experience feedback in the waste field called the LILW Forum which consists of representatives from all waste producers and the department for low- and intermediate-level waste at SKB. The purpose of the forum is to provide information and investigate questions related to low- and intermediate-level waste.

Programme

Since 2012 there is a joint group for waste from dismantling and demolition called the Type Description Group for decommissioning waste. The group, in which all waste producers are represented, is supposed to develop common strategies for handling of type descriptions for waste from dismantling and demolition.

In conjunction with the applications for an extension of SFR, a safety assessment will be done showing whether the placement of waste in both the existing part of SFR and the extension is safe in the long term. Based on the safety assessment prior to commissioning of the extended SFR, an assessment will be done showing whether an adjustment of the planned deposition strategy can further enhance the safety of the repository. SKB also intends to re-examine how low-level waste from dismantling and demolition can be directed to near-surface repositories according to the description in Chapter 7 and to continue working on acceptance criteria for SFL, which is dealt with in Section 6.5.

4.3.2 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, which 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. In addition to remedial and preventive maintenance, the maintenance programme includes a programme for identification, handling and prevention of age-related deterioration and damage.

In the spring of 2012, a study was concluded of the SFR facility whose purpose was to determine the status of its systems. The study covered the areas of electrical installations, construction, mechanical installations and plumbing and heating and resulted in action plans for each area plus an action plan for the area of fire protection, which was not initially included. These action plans describe measures that are ranked according to the importance of the item or system function with respect to operation, personal safety and facility availability.

Table 4-1. General deposition rules for waste to SFR.

Waste type	Distribution of the waste between the repository parts BLA	 BTF	 BMA	 Silo
	Overall dose rate requirements for waste packages in 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 BWRs	After individual consideration 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	Dried sediment after individual assessment	Sludge in concrete tanks	Dried sediment in trash and scrap moulds	–
Evaporator concentrates from BWRs and PWRs	–	–	Concentrates deriving from different drainage systems	–
Ion exchange resins etc produced at SVAFO	–	–	After individual assessment, sludge from water treatment	Older waste from Ågesta, R2, etc after individual consideration
Trash and scrap from BWRs	Normally trash and scrap from intermediate building, waste building, turbine and generator building. After assessment, also 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 produced by Studsvik Nuclear	–	Ashes from incineration of trash.	Ashes from incineration of trash	–
Trash and scrap produced by Studsvik Nuclear and SVAFO	Low-level scrap from R2, HCL, ACL, waste facility, decommissioned facilities, hospitals, institutions, ABB Atom (Westinghouse Electric Sweden), nuclear power plants	–	Intermediate-level scrap from R2, HCL, ACL, waste facility, decommissioned facilities, hospitals, institutions, ABB Atom (Westinghouse Electric Sweden), nuclear power plants	Smoke detectors containing americium-241

Key to table: 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 unit 2, HCL – Hot cell laboratory, ACL – Active central laboratory.

4.3.3 Information management in connection with interim storage and deposition

SKB and the waste producers have systems for registration and reporting of waste packages that comply with SSM's requirements on information management. Previously, each organization had its own information management systems. A new tool, Gadd, is being developed to serve as a common database for the nuclear power plants and SKB. The system is primarily being developed to be used by SKB and OKG, after which BKAB, FKA and RAB will also be included. Studsvik Nuclear and SVAFO are not currently included in the plans for Gadd. SKB is responsible for the development and administration of the new database.

In a first version of Gadd, OKG and SKB can administrate information on the low- and intermediate-level short-lived waste. For waste sent to SFR there are functions for management of transport and deposition. The Gadd database facilitates information management between the different organizations. Information on waste delivered to SFR from organizations that do not use Gadd is managed by entering a waste data file with waste package information into the Gadd system. 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 waste is deposited in SFR, particulars on when and where the individual packages are deposited are entered into the database. Unlike the previous database Triumf, Gadd includes a 3D visualization of the deposited packages in the silo and BMA.

When Gadd is commissioned, data will be migrated from the replaced databases and verified to ensure that all previous data are preserved and have not been distorted.

Computation functions included in Gadd are used for compiling waste data and estimating the volume, material and activity contents of SFR. SKB's previous calculation tool, Triumf NG, is a standalone system that requires annual transfer of data from the Triumf database. One improvement offered by Gadd is that migration is no longer necessary, since the calculation module is included in the database. Registration of data for forecasts and non-package-bound data is handled manually as in Triumf NG, but now with better user-friendliness. After comments from SSM regarding the handling of non-package-bound data, the calculation methods for distributing activity over the waste in SFR have been further developed so that they also include waste that is interim-stored at the waste producers.

Programme

After the initial commissioning of Gadd, the development work will continue so that BKAB, FKA and RAB can also be connected to the database. Functions for SKB and OKG will be expanded and refined. As demanded by SSM, SKB will manage traceability between individual waste packages and different versions of type descriptions, which will be introduced in a clearer fashion when all nuclear power companies are included in Gadd.

In a later version of Gadd, the database is also expected to manage information on the waste that arises in connection with dismantling and demolition of the Swedish nuclear facilities as well as waste that is classified as long-lived low- and intermediate-level and is intended to be deposited in the future SFL.

5 Extension of final repository for short-lived radioactive waste

SKB's plans for the extension of SFR are related in this chapter. The planned extension entails an increase of the facility's storage capacity by an estimated 110,000 cubic metres plus space for nine BWR pressure vessels. SFR's current capacity is about 63,000 cubic metres. The purpose of this chapter is to provide an idea of the current status of the planning so that the applications of the results of research, development and demonstration in different phases can be more easily grasped. Technology development for final disposal of low- and intermediate-level waste is described in Section 8.1. More detailed accounts of the extension and the activities in SFR will be provided in the applications submitted in the spring of 2014.

The focus for the upcoming phase is to obtain licences to construct and operate an extended facility and to carry out the detailed design of the facility.

Conclusions in RD&D 2010 and its review

The plans for extension of SFR were described in RD&D programme 2010. In its review of the programme, SSM noted that further clarification is needed as to which waste serves as the basis for determining the capacity of the extension of SFR, especially with regard to waste from the facilities in Studsvik. SSM also concluded that the timetable needs to be re-examined to ensure there is enough time for regulatory review of the safety analysis reports, which have to be reviewed and approved prior to construction, trial operation and routine operation.

As far as siting is concerned, SSM said that they felt that the supporting material presented by SKB was far too limited to prove that Forsmark is the best site for a siting of the final repository for waste from dismantling and demolition.

5.1 Consultations

Consultations between SSM and SKB regarding extension of SFR have been held on two occasions in 2011 and four occasions in 2012, and another consultation is planned in 2013. In the consultations, SKB has presented the siting issue, progress in design, technology development and the safety assessment report. Furthermore, the design and contents of the application under the Nuclear Activities Act for extension of SFR has been discussed. In the consultations, SSM has expressed viewpoints on the work being done for the final disposal and interim storage of low- and intermediate-level waste in an extended SFR. Furthermore, SKB has held consultations pursuant to the Environmental Code as a part of the work of compiling the environmental impact statement that must be appended to the applications, according to the Environmental Code and the Nuclear Activities Act.

5.2 Superordinate requirements on the facility

The existing SFR 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 until their release has no radiological consequences. SFR thus has no absolute containment function. The same basic safety functions apply for the extension of SFR as for the existing facility.

As far as is reasonable and warranted, the facility should be designed so that the risk of radionuclide releases and future doses to personnel is minimized. Taking into consideration the different types of waste, the rock vaults should be designed so that they provide the desired barrier function, in interaction with the waste packages.

The extension of SFR should be designed to accommodate all additional short-lived low- and intermediate-level operational waste and all short-lived waste from dismantling and demolition that is expected to arise in connection with the decommissioning of all nuclear power plants, including the Ågesta Reactor and the research reactors at Studsvik, as well as the decommissioning of Clink (central facility for management, interim storage and encapsulation of spent nuclear fuel). All waste from hospitals, research and industry that will be produced or currently exists in the facilities in Studsvik is 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 operating time of the nuclear power plants (50 years for Ringhals 1 and 2 and 60 years for Ringhals 3 and 4 and for the reactors in Forsmark and Oskarshamn), except for the quantities expected to be able to be deposited in near-surface repositories.
- All short-lived waste from dismantling and demolition of the existing nuclear power plants, including Ågesta and the research reactors R2/R0 in Studsvik.
- All short-lived operational waste from Clab and Clink.
- All short-lived waste from dismantling and demolition of Clink.
- Forecasted waste volumes from Studsvik Nuclear and SVAFO.
- Final disposal of large intact BWR pressure vessels, except for highly neutron-irradiated internals.

5.2.1 Quantity of short-lived operational waste

The quantity of short-lived operational waste that is planned to be deposited in SFR at its closure in 2076 is estimated today to be about 68,000 cubic metres (SKB 2013a). The estimated volume is the sum of the volume deposited up to and including 31 December 2012 and the forecasted volumes for future operation.

5.2.2 Quantity of short-lived waste from dismantling and demolition

The quantity of short-lived waste from dismantling and demolition that is planned to be deposited in SFR at its closure in 2076 is estimated today to be about 84,000 cubic metres (SKB 2013a).

The forecast is based on data from studies for decommissioning, decommissioning plans and data for the different facilities. In addition to the inventoried waste quantities, an estimate has been made of the quantity of secondary waste that might arise during dismantling and demolition.

5.2.3 Uncertainties in waste volumes

The estimated quantity of waste that is intended to be deposited in the future is surrounded by numerous uncertainties. Some uncertainties are difficult to predict. Changes in laws and political decisions, operating conditions and the closure dates of the different nuclear power from what is currently assumed cannot be ruled out.

The forecast is based on operating experience and knowledge concerning how waste production has fluctuated in previous years. As regards waste from dismantling and demolition, uncertainties in the inventoried waste quantity are presented in the decommissioning studies. Aside from the inventoried waste – which includes iron/steel, concrete and sand – a certain fraction of secondary waste is expected to arise in conjunction with dismantling and demolition.

Another factor that can affect the future waste quantities is to what extent different types of aftertreatment are performed. This mainly concerns the waste from dismantling and demolition, since there are at present no established procedures for how it is handled. Examples of this are what packing degree can ultimately be attained and whether further volume reductions can be achieved by melting of low-level process systems. The waste quantities to SFR are also determined by the radiological clearance work and the use of near-surface repositories in the future. The option of depositing the very low-level waste from dismantling and demolition in near-surface repositories instead of in SFR would greatly reduce the waste volume to BLA and thereby comprise the main alternative. At present, however, all material from dismantling and demolition that is not free-released and is classified as short-lived is included in the design volume for the extension of SFR. It is assumed that much low-level operational waste can continue to be deposited in near-surface repositories.

5.2.4 Design waste volumes

An evaluation has been done of how the forecasted waste volumes and identified uncertainties will be included in designing the capacity of the SFR extension. It has been decided that the design waste volume for the extension of SFR is 110,000 cubic metres plus space for nine BWR pressure vessels. The decision is based on an attempt to balance the requirement of disposing of all waste that might arise against the risk of oversizing the facility. In addition to the basic forecasts of waste from operation as well as from dismantling and demolition, the design waste volume also includes some 15,000 cubic metres to account for uncertainties. Figure 5-1 shows the results of the latest forecast of the volumes of waste from operation and from dismantling and demolition. The low-level waste that does not fit in the existing facility will be interim-stored at the waste producers until the extended repository is put into operation.

5.2.5 Interim storage of long-lived waste

The SFR extension will be designed to permit interim storage of long-lived waste (core components) from the nuclear power plants. According to a preliminary estimate, the greatest volume which the interim storage facility may have to hold is about 2,800 cubic metres. The interim storage facility will be designed with a view to the needs of the power companies and the date when SFL is planned to be ready to receive waste. Studies show that an interim storage facility in SFR is technically feasible. In the safety analysis report that will be included in the applications for an extension of SFR, the interim storage of long-lived low- and intermediate-level waste will be treated as a part of the operation.

5.3 Acceptance criteria for waste to SFR

Waste acceptance criteria have been worked out for waste in the extended SFR. The waste acceptance criteria have been compiled on the basis of SFR's design premises and safety assessment, requirements from national authorities and international standards and recommendations. The new waste acceptance criteria have been combined with criteria for the existing SFR. The result is a combined set of requirements for short-lived low- and intermediate-level waste destined for disposal as well as for long-lived low- and intermediate-level waste destined for interim storage in the extended SFR.

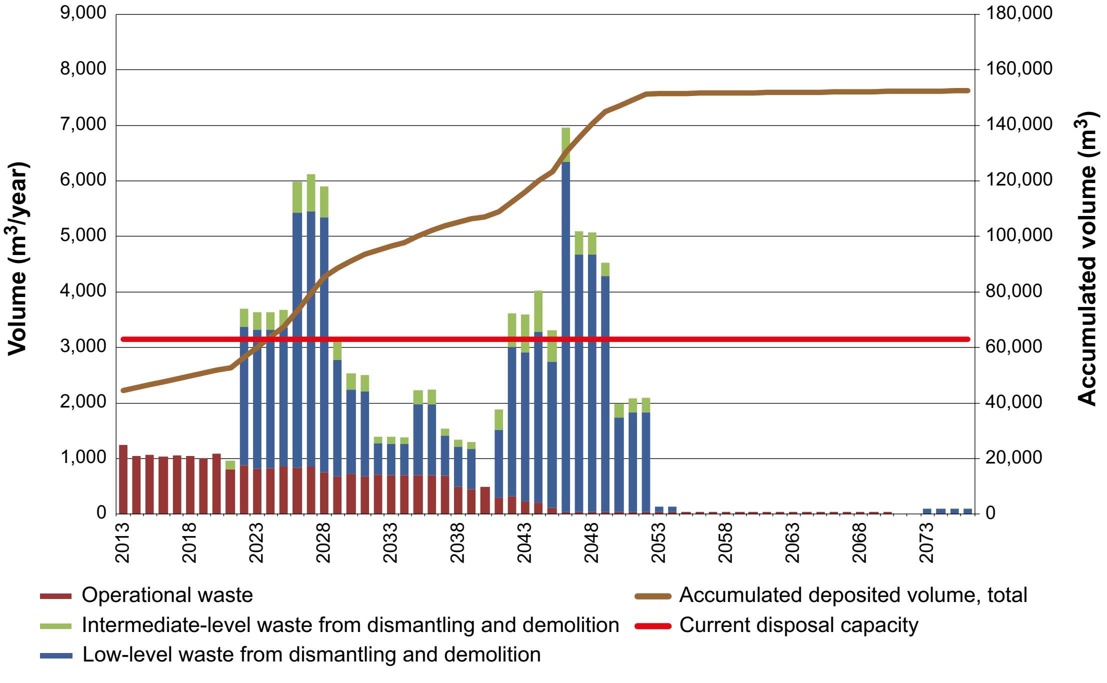


Figure 5-1. Forecast of volumes of short-lived waste from operation and from dismantling and demolition to SFR. The volume that arises during a year is read on the left-hand axis. The brown curve shows the accumulated volume of waste and is read on the right-hand axis (total about 155,000 m^3). Accumulated volume can be compared with current disposal capacity (red line, total 63,000 m^3).

5.4 Siting

A site study has been published in preparation for the applications for an extension of SFR (SKB 2013b). Siting a final repository for short-lived waste from dismantling and demolition adjacent to SFR appears logical for many reasons, such as the fact that the existing facility works well and the waste from dismantling and demolition is of a similar character to the operational waste.

The legislation requires a report on alternative sites for final disposal of short-lived waste from final dismantling and demolition. To satisfy these requirements, SKB has systematically compared different sites with respect to the following factors, which may have a bearing on the overall evaluation:

- Long-term safety.
- Technology for execution.
- Environment and health.
- Societal aspects.

Like the selected site, the alternative site shall offer good prospects for satisfying requirements on long-term radiological safety. Assessing this requires evidence in the form of geoscientific data and other information. Furthermore, the site shall offer good prospects for building and operating a final repository so that its establishment has a limited impact on the environment and human health. Robustness and efficiency in execution shall be striven for, and the costs of establishment and operation shall be reasonable. There shall also be good prospects of achieving societal acceptance for a final repository on the alternative site.

Based on experience from the SAR-08 safety assessment (SKB 2008a), the following site-specific safety-related factors have been judged to be of importance for final disposal of short-lived waste from dismantling and demolition:

- Low permeability in the bedrock.
- Low hydraulic gradient.
- Reducing conditions.
- Low seismic activity.
- Absence of ore potential.
- Low risk of wells being drilled.

Based on these requirements/preferences, general assessments can be made of long-term safety. SKB's previously investigated areas within the framework of the siting process for the Spent Fuel Repository (here called reference areas), see Figure 5-2, are possible objects of comparison, mainly due to good availability of geoscientific data but also because the purpose of the investigations in these areas has been to identify rock volumes suitable for final disposal.

A cursory examination of the safety-related factors indicates that most of the reference areas have rock conditions that could very well permit final disposal of short-lived waste from dismantling and demolition. None of the reference areas stands out as being obviously most suitable if only the safety-related factors are considered.

If other factors are also weighed in, a different picture is obtained. Several of the areas lack prospects for industrial establishment, and from a societal viewpoint it is highly doubtful whether any of the reference areas, with the exception of Simpevarp/Laxemar, is available as an alternative site. Several of the areas are located in municipalities that have declined participation in the siting process for the Spent Fuel Repository. In other cases the previous test drilling activities have met with local resistance. The possibilities of disposing of waste from dismantling and demolition on another site than Forsmark (at SFR) or Simpevarp/Laxemar are therefore highly uncertain. SKB's assessment is therefore that the only site that is a realistic alternative to Forsmark is Simpevarp/Laxemar.

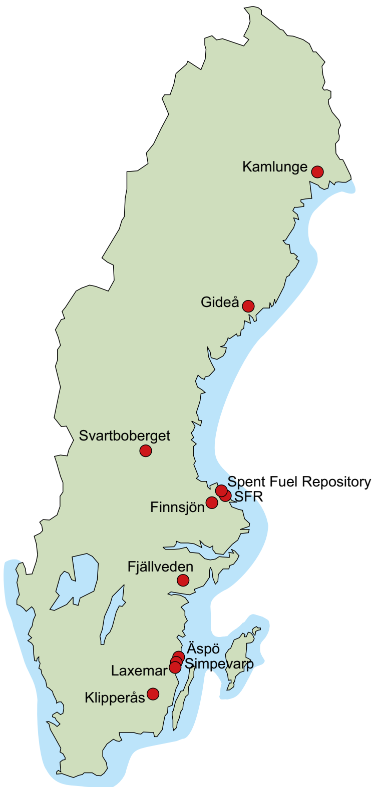


Figure 5-2. *Reference areas studied in the siting study.*

This is based on the following:

- Site investigation has been carried out on the Simpevarp/Laxemar site. This means that extensive data are available on the bedrock for assessing the suitability of the site for final disposal of waste from dismantling and demolition.
- Simpevarp/Laxemar has good prospects for technical feasibility with respect to rock construction and access to an existing transportation system and other infrastructure.
- Final disposal at Simpevarp/Laxemar entails limited impact on the environment and human health.
- Final disposal at Simpevarp/Laxemar can probably obtain societal acceptance.

In a comparison between the Laxemar and Simpevarp areas, Simpevarp is found to be the preferable alternative. The main reasons are shorter transport distances and less impact on the natural and cultural environments in the case of Simpevarp, combined with doubts as to the availability of land in the Laxemar area. The Simpevarp area, which also includes Ävrö and Hålö, is therefore deemed to be a reasonable alternative site for final disposal of waste from dismantling and demolition.

In a comparison between final disposal of short-lived waste from dismantling and demolition in an extension of SFR with the alternative of a separate final repository for short-lived waste from dismantling and demolition in Simpevarp, it is found that both alternatives offer potentially good prospects for long-term safety. The differences that are nonetheless found speak in favour of Forsmark. Similar conclusions are drawn from comparisons of factors related to human health and the environment as well as societal aspects.

The differences between the two alternatives are greatest when technical feasibility is compared. When establishment and operating aspects are compared, Forsmark's advantages are clearly revealed. Basically, this is because all final disposal of short-lived waste can be gathered at one site in the case of Forsmark. In the short term, this permits time and cost savings in the establishment phase, since e.g. the operations area and many of the required functions are already in place, while a repository in Simpevarp would require establishment of an operations area. The long-term efficiency gains of having one facility instead of two are clear, owing to the long operating times. Another essential advantage to be considered is better prospects for long-term continuity of the operation.

SKB's main conclusions are:

- The selected siting of the facility for final disposal of waste from dismantling and demolition enables the purpose of the activity to be achieved with the least possible intrusion and detriment to human health and the environment.
- No other site can be identified that is clearly better than the selected site and that is available for use with reasonable expenditures of time and money and within the desirable timeframe.

SKB therefore plans to extend the existing SFR in Forsmark to receive the future low- and intermediate-level operational waste and waste from dismantling and demolition.

5.5 Site investigations

SKB has conducted site investigations for the purpose of characterizing the rock in the area in Forsmark being considered for an extension of SFR from the viewpoint of constructability and long-term safety. The goal was to investigate a rock volume large enough to accommodate the entire extension. The investigations were begun in 2008 and concluded in 2011.

5.5.1 Results from investigations

To carry out the site investigation, the work was first divided into 1) investigations (in the field) and 2) modelling. From the viewpoint of results, the role of the investigation was to produce primary data and store them in a primary database. The role of modelling was to produce the discipline-specific site descriptive models and prepare the overall site description. Altogether, four percussion boreholes and eight cored boreholes were drilled and investigated, see Figure 5-3. Approximately 3,000 metres of drill cores were analyzed, and preliminary data indicate good rock quality in the area. The bedrock consists for the most part of metagranite and pegmatite with an average frequency of three to four open fractures per metre.

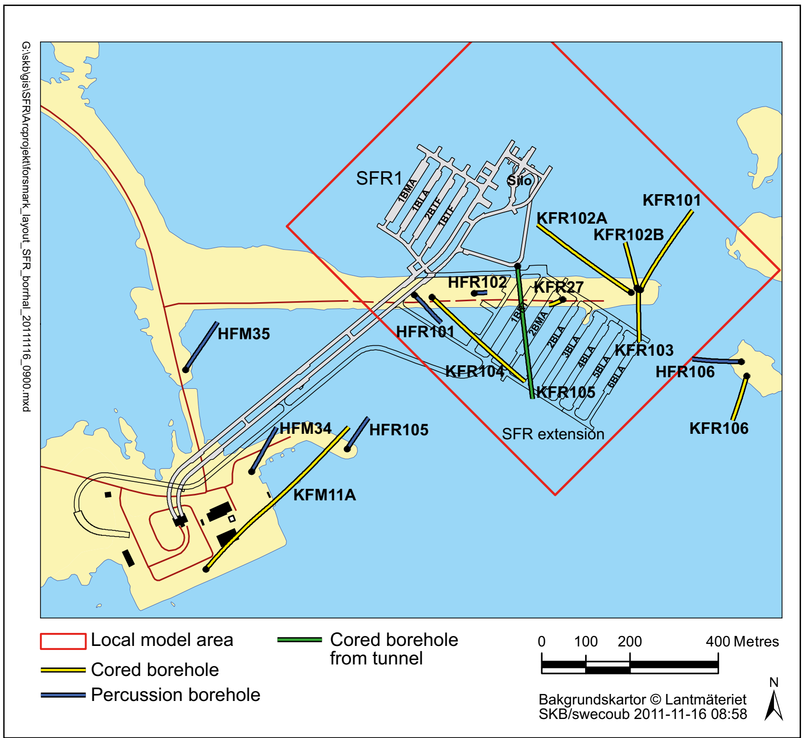


***Figure 5-3.** Map showing borehole locations for investigations for extension of SFR (KFM 11 A, HFM 34 and HFM 35 were not included in SFR's investigation programme). The figure shows an example of a preliminary layout for the extension's deposition area.*

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

5.5.2 Modelling and site description

Together with previously conducted investigations from the construction of SFR and from the site investigation for the Spent Fuel Repository, the field investigations that were carried out in the investigation programme comprise the basis for description of the investigated site, see Figure 5-4.

In general, the modelling project can be said to have comprised quality control of data, evaluation and analysis of primary data, three-dimensional modelling and reporting. The final result is a site descriptive model, SDM-PSU (SKB 2013b), for the investigated area. The site description comprises an integrated geoscientific account of the properties of the investigated rock volume and its relation to the regional environs. The site description is needed in the design work for positioning and configuring the final repository on the site. It also provides a basis for assessment of the long-term safety of the final repository. The site description is not limited solely 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, rock mechanical, hydrogeological and hydrogeochemical conditions on the site.

Four model versions have led up to the site descriptive model SDM-PSU (SKB 2013b). Data collection and modelling were structured in disciplines for efficient control, see Figure 5-5.

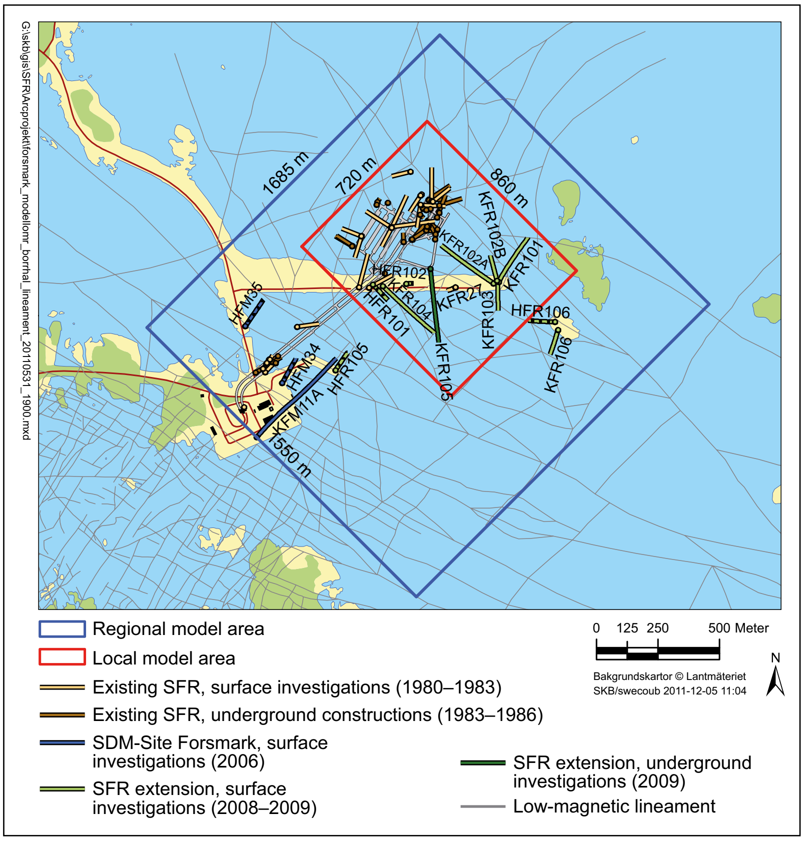


Figure 5-4. *Map showing boreholes from different time periods. The map also contains the regional model area for which updated models were developed. The local model area has a higher data intensity, which has permitted a higher resolution in the deformation zone model.*

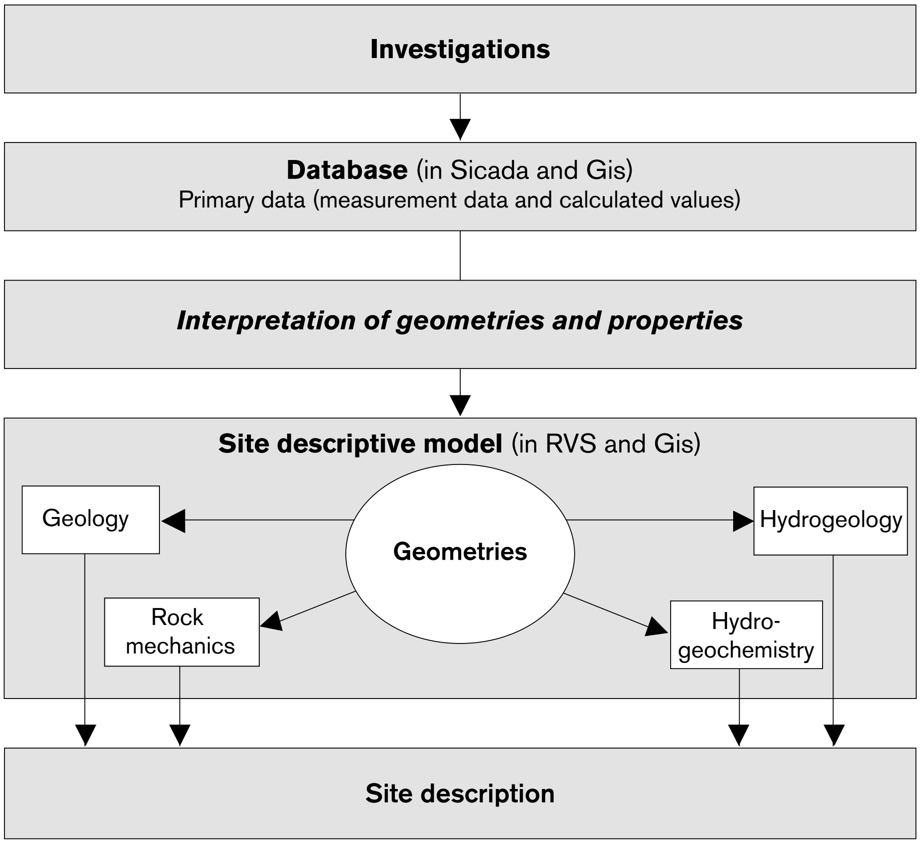


Figure 5-5. Illustration of the investigations' production of primary data and interpretation (modelling) of geometries and properties resulting in site descriptive models. The division into disciplines pertains to both investigations and modelling.

5.6 Work methodology for design

5.6.1 Facility parts

The facility parts can be divided into rock caverns underground, 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.

5.6.2 Stepwise execution

Up to the start of construction, design of the SFR extension takes place in two main phases: system design and detailed design, see Figure 5-6. System design is in turn broken down into a number of layout stages of gradually increasing maturity: Layout D0, 0, 1 and 2. The purpose of the system design phase is to define the facility to a sufficient degree so that operational and long-term safety can be analyzed and to prepare controlling material for the detailed design. In the detailed design phase, detailed design of the facility is carried out and tender specifications and construction documents are produced.

Layout D0 of SFR was completed during the summer of 2009. The reference design was based on the cross-section of the underground openings in the existing SFR and was dimensioned based on then-valid forecasts of the volume of waste from operation and dismantling and demolition. The goal of layout D0 was to provide a basis for determining the site-specific configuration of the repository, where the geological and hydrogeological conditions on the site are also taken into account.

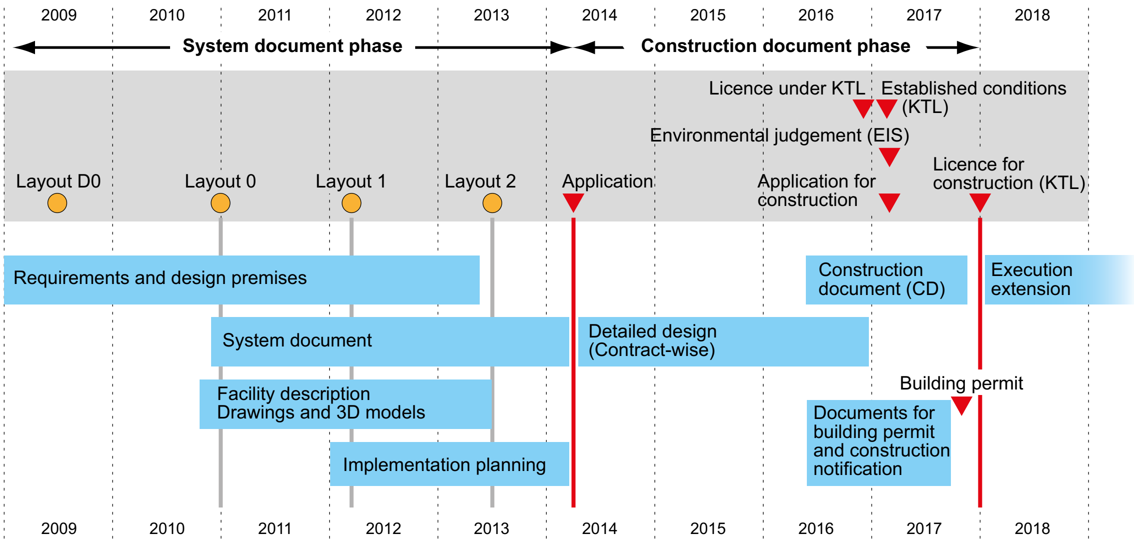


Figure 5-6. *General structure for production of design basis documents.*

Layout 0 represented an in-depth version of layout D0 and was based on studies of preliminary requirements and design premises for installations and equipment – e.g. ventilation, power supply, fire safety and waste quantities – and their impact on the layout of the rock caverns.

The purpose of layout 1 was, based on updated requirements and design premises as well as results from the site model, to serve as a basis for preparatory hydroanalyses for positioning the facility in the bedrock. In conjunction with layout 1, a number of alternative facility layouts were devised to permit a comparison of hydrogeological properties, which is an important premise for the long-term safety assessment. In conjunction with layout 1, a decision was also made to locate the main level for the extended portion at a depth of 120 metres below sea level. The extension was thus placed at a greater depth than in previous layouts. The main reason for locating the extension at a greater depth is that the facility has been reconfigured to avoid certain water-bearing structures within the area adjacent to the existing SFR where the extension was initially planned to be located (60 metres below sea level).

Based on the results of the hydrogeological modelling, and thereby on long-term safety, a facility layout was chosen as the main alternative for the continued work and the continued analyses. A 3D rock model was devised as an intermediate step between layouts 1 and 2 to serve as a basis for hydromodelling and assessment of long-term safety prior to submission of the applications.

Layout 2 is the layout that has been adopted as a reference design for the facility described in the safety analysis report. It also serves as a basis for the applications under the Nuclear Activities Act and the Environmental Code, Figure 5-7. The system design documents for the different installation categories (e.g. plumbing and heating, construction works and electrics) are also tied to layout 2 and then serve as a basis for detailed design. An important factor that influences the design of the facility and its long-term safety properties is the design of the engineered barriers. This work is described in Section 8.1.

In the detailed design phase, which follows the system design phase, detailed design of the facility is carried out. Tendering specifications, detailed technical descriptions for the different subcontracts (rock, plumbing and heating, electrics, telecoms etc), construction documents and documents for building permit application are then prepared. Start of construction is planned for the beginning of 2018.

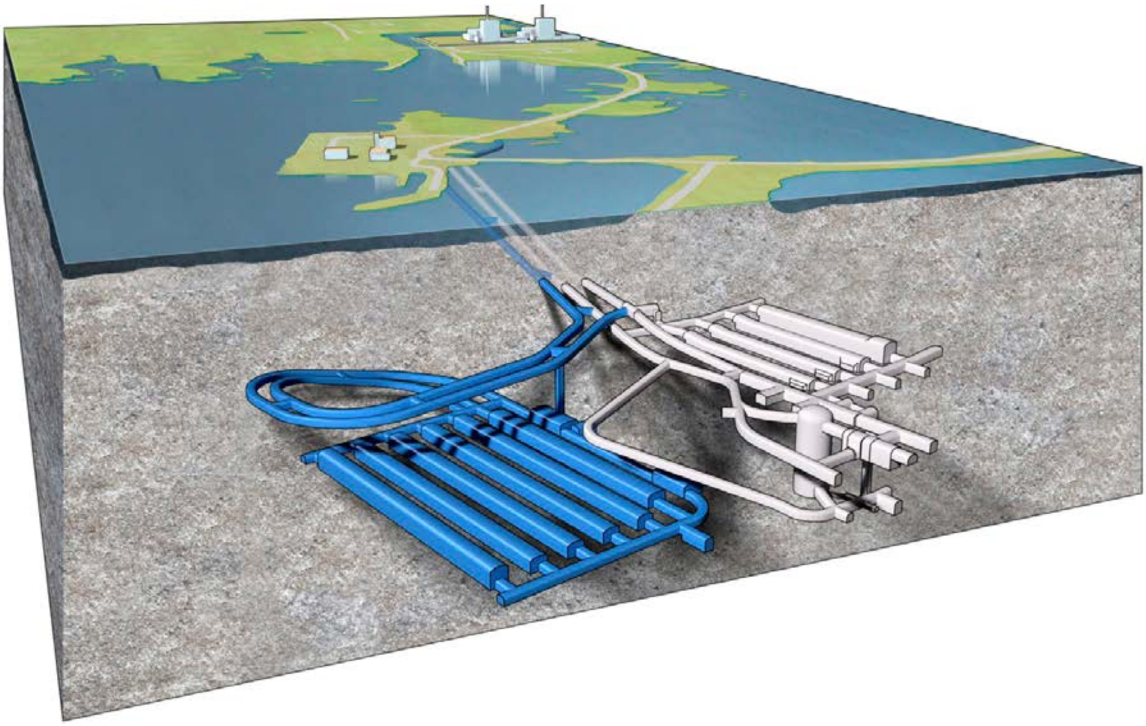


__Figure 5-7.__ Extended SFR according to layout 2, where the existing part is light grey and the planned part is blue.

5.7 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 Regulations concerning Safety in Nuclear Facilities, SSMFS 2008:1. 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.

A preparatory preliminary safety analysis report for the extended SFR will be prepared for the application under the Nuclear Activities Act for extension of SFR. The preparatory preliminary safety analysis report will then be updated and further detailed so it can be submitted for approval by SSM prior to the construction of the facility. An updated safety analysis report that is supposed to reflect the as-built facility will be prepared prior to trial operation. For SFR, this means that the safety analysis report for the existing SFR will at this point be replaced by the updated safety analysis report describing trial operation of the extended facility. Before the extended facility may then be put into routine operation, the safety analysis report must be supplemented with experience gained from trial operation. This means that the contents of the safety analysis report will change over time. The preparatory preliminary safety analysis report contains general and conceptual information, which is subsequently made more precise so that it can show in detail before trial operation how the requirements on the facility and its activities have been satisfied. In all phases, the safety analysis report must be approved by SSM.

6 Final repository for long-lived waste

Among SKB's repositories, the final repository for long-lived waste (SFL) is planned to be the last to be put into operation. Before SFL is put into operation, several important milestones must be passed, such as choice of repository concept and site, assessment of long-term safety, preparation of applications, construction, etc. SFL will be the smallest of the three final repositories, with an estimated disposal volume of about 16,000 cubic metres.

In this chapter, SKB presents its programme for disposal of the long-lived low- and intermediate-level waste. The chapter describes alternative courses of action for managing the waste that is planned to be deposited in SFL, as well as for planning, development and construction of the final repository. Furthermore, it describes the work of updating the reference inventory and the ongoing concept study for evaluating different strategies for final disposal of SFL waste. Plans for the development of waste acceptance criteria, assessment of long-term safety and the site selection process are also presented.

To satisfy the Government's requirements on an in-depth account of alternative courses of action for management and disposal of the waste and a hands-on presentation of repository layouts for SFL, SKB presents parts of the work that has been done within the framework of the concept study. The concept study will be presented at the end of 2013, and geological disposal is expected to be a suitable strategy for final disposal of SFL waste. The following sections describe SKB's plans for SFL as a geological repository.

The overall planning for SFL is presented as a part of the company's plan of action in Chapter 2. Section 4.2 deals with the origin of the long-lived low- and intermediate-level waste and the current situation for management of the waste. The technology development that is being pursued for SFL is described in Section 8.2, while the research programme for the long-lived low- and intermediate-level waste is described in Chapter 21.

Conclusions in RD&D 2010 and its review

The plans for the future work with SFL were described in RD&D programme 2010. In the review of the RD&D programme, SSM expressed wishes for a more in-depth account of different alternative courses of action for management of long-lived low- and intermediate-level waste. In particular, SSM wanted more information on premises and alternatives with respect to a possible earlier and stagewise construction of SFL. A concretized account of the repository layouts SKB has begun to develop was called for. Further, SSM recommended to the Government that SKB's work on the in-depth account should take place in consultation with SSM.

The Government's approval of RD&D programme 2010 was made subject to the condition that the reactor owners must, when preparing next RD&D programme, consult regularly with SSM in matters relating to long-lived low- and intermediate-level waste. In RD&D programme 2013, SKB should give a more in-depth account of different alternatives for management and final disposal of the waste as well as a concretized account of the repository layouts SKB has begun to develop. The account shall describe premises and alternatives with respect to a possible earlier and stagewise construction of SFL.

Current situation

The work of identifying possible final repository concepts for SFL has begun. The work is essentially following the methodology SKB has used earlier for system analysis of different solutions for disposal of spent nuclear fuel (SKB 1992, 2000a).

The work includes:

- Identification of strategies for management and disposal of long-lived low- and intermediate-level waste.
- Identification of different final repository systems or final repository concepts in order to implement a given strategy.
- Identification and choice of selection method and selection criteria.
- Evaluation and choice of repository concepts to work further with.
- Development and improvement of chosen repository concepts.
- Safety assessments and final selection of one repository concept.

In the first stage, different strategies for final disposal long-lived low- and intermediate-level waste are identified, for example sub-seabed disposal or geological disposal. Based on the strategies that comply with laws and international conventions, different final repository systems or final repository concepts have been identified, where a final repository concept consists of the set of facilities and components that is required to implement a given strategy. There may be different final repository concepts that realize a given strategy. For example, SFR and KBS-3 are two different final repository concepts for the implementation of geological disposal.

In parallel with the work of identifying strategies and repository concepts, the study has – with reference to laws, regulations and international conventions – identified requirements on the final repository system and its subsystems. These requirements have then comprised the basis for formulating evaluation factors that are used to compare the final repository concepts with each other. Descriptions have been formulated for the identified final repository concepts to permit a comparative evaluation. Further, different investigations have been carried out to shed light on the differences between the concepts. These investigations serve as a basis for the concluding evaluation. The evaluation takes the entire handling chain into account from the origin of the waste to the closed repository and thus includes not only the final repository itself, but also the need for special waste packages, conditioning methods, etc. The need for technology development as well as the construction and operating phases are also taken into account.

In the evaluation, an overall assessment is made of the final repository concepts with the support of evaluation factors linked to long-term safety, environment, technology, cost and time. The concept study presented at the end of 2013 will recommend at most two concepts for further evaluation of long-term safety.

In addition to the concept study described above, an updating of the reference inventory for SFL is also under way, and the results will be presented at the end of 2013. Current information concerning the inventory is presented in Section 6.4.

6.1 Consultations with SSM

Consultations between SSM and SKB concerning long-lived low- and intermediate-level waste have been held twice during 2012. In these consultations, SKB has presented the progress made in the concept study, the work of updating of the reference inventory for SFL and relevant research. SSM has also had an opportunity to have an early look at the account SKB has planned to give in RD&D programme 2013. On these occasions, SSM has been given an opportunity to express viewpoints on the work being done to manage the long-lived low- and intermediate-level waste as well as the plans for presentation of SFL in RD&D programme 2013.

6.2 Alternatives for management of long-lived low- and intermediate-level waste

The management of the waste that is planned to be disposed of in SFL is based on the waste and its properties, as well as the current planning premises in terms of operating time for existing reactors.

The different waste types planned to be deposited in SFL have different properties. Core components and PWR pressure vessels from the nuclear power plants consist primarily of steel and stainless steel. The control rods consist of stainless steel and neutron-absorbing material such as boron carbide. The activity at deposition in both includes long-lived radionuclides created by neutron irradiation. The legacy waste is to a large extent already conditioned by embedding in cement and contains other long-lived radionuclides, such as uranium and fission products. From a radiation protection viewpoint, core components and control rods comprise intermediate-level waste, for whose handling radiation protection is required, while large parts of the legacy waste can be handled without radiation protection.

Since all handling of the waste entails a risk for increased dose load, the handling chain should be planned as a whole. Planning of the handling chain should preferably be done even before the start of the activity that generates the waste. The situation is different for the waste destined for SFL, since much of the waste already exists. The aim of all handling must be to ensure post-closure safety. This long-term safety must then be weighed against the dose load to personnel during handling and operation.

Handling and conditioning of core components

The purpose of waste conditioning is to alter the properties of the waste to better meet the requirements on e.g. geometric dimensions, surface dose rate or properties that affect the long-term safety in a final repository.

The activity of the core components when they are removed from the reactor influences both handling alternatives and choice of final disposal method. The most demanding radionuclide from a radiation safety viewpoint during the first 70 years is cobalt-60. All handling requires radiation shielding, and any segmentation of core components is therefore preferably done in the pools at the nuclear power plants. Activation products such as chlorine-36, nickel-59 and molybdenum-93 are of great interest from a long-term safety viewpoint (SKB 1999). The release rate for activation products in the core components is controlled by the corrosion of steel and stainless steel.

In the case of core components, three main alternatives can be formulated for handling and conditioning:

1. Existing steel tanks (also called BFA tanks) are defined as final disposal packages.
2. The waste is packed in other standardized shielded waste containers.
3. The waste undergoes advanced conditioning, such as melting.

The steel tanks form part of a system that has been used by OKG and subsequently by FKA for lifting out segmented core components for dry interim storage. The steel tanks are available in different models with different wall thicknesses that can be chosen based on the activity of the waste. Due to the good radiation-shielding properties of the steel, the steel tank is a well-suited package for interim storage of core components. From a long-term safety viewpoint, the steel tank adds nothing, since no long-term containment function can be proven. If the steel tank is to be used as a final disposal package, the tanks should be opened and the waste grouted before the tanks are placed in the final repository. Studies indicate that it is advisable to let the waste decay for some time prior to grouting to avoid radiolysis. The steel tank alternative is deemed from a handling viewpoint to give a limited dose to the personnel.

The other alternative entails placing the waste in a shielded waste container that forms part of a standardized waste container system for SFL. Such a shielded waste container can be designed either solely from a radiation protection perspective, or from a perspective that combines radiation protection with long-term safety. A shielded waste container that is fabricated with e.g. full penetration welds could be credited with a safety function – containment – in the final repository. Waste containers in the system can be shipped in standard transport casks, lifted by the same lifting tool and stacked on top of one another to the desired height. This requires a transloading of existing interim-stored waste that is currently stored in steel tanks at FKA and OKG as well as additional waste from Barsebäck. This handling presumably requires a central transloading facility where steel tanks are opened and emptied. The waste is loaded into new packages, grouted and transported to an interim storage facility or directly to the final repository. The dose load to personnel is above all dependent on the decay time of the waste before handling. Compared with alternative 1, this handling entails the risk of a higher dose load, but with remote-controlled handling equipment, it should be possible to keep the dose load down. Keeping the dose load down requires segmenting the waste into pieces that fit both in a steel tank and in the standardized waste container system. Steel tanks that are emptied can after treatment (e.g. sand-blasting) be reused or cleared.

The metallic waste can be melted to reduce its volume as well as to reduce its surface-area-to-volume ratio. This method is mainly deemed to be suitable for pure metallic waste and not for control rods. According to the latest safety assessment for a final repository for long-lived waste (SKB 1999), releases of radioactive substances from the part of the repository with core components are mainly affected by the corrosion rate of steel. Reducing the ratio between surface area and volume is a way to reduce the release rate for radionuclides expressed as weight per unit time. Since the waste consists largely of thin steel sheets and tubes, melting to cylindrical ingots can greatly reduce the rate of release of activation products such as chlorine-36, nickel-59 and molybdenum-93.

Melting of low-level metallic waste is done today in a melting plant on the Studsvik site. SKB has contracted Studsvik Nuclear to investigate the prospects for melting intermediate-level metallic waste, such as core components. The conclusion is that up around 75 percent of the core components (and internals) can be melted provided that they are allowed to decay for at least 50 years (Huutoniemi et al. 2012). Studsvik Nuclear's assessment is that technology development is required to design a facility that is completely remote-controlled and that will not contribute higher doses to personnel than the existing melting plant during normal operation. However, the consequences in the form of dose to the personnel due to unforeseen events, such as loss of power supply or steam explosion, are so great that Studsvik Nuclear concludes that the activity levels in the waste must be reduced before it is conditioned by this method.

An advanced conditioning such as melting can only be justified if it is required to guarantee long-term safety. The method requires building a new facility, and due to the limited quantity of waste, the cost per unit weight of conditioned waste will be high. The dose load to personnel will in all probability be the highest of the three alternatives presented. The existing system where the waste is interim-stored in steel tanks pending conditioning can be used in the same way as today.

Handling and conditioning of PWR pressure vessels

The activity content of the PWR pressure vessels is higher than that of the BWR vessels. The more compact dimensions of the PWR pressure vessels places the walls of the vessel closer to the core, which affects the content of activation products. The activity is very unevenly distributed, with high induced activity nearest the core and much lower activity towards the head and bottom. The part nearest the core has such high activity levels from long-lived radionuclides that exceeds the limit for disposal in SFR.

The alternatives that have been arrived at for handling of PWR pressure vessels are:

1. The vessels are emptied, segmented, sorted and packed in steel tanks.
2. The vessels are emptied and deposited directly in a final repository.
3. The pressure vessels are deposited directly with core components and reactor internals still in the reactor pressure vessel (RPV).

The first alternative entails that the tank is emptied of its contents while it is still in the reactor containment. Core components and RPV internals are treated according to the above handling alternatives for core components. The RPV is then segmented and the parts are sorted according to activity content. The parts that are suitable for final disposal in SFL are placed in steel tanks intended for this purpose, while less radioactive parts are handled in other waste streams for deposition in e.g. SFR.

In the second alternative, the emptied PWR pressure vessel is handled in one piece and transported to SFL for deposition of the whole tank without segmentation.

In the third alternative, core components and RPV internals are left in the PWR pressure vessel when it is lifted out of the reactor containment. The RPV is handled in one piece and transported to SFL for deposition in one piece, without further emptying or segmentation. Ringhals's plan for dismantling and demolition is currently based on this alternative. This alternative assumes that core components and RPV internals are secured and that the RPV is shielded during transport.

The design of SFL is affected if Ringhals's three PWR pressure vessels will be deposited whole. The PWR pressure vessels are assumed to require similar engineered barriers as the core components, but will probably need their own rock vault. The ramp and connecting tunnels at repository depth need to be designed to accommodate whole tanks. The radiological and economic consequences of the different alternatives need to investigated.

Handling and reconditioning of the legacy waste

Four main alternatives can be formulated for the legacy waste, which is already conditioned:

1. The waste is handled without further conditioning.
2. The waste is packed in standardized waste containers intended for SFL.
3. The waste is retrieved and sorted.
4. Advanced reconditioning of the waste, for example by crushing and vitrification.

The first alternative, to handle individual drums and waste packages of various sizes, is time-consuming and inappropriate from a handling viewpoint. Different types of packages require different handling equipment and transport containers. The strength of individual packages must be taken into consideration when packing and stacking different types of packages in a rock vault.

The second alternative involves efficient handling with uniform and standardized handling systems. This alternative entails that the existing waste, which for example is already conditioned in 200-litre drums or 80-litre drums in five-hole moulds, is loaded into waste containers that comprise part of a standardized system. These waste containers can be transported in standardized transport containers, lifted by the same lifting tool and stacked on top of each other to the desired height. This alternative probably gives a lower dose load to personnel compared with the first alternative.

The third alternative entails that the waste is retrieved and sorted. This is done by breaking up or crushing the grouting material in order to free the original waste. The waste is characterized, documented and sorted into suitable fractions. The waste can then be conditioned once again in suitable waste packages. This procedure is similar to the one used by SVAFO during the period 1986–2002 to empty the active trough. At that time, a new plant was designed with a hot cell for handling the waste safely. Due to the much greater volumes involved in the management of legacy waste, a new plant needs to be built in order for this alternative to be feasible. Such a plant can be designed with a small dose load to personnel. The advantages of this alternative include the fact that the uncertainties in the inventory can presumably be greatly reduced, along with the disposal volumes.

The fourth alternative for management of the legacy waste entails that a new (for Sweden) type of conditioning plant is designed and commissioned. One conceivable method is to crush the conditioned waste in a closed chamber and then heat it to a very high temperature in order to create a glass-like end product that can be deposited in the final repository. In view of the heterogeneous composition of the legacy waste, several different conditioning processes may be required. The plant can presumably be designed so that the dose load to personnel is small, but leads inevitably to radioactive secondary waste such as filters from flue gas cleaning and waste from dismantling and demolition of the plant, which also requires disposal. This must be weighed against the expected benefit of the method in terms of long-term safety. Plants that condition waste in this way exist today in e.g. Belgium (BelgoProcess) and Switzerland (Zwilag).

Flows and shipments of long-lived low- and intermediate-level waste

The SFL waste is can be found today mainly at five different geographic locations: Barsebäck, Forsmark, Simpevarp, Ringhals and Studsvik. SFL waste from medical care, research and industry is shipped to the Studsvik site, where it is interim-stored.

All sites with SFL waste have their own ports at which SKB's transport ship makes regular calls. Evaluation of the environmental impact of shipments of SFL waste show that the siting of the final repository is the most important factor. If SFL is sited on or near the coast, shipments of the SFL waste can also go by sea, which causes less environmental impact than equivalent overland shipments. The environmental impact of shipments will be included in the assessment of the siting process. Since the siting process for SFL has not yet begun, it is not possible to describe how waste will be transported. The siting process for a conditioning plant also needs to analyze the environmental impact of this transport.

6.3 Alternatives for final disposal

SKB's current timetable for commissioning of SFL is described in Section 2.2, and Figure 2-4 illustrates the development steps leading to an operational repository for long-lived waste. The following section starts by describing the development steps for the purpose of illustrating possibilities and limitations in the development work and estimating the time the development steps are expected to take. Based on these assessments, SKB's timetable for commissioning and the two alternatives entailing deferred or stagewise commissioning of SFL are discussed.

The alternatives are based on the assumption that a geological repository is the point of departure for SKB's planning.

6.3.1 Development steps to a commissioned repository

SKB's plan for the development of SFL is based on the same methodology as that used for the Spent Fuel Repository: a stepwise and iterative process where assessments of long-term safety determine the choices of direction and the requirements on technology development.

Commissioning of SFL is the final step in a development process involving a number of steps: technology development, site selection, assessment of long-term safety, preparation of applications, design, construction and commissioning.

Technology development

The development of barriers requires feedback in the form of regular evaluation of the effects on long-term safety. Technology development up to a commissioned final repository is estimated to require at least a couple of iterations of development followed by safety assessment. Experience shows that such an iteration takes about six years. Technology development also requires interaction with the site selection process to arrive at a favourable combination of engineered barriers and site properties.

The parts of technology development that are linked to the design of the facility and equipment as well as operation can presumably be borrowed to a great extent from the development and operation of SFR. Handling of waste in SFL will probably not require special equipment; commercial standard equipment or modified standard equipment can be used.

Technology development will be pursued in parallel with the siting process. The extension of SFR that is now being planned will provide important experience for the technology development linked to SFL. Based on experience from the planning and development of technical solutions for similar facilities, SKB believes that the uncertainties in the time estimates for technology development are small in relation to other parts of the programme.

All things considered, technology development is not deemed to be time-critical for the establishment of the repository.

Site selection

The temporal aspects of the site selection process are among the most difficult to judge. The process includes local consensus-building that is both necessary and desirable. Experience from the site selection process for the Spent Fuel Repository gives an indication of the pace of the political process.

Current planning is based on the assumption that one of the municipalities that at an early stage are participating in a feasibility study for SFL also has the safety-related factors and the local acceptance for a siting. If this assumption does not hold true and it is necessary to contact a larger circle of municipalities at a later stage, site selection will presumably be delayed by a decade or so.

Previous experience shows that the purely technical part of a site investigation on one or two sites with test drilling and monitoring takes five to seven years. The results of the investigations need to be processed and interpreted prior to site selection.

Because the site selection process constitutes a time-critical part of the programme of building SFL, a study will be initiated during the coming RD&D period concerning how the site selection process should be conducted. The process SKB intends to carry out in order to select a site is presented in Section 6.7.

Assessment of long-term safety

Safety assessments comprise an important component of SKB's technology development and site selection process. In order to evaluate the consequences of technical solutions and choice of bedrock, more or less complete safety assessments need to be carried out. A safety assessment takes about three years to carry out, provided that essential parts of the basic research needed for the assessment have been done in advance. The initial part of the assessment follows directly after the technology development programme, and it is not believed possible to appreciably shorten the time required. A site-specific safety assessment is conducted when the site has been selected and serves as a basis for the applications to build the repository.

Safety assessments and the associated research are highly specialized disciplines where it takes a long time to acquire the necessary competence. The competence and methodology in the field that SKB has accumulated in its work with the other final repositories will also be of benefit to SFL.

Preparation of applications

Preparation of applications under the Nuclear Activities Act and the Environmental Code is interlinked with assessment of long-term safety and the site selection process. It will take about two to three years after site selection to carry out environmental impact assessments, compile documentation, etc. Parts of the work are done in parallel with the site-specific safety assessment.

Design

In its work with the SKB Spent Fuel Repository and the extension of SFR, SKB is gathering experience of the design and construction planning process that can also be used in the work with SFL. Some parts of the design work can be carried out in parallel with licensing, while others cannot be done until a decision has been made on execution. Systems project management and detailed design of the parts to be built at an early stage can be carried out during licensing. Other detailed design is then done at a suitable pace with respect to other construction planning. The design process can be estimated to take a couple of years from the time a decision has been made on execution. Design can be done in parallel with other construction planning. Viewed in a larger perspective, the time required for design is thus short and lies far ahead in time. There is little chance that this time can be appreciably shortened.

Construction and commissioning

Some construction planning can take place during licensing. After a licence has been obtained and a decision has been made on execution, construction can begin within a couple of years and be completed in six to seven years. The construction of accesses to repository level is time-critical, and there is little potential to step up the pace of the tunnelling work. The volume to be disposed of is relatively small, so the potential time gains achieved by driving parallel faces are limited.

One way to significantly shorten the time for construction of SFL is to co-locate it with an existing repository, e.g. SFR. The time-consuming process of tunnelling to repository level can then be shortened.

6.3.2 Timetable for commissioning of SFL

As shown in SKB's timetable for commissioning of SFL in Figure 2-4, the different steps described in Section 6.3.1 are carried out in logical sequence, starting immediately after the concept study is finished. If possible, activities are pursued in parallel to save time. The current plan leads to operation of SFL in about 2045.

As is evident from the preceding section, technology development is not deemed to be time-critical for the project. The chain of activities that are deemed to be time-critical for development and commissioning are:

- Site selection.
- Site-specific safety assessment.
- Preparation of application documents.
- Licensing.
- Design.
- Construction and commissioning.

The first part of the site selection process – the internal study of how site selection should be pursued – will be commenced during the upcoming RD&D period. The site investigation phase is expected to begin in around 2021 and last between five and seven years. Site selection can then take place in 2027, followed by parallel work with site-specific safety assessment and preparation of application documents lasting three years. Applications for licences to build SFL will be submitted in 2030, after which the licensing process will ensue. If licences to build SFL are obtained in 2036 or later, detailed design and construction planning will begin immediately thereafter. Start of construction is expected to take place a couple of years after the applications have been approved and a decision on execution has been taken. Construction will last six to seven years, after which commissioning will be followed by one year of trial operation.

All things considered, this summary shows that commissioning in 2045 is an ambitious but not unreasonable goal to achieve. It also shows that it is unlikely that the repository can be commissioned much earlier. The uncertainties in the timetable are mainly attributable to the site selection process, which is also judged to be time-critical.

If the core components can be disposed of immediately after dismantling and demolition of the NPPs, there is a possibility that the last core components can be disposed of in around 2055, if dismantling and demolition of Forsmark 3 and Oskarshamn 3 is begun in 2047 according to the current plan. The time for closure of the repository will then be dependent on the long-lived waste that arises during dismantling and demolition of Clink or the long-lived waste that is managed by Studsvik Nuclear AB.

Decommissioning of Clink can be done when all spent nuclear fuel has been encapsulated. According to the plan, dismantling and demolition will start in around 2073 and take five to seven years. According to the current plan, the waste from dismantling and demolition of Clink will be disposed of in SFR. If long-lived waste should arise, it will have to be disposed of in SFL. Closure of SFL can then be commenced between 2075 and 2080.

Assuming that waste from dismantling and demolition of Clink needs to be disposed of in SFL, the total operating time will be about 30 years. Deposition of both core components and waste from SVAFO and Studsvik Nuclear will take place during the first decade after commissioning. The relatively small additional waste volumes from non-nuclear activities will be deposited during the remainder of the repository's operating life. Any long-lived waste that has arisen during dismantling and demolition of Clink will be deposited during the final years of the repository's operation.

One advantage of early construction of SFL is that interim storage of the legacy waste can be concluded and, ultimately, SVAFO's activities can be wound up. With a final repository in operation, waste acceptance criteria can be formulated, which makes it possible for the NPPs to finally condition their waste in conjunction with dismantling and demolition.

A longer operating time creates greater challenges as far as guaranteeing the initial state of the repository is concerned, since the operating phase tends to exert stresses on the underground structures. The rock vaults that have been finished and filled with waste first have to be left open during the remaining operating time and be sealed when deposition in all repository parts has been concluded. Sealing individual rock vaults in an otherwise open repository and allowing them to fill with water is inadvisable, since the water flow through the sealed part can be affected by the fact that other parts are still being pumped out.

6.3.3 Deferred commissioning

The alternative of deferred commissioning is based on the planned closure time and is intended to shorten the operating time for the repository.

If closure is planned for around 2075, SFL should be commissioned not later than 2065, see Figure 6-1. Then deposition of the legacy waste and core components can be done first, which is expected to take seven to eight years. Any waste from Clink is then deposited and the work of closure of the repository can commence. The interim storage facilities that are already in operation today for core components will be used for interim storage until deposition can take place. The waste being interim-stored on Studsvik Nuclear's and SVAFO's sites will continue to be stored there.

If conditioning by melting of the metallic waste should be required to guarantee long-term safety and it is deemed advisable from a radiation protection viewpoint to let the core components decay until conditioning takes place, the core components will not be available for deposition until after a decay period of about 50 years. Barsebäck 1 was closed in 1999, which means that conditioning of its core components can take place in around 2050. Conditioning is done in campaigns as the reactors are shut down, after which the waste is ready for deposition. Conditioning and deposition of the last core components will then take place in around 2100. Deferred commissioning will then mean that SFL can be commissioned in around 2090, providing an operating period of ten years during which all waste will be deposited.

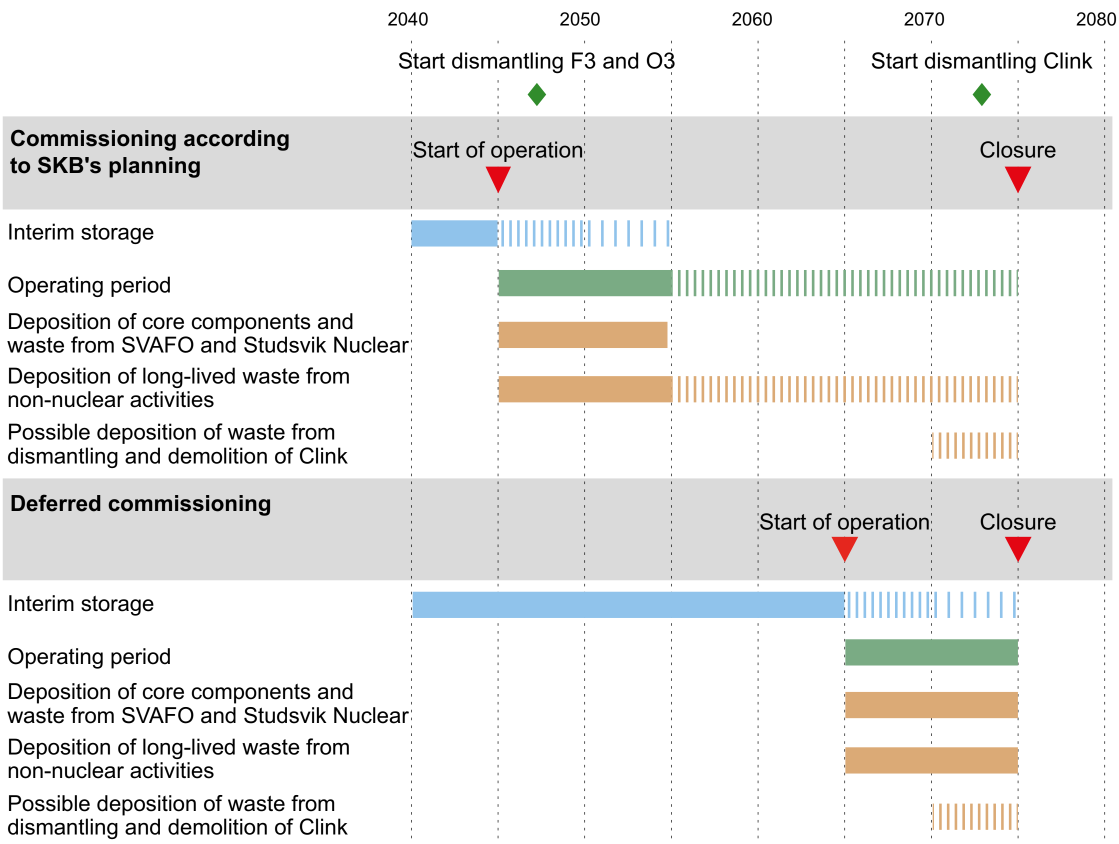


Figure 6-1. *SKB's timetable for commissioning and closure of SFL and the alternative of deferred commissioning. Operating times and deposition periods are marked, as is the planned start of dismantling and demolition of Clink and the two reactors that are taken out of service last.*

The advantage of a short operating period is that the stresses on the facility's structures are reduced. The costs of interim storage for a longer time have to be weighed against the reduced operating costs for the repository.

A great uncertainty lies in how SKB's mission will change over time, for example with regard to the management of radioactive waste from non-nuclear activities, which can affect the closure date. A disadvantage of linking the construction of the repository to a far-off planned closure date is the risk of time drift.

6.3.4 Stagewise commissioning

In the stagewise conditioning alternative, the rock vaults are built as the need for them arises. The legacy waste exists today and could be deposited first according to this alternative. The waste from dismantling and demolition of the reactors will not arise until the 2030s and 2040s and would then be deposited in a second stage of SFL.

It is not expected to be possible to put the first part of a stagewise built repository into operation earlier than SKB's current timetable for commissioning of SFL, i.e. in around 2045. The shortening of the construction period for a stagewise built repository is negligible in relation to the other time spans, since the disposal volumes are small. The development steps that are judged to be critical for the time to start of construction are site selection, assessment of long-term safety and preparation of applications. None is affected by the stagewise approach. The last core components will be taken out of Forsmark 3 and Oskarshamn 3 in around 2047, according to current plans. If this waste can be deposited without extensive conditioning, the last core components will thus become available relatively soon after commissioning according to SKB's current plans. This defeats the whole point of stagewise commissioning. If, on the other hand, extensive conditioning of the core components should be necessary in order to ensure long-term safety, the time of deposition can be postponed 50 years. In this case, stagewise commissioning becomes a viable alternative.

One advantage of stagewise commissioning is that the disposal volume can be adapted to changed conditions and new waste, which cannot be ruled out when the timeline for operation of the repository is so extended. Another advantage is that the engineered barriers in the later-built rock vaults will not be subjected to the stresses entailed by an extended operating time. A factor that speaks against stagewise construction is the relatively small size of the different repository parts, compared for example with SFR. The repository part that would then be built last is the one for core components, which represent a disposal volume of about 5,000 cubic metres. All rock vaults need to be left open until the end of the repository's operating period.

6.4 Updating of the reference inventory

According to RD&D programme 2010, SKB intended to prepare an updated reference inventory for SFL. The purpose was to obtain a more detailed inventory where generalized assumptions in the previous inventory from 1998 (SKBdoc 1416968) would be placed with reported waste quantities and updated determinations of radionuclide content.

Current situation

Figure 6-2 shows a general forecast of when long-lived low- and intermediate-level waste arises. The total volume has been calculated to be about 16,000 cubic metres. The volume has been calculated based on the following assumptions:

- Existing waste quantity is calculated from data reported by the waste producers and excerpts from the Draak database for interim-stored core components and control rods in Clab (SKB 2002). In the future, this database will be combined with the Gadd database to include both short-lived and long-lived low- and intermediate-level waste, see Section 4.3.3.
- Forecasted waste volumes are based on data reported by the waste producers and assumptions concerning periodic replacements of control rods (operating time 15−20 years) and neutron detectors (operating time 10 years) for BWRs.

- Waste quantities from dismantling and demolition are taken from the decommissioning studies (Griffiths et al. 2008, SKB 2013c, d, e) as well as volumes reported by Studsvik Nuclear and SVAFO.
- Studsvik Nuclear's forecast for operational waste extends up until 2045, so forecasted operational waste has been distributed evenly over the years up to and including 2045, when waste from dismantling and demolition of Studsvik Nuclear and SVAFO is also assumed to arise (even though the date for decommissioning of Studsvik Nuclear's facilities has not been determined).
- The disposal volume for core components and control rods has been calculated based on a packing degree of 1.1 tonnes per cubic metre, an expected inside volume of six cubic metres and an expected outside volume of 9.9 cubic metres, which is equivalent to the dimensions of a steel tank (including inner storage canister) with a wall thickness of 150 millimetres.
- Control rods from Ringhals PWRs are disposed of together with the spent nuclear fuel.
- Core components and RPVs from PWRs are segmented and deposited in the same way as core components from BWRs. The alternative of depositing the PWR pressure vessels whole (with or without core components) would reduce the total disposal volume by 400−800 cubic metres.

Legacy waste managed by SVAFO

The legacy waste that is managed by SVAFO was inventoried in 2012. The total waste volume is 6,500 cubic metres and consists mostly of cement-embedded waste in 200-litre drums, most of which have been placed in new 280-litre drums. Most of these drums have been subjected to radiographic and gamma spectrometric monitoring. This examination shows that a number of drums contain free liquid (which in some cases is assumed to be mercury). Even after this analysis, great uncertainty remains regarding both material composition and radionuclide content. The same also applies to other waste containing trash and scrap, since the waste is heterogeneous, which makes representative sampling more difficult at the same time as the exact origin of the waste is not known.

Relatively good information is available for the gamma-emitting radionuclides that could be measured directly as well as actinides whose total quantity is known (even though the exact distribution among individual waste packages cannot be known with the information that is currently available).

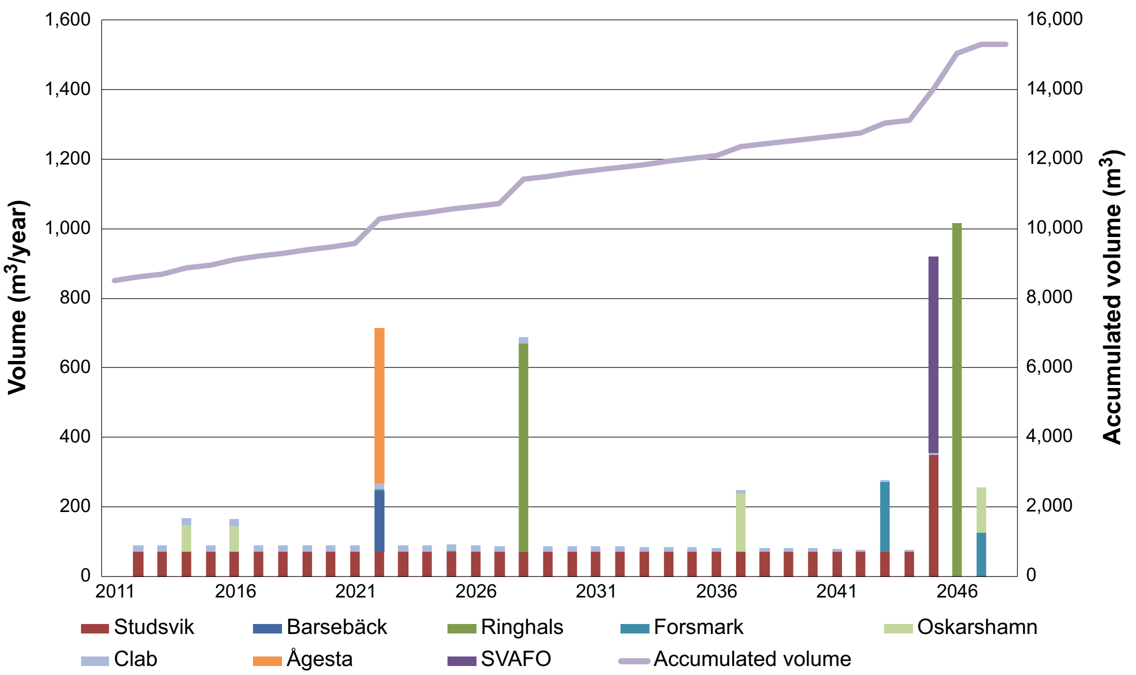


Figure 6-2. *Overall timetable showing when long-lived low- and intermediate-level waste arises. The bars show the waste that arises per producer and year (left-hand axis). The curve shows the accumulated volume (right-hand axis). Data for Clab includes only interim-stored control rods and core components.*

In the reported data, correlation factors for Studsvik Nuclear's waste in the Triumf NG database (SKB 2010b) have been assumed, giving a total activity in 2075 of $1.3 \cdot 10^{15}$ Bq. Data are currently lacking to justify the general use of these correlation factors. If the use of correlation factors is limited to drums containing ashes or sludge and ion exchange resins, the total activity of the waste is $1.0 \cdot 10^{15}$ becquerels (Bq) in 2075.

Waste from Studsvik Nuclear's activities and other Swedish research

Reported long-lived low- and intermediate-level waste from Studsvik Nuclear consists of drums containing ashes, sludge and ion exchange resins, as well as trash and scrap. The total volume of the existing waste is about 1,200 cubic metres, and the activity of the total radionuclide inventory is $1.3 \cdot 10^{12}$ Bq in 2075. Forecasted waste volumes up to and including 2045 and waste from dismantling and demolition of Studsvik Nuclear's facilities amount to about 2,700 cubic metres, but there is no activity information for this waste. Since Studsvik Nuclear's activities, compared with the activities of the NPPs, generates waste of a highly diverse nature, a long-term forecast for operational waste must be regarded as highly uncertain.

Waste from nuclear power plants

Long-lived low- and intermediate-level waste from the NPPs consists mainly of neutron-irradiated reactor internals. The radionuclide inventory has been determined from activity calculations done for the decommissioning studies plus data reported from the NPPs. Existing waste consists of interim-stored core components at the NPPs and in Clab plus interim-stored control rods in Clab. The forecast has been calculated based on remaining operating time and expected operating time for neutron detectors and control rods plus reported forecasts for replacement of large core components.

The combined activity of the core components in 2075 is $1.9 \cdot 10^{17}$ Bq and the disposal volume is estimated at about 4,600 cubic metres. Radionuclide content data from the decommissioning studies have been used for forecasted control rods and probes (they have been included in the activity calculations even though they do not constitute part of the actual waste from dismantling and demolition). Calculated activity from the decommissioning studies has also been used to supplement the inventory for components where activity data are lacking (mainly older control rods).

Programme

An updated inventory is planned to be completed by the end of 2013. The inventory will represent current knowledge concerning long-lived low- and intermediate-level waste based on data reported from all waste producers. This inventory will be updated continuously. An updated chemotoxic inventory will also be obtained from the work with the reference inventory.

Legacy waste managed by SVAFO

Development of the repository concept for SFL and acceptance criteria for long-lived low- and intermediate-level waste is under way. In parallel with this, SKB will clarify the need for supplementary studies to reduce the uncertainty in material and radionuclide content data. SKB will also request supplementary information on waste that has not yet been inventoried, plus a more detailed accounting of material quantities, particularly with respect to chemotoxic substances.

Waste from Studsvik Nuclear's activities and other Swedish research

The forecasts for the operational waste from Studsvik Nuclear's activities will be updated continuously and the radionuclide inventory will be able to be calculated when new nuclide vectors are calculated for the different facilities. In addition, radioactive waste will be produced at ESS (European Spallation Source) in Lund.

Waste from nuclear power plants

SKB intends to update the inventory for near-core components from the NPPs continuously in order to reduce the uncertainty in calculated radionuclide content. More detailed calculation methods and more accurate data on material composition and waste quantities will be used as they become available. New calculations will also be carried out to supplement data on the radionuclide inventory for older control rods where they are lacking today.

6.5 Waste acceptance criteria

It will not be possible to formulate acceptance criteria for conditioned waste to be disposed of in SFL until a decision has been made on the repository concept.

Programme

In parallel with the development of the repository concept, SKB will formulate acceptance criteria for waste to be deposited in a future SFL. The properties that will serve as a basis for the acceptance criteria will be identified based on acceptance criteria for SFR (see Section 4.3.1), properties of the existing waste and properties of the repository concepts. As details of the repository design are finalized, it will be possible to further define the acceptance criteria. Final acceptance criteria will be defined when the repository design has been determined. The acceptance criteria will also clarify which waste should be deposited in SFL, from an ALARA and BAT perspective.

6.6 Safety evaluation

SKB's method for development of SFL is an iterative process where technology development and research are followed by evaluation of the long-term safety of the repository. The evaluation of long-term safety that is planned for 2016 is not a complete safety assessment, since important components for making such an assessment are missing, for example site-specific data. The purpose of the safety evaluation is to choose SKB's main line among different repository concepts. Moreover, the safety evaluation will develop the set of requirements for the waste, the engineered barriers and the rock in order to support the work with waste acceptance criteria, continued technology development and the site selection process. The safety evaluation also constitutes a basis for identifying areas for further research and serves as support in selecting and prioritizing those areas where the greatest development needs exist.

In addition to the preliminary safety assessment from 1999 (SKB 1999), the concept study and the updated reference inventory – both of which will be presented at the end of 2013 – comprise important background material for the safety evaluation, along with results from completed research projects for SFL and other repositories. Site data will be taken from the site investigations for the Spent Fuel Repository. Other background material consists of international databases for features, events and processes in the final repository and SKB's own databases for features, events and processes from e.g. the safety assessments for the Spent Fuel Repository and the extended SFR.

6.7 Site selection process

SKB has previously established the basic requirements for siting of final repositories:

- Bedrock that enables the safety requirements to be met.
- Political and general acceptance in the concerned municipality and among nearby residents.

The basic principle for site selection for SFL is a stepwise site selection process following the same methodology as that used for the Spent Fuel Repository. In such a process, safety-related properties and local acceptance are crucial for siting, but other factors such as health, environment and societal resources are also important. The goal is to conduct an open and transparent process in collaboration with SSM and the municipalities, where the premises for different actors are clarified at an early stage and where the different steps in the process have been agreed upon and communicated.

Study of site selection process

An internal study of the process leading to the selection of a site for SFL will be initiated during the coming RD&D period. The study will identify competency requirements and an appropriate organization for the siting work and present a plan for how the site selection work will be conducted. Furthermore, the study will, after the safety evaluation has been completed, propose evaluation factors for site selection (siting factors), categorized on the basis of safety-related site characteristics, societal resources, technology for execution, and health and environment.

Feasibility study phase

Based on the results of the assessment of the long-term safety of the repository concept for SFL that is planned for 2016, preliminary requirements can be made on the site for the final repository. These preliminary requirements, in combination with the identified siting factors, provide premises for exploring the possibilities for siting based on both site characteristics and local acceptance. SKB's intention is to commence the local consensus-building process for SFL by means of a dialogue with municipalities and private citizens after the safety evaluation has been carried out.

SKB's strategy is to primarily target municipalities where good prospects are judged to exist for being able to ensure long-term safety and that have previously shown an interest in the siting of nuclear facilities. At the same time, SKB is positive to initiatives by individual municipalities to participate in feasibility studies, and such municipalities will be treated on an equal footing with those municipalities that are the subject of more targeted contacts. SKB does not rule out the possibility that additional municipalities will be contacted in this phase of the process. Those municipalities that choose to participate in a feasibility study will be investigated on the basis of the established siting factors. The work will be pursued in dialogue and collaboration between SKB and the concerned municipality.

Site investigation phase

Site investigations will be conducted in those municipalities that show an interest in hosting SFL and where, based on the feasibility studies, SKB judges that the prospects are good. The site investigations entail a stagewise characterization of proposed sites, where the site's safety-related characteristics, as well as environmental and societal factors, are investigated in greater detail. 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.

Programme

The siting work will be commenced during the coming RD&D period. An internal study will identify competency requirements and an appropriate organization for the siting work and, after a safety evaluation has been carried out, present a plan for how the site selection work will be conducted. Furthermore, the study will propose evaluation factors for site selection (siting factors), categorized on the basis of safety-related site characteristics, societal resources, technology for execution, and health and environment.

7 Near-surface repositories

This chapter explains how the question of disposal of very low-level, short-lived waste from dismantling and demolition of the nuclear power plants is planned to be managed. This waste comprises a relatively large portion of the waste from dismantling and demolition, but contains only a fraction of the radioactivity. Comparisons show that it can be advantageous to put it in a near-surface repository instead of depositing it in SFR. A conventional waste facility in combination with an interim storage facility can serve as a complement to such shallow land disposal. This provides greater opportunities for source separation of very low-level, short-lived waste from dismantling and demolition.

The nuclear power plants in Forsmark, Oskarshamn and Ringhals say in their decommissioning plans that a near-surface repository for the very low-level waste from dismantling and demolition shall be available, see Section 9.3. BKAB takes a positive view of the possibility of disposing very low-level waste from dismantling and demolition in a central near-surface repository. The licensees for the nuclear power plants are thereby of the opinion that the future main alternative for final disposal of the very low-level and short-lived waste from dismantling and demolition is shallow land disposal. At present, this waste is included in the applications for extending SFR in order to obtain some flexibility and freedom of choice.

Conclusions in RD&D 2010 and its review

In its review of RD&D programme 2010, SSM took a positive view of the initiatives that have been taken to obtain a better estimate of the quantities of waste from dismantling and demolition that are intended to be placed in a near-surface repository. However, SSM said that the information in the RD&D programme regarding the work and the feasibility studies that have been done was far too meagre for an assessment. For example, no reference was made to any report on the feasibility study regarding shallow land disposal. Nor was it clear when the strategic decision on shallow land disposal would be taken.

SSM considered it urgent that the intentions of the industry to dispose of very low-level waste by shallow land disposal be elaborated on and defined more concretely, particularly in view of the closely related need to determine the capacity of an extended SFR for waste from dismantling and demolition.

SSM said that near-surface repositories need to be seen as part of a larger system of various ways to dispose of waste. Guidelines need to be developed by the industry regarding what waste can and should be disposed of by shallow land disposal. In SSM’s opinion, waste that can be recycled or reused in an environmentally and radiologically safe and cost-effective way should not be deposited in a final repository. Several reviewing bodies shared this assessment. As SSM stated in the review of RD&D programme 2010, the guidelines for distribution of waste between the different repository parts in SFR should also include near-surface repositories and SFL.

According to the review statement regarding RD&D programme 2010, the Swedish National Council for Nuclear Waste could not see any viable reasons not to allow the principles of conservation and sustainability to be fully applied in conjunction with the treatment of radioactive waste. This was particularly true of very low-level waste deposited in the near-surface repositories at the nuclear power plants and on the Studsvik site.

Current situation

Since RD&D programme 2010, the low-level waste from dismantling and demolition that is intended to be disposed of in SFR has been examined to determine how large a fraction can be disposed of in a near-surface repository. The waste inventory from the studies of decommissioning of the nuclear power plants in Forsmark, Oskarshamn and Ringhals, where near-surface repositories are in operation, has been analyzed. The results suggest that about half of the low-level waste from dismantling and demolition may be eligible for final disposal in near-surface repositories such as those that exist today for operational waste instead of being deposited in SFR.

Today there is no licence to dispose of low-level waste from dismantling and demolition in near-surface repositories, so the extension of SFR is being designed for all this waste.

Programme

During 2013, SKB started a project aimed at gathering data as a basis for a decision on whether a near-surface repository can provide an alternative for disposing of portions of the waste from dismantling and demolition in an environmentally and radiologically safe and cost-effective manner, and whether conventional waste facilities can be utilized in any way.

If shallow land disposal proves to be an advantageous disposal alternative, a decision must be made whether SKB should operate a central near-surface repository or whether the near-surface repositories that exist locally today at the nuclear power plants should be extended and used.

The goal is that the fundamental questions should be answered by early 2014 and that the project can be concluded and decisions on future policy taken during the current three-year period.

8 Technology development for final disposal of low- and intermediate-level waste

8.1 Final repository for short-lived radioactive waste

Waste arising during operation, maintenance and decommissioning of the Swedish nuclear power plants is deposited in SFR. The repository consists today of four rock vaults for different types of waste – one rock vault for low-level waste, 1BLA, one rock vault for intermediate-level waste, 1BMA, two rock vaults for concrete tanks, 1BTF and 2BTF – plus a silo. The total existing disposal volume is about 63,000 cubic metres. An extension of SFR by 110,000 cubic metres to make room for the waste from decommissioning of the Swedish nuclear power plants and with room for nine BWR pressure vessels is currently being planned and designed. Altogether, the plans call for six rock vaults: four for low-level waste, 2–5 BLA, one for whole BWR-reactor pressure vessels, BRT, and one for intermediate-level waste, 2BMA. Virtually all waste will arrive at SFR packaged. The exception will be large components, such as whole reactor vessels. The waste is treated and adapted to the waste containers that can be handled in SFR. The different types of waste packages that can be, and are planned to be, handled in SFR are: ISO containers (full and half height), concrete tanks, steel drums, concrete moulds, steel moulds and tetramoulds. The tetramould is a package intended to be used for waste from dismantling and demolition with outside dimensions of 2.4×2.4×1.2 metres (width×length×height).

The engineered barriers and the technology development underlying the detailed design of the barriers are described in the following sections.

8.1.1 Overview of engineered barriers

The purpose of the engineered barriers in SFR is to prevent or retard the release of radionuclides to the surrounding environment. Different requirements are made on the choice of barriers depending on the properties of the waste. In BLA, safety is based on the limited nuclide inventory, so there are no engineered barriers there. In BMA, the primary engineered barrier is a concrete structure in which waste packages are placed. The space between the concrete structure and the rock is backfilled with crushed rock at closure. In the silo, the engineered barriers consist of both a concrete structure in which the waste has been grouted and a bentonite fill between the concrete structure and the rock wall to further reduce the water flow through the repository. In the rock vaults for concrete tanks (1BTF and 2BTF), the engineered barriers consist of the concrete tanks and the surrounding grout. 1BTF also contains ash drums, which are progressively embedded in grout. The space between the grout and the rock is backfilled with crushed rock at closure.

In order for the engineered barriers to fulfil their expected long-term function, accuracy is required when choosing materials and methods used in their construction. This section describes the development of design and construction methods to meet the requirements on the long-term properties of the engineered barriers. The research being done on development of materials is described in Chapter 22 ("Concrete barriers") and Chapter 25 ("Buffer and backfill").

Conclusions in RD&D 2010 and its review

The technology development for the SFR extension was described briefly in RD&D programme 2010.

In its review, SSM said that the description of the programme was much too brief to permit a judgement. According to SSM, if the future extension of SFR is to satisfy the requirement on best available technology, it may be necessary to strengthen the barrier function of BLA-like repositories.

Programme

The design of the engineered barriers in the extension of SFR will differ from the design of the barriers in the existing facility. In order to ensure that the planned solutions can be implemented, the following development needs have been identified:

- 2BMA, use of unreinforced concrete in barrier.
- 2BMA, development of concrete recipe for unreinforced concrete in barriers.
- Development of grout and grouting method in 2BMA.

In addition to development of new materials and methods for construction of the extension of SFR, experience from the existing facility has also indicated needs for improvement, such as post-construction inspection and control of the environment in the repository. The latter includes control of humidity and preventing leaking groundwater from dripping on technical installations and barriers by installation of e.g. a waterproofing membrane.

8.1.2 Engineered barriers for the silo

Conclusions in RD&D 2010 and its review

Technology development linked to the silo was not described in RD&D programme 2010. SSM had nothing to say in its review of technology development. It was, however, noted elsewhere in the review that SSM took a positive view of SKB's plans for revision and remedial measures in response to the discovery of contaminated groundwater in a number of shafts in the concrete silo.

Current situation

During the period 2011–2013, SKB commissioned a number of studies linked to the presence of groundwater in the silo. These studies examined how the silo would be affected by the presence of water as well as the probability that water was present in the silo during grouting.

The experimental investigations that have been conducted have shown that the grout separates if water is present in the shaft during grouting. This separation leads to the formation of a cement-rich layer at the top and an aggregate-rich layer at the bottom of the grout column. The hydraulic and mechanical properties of these two layers have been further investigated in a number of studies, where the main focus has been the hydraulic properties of the cement-rich layer. These properties determines whether the grout in question satisfies the requirements on gas permeability.

The studies have shown that the hydraulic conductivity of the cement-rich layer varies slightly from sample to sample, but that it is on average above the required level in those samples considered to best represent the cement-rich layer that forms during grouting in the water-filled column.

Programme

SKB does not at present have any plans to conduct further experimental studies in this area, but will compile and evaluate the results of the completed studies.

8.1.3 Engineered barriers for intermediate-level waste in existing SFR

BMA is a 140-metre-long concrete structure divided into 15 cells. Conditioned waste in concrete or steel moulds and steel drums is deposited in these cells, see Figure 8-1.

When SFR was constructed, the facility was designed for a planned operating time of about 30 years. The reason for this was the decisions that had been made that SFR should be closed in about 2010, when the Swedish nuclear power plants were to be phased out. New decisions have since been made to extend the operating times of the nuclear power plants, which means SFR needs to be kept open longer.

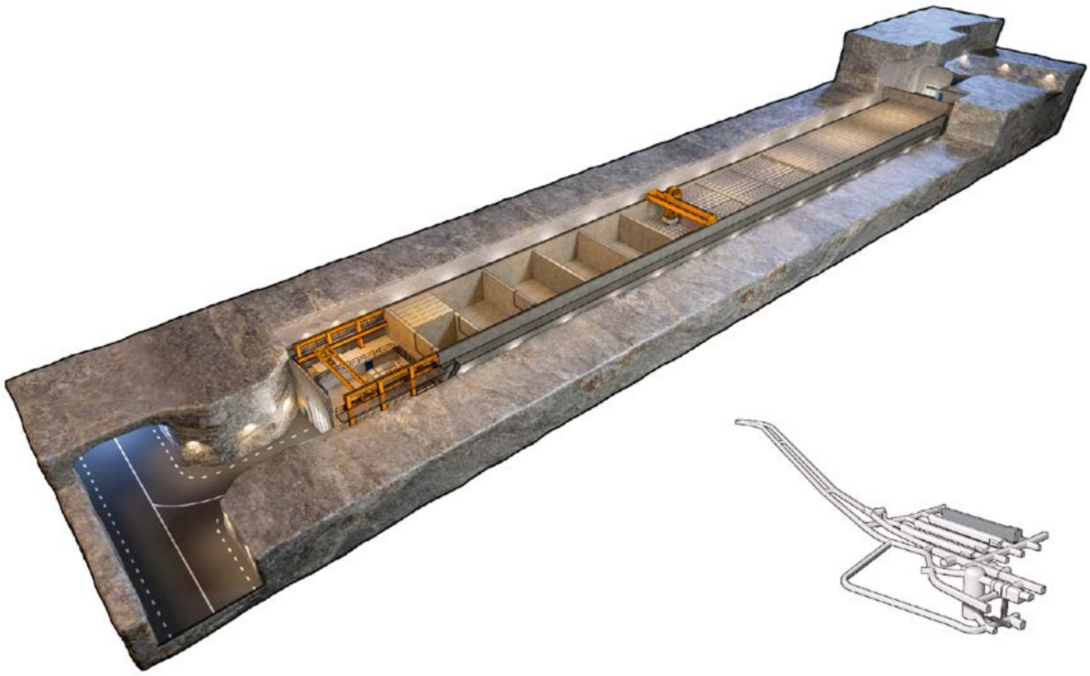


Figure 8-1. 1BMA – Rock vault for intermediate-level waste in existing SFR.

Conclusions in RD&D 2010 and its review

In RD&D programme 2010, SKB's plans for an investigation programme was described. The purpose of the programme was to determine the status of both systems and concrete structures. The study would also indicate needs for repairs and propose when they should be undertaken. SSM did not express any viewpoints on this in thier review.

Current situation

During 2010 and 2011, an extensive investigation programme was conducted of the status of the concrete structures in BMA. The findings of this investigation programme will also serve as a basis for designing the extension of SFR.

The study revealed damage caused by corrosion of steel reinforcement. The corrosion has been caused by enrichment of chlorides in the surface layer due to dripping of chloride-containing groundwater onto the concrete structure. The damage was found both on the engineered barriers and on other structural parts where steel parts had been cast in concrete, such as base slabs.

In addition to damage caused by corrosion of embedded steel, penetrating fractures were also noted in the structures. The fractures were created when the structures were built and stem from cooling movements during casting in combination with shrinkage of the concrete.

An investigation of embedment of the waste in BMA was conducted in 2012. The purpose was to explore the possibilities of using the grout that is used today in the silo for embedment of the waste in BMA as well, and to determine whether the method was suitable for this purpose. The study showed that neither the grout nor the method was suitable for embedment of waste in BMA (Pettersson and Thunberg 2012).

Programme

As mentioned above, the status assessment of BMA revealed damage of a kind that needs to be repaired. The choice of time for the repair measures depends on the nature of the damage. Repairs that are needed to maintain safe operation should be done promptly, while repair that could be of importance for post-closure safety can be done any time before closure of the repository. SKB's programme for repairs of BMA and other technology development that need to be done before closure are presented below.

Dealing with penetrating fractures

Penetrating fractures were noted both in the barrier walls and at the casting joints in the barrier's bottom slab. These fractures risk short-circuiting the function of the permeable material around the barrier structures and need to be repaired before closure. Methods for this include e.g. drilling followed by injection of a fine-grained grout. Even though the method is fairly straightforward, opportunities to verify the results are limited, particularly in the case of the bottom slab. SKB is currently investigating different methods for repairing fractures and is determining when repairs should be done.

Repair of damage caused by corrosion of embedded steel

The visible damage to the concrete barriers can mostly be attributed to the fact that chloride-containing groundwater has dripped onto the structures, leading to enrichment of chloride ions in the surface layer of the concrete. With time, the chloride ions have penetrated into the concrete, reached the reinforcing steel and nullified the passivity to corrosion otherwise created by the high pH of the concrete pore water. The corrosion process leads to the formation of corrosion products, whose volume is greater than that of the original material. This causes a pressure inside the concrete, and if the tensile strength of the concrete is exceeded, concrete spalling will occur and the reinforcement is exposed, see also Section 20.2.19.

Repair of concrete structures that have suffered corrosion of the surface reinforcement can be carried out by removal of the damaged concrete along with some of the undamaged concrete. This can be done by water blasting the damaged concrete, after which a new, fairly thick layer of a dense concrete is applied. In addition to creating a rough surface suitable for application of a concrete overlay, water blasting also remove corrosion products and chloride from the existing reinforcement. This provides good conditions for the overlay material. Since a waterproofing membrane has been installed in BMA, no new accumulation of chloride ions is expected on the concrete, so the repair will most likely last.

Improvement of the climate in the facility

Damage to the concrete structures in the facility caused by corrosion can largely be attributed to the fact that the climate in the facility is unfavourable for iron and steel. The combination of high humidity and high chloride concentrations creates a favourable environment for corrosion.

The waterproofing membrane that was installed in 1BMA in 2010 protects the concrete structures against dripping groundwater and should stop further enrichment of chloride ions in the concrete. The chloride that has already penetrated into the concrete will remain and affect both the reinforcement and the concrete. This requires thorough repairs. Installing climate control equipment can reduce the humidity in the facility and reduce the effect of the elevated chloride concentration in the concrete.

Development of grout and grouting method for BMA

As mentioned above, there are problems regarding the choice of both grouting material and grouting method for embedment of BMA (Pettersson and Thunberg 2012). For this reason, a new grout and a new application method will be developed.

The requirements on materials and methods are based on the rate of gas formation in 1BMA, the distance between the moulds, the height of the mould stack and the swelling properties of the waste. The monolith that will be formed when the waste is grouted must be strong enough to withstand the unilateral water pressure that arises when the repository becomes water-saturated after backfilling and closure. The time when this technical development will be done has not fully been determined, but the goal is that it should be carried out during the next three years.

8.1.4 Engineered barriers for low-level waste in extended part of SFR

Four rock vaults for short-lived low-level waste, 2-5BLA, are planned to be built in the extension of SFR. Waste from the decommissioning of the Swedish nuclear power plants will be deposited in the planned rock vaults. The waste, most of which will be deposited in ISO containers of different sizes, consists primarily of materials from dismantling and demolition in the form of steel, concrete and sand.

Conclusions in RD&D 2010 and its review

This area is described only briefly in RD&D programme 2010. SKB made the judgement that most of the waste that will be placed in an extended SFR will be low-level and will be deposited in repository parts with structures similar to those in today's rock vault for low-level waste. SKB also said that changes of the barriers in the repository part and conditioning of the waste will be reconsidered, among other things with regard to the prospects for backfilling. This will be done based on operating experience and experience from the most recent safety assessment for SFR (SKB 2008a).

In its review, SSM noted that strengthening of the barrier function for the future extension may be necessary in order to meet the requirement on best available technology.

Current situation

Since RD&D programme 2010, SKB has carried out a number of studies concerning the choice of barrier design for rock vaults 2–5BLA. A number of alternative designs have been discussed, each with its own advantages and challenges. In this work, which is not yet finished, the present-day solution for 1BLA has been compared with designs similar to those in 1BMA and the alternative of filling the waste packages with cement-based mortar.

The technical solution that is used today for 1BLA can be considered well-known and could also be used in 2–5BLA.

One of the alternatives that has been studied is a 1BMA-like structure that offers both radiation protection during operation and improved barrier properties after closure. However, this design entails a number of technical challenges of which withstanding a unilateral water pressure equivalent to a water column of 150 metres is considered to be the most difficult. This can be dealt with by making the walls strong enough to withstand this pressure or by filling the waste packages with concrete. Alternatively, the concrete structures can be built so that water can flow into the repository in a controlled manner and thereby create a counterpressure during the most critical phase. Of these alternatives, controlled water inflow has been found to be the most promising.

The third alternative that has been studied is to fill the waste packages with cement-based mortar to stabilize the waste and create a large sorption surface. This solution is not without its challenges either, since the waste packages must be filled so that mortar and waste are mixed homogeneously and no large voids are left through which the water can flow. This alternative also requires that the proportion of reactive metals be kept low.

Programme

The design of 2–5BLA is described in the applications for a licence to extend SFR based on a larger body of data concerning pre- and post-closure safety.

8.1.5 Engineered barriers for intermediate-level waste in extended part of SFR

Waste from decommissioning of the Swedish NPPs will be emplaced in the planned rock vault for short-lived intermediate-level waste, 2BMA. The waste, which will for the most part be deposited in steel and concrete moulds, consists primarily of materials from dismantling and demolition in the form of steel and concrete. The total disposal volume in 2BMA has been estimated at 22,000 cubic metres.

Conclusions in RD&D 2010 and its review

Technology development for the engineered barriers in 2BMA was described only briefly in RD&D programme 2010. In its review, SSM took a positive view of the studies that were planned, for example regarding the addition of a bentonite barrier in a future 2BMA.

Current situation

Since RD&D programme 2010, SKB has commissioned numerous studies concerning possible designs of the future 2BMA. This work has drawn heavily on experience from the investigation programme carried out in the existing facility. Among the most important lessons learned are:

- Separate the overhead crane runway from the engineered barriers.
- Increase the spacing between moulds to facilitate grouting.
- Divide the long repository structure into freestanding sections to reduce the risk of fracturing during construction.
- Build the structure of unreinforced concrete.
- Cast floor and walls at the same time to reduce constraint and fracturing at the transition between them.
- Do not allow leaking groundwater to drip on the engineered barriers.

In addition, questions arise concerning the choice of concrete composition for the barriers, the temperature during construction and the grouting method.

Based on these premises and experience from 1BMA, it has been decided to build the engineered barriers in 2BMA in the form of caissons. This entails freestanding concrete structures with approximate dimensions of 16×16×8 metres spaced at a distance of about 1.5 metres, Figure 8-2. The freestanding concrete structures are planned to be constructed of unreinforced concrete. The size of each caisson is adjusted so that the tetramoulds are spaced at a distance of about 100 millimetres to permit grouting. The overhead crane runway will be mounted on pillars which are separated from the barrier structures. With the waste quantities assumed for disposal in 2BMA, 14 caissons will be needed, resulting in a disposal vault with a total length of about 275 metres.

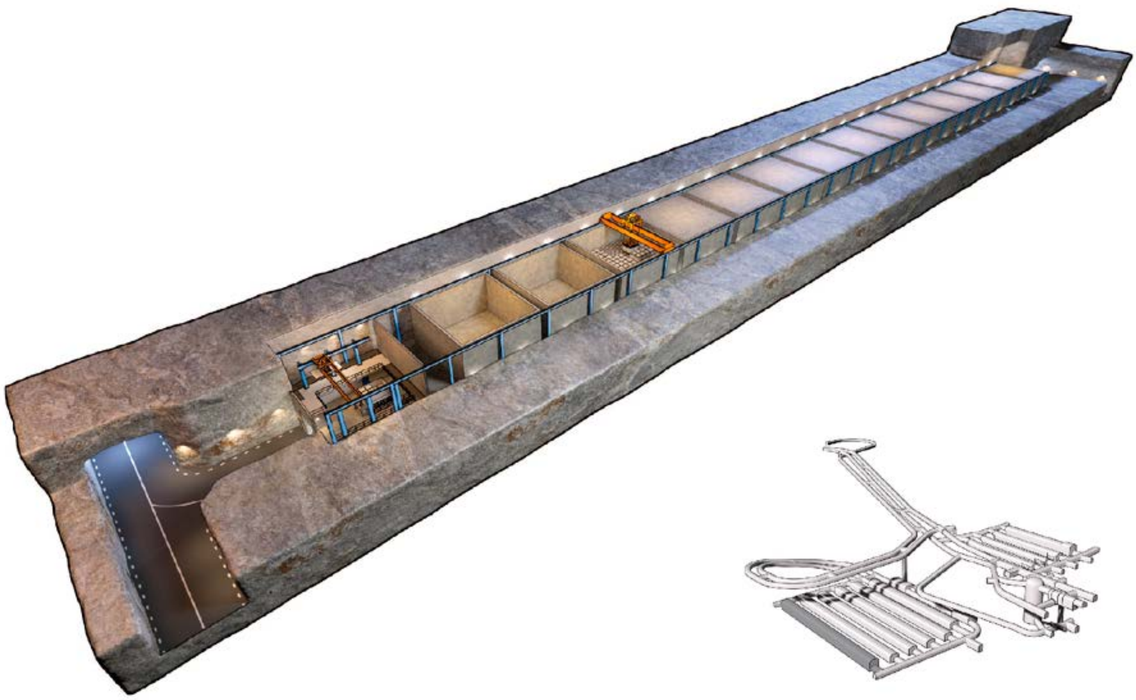


Figure 8-2. *2BMA – Rock vault for intermediate-level waste in extended part of SFR.*

Programme

In accordance with what is said above, a number of possible improvements have been identified for the new 2BMA compared with 1BMA. Several of these require technology development in order to be realized. SKB's technology development programme for the 2BMA repository part is presented below.

Casting of barriers of unreinforced concrete with the dimensions indicated above requires thorough preparations. SKB therefore plans to build such a structure on a full scale to verify method, execution and result. A quality control programme for design and construction will be developed to ensure that the requirement specification is fulfilled. This work will be started when the development of the concrete composition for the caissons has been commenced.

In order to ensure that requirements on the repository's long-term safety are met, the concrete structure must contain as few fractures as possible. There are several methods for achieving this, but they cannot all be combined with other design requirements. A carefully designed concrete is needed, but above all very low shrinkage and good workability and density. This work has been initiated and problem formulation is under way. A development plan will be prepared.

Grouting in 2BMA essentially serves two purposes: 1) to exert, together with the waste packages, resistance to the water pressure that arises due to resaturation of the repository, and 2) to minimize the advective water flow around the waste packages.

The grout used in the silo is designed for good fluid properties in 70-millimetre-wide gaps. For this reason, and on the strength of study results (Pettersson and Thunberg 2012), it has been decided that the distance between the tetramoulds in 2BMA must be 100 millimetres in order for the existing grout to be used.

Handling of this grout requires some caution due to its tendency to separate. It should therefore not be pumped too vigorously or released from a height equivalent to the whole mould stack.

Whether the solution entails modifying the method or adjusting the recipe will be a question for a future study to determine. To start with, the requirement specification will be determined and technology development carried out, supported by the work being done on grout for 1BMA.

8.1.6 Engineered barriers for reactor pressure vessels in extended part of SFR

A rock vault for disposal of whole BWR pressure vessels, called BRT, is being planned in the extended part of SFR, see Figure 8-3. The PWR pressure vessels will be disposed of in SFL and will be interim-stored until SFL has been put into operation.

The layout of a BRT rock vault is based on the assumption that the RPVs are placed in a long row in a rock vault about 14 metres wide, 12 metres high and up to 210 metres long. The engineered barrier is the concrete in which the RPVs are embedded. The space between the grouting and the walls of the rock vault is filled with a permeable material such as macadam. To minimize the void volume in the repository, it is also possible to fill each reactor vessel with a cementitious mortar. The calculated waste volume for BRT has been estimated at 20,000 cubic metres.

Conclusions in RD&D 2010 and its review

RD&D programme 2010 did not describe any work linked to the BRT rock vault. In its review, SSM did not express any viewpoints on this.

Current situation

Since RD&D programme 2010, SKB has conducted a series of investigations concerning choice of barrier design for the BRT disposal vault, and a number of alternative solutions have been discussed, each with its own advantages and challenges. Considering the advantages of an environment where the corrosion rate can be kept low, it has been decided to use cementitious material in the engineered barriers in this rock vault. The exact design and method of construction has not yet been clarified.

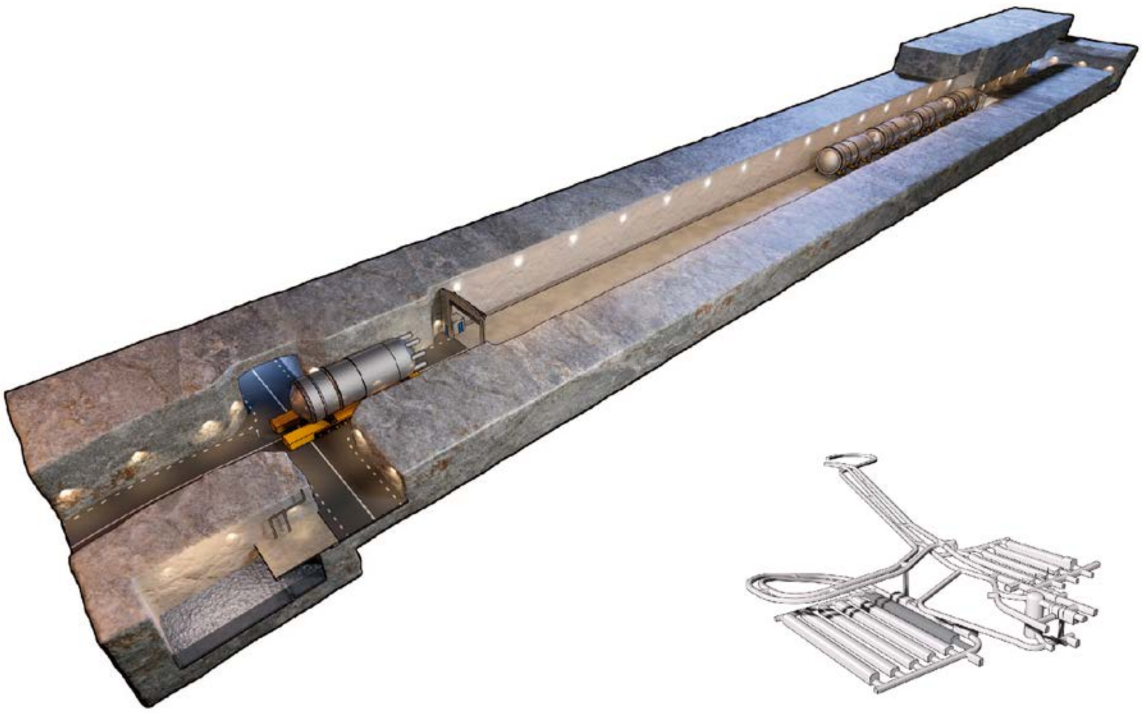


Figure 8-3. BRT – Rock vault for reactor pressure vessels in extended part of SFR.

Programme

SKB's programme for development of technical solutions for BRT includes a number of areas, the most important of which are:

- Study of need for internal grouting in reactor vessels.
- Development of material and method for internal grouting in RPVs, where applicable.
- Design of method for execution of external grouting.
- Development of method to prevent the RPVs from floating up during grouting in the rock vault.

8.1.7 Closure

When operation of SFR is concluded and measures included in the decommissioning plan have been carried out, all parts of the underground facility shall be closed and sealed. This entails filling the underground facility with material whose purpose is to reduce the mobility of the radionuclides and prevent access to the waste. Closure entails backfilling of rock vaults, installation of plugs consisting of mechanical plugs and hydraulically tight sections, and backfilling of access ramps and tunnel system. The different parts of SFR will be backfilled in different ways and with different materials to achieve the desired overall function.

Natural materials that are mechanically and chemically stable over a long time will be used for the closure components.

The process of closure is assumed to be carried out during a relatively short span of time, where the rock vaults that are being filled with waste will remain open until the entire SFR has been closed. The closure rate is dependent on water flows and pressure distribution, which means that more in-depth studies of these parameters will be conducted. Developing a strategy for closure of SFR is an ongoing project that will continue.

Conclusions in RD&D 2010 and its review

SKB described the closure strategy only briefly in RD&D programme 2010, stating merely that they intended to develop the strategy for closure of the existing facility and to devise a strategy for the extended part of the facility. In its review, SSM did not express any viewpoints on this.

In its review of the 2008 safety assessment (SKB 2008a), SSM requested a more well-defined closure strategy. Since that time, SKB has worked continually to clarify how closure is to be carried out. This work will be described in a closure plan for SFR, which will be included in the application under the Nuclear Activities Act for a licence to extend SFR. The reference design for the closure will be described in the closure plan.

Current situation

Since the publication of RD&D programme 2010, SKB has worked with a conceptual closure strategy for an extended SFR. Figure 8-4 shows an example of how the closure can be designed. The design of the closure will be described in a closure plan that will be included in the application under the Nuclear Activities Act for a licence to extend SFR.

Plugs will be installed in the connection between the rock vaults and the adjoining tunnels. All plugs are made of two main components: mechanical plugs and hydraulically tight sections consisting of bentonite clay. The purpose of the plugs is to reduce the water flow through the rock vaults. When bentonite clay expands on water saturation, a constraint is needed to maintain its density and low hydraulic conductivity. The mechanical plug can consist of either concrete or transition material that conveys the swelling pressure from the tight section to backfill material in the adjoining tunnel system. The latter solution is called an earth dam plug.

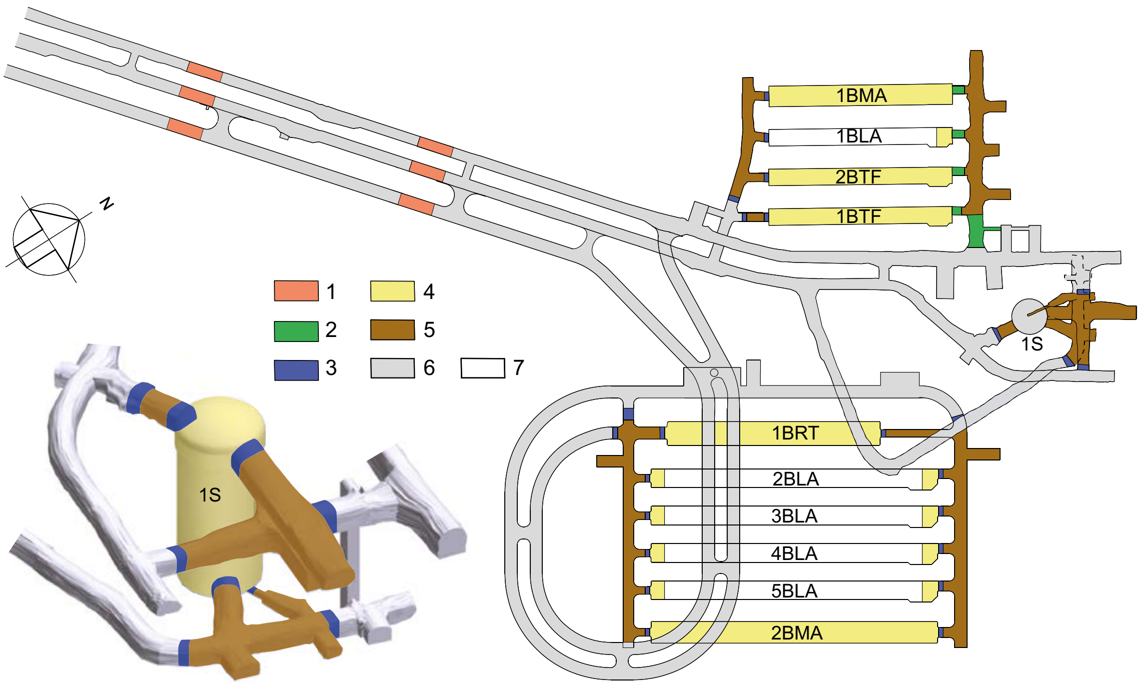


Figure 8-4. *Reference design for closure of SFR including extended part with detailed view of silo. 1) Plugs in access ramps. 2) Transition material. 3) Mechanical plug of concrete. 4) Backfill material of macadam. 5) Hydraulically tight section of bentonite or bentonite mixture. 6) Backfill material in access ramps and tunnel system. 7) Non-backfilled openings.*

Programme

SKB is planning for continued technology development of concrete plugs to achieve a robust design that does not require extensive rock works. One alternative that will be studied is to use concrete plugs where friction against the rock wall absorbs the load. When the final layout of the extension has been determined, a more precise calculation of the dimensions of each concrete plug may be required.

SKB intends to develop the concept of the earth dam plug and the transition material. The evolution of the earth dam plug over time needs to be studied and analyzed. The work that needs to be done mainly involves calculations, parameter studies and modelling.

A more detailed analysis of how the bentonite in the tight sections will be designed and installed needs to be performed. The possibility of achieving sufficiently high density in the hydraulically tight sections with bentonite pellets or granulated bentonite will be studied.

8.1.8 Borehole sealing

There are a large number of investigation boreholes in the area around SFR. Some boreholes have already been sealed according to established practice, while others are still open. It has been decided that all boreholes that are open today need to be sealed.

The purpose of sealing the boreholes is to restore the hydraulic properties of the bedrock so that the boreholes do not provide a flow path for groundwater and thereby do not contribute to radionuclide transport to the ground surface or the seafloor. The method of borehole sealing is based on technology taken from the Spent Fuel Repository and will have to be adapted to the conditions at SFR.

Conclusions in RD&D 2010 and its review

The closure strategy was described in general terms in RD&D programme 2010. SKB said that they intended to develop the strategy for closure of the existing facility and to define a strategy for the extended part of the facility. In its review, SSM did not express any viewpoints on this.

Current situation

SKB has analyzed scenarios for the Spent Fuel Repository where boreholes do not have the same hydraulic conductivity as surrounding rock. These analyses suggest that the requirements on the tightness of the borehole seal can be relaxed (Luterkort et al. 2012).

Programme

Sealing of the borehole at SFR is based on the same concept as closure of the Spent Fuel Repository, which entails filling the borehole with compacted bentonite. In order for the bentonite to be installed under controlled forms, preparation of both borehole and bentonite is required. There is still a need for development of the methods for borehole sealing. SKB therefore intends to review the concept and modify it for use in SFR. A study must also be done of which boreholes have to be sealed before the extension of SFR is carried out.

8.2 Final repository for long-lived waste

Technology development for SFL includes methods and solutions for conditioning of waste, waste containers, engineered barriers for the final repository, and closure/sealing.

The different repository designs for a geological final repository for long-lived low- and intermediate-level waste that SKB is considering are described in Section 8.2.1. The series of waste containers which SKB has developed on the conceptual level to hold different types of long-lived low- and intermediate-level waste are presented in Section 8.2.3. A study of technical solutions for segmentation of BWR control rods is presented in Section 8.2.5.

8.2.1 Engineered barriers

The purpose of SKB's work is to guarantee long-term safety for man and the environment by disposing of the radioactive waste in a safe manner. To achieve this, different types of waste need to be disposed of in different ways. SKB applies this principle by having different repositories for different types of waste, for example a repository for spent nuclear fuel, and by creating barriers with different properties in different parts of the same repository, such as in SFR. In accordance with this principle, the different rock vaults planned for SFL will be adapted for specific types of waste.

In a concept study for SFL, SKB has developed several different barrier concepts in order to evaluate the repository's long-term safety function, environmental impact, technical feasibility and cost. At the end of the concept study, the barrier concepts will be compared to identify suitable solutions for the various waste fractions. The barrier concepts that are judged to have good prospects of satisfying the safety requirements will be evaluated between 2014 and 2016 with respect to long-term safety. Only after the safety evaluation is completed will main alternatives be chosen.

In order to permit a comparative assessment of the barrier concepts, they have been developed to an equivalent conceptual level. Each barrier concept has been designed as an individual rock vault to permit evaluation of factors linked to constructability and operation. Four different barrier concepts are described in the following sections, three of which comprise individual engineered barriers – of concrete, bentonite and crushed rock/gravel, respectively – and the fourth is a combination of the three barriers in a technical multi-barrier.

In the account of the three barriers concrete, bentonite and crushed rock/gravel, a rock vault has been chosen as the geometric configuration of the disposal room, but other geometric forms, such as a silo, cannot be ruled out. A silo has been chosen for the multi-barrier repository, but a rock vault is possible in this case as well.

A prerequisite for the final repository is a site with suitable bedrock for building a sufficiently large rock vault or silo. A repository depth of 300−500 metres has been assumed in the study.

Repository design based on engineered barrier of concrete

The purpose of an engineered barrier of concrete in a geological repository is to limit both the groundwater flow through the waste and the diffusion of substances to and from the waste. By using cement and concrete, a barrier against advective flow is created, making diffusion the principal process for transport of substances away from the waste. At the same time, the diffusion rate in cement and concrete of a number of important radionuclides is low, and sorption is good. Furthermore, the concrete creates an alkaline environment in the repository, creating a passivating layer on metallic waste. This in turn leads to a lower corrosion rate. The dissolution rate of the radionuclides bound in the metal is thus limited by a high pH in the repository.

In the rock vault, a concrete structure with concrete walls is erected to provide radiation shielding during operation, see Figure 8-5. The repository is divided into different cells on which concrete lids can be placed after waste has been deposited in the cell. The size of each cell is about 10×15 metres. The thickness of the concrete walls is about 0.5 metre. The waste is deposited by means of an overhead crane whose runway rests either directly on the rock or on separate pillars. Waterproofing membrane is installed to divert the drainage water and protect the concrete structures during operation. The more detailed design of the rock vault may be based on the solutions developed for 2BMA in SFR, with freestanding unreinforced concrete caissons fabricated by slip-form construction.

The entire excavated rock volume in the concrete repository is backfilled with concrete after deposition. This results in a solid concrete monolith encasing the waste.

Safety functions and release-limiting factors are:

- Limited flow of groundwater through the concrete.
- Limited diffusion through the concrete.
- Sorption of radionuclides.
- Low corrosion rate for metals.

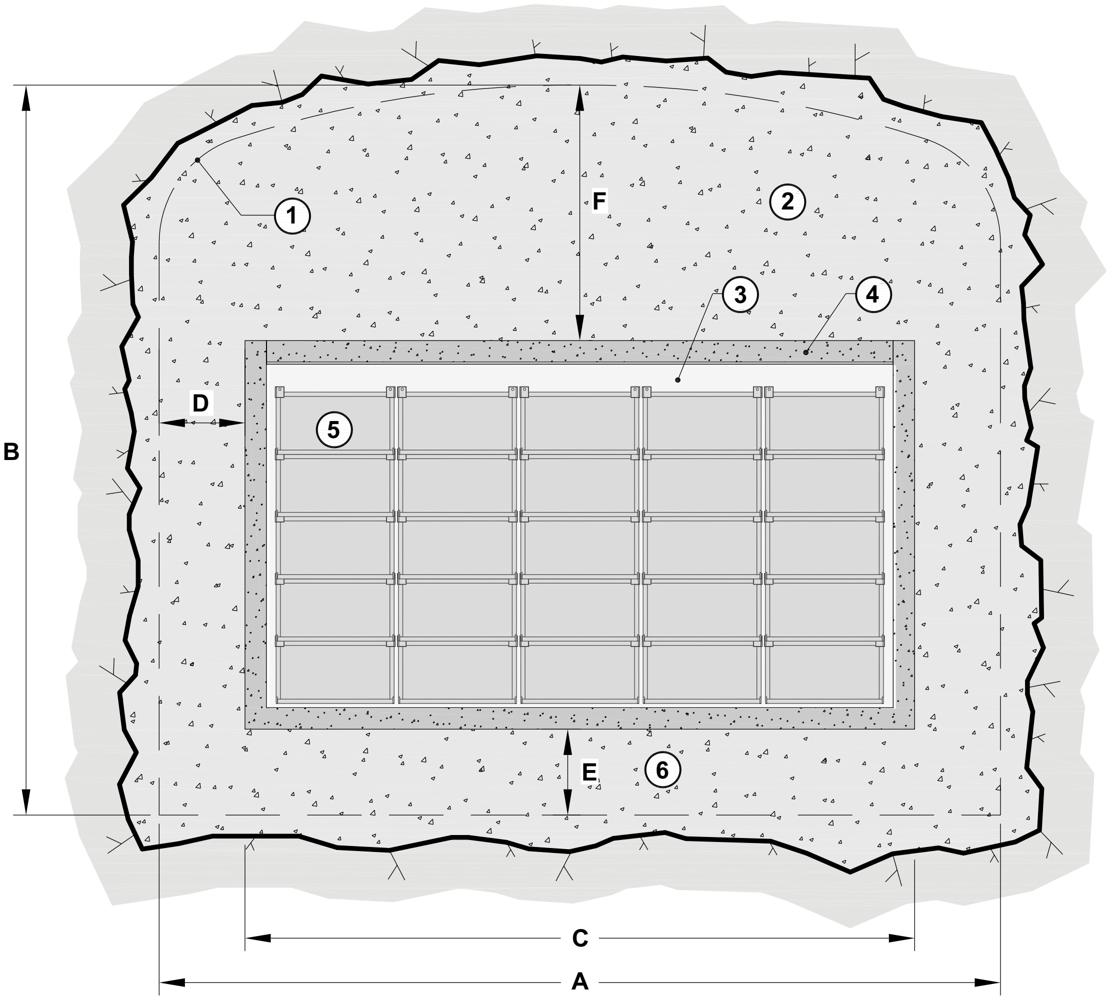


Figure 8-5. *Cutaway view of rock vault with concrete barriers. 1) Theoretical tunnel contour. 2) Unreinforced concrete. 3) Grout. 4) Concrete structure for operation (0.5 m). 5) Waste packages. 6) Unreinforced concrete. Dimensions: A = 20 m, B = 17 m, C = 16 m, D = 2 m, E = 2 m, F = 5–10 m.*

Barriers in the repository:

- Walls, floor and lid in cellular structure: concrete (reinforced or unreinforced), thickness about 0.5 metre.
- Filling of voids: grout between waste packages and between waste packages and cell walls, thickness about 0.1 metre.
- Backfill between cellular structure and rock (sides and bottom): concrete about two metres.
- Backfill above cellular structure: concrete 5−10 metres.

Long-term safety is dependent on, among other things, how well the concrete barriers limit the advective flow of groundwater through the waste volume. The hydraulic conductivity through the concrete is highly dependent on the presence of fractures in the concrete. Reinforcement corrosion is generally a common cause of fractures in concrete. To reduce the risk of fractures, an unreinforced concrete structure is being considered in 2BMA for the extended SFR (see Section 8.1.5). An unreinforced structure is capable of withstanding much less load. During the water saturation phase, after closure, the unilateral hydrostatic pressure in the rock at 300–500 metres depth will be 3–5 megapascals (MPa). In a concrete structure, the load must be absorbed by the waste, the waste packaging and the surrounding grout in order that the concrete structure will not be crushed. As the repository becomes water-saturated, the pressure difference is equalized. When unreinforced concrete is used, the waste and its packaging will have to withstand greater loads than when reinforced concrete is used, which must be balanced against the expected improvement in long-term safety.

A large waste fraction consists of metallic waste such as core components. The concrete repository creates an alkaline environment, which results in a lower corrosion rate for steel and stainless steel than a more neutral environment. Corrosion of metallic waste gives rise to gas. The gas generated during the post-closure phase must be able to leave the repository without adversely affecting the repository's barriers. Technical solutions may need to be developed to handle the gas transport through the barrier.

Repository design based on engineered barrier of bentonite

The purpose of an engineered barrier of bentonite in a geological repository is to limit the groundwater flow through the waste and the diffusion of substances to and from the waste. The use of large quantities of swelling bentonite limits the groundwater flow and leaves diffusion as the main process for transporting substances away from the waste.

A concrete structure with a concrete slab mounted on pillars is built in the rock vault. Concrete walls are raised on the slab for the primary purpose of providing radiation shielding during operation, see Figure 8-6. The structure is divided into cells in the same way as in the rock vault with a concrete barrier, and waste deposition also takes place in the same way by overhead crane. The function of the concrete structure is limited to the operating period, to bear the load of the waste and to provide radiation protection. No bentonite is installed before closure.

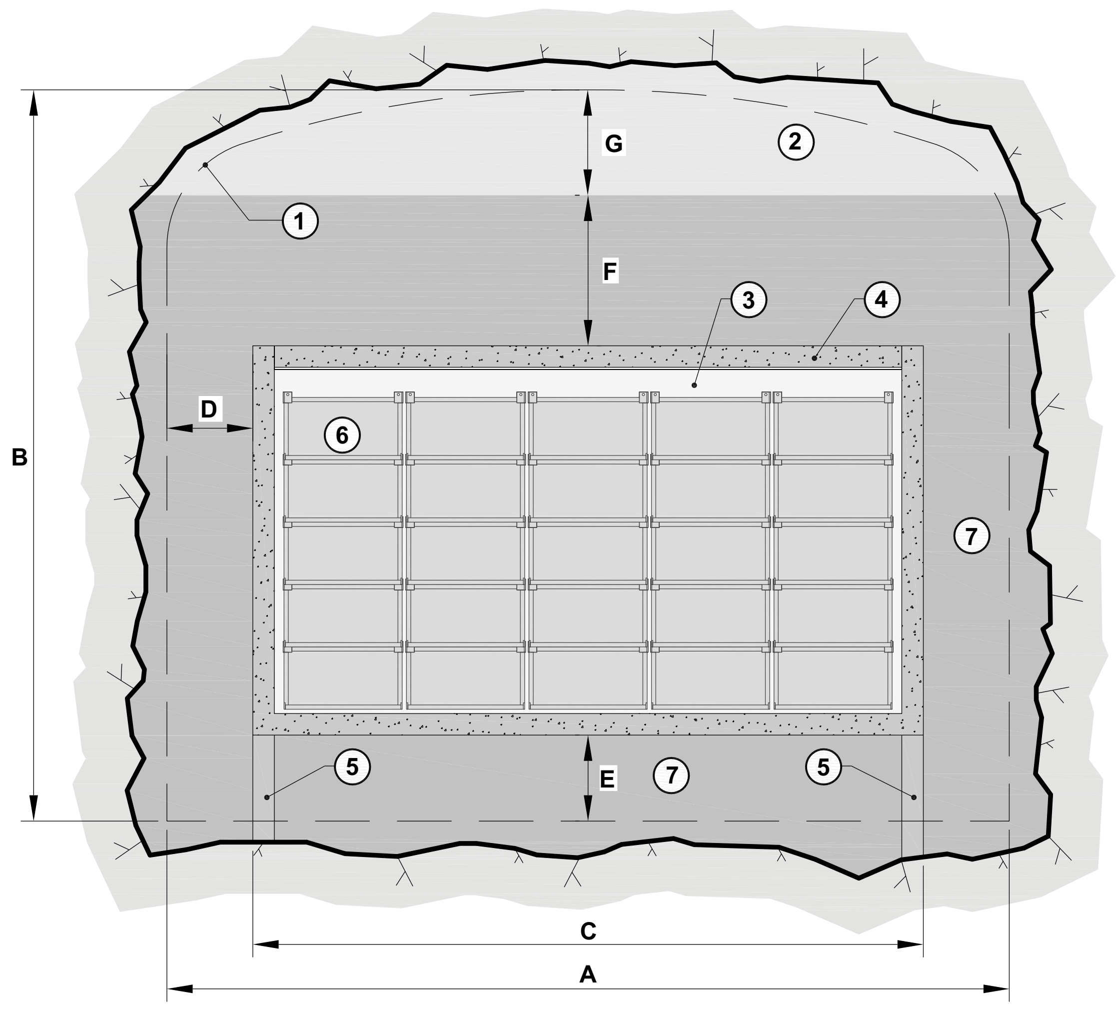


Figure 8-6. *Cutaway view of rock vault with bentonite barrier. 1) Theoretical tunnel contour. 2) Bentonite pellets. 3) Grout. 4) Concrete structure for operation (0.5 m). 5) Granite pillars. 6) Waste packages. 7) Bentonite blocks. Dimensions: A = 20 m, B = 17 m, C = 16 m, D = 2 m, E = 2 m, F = 3–4 m, G = 2–3 m.*

No later than at closure, the waste packages are embedded in grout and a concrete lid is placed on top of the cells. Bentonite blocks are placed beneath the concrete structure, as well as on the sides and on top. The remaining space – between the bentonite blocks in the walls and the walls of the rock vault, as well as at the top of the rock vault – is filled with bentonite pellets.

Due to the limited use of concrete in relation to the quantity of bentonite, the impact of the concrete on the swelling properties of the bentonite is judged to be limited.

Safety functions and release-limiting factors are:

- Limited flow of groundwater through the bentonite.
- Limited diffusion through the bentonite.
- Sorption of radionuclides.

The concrete structure fills important functions during the operating period, but is not credited with any safety function in the assessment of the long-term radiological safety of the concept.

Barriers in the repository are:

- Filling of voids: grout between waste packages and between waste packages and cell walls, thickness about 0.1 metre.
- Backfill at sides and bottom: bentonite blocks with a dry density of 1,600–1,700 kg per cubic metre (kg/m^3), about two metres.
- Backfill on top: bentonite blocks with a dry density of 1,600–1,700 kg/m^3, 3–4 metres.
- Top fill: bentonite pellets with a dry density of about 1,000 kg/m^3.

A dry density for the bentonite of 1,600–1,700 kg/m^3 is assumed as a basis for designing the barriers. Blocks of high-quality bentonite can then be fabricated using standard methods. The remaining empty volume, mainly between blocks and the walls of the rock vault, will be filled with bentonite pellets, which have a lower average dry density. The bentonite blocks are stacked to a width of about two metres, and the remaining gap to the walls of the rock vault is estimated to be 0.2 metre on average. This volume is filled with pellets with a dry density of about 1,000 kg/m^3. If the average dry density of the blocks is 1,700 kg/m^3, the average density at the walls will be 1,550 kg/m^3. Approximately the same relationship is expected to exist between blocks and pellets for backfilling of the volume underneath waste. On top of the waste, the height that is filled with pellets is considerably greater: 2–3 metres in the basic design. This means that the average dry density will be approximately 1,300 kg/m^3. The density gradient in the volume is expected to persist even after homogenization. The difference in swelling pressure in the volume due to the density variation needs to be further investigated.

The continued evolution of the bentonite barriers also needs to take into account how gas evolution in the waste is to be handled. It may be necessary to install valves that allow gas to pass through the saturated bentonite without increasing water transport through the waste. Another alternative is to investigate how low the density in the bentonite needs to be to allow the gas to be transported through the clay without negative effects, while at the same time being high enough to ensure the long-term safety of the barrier. The design of the operational structures needs to be studied in greater detail in order to limit the impact of the concrete on the long-term function of the bentonite.

Repository design based on engineered barrier of crushed rock/gravel

The purpose of an engineered barrier of crushed rock/gravel in a geological repository is to limit the groundwater flow through the waste by creating a hydraulic cage around the waste. The use of large quantities of a high-permeable material such as crushed rock/gravel creates a transport pathway for the groundwater around the waste, which limits the advective flow through the waste and thereby the outward transport of substances from the waste.

A concrete slab resting on a bed of crushed rock/gravel is laid in the rock vault, see Figure 8-7. This bed comprises part of the hydraulic cage. Expansion joints are created in the concrete slab to permit small movements. Roof and walls in the rock vault are sealed with shotcrete. In this solution as well, an overhead crane whose runway rests either directly on the rock or on separate pillars is used for waste deposition.

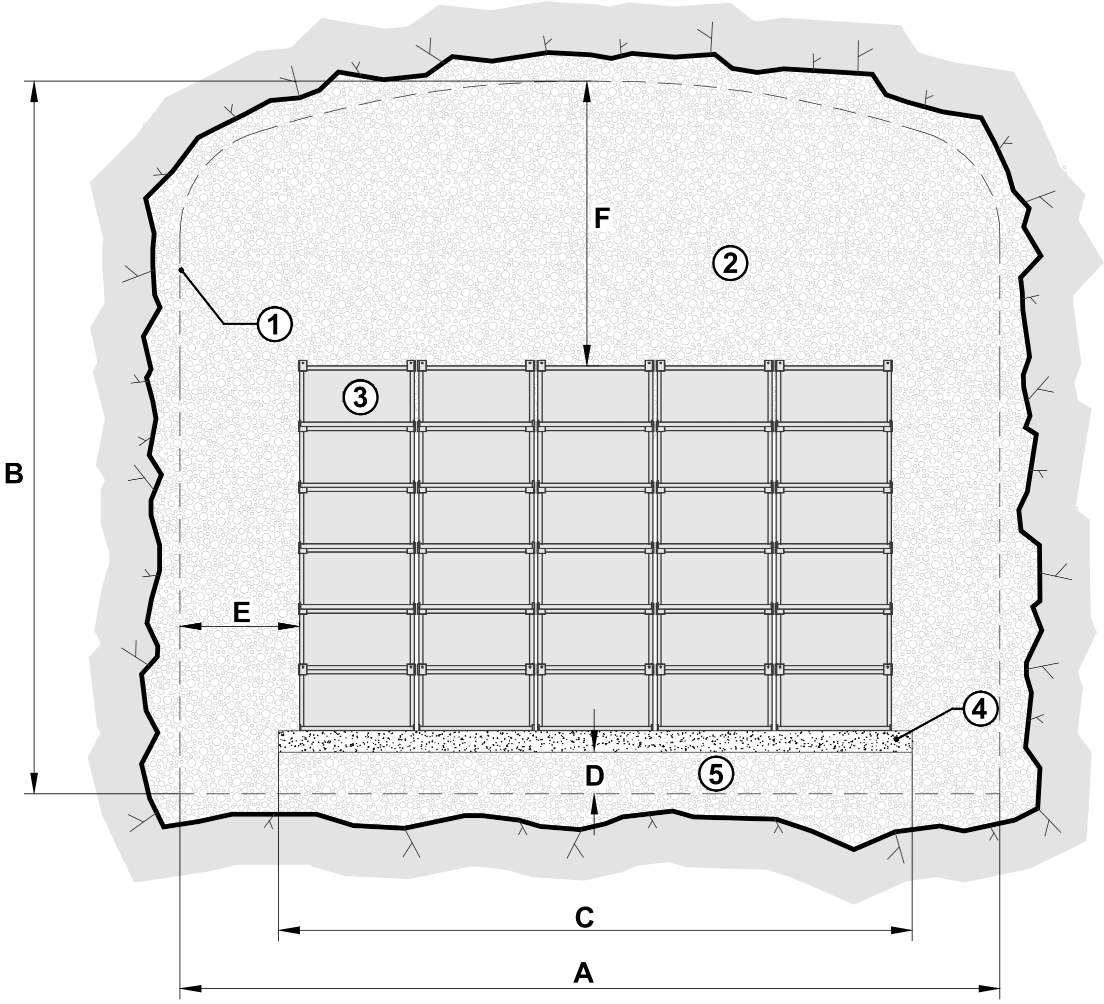


Figure 8-7. Cutaway view of rock vault with barrier of crushed rock/gravel. 1) Theoretical tunnel contour. 2) Crushed rock. 3) Waste packages. 4) Concrete slab (0.5 m). 5) Shot rock and macadam. Dimensions: A = 20 m, B = 17 m, C = 15 m, D = 1 m, E ≈ 2.5 m, F = 5–10 m.

At closure, the space between the waste and the rock is filled with crushed rock/gravel. The function of the concrete slab is limited to the operating period, as a bearer of waste packages.

Safety functions and release-limiting factors are:

- Limited flow of groundwater through the waste.
- Sorption of radionuclides.

Barriers in the repository consist of crushed rock/gravel with uniform particle size and a hydraulic conductivity of 10^{-5} metres per second or more:

- Bed of crushed rock/gravel beneath the waste: one metre.
- Backfilling with crushed rock/gravel on the sides: about 2.5 metres.
- Backfilling with crushed rock/gravel on top: 5–10 metres.

Repository design based on engineered multi-barrier of concrete/bentonite/crushed rock

The purpose of an engineered multi-barrier of concrete, bentonite and crushed rock in a geological repository is to limit the groundwater flow through the waste and the diffusion of substances to and from the waste. This is achieved by a combination of a hydraulic cage (crushed rock), a material with low permeability (bentonite) and a material with low diffusivity (concrete).

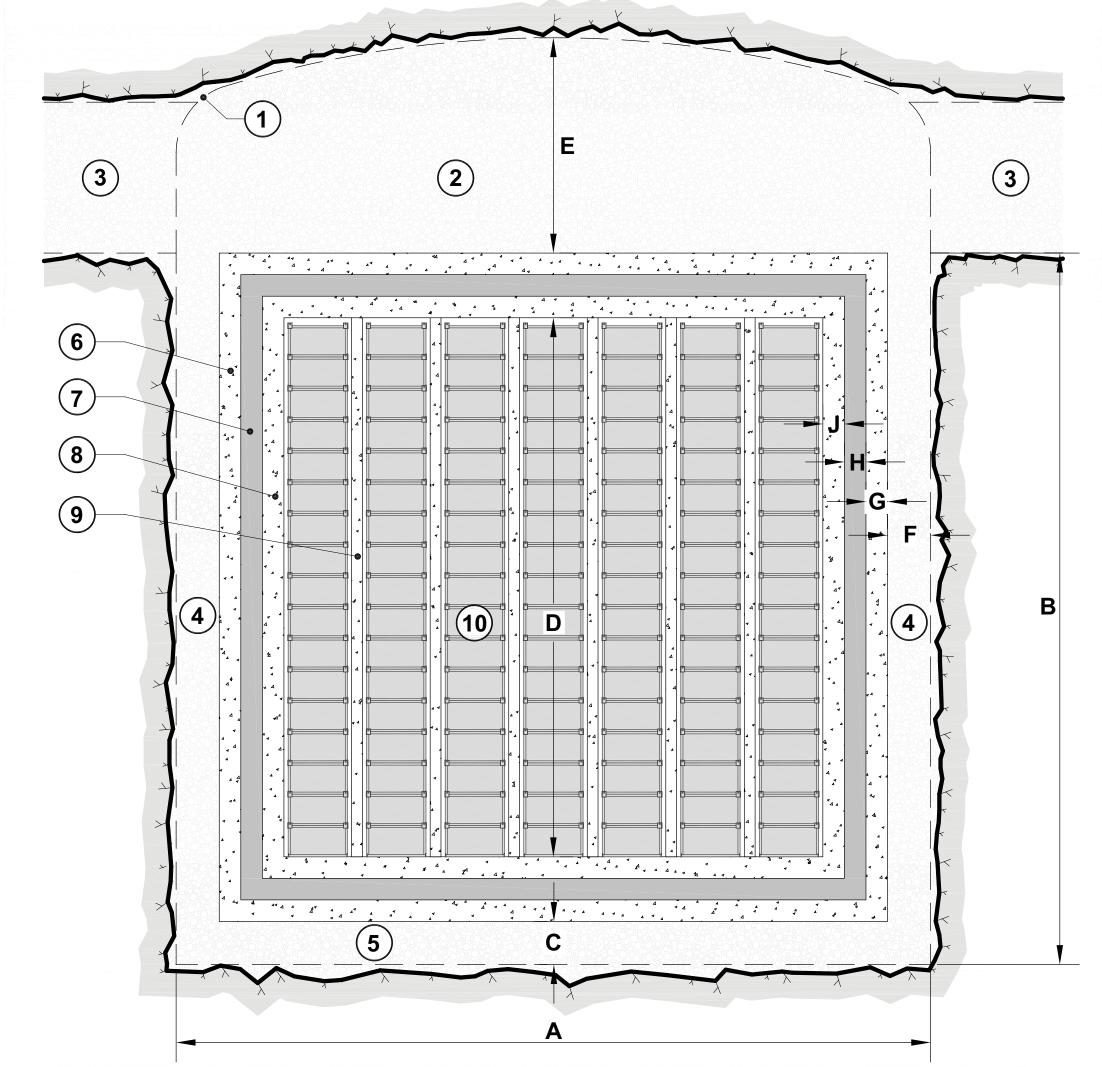


Figure 8-8. *Cutaway view of silo with barriers of concrete, bentonite and crushed rock. 1) Theoretical rock cavern contour. 2, 3, 4, 5) Crushed rock. 6) Reinforced concrete. 7) Bentonite blocks. 8) Concrete. 9) Concrete shaft walls (0.5 m). 10) Waste packages. Dimensions: A = 35 m, B = 33 m, C = 2 m, D = 25 m, E = 5–10 m, F = 2 m, G = 1 m, H = 1 m, J = 1 m.*

The geometric configuration is a double-walled cylinder of concrete placed in crystalline rock. The space between the bentonite walls is filled with bentonite along the lateral surface of the cylinder and a mixture of sand and bentonite along its base surface. The space between the outer concrete cylinder and the rock is filled with crushed rock. The thickness of the concrete walls is one metre, and the bentonite layer is also approximately one metre. At closure, a lid of concrete/bentonite/concrete is placed on the silo and the space above the silo is filled with crushed rock. Figure 8-8 shows a cutaway view of a silo with barriers of concrete, bentonite and crushed rock.

The interior of the silo is divided into separate shafts in the same way as today's silo in SFR. The size of each shaft is about 2.8×2.8 metres to allow some space between waste packages and wall. The thickness of the inner concrete walls is about 0.5 metre. An overhead crane is used for waste deposition, just as in the silo in SFR. Waterproofing membrane is installed to divert the drainage water and protect the concrete structures during operation.

Safety functions and release-limiting factors are:

- Limited flow of groundwater through the waste, since the flow is channelled through a hydraulic cage.
- Limited flow of groundwater through the concrete and the bentonite.
- Limited diffusion through the concrete and the bentonite.
- Sorption of radionuclides.

Barriers in the repository are:

- Filling of voids: grout between waste packages and shaft walls, thickness about 0.1 metre.
- Inner concrete silo: reinforced or unreinforced concrete, thickness about one metre.
- Bentonite layer: bentonite, thickness about one metre.
- Outer concrete silo: reinforced concrete, thickness about one metre.
- Fill around outer concrete silo: crushed rock, thickness about two metres.
- Beneath outer concrete silo: drained layer of crushed rock, thickness about two metres.
- Backfilling of top part: crushed rock, thickness 5−10 metres.

During the water saturation phase, after closure, the unilateral hydrostatic pressure in the rock at 300–500 metres depth will be 3–5 MPa. The inner silo is then expected to be supported by the deposited waste and its embedment. How the outer concrete silo is affected by the unilateral hydrostatic water pressure as well as by the swelling pressure exerted by the bentonite placed between the concrete cylinders needs to be studied.

Technical solutions may need to be developed to handle gas transport through the barriers.

Programme

After the concept study is presented at the end of 2013, the technology development work will focus on the solution or solutions to be assessed in the safety evaluation.

The safety evaluation of concepts presented in 2016 will define requirements on the properties of the engineered barriers. Based on these requirements, a more detailed description of the design can be prepared.

8.2.2 Closure

The technical questions surrounding plugs and backfilling are expected to be similar for SFL as for SFR. The programme for further technology development of plugs and backfilling of rock vaults described in connection with the extension of SFR (and in Section 8.1.7 "Closure") therefore fills the needs that have been identified today for SFL.

8.2.3 Waste containers

In designing a handling system for waste, all parts in the handling chain from the waste producers to the final repository need to be taken into consideration to find as favourable a solution as possible overall. An important component for developing the handling chain for the waste to be deposited in SFL is identifying suitable waste containers for the final repository. The development of waste containers also supports the work of developing waste acceptance criteria, the layout of the repository parts and the handling equipment.

A study has been conducted to develop a series of waste containers for SFL waste on the conceptual level. The SFL waste exists in many different forms and packagings, both as conditioned waste and as interim-stored retrievable waste. The purpose of the study has been to develop waste containers with dimensions that comprise a modular system for transport and deposition in SFL, based on the existing waste and its packagings. The goal is to develop a series of waste containers that permits rational handling at all stages by means of a common handling system consisting of e.g. lifting tool and transport containers. The results of the study will be presented in a report that will be published in 2013 and is described in general terms in the following section.

Conceptual designs for the following five waste containers are presented:

- Waste container for four standard moulds.
- Waste container for 16 drums of 200 litres each on drum trays.
- Waste container for 16 drums placed in 280-litre protective drums.
- Shielded waste container for core components.
- Long-term durable shielded waste container for core components.

Dimensions, volumes and weights of all waste containers are shown in Table 8-1.

Premises

The general premises for the study, aside from the waste, are as follows:

- It has been assumed that the waste containers for SFL shall, if possible, have the same outside dimensions (length×width) in order to facilitate handling and transport by ATBs (waste transport containers). The gross shipping weight (ATB including load) is limited to 120 tonnes.
- All waste containers must be stackable.
- The waste in all waste containers shall be embedded in concrete.
- After deposition in SFL, all waste packages must be able to be embedded in concrete if the long-term safety of the repository so requires.
- Lifting devices and lifting tools must be designed with overstrength or redundancy.
- The shielded waste container for core components and the long-term durable shielded waste container shall be made of steel plate with a thickness of 100 millimetres.
- The shielded waste container for core components shall be provided with a screwed-on lid.
- The long-term durable waste container shall be fabricated with full-penetration welds. This also applies to the lid welds.

Conceptual design

A basic premise for the study is that all waste containers shall have the same outside dimensions as regards length and width and that it shall be possible to handle the waste containers with the same lifting tool.

Since the same lifting tool should be able to be used for all waste containers, a new type of lifting lug located at the corners of the waste containers has been developed. The lifting lug consists of a bent plate (see Figure 8-9), which also serves as a guide for stacking the waste containers and for the lifting tool. The lifting lugs, which have the same design on all waste containers, are designed to lift the heaviest one. The design-basis assumption is that only two (diagonally opposed) of the four lifting lugs bear the load (overstrength factor).

Figure 8-10 illustrates the principle of the proposed lifting tool. The lifting tool consists of a frame fitted with four horizontal pins located in the corners of the frame. The pins are hydraulically operated.

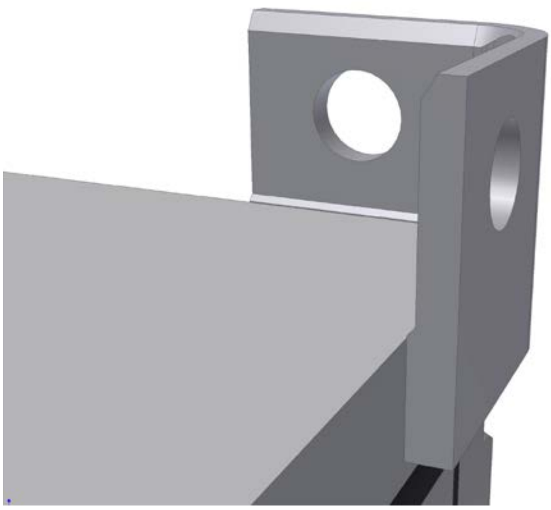


Figure 8-9. *Lifting lug for waste container for SFL.*

Figure 8-10. Lifting tool.

Waste container for moulds

The waste container for moulds shall be able to be loaded with four standard moulds, see Figure 8-11. To facilitate grouting, the following assumptions have been made:

- Distance between moulds: 50 millimetres.
- Distance between mould and inside of waste container: 20 millimetres.
- Distance between mould and top edge of waste container: 80 millimetres.

The waste container consists of a welded framework of square tubing. The sides consist of corrugated plates that are welded to the framework. The plates are corrugated to prevent deformation during grouting with concrete. The bottom plate consists of a smooth plate with stiffeners positioned like a cross inside the waste container, see Figure 8-12. The stiffeners also act as guides for the moulds when they are placed in the waste container. The bottom plate lies flush against the bottom edge of the framework so that air pockets will not form on the underside of the waste container during stacking and subsequent grouting with concrete. Guides for the moulds are also located along the outer sides of the waste container.

Figure 8-11. Waste container for four moulds, loaded. Reinforcement for concrete grouting visible.

Figure 8-12. *Wate container for four moulds, empty. The stiffeners for the bottom plate and the guides for the moulds are visible.*

The waste container will not be provided with a lid. The concrete surface after grouting of the load will therefore be in level with the top edge of the framework. The concrete surface above the moulds can be provided with reinforcement to reduce fracturing during handling of the waste containers.

Waste container for drums

The waste container for drums shall be able to be loaded with four drum trays, each with four 200-litre drums, see Figure 8-13 and Figure 8-14. The design of the waste container is basically the same as that of the waste container for moulds. The difference is the height, which is adjusted to the height of the largest drum, and the fact that the guides are adjusted to the slightly larger dimensions of the drum trays. The nominal spacing between drum trays for grouting is 40 millimetres.

Waste container for drums in protective drums

A large number of 200-litre drums from SVAFO have been placed in 280-litre protective drums. The waste container shall be able to be loaded with 16 drums placed in protective drums, see Figure 8-15.

The design of the waste container is basically the same as that of the waste container for drums. The difference is the height and the inside dimensions of the waste container, which are adjusted to the dimensions of the protective drums.

Figure 8-13. *Waste container for four drum trays (16 drums), loaded. Reinforcement for concrete grouting visible.*

Figure 8-14. Waste container for four drum trays (16 drums), empty. The stiffeners for the bottom plate and the guides for the drum trays are visible.

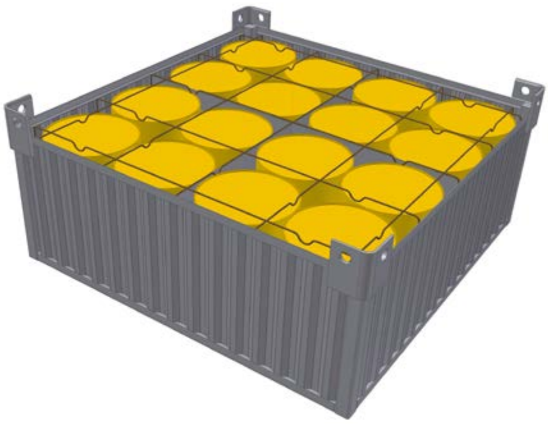


Figure 8-15. Waste container for drums placed in protective drums, loaded. Reinforcement for concrete grouting visible.

Shielded waste container for core components

In terms of radiation shielding, the waste container for core components has the same function in terms of radiological safety as the existing steel tank (also called BFA tank), but its dimensions are adjusted to other waste containers described in this section. It does not have a inner canister for the waste.

The shielded waste container for core components (see Figure 8-16) has the same outside dimensions as the waste container for moulds. The waste container has sides made of 100-millimetre-thick steel plate and is provided with a lid that is screwed on with screws. The in-plane inside dimensions of the waste container (length×width) are the same as for the waste container for moulds. The inside height is smaller due to the thicker bottom and the lid.

The waste container's side plates and bottom are joined by welding. Based on experience from similar structures (for example the steel tanks that are used today for interim storage of core components), it is assumed that the welds will be made by conventional welding technology, with a weld depth of 10−15 millimetres. Owing to the thickness of the material, an elevated working temperature is required (100–200°C), with subsequent stress-relief annealing (550−650°C).

After welding, the contact surface for the lid is face-milled. No seal will be used between the lid and the waste container; the sealing surfaces are steel on steel. Waste placed in the waste container is embedded in concrete up to the level of the bottom edge of the lid. After embedment, the lid is fitted to the waste container with screws. The waste container is provided with guide studs for the lid, and the lifting lugs provide rough guidance. The lid is handled with lifting eyes that are removed after fitting.

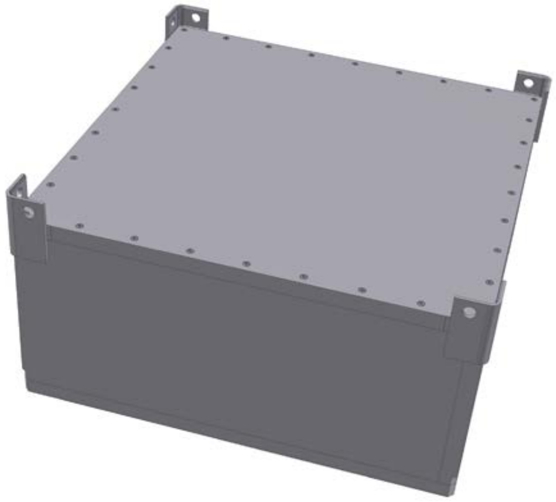


Figure 8-16. Shielded waste container for core components with lid fitted.

Long-term durable shielded waste container for core components

A long-term durable shielded waste container for core components has the same function in terms of radiological safety as the existing steel tank (also called BFA tank) and the shielded waste container for core components described above, but is fabricated in such a way that it can be credited with a long-term containment function in the final repository. The welds joining the waste container's plates are critical and must meet the same requirements on corrosion resistance as the plates themselves. The waste container is presented as a possible alternative for handling core components in the event a containment function is striven for in the waste container.

The long-term durable shielded waste container (see Figure 8-17) shall have the same outside dimensions as the waste container for moulds. The waste container has sides made of 100-millimetre-thick steel plate and is provided with a lid that is welded to the waste container. All welds shall be full-penetration welds, and the goal is that the weld should have the same corrosion properties as the parent metal.

For accessibility during welding of the lid, the waste container's lifting lugs are attached to the lid. There are lifting lugs on the inside of the waste container for handling the waste container before the lid is fitted, see Figure 8-18. The lifting lugs shall be designed to handle a loaded waste container. A lifting tool with "swivel clubs" is used for handling. This lifting tool is designed as an add-on module to the regular lifting tool (which is shown in Figure 8-10).

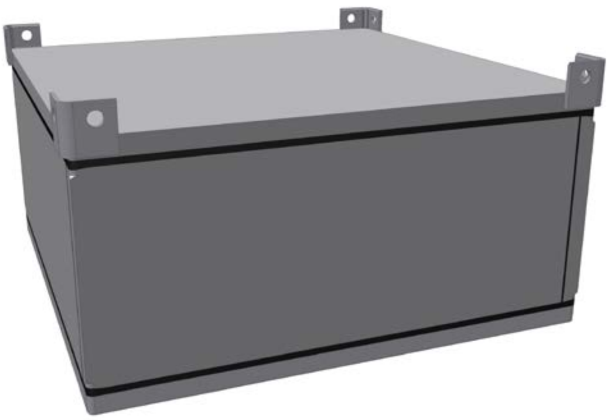


Figure 8-17. Long-term durable shielded waste container for core components with lid fitted.

Figure 8-18. *Lifting lugs inside long-term durable shielded waste container for core components.*

Information has been obtained regarding the geometric limitations of different types of welding equipment for joining the waste container's sides and bottom and for fitting the lid. It should be possible to join the sides of the waste container by means of submerged arc welding, which must then be done in a horizontal position. The use of narrow gap welding reduces the required welding volume and thereby any welding residual stresses. Welding of lid and bottom can be done with a traditional welding profile, for example a J-joint in accordance with SS-EN ISO 9692-1:2004 (SIS 2004), where robotized MIG (Metal Inert Gas) welding can be used.

Due to the metal thickness, all welding work must be done at an elevated working temperature (100−200°C) to reduce the risk of hydrogen embrittlement and normally also be followed by stress-relief annealing (550−650°C). Stress-relief annealing will not likely be possible after the lid is welded on, since the waste container is full and the waste is embedded in concrete. One option is to maintain the elevated working temperature for a while after welding to obtain a more favourable cooling process and reduce the risk of phase transformations in the material. The judgement is that omitting stress-relief annealing would not affect the corrosion properties of the material. After welding, nondestructive inspection can be performed to verify the quality of the weld. It should be pointed out that the grout must be fully hardened before the lid can be fitted and welded on.

Stacking

Stacking of the waste containers is facilitated by the fact that the lifting lugs located in the corners of the waste containers also serve as guides. With regard to the number of waste containers that can be stacked on top of each other, scoping calculations have been carried out based on the assumption that all waste containers will be filled with concrete and that they will be embedded in concrete at deposition.

In the case of the waste containers for moulds and drums, which are design-basis factors when stacked, it is assumed that the load is only absorbed by the surface area of the embedding concrete inside the waste containers. The load-bearing capacity of the waste container itself is not taken into consideration. Conservative scoping calculations show that at least 20 waste containers can be stacked on top of each other without any grouting outside the waste containers, and that at least 40 waste containers can be stacked on top of each other if grouting is done outside the waste containers. More detailed calculations that also take the load-bearing capacity of the waste containers into account should show that even more waste containers can be stacked. Stacking in connection with interim storage is thus not limited by strength-related aspects, but by stability- and handling-related aspects.

Dimensions and data

Table 8-1 presents dimensions, volumes and weights for the five different waste containers that have been discussed.

Table 8-1. Compilation of dimensions and data for waste containers.

		Unit	Waste container for moulds	Waste container for drums	Waste container for drums in protective drums	Shielded waste container for core components	Long-term durable shielded waste container for core components
Outside dim.	(LxW)	mm	2,690×2,690	2,690×2,690	2,690×2,690	2,690×2,690	2,690×2,690
Outside dim.	(H)	mm	1,296	980	1,050	1,296	1,296
Inside dim.	(LxW)	mm	2,490×2,490	2,490×2,490	2,630×2,630	2,490×2,490	2,490×2,490
Inside dim.	(H)	mm	1,284	968	1,038	1,096	1,096
Volume	(outside)	m^3	9.38	7.09	7.6	9.38	9.38
Volume	(inside)	m^3	8.6	6.48	7.34	6.8	6.8
Volume	(load)	m^3	6.91	3.33	3.33	2.55	2.55
Volume	(concrete)	m^3	1.69	3.16	4.02	4.25	4.25
Weight	(empty)	kg	1,755	1,602	1,480	20,300	20,300
Weight	(load)	kg	20,000	8,000	8,400	20,000	20,000
Weight	(concrete)	kg	3,377	6,312	8,031	8,495	8,495
Weight	(total)	kg	25,200	16,000	18,000	48,800	48,800

Programme

The further development of long-term durable waste containers includes trial fabrication. The applicability and practical feasibility of the joining method will be examined. The properties of the joint will be characterized and different quality measures of these properties will be evaluated. The purpose of the programme is to learn more about the joining of thick-walled materials for final disposal applications in general. The findings are thus applicable not only to the long-term durable waste container presented above, but also to the existing steel tank for interim storage of core components. This work will be started during the current three-year period.

The evaluation of long-term safety that will be presented in 2016 will stipulate requirements on the waste and waste containers, for example on mechanical properties. Based on these requirements, waste containers can be further developed and finally adopted as preliminary final disposal packages for SFL. Development of lifting tools is determined by the choice of waste containers.

In order to transport long-lived waste in containers described in this section, new transport containers need to be developed and licensed. Needs and requirements on these transport containers need to be studied, which will begin after preliminary final disposal packages have been adopted for SFL.

8.2.4 Handling equipment for core components

FKA and SKB have jointly developed handling equipment to segment and lift out core components and reactor internals from reactor pools to steel tanks. The steel tank's storage canister is lowered into the pool, where it is filled with segmented core components and reactor internals. The storage canister is then lifted under a radiation-shielding hood to a steel tank in the reactor hall. The tank and the waste are dried by means of a vacuum pump system before the lid is fitted and the tank is transported out in a transport container.

The handling equipment for core components is in operation and has been used by FKA to lift out segmented core components and RPV internals to steel tanks. The core components stem from power increases in all three units during the 2000s, and the equipment is planned to be used by BKAB to lift out segmented core components during dismantling and demolition of the reactors in Barsebäck.

8.2.5 Segmentation of control rods

SKB has conducted a feasibility study to examine the possibilities of segmenting BWR control rods in Clab. Segmentation of control rods can be done for the purpose of obtaining more compact wet interim storage in Clab's pools, or more compact dry interim storage on another site. Control rods of various models have been brought to Clab over a long period of time. For the purposes of the study, they can be subdivided into control rods from ABB Atom/Westinghouse and control rods from General Electric, based on their design.

The control rod handles are generally cut off at the nuclear power plants for transport to Clab. The remaining part is cruciform in shape with a length of 4.2 metres and a width of 0.3 metres.

Studied cutting sequence:

- Step 1: The lower part of the handle/stem is cut off with a horizontal cut in the lower part of the control rod blade. The cut-off stem is grasped by a lifting tool and carried to a suitable scrap storage canister.
- Step 2: The cruciform control rod is cut with a band saw into four separate blades. This is planned to be done by a vertical cut in the centre of the cruciform control rod. The separate blades are grasped and handled by a lifting tool and placed either in a transloading position or directly in a storage canister for continued storage in Clab.

The storage canister is designed for normal handling in Clab and is envisioned as having internal dimensions of 760×760 millimetres and a height that accommodates the full length of the blades, 4.2 metres. A theoretical packing degree of 500 blades, i.e. 125 control rods, per storage canister can then be achieved. A more reasonable estimate, based on previous experience and weight limits for handling equipment, would be around 90 control rods per storage canister.

To permit interim storage in e.g. a steel tank, the length of the blades needs to be reduced. This can be done by adding a third segmentation step to the sequence described above:

- Step 3: Cutting of the blades to suitable length.

Since segmentation may be done in a pool where other equipment is also present, some kind of shielding should be installed around the cutting area to prevent debris and chips from being spread over large areas and to other equipment in the pool. Such shielding is used as standard at the nuclear power plants when core components and reactor pressure vessel internals are segmented. Water purification in the shielded-off area is also expected to be possible by established methods.

Control rod blades from ABB Atom/Westinghouse consist primarily of 8−10-millimetre-thick plates in which holes have been drilled horizontally to contain boron carbide powder (in certain positions boron carbide rods) or hafnium. When they are cut, one or two drilled holes are punctured, which means that a limited amount of boron carbide can be released into the water. Tests show that there is no pressure in the control rods. The design of the control rod blades from General Electric is different in that it consists of vertical tubes filled with boron carbide powder that are encased in a stainless steel shell. When the blade is cut to a suitable length, all channels in the blade are thereby affected. A suitable method for cutting the control rod blades remains to be determined.

There is a risk that tritium will be released when the control rod blades are cut. Tritium is formed from the boron carbide in the control rod blades during operation. The studies that have been done show low releases of tritium, which can be dealt with by means of e.g. a hood to collect the gas.

9 Responsibility, planning and technology for decommissioning of nuclear facilities

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

Figure 9-1 shows an overview of SSM's requirements regarding decommissioning during the life of a nuclear power plant. Decommissioning includes defuelling, shutdown operation and dismantling and demolition. Defuelling is the activity from final shutdown of the nuclear power reactor until all fuel has been removed from the plant. In cases where dismantling and demolition will not be commenced immediately after defuelling operation, a period of shutdown operation begins. Activities required for cleared of the plant from regulatory control are carried out during dismantling and demolition. After SSM has approved an application for release of the plant, it can be under the Nuclear Activities Act and the Radiation Protection Act.

According to the Environmental Code, an environmental impact statement must be submitted both before final shutdown of the facility and as a part of the application for a dismantling and demolition licence, see Figure 9-1.

To comply with Chapter 9 in SSMFS 2008:1, the following are required:

- Decommissioning plan and strategy.
- Measures in conjunction with final shutdown and shutdown operation.
- Measures in conjunction with dismantling and demolition.
- Documentation and decommissioning report.

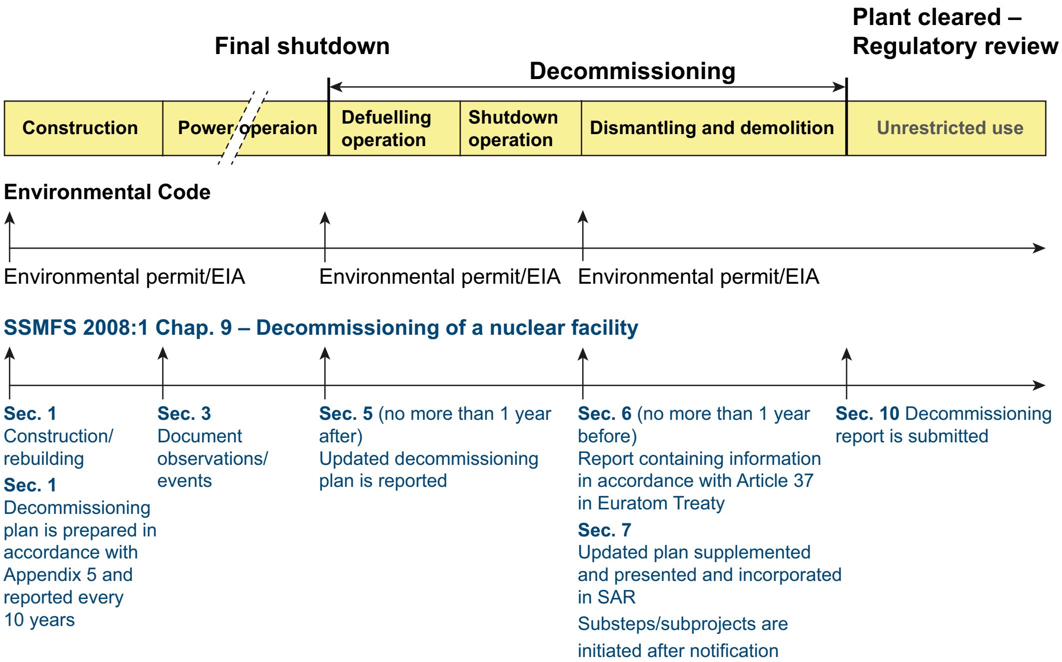


Figure 9-1. Overview of SSM's requirements regarding decommissioning during the life of a nuclear power plant.

Conclusions in RD&D 2010 and its review

It was pointed out in SSM's review of RD&D programme 2010 that decommissioning needs to be described more exhaustively and in greater detail. SKB's plans in relation to the activities of the nuclear power companies need to be clarified and SKB's flexibility in relation to changes in the plans for the nuclear power plants needs to be explained. Furthermore, the account in the RD&D programme needs to show that it is the licensees for the different plants who stand behind this account and clarify whether it is the account in the RD&D programme or the decommissioning plan that takes precedence in the event of discrepancies.

The Government decision regarding RD&D programme 2010 stated that the reactor owners shall present plans and strategies regarding decommissioning of the nuclear power plants (including Ågesta) more exhaustively and stipulate which tasks have been delegated from the reactor owners to SKB. The reactor owners shall consult continuously with SSM in matters concerning decommissioning plans and decommissioning studies.

Further, the Swedish National Council for Nuclear Waste said in its comments that it would like to have a clearer account of the role of the Decommissioning Group and clarification regarding the possibility of conditioning waste from dismantling and demolition for the purpose of reducing the waste volume and permitting recycling of materials.

Current situation

Decommissioning studies were completed in 2013 for the NPPs in Forsmark (SKB 2013c), Oskarshamn (SKB 2013d) and Ringhals (SKB 2013e), while the study for the NPP in Barsebäck (Griffiths et al. 2008) was finished earlier. The studies present estimates of waste quantities, timetables and costs for decommissioning of the plants in question. This serves as a basis for determining capacities in SKB's final repository system and fees to be allocated to the Nuclear Waste Fund. With this work, activity calculations have been carried out for all reactor units in order to create a waste inventory for dismantling and demolition. A comparative analysis has been done of these studies that presents and explains qualitative and quantitative differences (Paul 2013).

During 2012, Vattenfall and SVAFO completed a new study for the decommissioning of Ågesta combined heat and power plant (CHP). For further information see Section 9.3.6.

The clearance manual (SKB 2011d) was completed in 2011, and training regarding clearance in accordance with the manual is being provided in the industry. The manual has primarily been written from an operational perspective, and the subject needs to be further elucidated from a decommissioning perspective, see Section 9.3.8.

The OECD-NEA, in cooperation with the IAEA and the EU, has produced an updated cost structure for estimating the costs of decommissioning of nuclear facilities (OECD/NEA et al. 2012). The new structure is being used for cost estimates for the nuclear power plants in Forsmark, Oskarshamn, Ringhals, Barsebäck and Ågesta, as well as in SKB's Plan work.

9.1 Consultations with SSM

Since RD&D programme 2010, the licensees for the nuclear power plants (including Ågesta) have, together with SKB, held consultations with SSM on the subject of decommissioning. Matters mentioned in the conclusions in RD&D programme 2010 have been discussed, and SSM has been given general information on the account of decommissioning planned by SKB for RD&D programme 2013.

9.2 Division of responsibilities

The licensee of a plant is responsible for compliance with SSM's requirements in accordance with Figure 9-1.

According to the Nuclear Activities Act, the licensees for the nuclear facilities in Sweden are responsible for safely decommissioning the radioactive parts of the facilities (SFS 1984:3). 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 stipulates that a licensee shall calculate the estimated cost of decommissioning of nuclear power plants (SFS 2006:647).

The licensees – Barsebäck Kraft AB, Forsmarks Kraftgrupp AB, OKG Aktiebolag and Ringhals AB – are responsible for decommissioning 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 Nuclear 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 (SFS 1984:3).

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

SKB is tasked with assisting the licensees (RAB, OKG, FKA and BKAB) in fulfilling their obligations under Section 11 of the Nuclear Activities Act. The responsibility includes performing the necessary research and development activities required to fulfil the obligation to adopt all measures needed to safely decommission facilities when the nuclear activity will no longer be conducted.

As a part of this assignment, SKB shall participate in planning and executing the decommissioning of the NPPs and shall primarily coordinate the use of general methods and procedures for the decommissioning work including calculation of waste volumes, radionuclide inventory and costs.

Under today's legal requirements, it is the licensees who bear full responsibility for their decommissioning, while SKB is responsible for the above assignment, which is coordinated nationally.

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

SKB has not officially been contracted by the licensee of Ågesta for any assignment related to decommissioning.

9.3 Planning for decommissioning and development of methods

SKB's objective is to be able to manage the radioactive waste from dismantling and demolition of the nuclear facilities in accordance with the their decommissioning plans. The plans for the nuclear power plants and SKB's facilities are updated regularly in dialogue between the concerned parties. The planning work is an iterative process in which competence and understanding progressively increase and the decommissioning studies are updated and refined.

9.3.1 Decommissioning strategies

Three different strategies for decommissioning are often referred to in international contexts as Direct Dismantling, Safestore and Entomb. Decommissioning of the nuclear power plants in Sweden is for the most part planned to be done in accordance with the Direct Dismantling strategy. This entails that all radioactive components and buildings are decontaminated and/or dismantled shortly after shutdown and the waste is either transported and deposited in a final repository or is packaged and transported to an already established interim storage facility. This will not be completely applicable to the nuclear power plants in Barsebäck and Ågesta, since they have already been taken out of service and are in shutdown operation. Each licensee's decommissioning plan and strategy are described in greater detail in the following sections. Figure 9-2 shows a general reference timetable based on the licensees' planning.

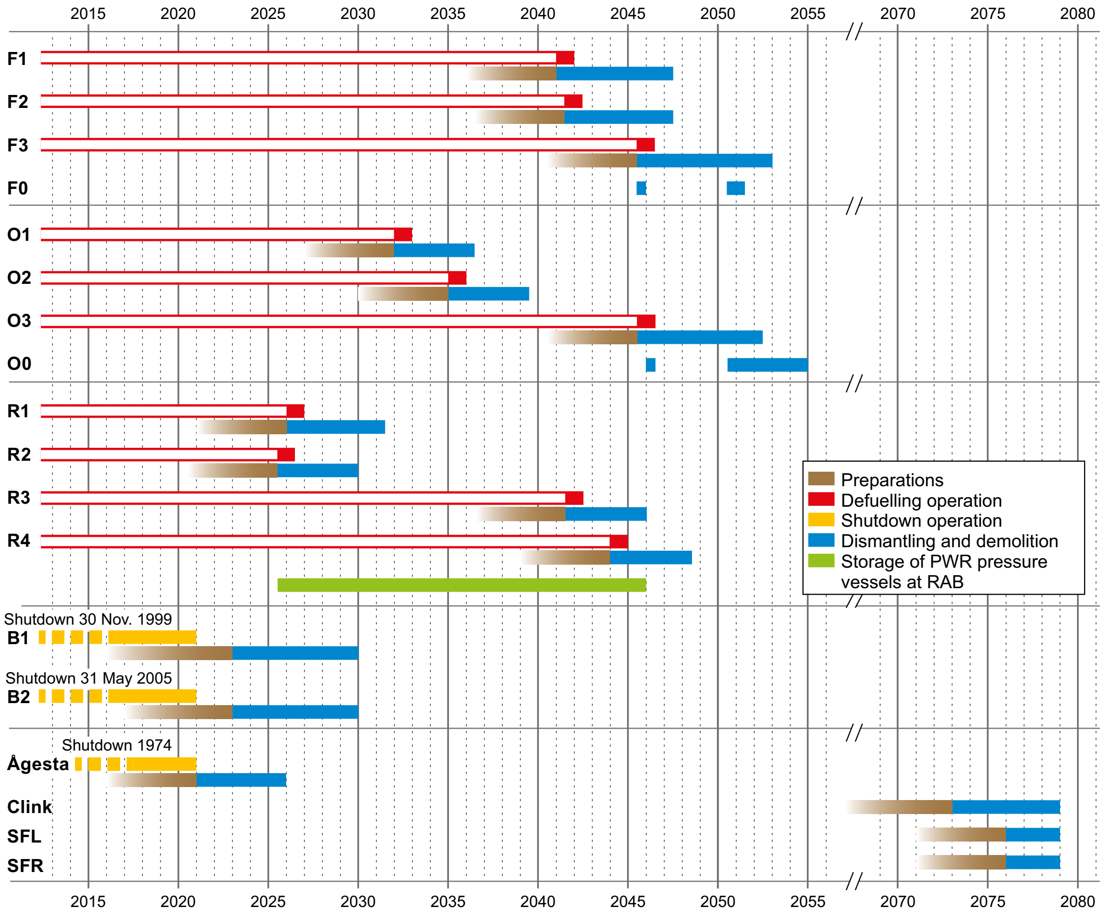


Figure 9-2. *Reference timetable for decommissioning of nuclear facilities (F0 and O0 are common facilities on the sites that are described separately).*

9.3.2 Barsebäck Kraft AB's planning for decommissioning and development of methods

E.ON Kärnkraft Sverige is the owner of the Barsebäck NPP, and Barsebäck Kraft AB is the holder of the nuclear licence.

According to BKAB's current planning, a final repository for the short-lived low- and intermediate-level waste from dismantling and demolition shall be ready to receive the waste, the preparations shall be completed and all licences shall have been obtained before dismantling and demolition are begun (Lorentz and Pålsson 2012). During 2013, BKAB is exploring the possibilities of dismantling and demolition without access to an extended SFR. One of the prerequisites in order for this to be optimized and carried out earlier is that an interim storage facility is available for the waste in 2018–2019. Further, dismantling and demolition of Barsebäck 1 (B1) and Barsebäck 2 (B2) is planned to be done as a common project, and the requirements are planned to be clarified before the project is started. The final goal of decommissioning is that the Barsebäck NPP is cleared in accordance with regulatory requirements.

BKAB's objective for decommissioning is that it should be safe, rapid and cost-effective. Potential risks must be eliminated or reduced. This is achieved by limiting the dose, which is mainly accomplished by carrying out system decontamination and dismantling large components. Furthermore, good logistics is facilitated mainly by detailed characterization, a clarified set of requirements, good planning, approved type descriptions for the different waste types and preparing SFR to receive low- and intermediate-level waste from dismantling and demolition. Cost-effectiveness is achieved by clear project management and an efficient timetable for dismantling and demolition to a cleared facility.

The time frame for a finished final repository with room for waste from dismantling and demolition will enable BKAB to conduct extensive feasibility studies. Furthermore, international experience feedback can be analyzed and drawn upon to achieve safer, faster and more cost-effective decommissioning.

To optimize handling and thereby minimize the dose, BKAB does not intend to treat the resulting radioactive waste to any great extent at the Barsebäck plant. However, low-level waste from dismantling and demolition may be sent off-site for melting and incineration. Otherwise the short-lived low- and intermediate-level waste will be packaged in the intended waste containers and sent directly to the final repository. Core components (long-lived waste) will be interim-stored until SFL is commissioned.

The total waste quantity, both radioactive and non-radioactive, has been estimated by means of an inventory aimed at quantifying volumes and masses. Dismantling, demolition, waste management and transport will be carried out as industrial processes with clear production lines. Low doses to personnel are prioritized, which means that radioactive parts will be dismantled and hauled away in large units. It further means that clearance measures such as decontamination will only be performed when a reasonable recycling value can be identified. This may lead to slightly larger volumes of radioactive waste, but is offset by lower radiation doses to personnel, a shorter timetable and a lower total use of resources. The plan includes removing the RPVs in one piece without internals, since they will be disposed of whole. A near-surface repository for very low-level waste will not be established at the Barsebäck plant.

The quantity of radioactive waste to be disposed of in SFR is estimated at about 18,000 tonnes. The total waste package transport need is estimated at two whole RPVs, 800 ISO containers, 300 steel moulds and 80 steel tanks. Shipments to SFR are expected to be evenly distributed over the years dismantling and demolition are in progress. The quantity of radioactive waste 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. All other material in controlled areas is judged to be clearable, and the volumes of this material have not been estimated yet.

Segmentation of reactor internals is one of the most time-consuming activities in a decommissioning project. BKAB has established a project that includes segmentation and packaging of vessel internals that are kept in the pools as well as those remaining in the vessels of B1 and B2. Storage will take place in a newly built storage facility for internals in a guarded area, similar to the one that has been built in Forsmark. Construction of this storage facility is planned to be started during the spring of 2014 and is estimated to take about seven months. Segmentation and transport of waste to the storage facility will begin during the spring of 2015 and be concluded in early 2017. Transport of the steel tanks to SFR will take place when the extended final repository is in operation.

BKAB has conducted a number of studies for decommissioning, the most important of which cover:

- Removal and dismantling of whole reactor pressure vessel.
- Segmentation of RPV and internals.
- Activity inventory during dismantling and demolition of Barsebäck 1 and 2.
- Survey and categorization of plant and environs with regard to radioactive contamination.
- Management of large components such as turbine, condenser, reheater and preheater.
- Demolition of reactor building.
- Management of reactor internals.

In addition, a 3D model has been created of the plant that includes land and buildings.

Ongoing studies are:

- Adoption of requirements for decommissioning.
- Demolition and waste logistics.
- Demolition of turbine building and other buildings.
- Decontamination of building structures.

BKAB's planned studies for the upcoming RD&D period are:

- Model for a clearance programme.
- Preparations for dismantling of service systems.
- Organization for dismantling and demolition.
- Decontamination of tubing and pressure vessels in the waste facility.
- Communication plan for stakeholders.

The work of compiling supporting material for an application for a regulator permit for dismantling and demolition will begin during the period.

BKAB is acquiring competence for dismantling and demolition by participating in the work of the following organizations in the decommissioning field:

- Liaison groups for decommissioning issues relating to nuclear facilities established on SKB's initiative. An example of this is SKB's Decommissioning Group.
- Standardization work in SIS/TK 405 Nuclear Energy and, by extension, the International Organization for Standardization, ISO.
- Collaboration that has been established with SVAFO.
- Experience exchange with E.ON and Vattenfall in Germany. The focus is on safety, decommissioning costs, decommissioning methods and clearance from regulatory control.
- In 2007, BKAB was admitted as a full member of the OECD-NEA's Technical Adviser Group, which meets twice a year and exchanges experience on different international decommissioning projects.
- Membership in the OECD-NEA's Working Party on Decommissioning and Dismantling (WPDD).
- Membership in the EPRI's Decommissioning Programme. The membership gives an opportunity to obtain support for minimizing risks and costs for decommissioning by improved planning and experience feedback from completed projects.

The overall planning is currently dominated by a number of feasibility studies that will be combined into one assignment for preliminary design. Preliminary design will include an in-depth analysis and evaluation of the results of the feasibility study. Examples of the content of the final report from preliminary design are schematic solutions, procurement procedures, economic calculations based on budget quotes, safety and risk analyses, environmental aspects, waste plans, dose budgets and timetables for decommissioning.

The results from preliminary design will provide a basis for the design work prior to the start of dismantling and demolition. This work includes developing a project organization, making supplier assessments, issuing invitations to tender, procurement of contracts, quality assurance, performance of risk analyses and licensing matters.

An established set of requirements for free release of the Barsebäck NPP is a prerequisite for being able to execute dismantling and demolition in as safe, environmentally sound and efficient a manner as possible so that the nuclear license for the plant ceases to apply. A dialogue between SSM and BKAB should take place in advance in order to establish the requirements. SSM's set of requirements and regulatory framework for release of the Barsebäck NPP from regulatory control has not yet been fully established. A clearance regulation for materials, buildings and land has been produced by SSM. The regulation does not specify any clearance levels for land; this is to be determined by SSM in each case. The following shall be established in a dialogue between BKAB, other licensees and SSM:

- The radiological criteria for clearance of land.
- The monitoring programme for clearance of the plant.
- The clearance process associated with decommissioning and dismantling of a nuclear facility.

9.3.3 Forsmark Kraftgrupp AB's planning for decommissioning and development of methods

FKA's decommissioning plan includes all three reactor units in Forsmark and defines the general premises for future decommissioning (FKA 2013). With the given general premises and as a complement to the decommissioning plan, SKB and FKA have, with the help of Westinghouse, conducted an integrated study for decommissioning of the entire plant (SKB 2013c). The purpose of the study is to determine waste and activity quantities, timetable and organization. On this basis, the costs of dismantling and demolition have then been calculated.

A fundamental planning premise is that the time from cessation of electricity production until the site is cleared shall be minimized as far as possible from the viewpoints of safety, feasibility and cost. Feasibility studies and detailed design for dismantling and demolition must then be carried out in good time before electricity production ceases.

After final shutdown, the reactor unit is being defuelled for approximately one year before dismantling and demolition begins. During this period, measures are adopted to reduce the quantity of activity in the plant, for example by removal of nuclear fuel, segmentation of reactor internals and system decontaminations. This is done for the purpose of being able to begin dismantling and demolition immediately after defuelling in order to achieve a time-optimized decommissioning.

Other fundamental planning premises are that near-surface repositories are available for deposition of very low-level waste from dismantling and demolition and that the RPVs can be transported and disposed of in one piece.

The goal of decommissioning is to remove the radioactive material and clear the plant from regulatory control. This is considered to have been achieved when there are no longer any restrictions from SSM on other use of land or buildings. The ultimate goal of decommissioning is that the site can be used for other industrial activities.

FKA's planning premise is that the reactors will be operated for 60 years, which means that Forsmark 1 will be operated until 2040, Forsmark 2 until 2041 and Forsmark 3 until 2045. FKA's planning for future decommissioning is to continue its collaboration with other licensees and SKB. FKA is participating in SKB's Decommissioning Group and Vattenfall's strategic cooperation forum in the field of decommissioning.

Advice and directives from SKB are also applied as a basis for the ongoing planning work within FKA. The report *Avveckling och rivning av kärnkraftblock* ("Decommissioning and dismantling of nuclear power units") (SKBdoc 1359832) deserves special mention as the document that embodies a consensus and guide for FKA, OKG and RAB with regard to the further planning of technology, methods and organization.

FKA's ambition and goal is that decommissioning should be safe, reliable and cost-effective. To achieve this goal, the planning is currently being focused on three areas:

- Maintain and develop competence by ensuring a high level of in-house expertise. Key competencies shall exist in-house, and generational changes shall take place without loss of important experience.
- Decommissioning planning that is revised regularly and kept up-to-date on the basis of new knowledge and new experience for the purpose of having an optimal and cost-effective decommissioning plan at any given time.
- Build confidence internally and externally by communicating relevant information regarding ongoing and planned work on decommissioning, dismantling and demolition of the reactor units. This is done methodically and systematically, both internally within FKA and externally.

9.3.4 OKG Aktiebolag's planning for decommissioning and development of methods

In conjunction with the implementation of power increases, OKG has developed and verified its methodology for managing large components. For example, reactor internals have been segmented, packaged and transported to OKG's rock cavern for waste (BFA) for interim storage.

Moreover, large components such as the turbine from O3 have been removed, packaged, transported, melted and cleared. Logistics and planning of the whole flow has been verified. This will continue to be developed in ongoing power increase projects.

The Oskarshamn plant has continued its development of primary circuit system decontamination and has collaborated with the nuclear power plants in Barsebäck and Forsmark on this. The industry has thereby accumulated a large body of experience from decontamination of ASEA BWRs. The decontamination factors that have been obtained in large system decontaminations have been considerably improved over the years and resulted in a reduction of activity by 95–99 percent.

OKG's planned time for all reactor units is 60 years (OKG 2013). This means that the final shutdown of O1 will be in 2032, O2 in 2034 and O3 in 2045. Decommissioning will proceed over a 20-year period.

Studies and experience will serve as a basis for the feasibility study and detailed design work that is done prior to decommissioning before electricity production ceases. Procurement of suppliers will be carried out in good time before decommissioning.

The time from the cessation of electricity production until the site is cleared will be minimized by good advance planning and establishment of waste streams to final repositories.

Already during defuelling for a reactor unit, measures will be adopted to reduce the quantity of activity and thereby reduce the requirement level in the plant. This will be done so that safety and operation for the units that are in service will not be jeopardized. Examples of such activities are removal of nuclear fuel, segmentation and removal of reactor internals and system decontamination. This means that certain dismantling and demolition operations can be started once the unit is shut down.

Defuelling is estimated to last approximately one year. No duration is planned for shutdown operation or re-establishment. This is a consequence of the fact that initial planning and design prior to decommissioning makes it possible to start dismantling and demolition simultaneously with measures during shutdown. OKG assumes that near-surface repositories equivalent to the already existing ones will be available for deposition of waste from dismantling and demolition. Reactor pressure vessels are planned to be removed intact from the plant, without internals, to be transported and disposed of as large, whole components. The ultimate goal for OKG's plant is that the site can be cleared and used for other industrial activities.

OKG's planning for future decommissioning involves continuing its collaboration with other licensees and SKB to discuss decommissioning issues. This collaboration takes place with SKB as the primary convener. Cooperation with E.ON for exchange of experience and expertise in the area is another important activity. For example, a competence matrix is being prepared showing what competencies E.ON, BKAB and OKG with subcontractors (if any) possess in the area. OKG will also follow the decommissioning of the Barsebäck NPP in order to benefit from the lessons and experience it provides, particularly in preparation for the decommissioning of the O2 reactor.

In parallel with the site-specific decommissioning study (SKB 2013d), the Oskarshamn NPP's decommissioning plan has also been updated during 2013 (OKG 2013). The decommissioning plan will be revised and refined regularly as the time for shutdown and subsequent decommissioning approaches.

OKG will carry out system decontaminations in its units and attend system decontaminations at the reactors in Ringhals and the pools in Clab during the coming RD&D period. Experience from these activities will be put to use in the decommissioning planning and contribute to maintaining and developing competence in the area.

9.3.5 Ringhals AB's planning for decommissioning and development of methods

RAB is currently planning for safe operation of the reactors R1/R2 for 50 years and R3/R4 for 60 years, which means continued operation of all reactors for another 13–30 years (R2 will be shut down in 2025, R1 in 2026, R3 in 2041 and R4 in 2043). The goal of the subsequent decommissioning is to remove radioactive material and restore the site so that it can be used for other industrial activities.

RAB's strategy is that decommissioning should be safe, optimized and cost-effective. To achieve these goals, the decommissioning planning is currently being focused on a number of strategic areas. These areas include ensuring in-house competence key competencies in the decommissioning field are available internally and generational changes can take place without the loss of important experience and expertise. In addition, the decommissioning plan is regularly revised and optimized on the basis of new knowledge and experience so that the decommissioning plan is optimal at any given time. In addition, open and relevant information is provided in a targeted and structured manner on current and future work in decommissioning planning to build confidence both internally and externally.

Crucial prerequisites for successful decommissioning have been identified. They are discussed in the following three paragraphs and embrace the categories documentation, waste and activities. RAB is currently working resolutely to ensure that these prerequisites are fulfilled in a reasonable time before the decommissioning of the first unit is commenced.

A 3D model of the end-state of each reactor unit and of the NPP as a whole will be available at the time of final shutdown. Operating data, historical events and operational experience relevant to decommissioning will be documented. A detailed radiological inventory will be compiled for each plant at final shutdown. This inventory will also be integrated in the 3D model, permitting effective dose optimization.

Cavities will be filled with conventional waste from dismantling and demolition up to one metre below the ground surface. A near-surface repository will be available for deposition of very low-level waste from dismantling and demolition. SFR has been designed with capacity to receive all short-lived waste from dismantling and demolition. Clearance will take place according to procedures suitable for the volumes, activity levels and material flows expected from dismantling and demolition. Interim storage of waste packages may occur as needed. All RPVs are planned to be disposed of in one piece, the BWR pressure vessel in SFR and the PWR pressure vessels in SFL. RAB's plan is that it will be possible to dispose of all PWR internals in the RPV. When nuclear activities have finally ceased on the site, remaining buildings and land shall be able to be used for other industrial activities. Decisions will be made in such good time before shutdown that detailed decommissioning planning can take place during the remaining operating period up to final shutdown. All fuel will be removed from the plant within one year after final shutdown. Decommissioning of each reactor will be done within projects and managed by RAB. RAB's goal is to gradually acquire the necessary competence in the decommissioning field and to be a competent outsourcer of the decommissioning projects. The projects will be well-defined and well-planned by the client (RAB) at the start. RAB's main responsibility as client is to provide the infrastructure for the projects. Experts with knowledge of decommissioning and its planning as a whole will be on hand at RAB. Work within the projects will mainly be executed by outside contractors. Radiation protection and physical protection will be adapted to the prevailing risk level during the different phases of decommissioning.

During the coming RD&D period, the focus of RAB's development work in the decommissioning field will be on the following areas:

- Competence assurance and acquisition – RAB has begun competence acquisition in the field by establishing a Decommissioning Planning Group. The group's goal is to make use of relevant experience from RAB's own operating personnel as well as experience from national and international decommissioning projects. The goal is to acquire competence so that RAB becomes a competent outsourcer of decommissioning projects.
- Plant inventory – an inventory of components, buildings and systems will be done with the goal of ending up with a body of data with fewer uncertainties as a basis for the next determination of waste volumes from dismantling and demolition.
- Radiological inventory at dismantling and demolition – RAB intends to contribute to competence acquisition in the areas of nuclide inventory modelling, source term determination and determination of nuclide vectors. Further, RAB will ensure that a gradually improved radioactivity characterisation and a calculation model for RAB's plants are available.
- Clearance – the requirements for clearance of decommissioning waste will be key parameters in determining how decommissioning is carried out. Reasonability, BAT and ALARA are assumed to be crucial parameters in developing these requirements. A body of practice will be accumulated regarding clearance measurements during decommissioning.

- Waste type descriptions – Ways to convert existing waste type descriptions so that they also include waste from dismantling and demolition will be investigated. RAB will support SKB in this work.
- Deposition of RPVs in one piece with internals – It is assumed that BWR pressure vessels and PWR pressure vessels will be deposited intact. Approximate analyses of the consequences of depositing the RPVs with internals inside the vessel will be carried out.
- Shallow land disposal of waste from dismantling and demolition – RAB's planning premise is that a near-surface repository will be able to receive all waste from dismantling and demolition that falls within the category of very low-level waste and that otherwise meets today's requirements on shallow land disposal. The repository is currently planned to be located adjacent to RAB's plants, but it is assumed that a national near-surface repository for waste from dismantling and demolition will be able to replace the local one without otherwise affecting the decommissioning strategy for this waste.
- Interim storage facilities may be able to contribute to an optimal decommissioning strategy by allowing the waste to decay for some time. This is dependent on the degree of contamination and radionuclide composition of the system and building surfaces after decontamination and additional waste treatment. In the years to come, RAB will therefore review the premises for interim storage of a) intermediate-level waste that is close to the limit value for reclassification to low-level waste, and b) low-level waste that is close to the limit value for shallow land disposal.

9.3.6 Ågesta's planning for decommissioning and development of methods

Vattenfall AB has contracted SVAFO to oversee ongoing shutdown operation and decommissioning planning for the Ågesta NPP. All fuel and heavy water have been removed from the facility. Two of four steam generators were previously removed and treated in the melting plant at Studsvik, after which the ingots were cleared. Most of the remaining radiological activity is in the RPV and its internals. Otherwise, most of the waste from dismantling and demolition comes from the two steam generators, the controllers and the primary systems. The diesel generator and batteries have also been removed. Otherwise, most systems and equipment are intact, for example the rock drainage system is still in operation. Inspection and service are carried out in accordance with issued radiation protection conditions, and some equipment, such as lifting devices, is maintained to facilitate future dismantling and demolition of the plant.

Background data in the form of expected waste amounts and radiological activity from the decommissioning of Ågesta have been collected for the SFR Extension Project. The data were submitted to SKB at the end of 2010.

An application for clearing most of the buildings and areas outside the reactor containment in Ågesta was submitted to SSM in 2010.

During 2011–2012, Vattenfall had a new study done for decommissioning of Ågesta (Lindow 2012). The study includes a general timetable and cost estimate for dismantling and demolition of all building parts in the rock cavern (including connecting tunnels) as well as the control room and switchgear building. The study for Ågesta was sent to SSM in the autumn of 2012.

Shutdown operation will continue until the start of dismantling and demolition. Dismantling and demolition is planned to start before the end of 2020. The decommissioning strategy for Ågesta was updated in 2012, and the decommissioning plan was updated in 2013 (Hedvall 2013). The decommissioning plan is expected to be updated periodically up to the start of dismantling and demolition, when a final decommissioning plan will be presented. The goal of dismantling and demolition of the Ågesta combined heat and power plant is a cleared plant (including the land and the rock cavern), which means that the radioactive waste has been removed.

Earlier decommissioning of Ågesta is probably possible, but would entail that the waste from dismantling and demolition would have to be transported to and interim-stored on another site. Depending on when SFR becomes available for deposition of waste from dismantling and demolition, this may be an alternative, with links to the studies of logistics and preparation of a waste plan that will take place during the coming RD&D period. Decommissioning of Ågesta also needs to be analyzed with respect to other possible ongoing decommissioning projects in the industry and their impact on resource availability at suppliers and other concerned parties.

During the years up to the start of dismantling and demolition of Ågesta, feasibility studies and planning will take place. A working group has been formed in 2013 to work with feasibility studies and decommissioning planning.

In the current decommissioning plan and the study for decommissioning of Ågesta, segmentation of the RPV is presented as the main alternative. Transport and deposition in the final repository of a RPV in one piece is described as an alternative scenario. The alternative of disposal of an intact RPV is considered preferable from e.g. an ALARA perspective. An in-depth study of the handling of the RPV will therefore be conducted during the period up to 2017.

Logistics is an important issue in the planning of dismantling and demolition. Unlike other reactors, Ågesta is not located on the coast. The waste from dismantling and demolition will have to be transported by truck on public roads. A further study of the waste shipments is needed to ensure that a safe and smooth logistics solution can be found. Logistics studies will be conducted during the coming years.

Optimization of the waste stream during dismantling and demolition of Ågesta is an area that will be studied in greater depth during the years prior to the start of decommissioning. Factors that will be taken into account in this optimization include the possibilities of on-site decontamination in Ågesta or at SVAFO in Studsvik and treatment of the waste externally. Clearance of waste is another important aspect, including the requirements for such clearance, that will be further studied during the coming years.

Measures will be adopted to avoid radiation doses to the environment and to minimize dose contributions to personnel, in accordance with the ALARA principle. A radiological survey will be conducted up to 2017 to provide better data as a basis for the decommissioning planning.

Work will be pursued on an environmental impact statement, a decommissioning plan, a safety analysis report, a waste plan and a report pursuant to Article 37 of the Euratom Treaty during the period up to 2020.

Vattenfall and SVAFO are participating in SKB's Decommissioning Group and thereby in the development work being pursued in the industry on decommissioning matters. SVAFO is also participating in the OECD-NEA's international forum for experience feedback from international decommissioning projects.

Experience from previous decommissioning projects shows that the number of measurements (material samples, air measurements, dose rate measurements etc) that are needed has increased since the 1980s. Experience also shows that unnecessary measurements are made when a sufficiently clear picture is not available of previous results. Better tools for documentation and visualization of measurement results are needed. Development of documentation and visualization tools is therefore included in decommissioning planning for Ågesta.

9.3.7 Planning for decommissioning of SKB's nuclear facilities

Clink

SKB is the licensee for Clab and will continue to be so when the integration of the planned encapsulation facility is finished and the plant is renamed Clink. The decommissioning plan for Clink (Hallberg and Eriksson 2008) will be updated in 2013 in conjunction with the compilation of supplementary material for the application for Clink. Clink will be decommissioned when all spent nuclear fuel has been encapsulated and deposited in the Spent Fuel Repository. The timetable depends on when the last nuclear power reactor is taken out of service. According to current planning, decommissioning of Clink could be commenced in around 2070 and be concluded within 5–7 years.

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 dismantling and demolition with a low dose to personnel, and the quantity of short- and long-lived radioactive waste that arises is expected to be limited. According to current plans, waste from dismantling and demolition will be sent to SFR for final disposal.

The goal of decommissioning is to remove all radioactive material and clear the plant. This means that all buildings, including equipment and land, will be cleared from regulatory control.

During 2013, SKB has conducted a study for decommissioning of Clink in order to provide waste inventory data as a basis for the extension of SFR and a cost estimate for Plan 2013.

SFR

During 2012–2013, in preparation for its application under the Nuclear Activities Act for extension of SFR, SKB has prepared a new decommissioning plan for SFR.

Decommissioning of SFR will begin when the main activity ceases, not to be resumed. Decommissioning will continue until the above-ground facility has been cleared and there are no radiological reasons to prevent the establishment of another industrial activity on the site. The parts of the facility that may be demolished in connection with decommissioning (the above-ground parts) are regarded as conventional, since they do not contain any radioactive material. A radiological survey of the facility will have to be done in order to rule out possible contamination of building parts that have been in contact with waste packages during operation, for example the terminal building. The goal of decommissioning is by definition a cleared facility. How far demolition should be carried beyond that depends mainly on how the site will be used in the future.

The timetable for decommissioning of SFR is linked to when the last currently existing nuclear power plants and SKB's other nuclear facilities are dismantled and cleared. Current plans call for 50–60 years' operation of the nuclear power plants and a few more years for Clink. Demolition of SFR could thereby be commenced in the mid-2070s.

SFL

No decommissioning plan has yet been prepared for SFL, since the design of the facility is in the concept stage.

Spent Fuel Repository

A decommissioning plan has been prepared for the Spent Fuel Repository and is 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 (Hallberg and Tiberg 2010).

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 and demolition of the surface part. Closure of the underground part is a part of the repository's barrier function and is of importance for long-term safety. SKB's work with closure is described in Section 13.2.4 (Closure).

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

9.3.8 SKB's planning for decommissioning and development of methods

Studies of decommissioning of NPPs have been conducted under SKB's auspices in cooperation with the nuclear power companies for more than 20 years. The studies have previously been based on the designation of reference plants for the BWR and PWR plants, respectively. Detailed decommissioning studies have been conducted for the reference plants, and the results have thereafter been applied to other plants. Strategies and technology have been compiled, and they are assumed to be the same for all the nuclear power units. In order to be able to determine the capacity of SKB's systems and prepare cost estimates for Plan 2013, site-specific decommissioning studies have been conducted in cooperation between SKB and the nuclear power companies. Capacities and costs are therefore now based on specific assumptions for each plant instead of the use of reference plants.

FKA, OKG and RAB state in their decommissioning plans that a near-surface repository for the very low-level waste from dismantling and demolition shall be available. A project has been started during 2013 to enable a decision to be made on a strategy in this matter, see Chapter 7.

From a national viewpoint, there is a need to coordinate decommissioning between the nuclear facilities. Further coordination is needed to ensure that the whole chain from decommissioning planning to final disposal of the waste is optimized. A Decommissioning Group with representatives from the nuclear power companies, Studsvik Nuclear AB, SVAFO and SKB was formed at the start of the 2000s, when the need emerged for a forum for dealing 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 and, for example, discuss the choice of different technical solutions for treatment and handling of the waste.

Both planning for decommissioning in the industry and SKB's work with designing the capacity of final repositories will intensify considerably during the coming RD&D period. As this happens, groupings between the nuclear power plants and SKB will be reconsidered to differentiate between informative and decision-making forums.

Decommissioning issues which SKB will discuss together with the nuclear power plants during the coming RD&D period will largely be of a technical nature. This includes work on waste type descriptions for the future waste from dismantling and demolition, adoption of waste containers for the waste and their acceptance criteria, and reconsidering the option of shallow land disposal of the very low-level waste. Another important and urgent question is what requirements apply to decommissioning, both during the planning stage and during future dismantling and demolition. Two concrete areas SKB will be working with are updating and revising the guide for preparation of a decommissioning plan (SKB 2004) and clarifying the clearance process in connection with decommissioning.

SKB is keeping track of and participating in the international development work being pursued in the areas of decommissioning and technology for dismantling and demolition through its participation in the OECD-NEA's cooperative programme and the IAEA's programme. The work is being pursued within the OECD-NEA in the Working Party on Decommissioning and Dismantling (WPDD) and within the IAEA in the International Decommissioning Network (IDN). SKB will continue to participate in these networks and has an opportunity to deepen the exchange of information now that site-specific decommissioning studies are available for all of Sweden's nuclear power plants.

One of the most important activities in connection with dismantling and demolition of a nuclear facility is the subsequent clearance of the site from further regulatory control from the viewpoint of radiation safety.

SSM's clearance regulations for materials, premises, buildings and land (SSMFS 2011:2) requires that the licensee carry out inspections in a structured manner to demonstrate that the presence of radioactive substances is not higher than the clearance levels indicated in the regulations or in another manner by SSM. SSMFS 2008:1 also contains important premises for satisfying the requirement on clearance.

A great deal stands to be gained by a well-defined clearance process, approved by all parties. For example, it engenders high confidence, since the licensees can show in detail what clearance procedures they have followed and present reliable results from measurements made. Other advantages are cost-effectiveness and the ability to keep to the timetables, since the risk of unforeseen events and thereby unforeseen costs decreases.

Another important motivation is to conserve the resources that have been set aside for decommissioning of the nuclear power plants by means of an effective clearance process.

A project was initiated in 2013 under SKB's auspices with the goal of developing a well-defined clearance process, accepted by the nuclear industry, for dismantling and demolition of Swedish nuclear facilities. The intention is to present the results in a public SKB report in the form of a manual to which licensees can refer in their planning and execution of decommissioning. This work is expected to be concluded in early 2015.

Part III

Spent nuclear fuel

10 Overview

10.1 Introduction

The focal point and purpose of SKB's activities is safe management and disposal of the spent nuclear fuel. The overall plan of action for designing and building future facilities in the final repository system is described in Section 2.3 "Plan of action for spent nuclear fuel".

Part III of RD&D Programme 2013 focuses on the technology development that is needed in order to design, build and operate the final repository system for spent nuclear fuel in a rational manner while satisfying the requirements on long-term safety, low radiation dose during operation of the facilities and a good external environment. As is evident from the applications submitted for the Spent Fuel Repository and Clink, a reference design that satisfies the design premises for the KBS-3 system has been adopted (SKB 2011a, c). At the same time, a feasible way towards production and an inspection programme has been found. Continued technology development is being done to proceed from the basic solutions outlined in the applications to solutions that are tailored to an industrialized process with specific requirements on quality, cost and time. A large part of the remaining development work consists of building up the production system with its quality assurance.

This chapter describes how technology development is managed to ensure that technology conforming to these requirements is delivered as needed during design, construction and commissioning of the final repository system's facilities. Furthermore, an overview is presented of the goals and scope of the technology development pertaining to the different barriers in the Spent Fuel Repository. The concrete development work for the different barriers aimed at ensuring the long-term safety of the final repository is described in subsequent chapters. Upgrading of Clab is described in Chapter 2.

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 be conducted at these facilities. In addition, certain tests can, in cooperation with Posiva, be conducted at Posiva's hard rock facility Onkalo, situated in Olkiluoto in Finland. 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 access to the knowledge and resources needed to carry out development work for SKB.

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 at repository depth is much lower than in the bedrock on Äspö. These and other differences need to be taken into account in the development work, but the Äspö HRL nonetheless remains an important resource for research, development and demonstration requiring an underground environment at the relevant depth. Furthermore, SKB is intensifying its cooperation with Posiva, and certain experiments may be conducted in Onkalo, where the rock conditions in many respects resemble those in Forsmark. The development work in many areas – such as fabrication and inspection of canisters, boring of deposition holes and development of transport vehicles and handling machines – is more or less independent of differences in rock conditions.

10.2 Basic principles for technology development

The structure with division of technology development into a number of production lines which SKB presented in RD&D Programme 2007 and in RD&D programme 2010 remains and has been further developed. This structure entails that the development work is conducted in production lines for fuel, canister, buffer, backfill, closure and underground openings. The production lines for buffer, backfill and closure share a number of common issues, so the development work for these production lines is integrated. This is also the reason why these lines are now presented in the same chapter, Chapter 13. Furthermore, technical systems are being developed for e.g. logistics and machines that are unique for the final repository and that are therefore not available on the market. This is described in a separate chapter, Chapter 15.

10.2.1 Design premises

The design premises comprise requirements which the KBS-3 facilities with their barriers must satisfy in order to ensure safety both during operation and after closure. The design premises specify e.g. what mechanical loads the barriers must be able to withstand, limitations concerning the composition and properties of the barrier materials, acceptable deviations in the dimensions of the barriers, and acceptance criteria for the various underground openings.

An initial set of design premises and other requirements is specified in the applications for construction of the Spent Fuel Repository and the encapsulation part of Clink. Design premises relating to long-term safety are presented in the report (SKB 2009a). Design premises relating to operational (pre-closure safety) are presented in Chapter 3 of SR-Operation for the final repository (SKBdoc 1091554) and Chapter 3 of the PSAR for Clink. Design premises for the different production lines are presented in Chapter 2 of the production line reports (SKB 2010e, f, g, h, i). In addition, overall design premises for the Spent Fuel Repository are presented in the repository production report (SKB 2010c). The requirements that are made on the choice of fuel assemblies for encapsulation are presented in the spent fuel report (SKB 2010d).

It is not possible to specify all detailed design premises for a given product or process from the start. Requirements, technology development and safety assessment must instead be formulated as the work proceeds. A revision of the design premises that were presented in the applications is now being done based on experience from the work with the production reports and the safety assessments that were done prior to the applications, as well as relevant viewpoints that have emerged during licensing. A cross-check is made between design premises for operation, post-closure safety and different production lines for the different barriers so that the reference design for the KBS-3 system and its subsystems agrees with the design premises.

The basic principles for evaluating design premises that pertain to several barriers in the Spent Fuel Repository are:

- Taken together, the design premises shall lead to compliance with requirements related to the safety of the entire Spent Fuel Repository.
- The design premises shall be practically achievable and verifiable for all concerned barriers.
- Design premises that entail simple, robust and effective solutions are preferred.

These principles are used to weigh together requirements for fuel, canister, buffer, backfill, closure and underground openings. The revised design premises serve as a basis for the preliminary safety analysis reports which SKB compiles prior to the start of construction of the Spent Fuel Repository and the encapsulation part of Clink. They will be presented to SSM when the PSAR is submitted.

Further revision of the design premises will be done in response to the conditions issued during the licensing process and in conjunction with the updating of the safety analysis reports. More detailed specification or re-appraisal of the relative importance of requirements between different systems may also need to be done during detailed design or prior to implementation.

10.2.2 Quality control and inspection

"Quality control and inspection" refers to the measures that need to be taken to provide assurance that the requirements made on the facilities during operation and after closure of the Spent Fuel Repository are satisfied. The goal is that the results obtained should conform to acceptable values for properties that contribute to safety and radiation protection.

Planned production as well as quality control and inspections in the production of the barriers for long-term safety have been described in general terms in the production reports. As development of production and testing methods progresses, the work with quality control and inspection will also progress. Systems for quality control and inspections will be established and implemented to quality-assure the production of the barriers.

A number of important activities in this process are to:

- Establish principles for safety and quality classification.
- Establish what is to be quality-controlled and -inspected, when quality control and inspections are to be performed and by whom in terms of first, second and third parties.
- Establish and qualify processes, methods, equipment and personnel for manufacturing and installation, testing and inspection.
- Establish the procedures that are to be applied in production to make sure that the KBS-3 repository satisfies quality requirements.

10.2.3 Delivery control model

To control the research, development and demonstration work needed to commission the facilities for encapsulation and final disposal of spent nuclear fuel, the work needs to be structured with milestones for results and progressive lock-in of premises. In order to obtain a common structure and nomenclature for this purpose, SKB has developed a delivery control model. The model was presented in RD&D Programme 2010 and has now been applied and developed within the various technology development projects that are being pursued or planned.

According to the delivery control model, technology development is subdivided into four phases:

- Concept phase.
- Design phase (system design and detailed design).
- Implementation phase.
- Operation and maintenance phase.

For each phase there is a specification of what should have been achieved and what should therefore serve as a basis for a decision to proceed to the next phase of development. Section 10.3 explains how far technology development must have come when some of the most important milestones are reached in the remaining work of designing and building the facilities in the KBS-3 system. The concept phase was passed in most cases when the applications were submitted.

The purpose of the design phase is to produce a design of the system, to verify that it satisfies the requirements, and to formulate proposals for production, implementation, operation, inspection and maintenance of the system. The reference design from the concept phase and design premises and other requirements serve as the starting point for the work in the design phase. The result of the design work is a more detailed, or sometimes revised, reference design. After completed system and detailed design, decisions are made to update the reference design. The design phase may be iterative since it may turn out that the proposed solution does not satisfy the requirements or that it cannot be produced or inspected in a cost-effective way.

The purpose of the implementation phase is to build and complete the production system and to fabricate, install and test developed systems in the facility and its organization. During implementation, fabrication processes and testing methods, with associated equipment and personnel, are qualified to verify that they can perform their tasks. The operating organization where the system is to be introduced is responsible for implementation with support from technology development. The implementation phase is concluded by a commissioning test aimed at showing that the systems installed in the facility, with organization and quality control systems, satisfy stipulated requirements and design premises. After the implementation phase is concluded the system can be considered to be industrialized.

Continued development is needed during the operation and maintenance phase. This is driven primarily by experience from operation and maintenance and the need for continuous improvements.

10.3 Need for technology development

Technology development shall deliver, on time, the industrialized technology needed to commission the Spent Fuel Repository and Clink. In the applications submitted for final disposal and encapsulation of spent nuclear fuel, SKB has described the reference designs of barriers for long-term safety and a repository layout that is technically feasible and can satisfy stipulated requirements and be verified against them. Detailed designs that are adapted to an industrialized process and comply with specific requirements on quality, efficiency and costs still need to be delivered and implemented. The repository layout must be adapted to the detailed local rock conditions that are encountered when the repository is built.

10.3.1 Overall needs

In the short term, the goal of technology development is to ensure that the technology needed to begin construction of the Spent Fuel Repository and the encapsulation part of Clink is available prior to the start of construction. In the case of the Spent Fuel Repository, this mainly refers to investigation methods and technology for construction of the repository accesses. This material is also needed for the document on handling of matters relating to nuclear safety prior to the start of trial operation, i.e. during construction of accesses, the central area and the first deposition area, Suus (abbreviation for "safety during construction of the final repository"), which SKB must prepare prior to the start of construction. Technology development is also needed for the systems that must be in place in the repository area in order for SKB to be able to present and get approval of the PSAR prior to the start of construction. The development needs linked to these short-term goals are discussed in RD&D Programme 2010, and a large number of technology development projects have since been initiated, as discussed in the sections on the current situation in Chapters 11 to 15. Most of these technology development projects are expected to be concluded during 2013 other 2014.

The time-critical steps of the detailed design work for systems in the encapsulation part of Clink will begin during the coming RD&D period. Moreover, detailed design will begin of the technical systems that need to be finished prior to detailed design of the repository area in the Spent Fuel Repository: the bentonite production building and the canister factory. This development work is expected to take several years and therefore needs to be started in good time before the detailed design of the different facility parts can begin. Additional tasks may follow from SSM's review of the applications under the Nuclear Activities Act.

The following section explains how far technology development must have come when some of the most important milestones are reached in the remaining work of designing and building the facilities in the KBS-3 system. Figure 10-1 provides an overview. It should be noted that several of the milestones lie beyond the coming RD&D period, but that due to the sometimes long development times, certain development activities need to be commenced now.

10.3.2 Retrieval

In Sweden there are no provisions in laws or other statutes requiring, for example, that it must be possible to retrieve spent nuclear fuel that has been emplaced in a final repository. According to SSM's general advice on the application of the regulations (SSMFS 2008:21) concerning safety in connection with the disposal of nuclear material and nuclear waste, measures can be adopted with the primary aim of facilitating the retrieval of disposed canisters. However, such measures must not lead to a deterioration in the long-term safety of the repository. There may be situations where pre-closure retrieval is necessary.

SKB deems that it is possible to retrieve canisters from the planned Spent Fuel Repository both before and after closure, but that doing so will be more laborious and probably more complicated after closure. The encapsulation part of Clink is designed so that it is possible to retrieve canisters containing fuel for re-encapsulation.

Retrieval is planned to be possible as a means of dealing with any defects that may arise or be detected during the deposition sequence. A more detailed and specific account of this will be prepared during the coming RD&D period.

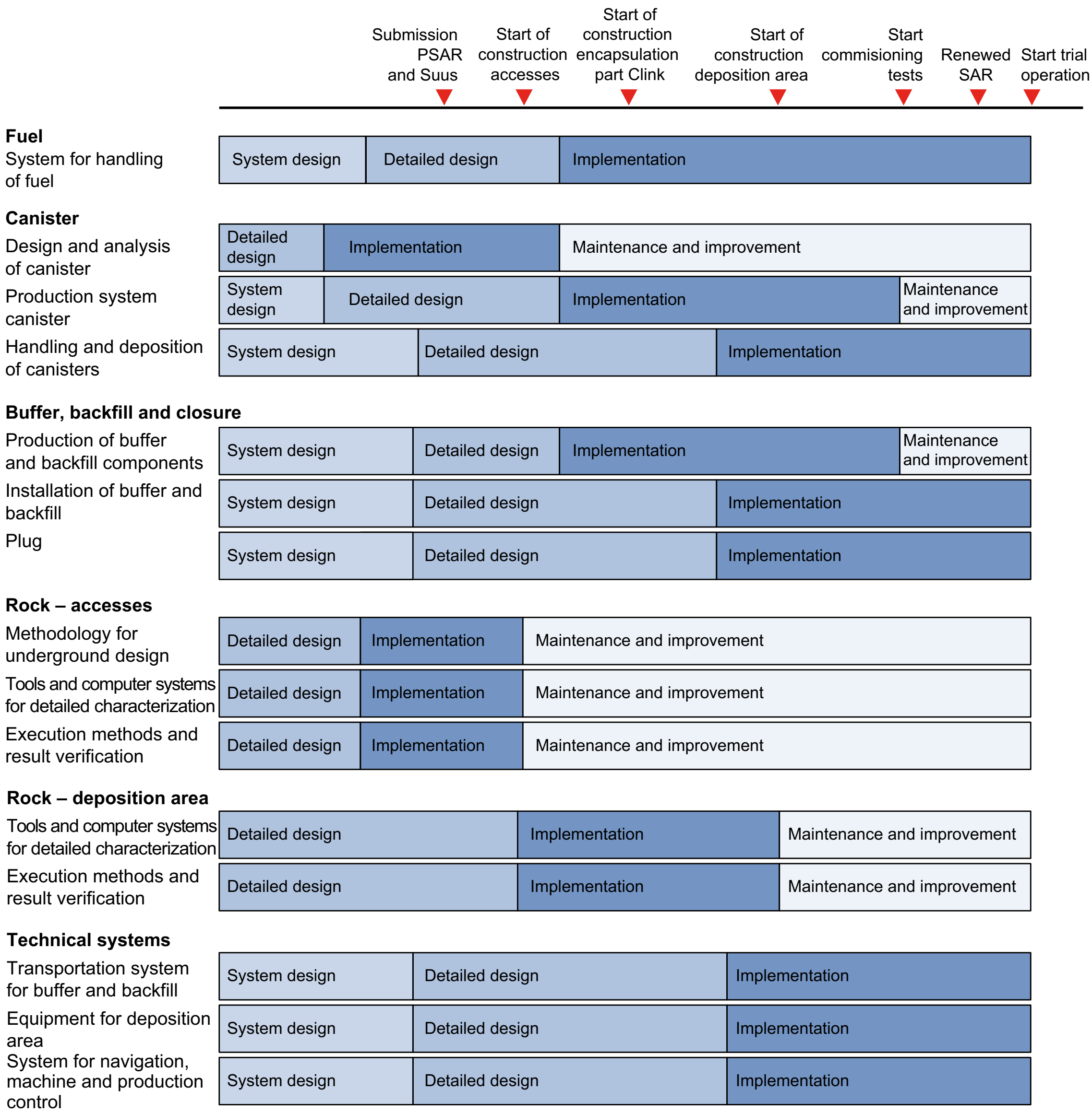


Figure 10-1. *Overview of how far technology development must have come when some of the most important milestones are reached in the remaining work of designing and building the different facilities in the KBS-3 system. Note that several of the future milestones lie beyond the coming RD&D period, but are shown here to provide a complete picture of technology development.*

10.3.3 Horizontal deposition

Together with Posiva, SKB is studying whether horizontal deposition (KBS-3H) can constitute an alternative to vertical deposition. The current project period lasts until 2016, after which an evaluation will be done. Decisions and planning for continued development of KBS-3H will follow this evaluation.

10.4 Milestones

This section describes important milestones where deliveries from technology are needed and what needs to be achieved by the time these milestones are reached. More details, as well as plans, are presented in Chapters 11–15.

10.4.1 Basis for PSAR prior to start of construction

Fuel

Prior to the PSAR for Clink and the Spent Fuel Repository, the following must be ready:

- Documentation of fuel. During the coming RD&D period the existing database will be evaluated, updating needs identified and requirements established.
- The database of calculated source terms and radiation characteristics, which gives the radiological properties of general fuels with different burnups and decay times, will be updated and augmented. The database will be incorporated in SKB's established and quality-assured tool for guaranteeing the quality of data management and user-accessibility.
- Evaluation of the chosen method for drying the fuel, Forced Gas Dehydration (FGD), will be completed.
- A deepened and broadened criticality analysis will be done with the overall goal that criticality analysis, including calculation of burnup credit and BA (Burnable Absorber) credit (see Section 11.9 "Criticality"), for all SKB's present and future facilities should be done according to the same principles, with the same modern calculation tools and with the same methodology.

Canister

Revised design premises will serve as a basis for updating of the production report for the canister. A PSAR prior to the start of construction means that the design phase must be essentially concluded, which means that the following must be ready:

- The canister's reference design shall be verified against the design premises.
- Proposed manufacturing methods shall be shown to be able to produce canister components and canisters according to the chosen reference design.
- Testing processes and inspections shall be clarified and shown to permit verification of compliance with the design premises.

Buffer, backfill and closure

Revised design premises will serve as a basis for updating of the production reports for buffer, backfill and closure. This work includes a product and process survey to determine what is important for safety and preparation of a preliminary quality plan that identifies the need for quality assurance measures.

The system design work will continue until submission of the PSAR.

Rock

The PSAR and Suus documents must contain descriptions of requirements, methodology, execution and verification of results for all rock work done in the final repository. In the case of the PSAR, such descriptions must also be provided for rock works in the deposition area, selection of deposition tunnels and deposition holes and how the results of rock works and selection can be verified against the design premises. These descriptions must provide a sufficiently clear picture of the work and the final result, but development of details can continue until detailed design of the deposition area begins.

Technical systems

Prior to the PSAR, material needs to be submitted from the ongoing development of technical systems for updating of production reports and preparation of system descriptions.

10.4.2 Start of detailed design of encapsulation part of Clink and canister factory

At the start of detailed design of the encapsulation part of Clink, the component technical systems must have essentially passed the detailed design phase. This means that:

- Technology development for decay heat measurements must have passed the concept phase, system design and detailed design.
- The chosen drying method must be verified and validated so that it meets the requirements made on the fuel.
- The work initiated by SKB to introduce safeguards in the design of the encapsulation part of Clink and the final repository must be completed.

At the start of detailed design of the canister factory, technology and methods for production of canisters must be fully developed and work on an industrial scale.

10.4.3 Prior to start of construction of encapsulation part of Clink and canister factory

Prior to start of construction of the encapsulation part of Clink and the canister factory, those systems that have undergone detailed design in accordance with Section 10.4.2 shall have been procured and plans for qualification shall have been established and incorporated in the plans for construction.

10.4.4 Basis for detailed design of accesses

Before detailed design of the Spent Fuel Repository's accesses can start, the Observational Method for underground construction must be implemented and a detailed characterization programme for ramp and shafts must be available.

10.4.5 Passage of level under top seal

Below the level of the top seal on the Spent Fuel Repository, the design premises stipulate requirements on the permeability of the installations intended to seal the repository at depth. This in turn imposes other requirements on rock work and the excavation-damaged zone (EDZ) beneath the top seal. This means that it must be verified that excavation methods, inspection programmes, and methods for rock support and grouting satisfy the requirements that apply to the level under the top seal.

10.4.6 Basis for detailed design of production building

Detailed design of the production of buffer and backfill shall be completed as a basis for detailed design of the production building. This means that the following must have been done:

- Requirement specifications on material must have been determined.
- A decision must have been made on the pressing technology for buffer blocks.
- Manufacturing methods that work on an industrial scale for production of bentonite blocks and bentonite pellets for buffer and backfill must have been determined.
- Inspection methods that work on an industrial scale must have been determined.
- The principles for quality control and inspection must have been determined.

The equipment that is needed in the production building consists almost entirely of standard machines. Some studies nevertheless need to be conducted to obtain data as a basis for requirement specifications and choice of equipment. This is done within the production lines for buffer and backfill.

10.4.7 Basis for detailed design of deposition area

Installation methods and methods for testing and inspection of buffer and backfill must have been designed in detail and verified prior to detailed design of the deposition area. For development of special machines, prototypes must have been produced and tested. To verify that the installation of buffer and backfill works as intended within the requisite timeframes, full-scale tests need to be conducted under conditions prevailing underground.

The following must be finished for rock-related issues:

- Verification of site model and site understanding.
- Test to make sure that the criteria for selection of deposition holes are appropriate.
- Deposition hole bored and machining of bevel in Forsmark rock tested.
- Rock excavation methods including rock support and grouting determined.
- Method for machining of level floor under Forsmark conditions determined.

10.4.8 Start of commissioning tests of the KBS-3 system

Technical systems that are needed in Clink must have been purchased, fabricated, installed, tested and qualified prior to commissioning tests of the KBS-3 system. Before commissioning tests can be done, methods and sub-processes for excavation of the Spent Fuel Repository must have been devised and qualified.

The deposition system must be put into operation before commissioning tests can be undertaken, which means that technical systems for handling and transport of canisters, buffer and backfill must have been fabricated, installed and tested. The systems will undergo integration tests to ensure that equipment and technical systems work together as intended before the commissioning tests. Qualifications of processes with appurtenant equipment, personnel and suppliers must have been completed and documented.

A system for quality control and inspection of canister manufacturing, production of buffer and backfill components, handling and installation of canister, buffer and backfill and the rock construction process must be implemented.

10.4.9 Basis for SAR

Before a licence can be obtained for trial operation of Clink and the Spent Fuel Repository, a renewed SAR must be submitted. Before an operating licence can be obtained, a supplemented SAR must be prepared and submitted to SSM. Results and experience from the implementation phase must be presented in this updated SAR, including the results of the commissioning tests in each facility. This means that the production reports will be updated with results from full-scale tests, qualifications and commissioning tests.

11 Technology development, fuel handling

This chapter deals with the technology development that is planned for handling of the spent nuclear fuel prior to final disposal. Issues that are of importance for the post-closure safety of the Spent Fuel Repository, such as fuel dissolution, are discussed in Chapter 23 "Fuel".

11.1 Requirements and premises

The spent nuclear fuel consists of fuel assemblies from the twelve Swedish nuclear reactors. Figure 11-1 shows fuel assemblies from a BWR reactor and a PWR reactor. In addition to the fuel from the twelve reactors, minor quantities of miscellaneous spent fuel 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.

RD&D Programme 2010 described how SKB has continued the work of specifying design premises for the Spent Fuel Repository and Clink during operation and the post-closure safety of the repository. The results of this work in terms of requirements related to the fuel were presented in the fuel report (SKB 2010d, Chapter 3), which is included in the applications for the Spent Fuel Repository. The most important of the design premises linked to the fuel are:

- The fuel's decay heat affects the temperature in the final repository. The current requirement is that the temperature in the buffer may not exceed 100°C. The fuel assemblies enclosed in a canister must therefore be chosen with respect 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.
- Radiation on the outside of the canister can lead to corrosion of the copper canister. The maximum permissible radiation dose rate on the canister surface is one gray per hour (Gy/h). For combinations of fuel assemblies that can be accepted for encapsulation with respect to the decay heat requirement, the surface dose rate on 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.
- 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.
- Criticality may not occur in the canister under any circumstances. Fuel assemblies to be encapsulated shall be selected with respect to enrichment, burnup, geometric configuration and materials in the canister so that criticality will not occur during handling and storage, even if the canister should, hypothetically, be filled with water.

All fuel parameters that affect the handling of the spent nuclear fuel and can affect the safety of the Spent Fuel Repository need to be known and documented. The fuel suppliers and the reactor owners have therefore developed calculation and inspection programmes for the parameters. The programmes are quality-assured and approved for their purpose by SSM.

The general rule is that the design premises that are linked to the fuel are revised as a part of the revision described in Section 10.2.1 "Design premises". The following design premises are planned to be revised:

- The requirement on the quantity of water in the canister will be revised and linked to whether residual water could adversely affect the canister's containment capability.
- The fuel will be divided into ordinary fuel and fuel requiring special handling. In other words, miscellaneous or damaged fuel will be handled in a separate process where special solutions are devised and analyzed.

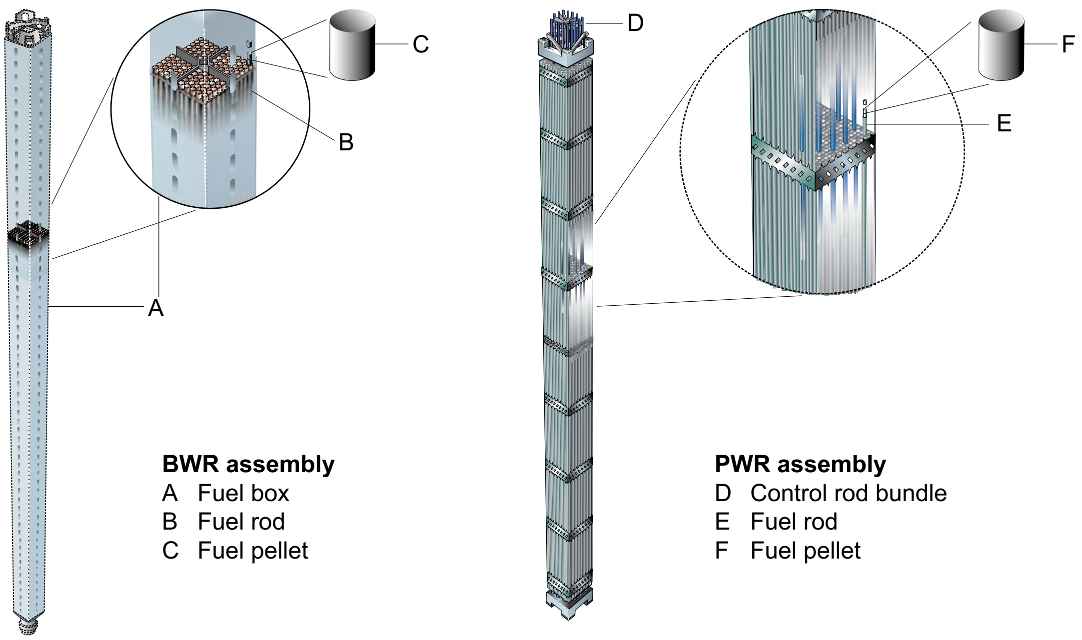


Figure 11-1. *Fuel assembly from a BWR reactor (left) and a PWR reactor (right).*

11.2 Current situation and programme

Conclusions in RD&D 2010 and its review

In RD&D Programme 2010, SKB gave an account of its development plans with regard to decay heat and radiation, water and water vapour, criticality and safeguards. In its review of the programme, SSM stated the following:

- SSM judged that SKB had an appropriate development programme with regard to decay heat and radiation and that the presented results are credible. The Authority pointed out that an account of the contributory gamma radiation heat should be included in future support documents. It was further noted that the two measurement methods used by SKB are not independent of each other. The agreement between the results of the measurement methods is therefore not considered to be a validation of the results.
- SSM found that SKB's technology development plan regarding water and water vapour should be improved and that a criterion for the drying process should be established. The Authority also noted that SKB had not reported data on damaged fuel and associated drying mechanisms.
- SSM found that SKB's programme for measurement of decay heat in new fuel types is good. Moreover, the Authority concluded that the criticality analyses should be supplemented with respect to the fuel configuration in the canister and PWR fuel that is reactive. All fuel handled by SKB shall comply with the criticality conditions.
- Regarding safeguards, SSM had a number of recommendations for improvements.

Summary of current situation and programme

Since RD&D Programme 2010 was submitted, SKB has striven to increase consistency in the treatment of technical issues related to the spent nuclear fuel. SKB is constantly gathering knowledge and developing methodology and technology related to the handling of the nuclear fuel.

Specific subareas where developments related to the handling of fuel are described are presented in Sections 11.3–11.11. These subareas are:

- Fuel requiring special handling.
- Documentation of fuel.

- Activity inventory and source terms.
- Radiation protection and dose estimates.
- Decay heat.
- Water and water vapour.
- Criticality.
- Fuel and canister optimization.
- Safeguards.

SKB's programme for handling of fuel comprises several parts, from requirements on information on the properties of the fuel before it is used in the fuel cycle to devising a programme for safeguards that is internationally approved. During the coming RD&D period, development work will be carried out pertaining to decay heat determination and criticality verification for all facilities and shipments included in the KBS-3 system. Development of safeguards is an area where SKB works in close cooperation with international bodies.

SKB will also handle nuclear fuel fabricated from reprocessed spent nuclear fuel, called MOX fuel (Mixed Oxide Fuel). This fuel will be taken into account in the different subareas.

11.3 Fuel requiring special handling

Current situation

SKB has decided that fuel requiring special handling will be studied and reported on separately. This category includes damaged fuel and miscellaneous spent fuel, and it will be handled in one or more separate processes where special solutions will be devised and analyzed.

Programme

There are many different types of fuel damage that require special methods to be managed in the KBS-3 system. For example, the damaged rods can be encapsulated individually or together in a control rod box. In order to ensure that all damaged fuels can be managed in the KBS-3 system, an inventory of damaged fuel and a plan of action for its management will be prepared during the coming RD&D period.

The proper steps to take with damaged fuel at the nuclear power plants, Clink or other facilities will then be determined.

11.4 Documentation of fuel

The database that is used today to document the inventory of spent nuclear fuel is called Pluto. Pluto has certain limitations regarding what information can be stored, user-friendliness, search options and automated checks. It will therefore either need to be replaced by a new database or improved to meet the requirements in the aforementioned areas. During the coming RD&D period, Pluto will be evaluated, updating needs identified and requirements established.

11.5 Activity inventory and source terms

Current situation

A company-wide report with calculated activity inventory and source terms has been prepared as a basis for the applications (SKBdoc 1179234). This report presents nuclear properties for general fuel types with different burnups and decay times. Data are included on fuel assemblies from PWRs and BWRs as well as BWR-MOX fuel.

Programme

The first upgrade of activity inventory and source terms will be done in 2013. Further needs will be identified in current projects and possible supplementary data will be added as needed. The report where this is presented will undergo quality assurance and be incorporated in SKB's quality-assured tool for ensuring the quality of data management and user-accessibility.

11.6 Radiation protection and dose estimates

Current situation

The KBS-3 method includes many different steps where it is very important to know the dose rates that are emitted by the units being handled. This includes, for example, handling of fuel assemblies in Clink, transport of filled canisters and canister handling in the final repository. Since RD&D Programme 2010, nuclear data on nuclear fuel assemblies and canisters with spent nuclear fuel have been gathered and used in analyses concerning radiation protection and dose estimates. These data are used in, for example, the applications for the final repository's safety assessment for operation.

Programme

Methods and calculation programmes for calculating dose rates and dose loads are internationally validated and verified and based on e.g. particle transport calculations of the Monte Carlo type. SKB intends to continue using these calculation methods and to develop and adapt the calculation models to SKB's situation and needs and apply them in an increasing degree of detail as the needs arise. This applies in particular when technology development enters the design phase.

11.7 Decay heat

Current situation

Since the middle of the 1990s, SKB has followed the development of calculation programmes for decay heat in spent nuclear via cooperation with both Studsvik Scandpower and Oak Ridge National Laboratory. The calculation methods that exist today are based on operating histories from the operators of the reactors. In order to calculate decay heat, it is therefore important to make sure that the data are correct. This can be done either by administrative quality assurance of the data or by verifying measurements of the fuel prior to encapsulation.

A calorimeter for measurements of decay heat in individual fuel assemblies has been in use at Clab since 2003, and measurements have been performed on more than 200 BWR and PWR assemblies (SKB 2006a). The results of these measurements are used to understand the properties of the fuel and as a basis for the development of calculation codes that is being done in collaboration with Oak Ridge National Laboratory (USNRC 2010a, b). Since the measurements are time-consuming, an alternative method, high-resolution gamma spectroscopy, has been developed in cooperation with Uppsala University (SKB 2006a).

Programme

SKB will continue during the coming RD&D period to perform calorimetric measurements and participate in joint international efforts to refine present-day calculation models. SKB will continue to develop the method of determining decay heat via high-resolution gamma (or gamma-ray) spectrometry. The method will be developed and complemented with neutron measurements to improve the results. The measurements will be done in cooperation with Uppsala University and the Los Alamos National Laboratory.

11.8 Water and water vapour

Current situation

During the design of the encapsulation part of Clink, two methods for drying of fuel have been studied: vacuum drying and drying with hot air. Both methods have been found to satisfy the requirements on drying of undamaged fuel. Regarding damaged fuel, there are uncertainties as to the efficacy of the drying method in expelling the water through pinholes in the fuel cladding. Vacuum drying is the reference method that is presented in the application for Clink.

A project was started in late 2009 for the purpose of finding an alternative method for more efficient drying of fuel. In early 2010, a market survey was conducted of methods for drying of spent nuclear fuel. The methods and systems that were found in the market survey were variants of vacuum and hot-air drying, plus a system called Forced Gas Dehydration (FGD). FGD is a method developed by Holtec International and is a variant of Forced Helium Dehydration (FHD), which is licensed by the Nuclear Regulatory Commission in the USA.

Vacuum and hot-air drying have been considered previously by SKB and their weaknesses have long been known. Experience of the strengths and weaknesses of these methods, as well as newer experience from FGD, leads to the conclusion that FGD has advantages over the other drying methods. They include both how effectively the fuel can be dried and how gentle the drying is on the fuel.

Programme

SKB has chosen to proceed with the FGD method. FGD is a two-step process where the first step involves vaporizing water, moving it and condensing it out. Bound water is then expelled from e.g. damaged fuel in a process similar to freeze-drying. The feasibility study shows that FGD is preferable for fuel drying since it is able to expel water from even slightly damaged fuel. Further design work will assume that FGD is used for fuel drying in the encapsulation part of Clink and that the blowdown gas is nitrogen.

In the next phase (system and detailed design), the drying system will be further adapted to the needs of the facility (cooling of equipment, treatment of process water, treatment of nitrogen, interface with other systems, adaptation for 60 years' operation, etc). The system will be designed to European codes and standards so that it can be CE-marked.

A remaining task is to determine what size of defects can be expected on Swedish nuclear fuel, i.e. how small holes are relevant to take into account and test that a drying method can remove bound water from.

11.9 Criticality

Current situation

In the applications for Clink and the Spent Fuel Repository, credit has been taken in the criticality analyses for the decrease in reactivity that occurs when the fuel is irradiated in the reactor and burnup increases (burnup credit). In the case of Clab, burnup credit is not taken today for PWRs, but BA credit is taken for BWRs. This means that credit is taken for the fact that BWR fuel contains a reactivity-reducing substance (mainly gadolinium-155) that burns up during the early period in the reactor, a so-called Burnable Absorber (BA).

Programme

The method used by SKB for burnup credit in the applications for the Spent Fuel Repository follows a method developed by Oak Ridge National Laboratory. SKB intends to implement this methodology during the coming RD&D period for criticality calculations for all of SKB's facilities and the transportation system.

The overall goal of SKB's programme regarding deepened and broadened criticality analysis is that criticality analyses for all of SKB's present and future facilities should be done according to the same principles. This includes both methodology and calculation programs as well as the analysis itself, including burnup credit and BA credit. Development of the methodology and validation of calculation programs shall comply with modern internationally accepted standards and requirements.

Furthermore, SKB shall ensure that it has the necessary competence regarding criticality. On a more detailed level, the programme has the following content:

- Validate the calculation tools SKB uses for criticality analysis.
- Describe the methodology and the principles SKB uses for burnup credit.
- Update the criticality analysis for the encapsulation part of Clink.
- Update the criticality analysis for the Spent Fuel Repository.
- Ensure that the competence and the tools needed to perform criticality analyses for Clab, Clink and the Spent Fuel Repository are available within the Vattenfall Group.
- Update the criticality analyses for Clab with modern calculation tools and for the fuel types that are most reactive today.

11.10 Fuel and canister optimization

Current situation

When design premises have been determined for the encapsulation part of Clink and the Spent Fuel Repository, the need to optimize fuel and canister handling has emerged. This has to do with which fuel assemblies should be placed in which canister. An initial optimization has been done and reported in the fuel report (SKB 2010d). The primary target of the optimization was the canister's total decay power/heat and to minimize the number of canisters to be disposed of.

Programme

The degree of detail in the design premises has increased, and a new and more detailed optimization is required. The number of parameters included in the optimization should be increased for one thing. There is therefore a need to revise and supplement the account of fuel and canister optimization.

The goal is to fill the positions for fuel assemblies in the canisters prior to deposition without making their handling too complex or requiring too many lifts. Requirements on criticality and decay heat must always be complied with. Several different aspects of the KBS-3 system must be taken into account, for example:

- Optimization of fuel content.
- Handling in Clink.
- The Plan report (prediction of future fuel use and economic optimization).
- The Spent Fuel Repository's deposition layout.

11.11 Safeguards

Current situation

Safeguards enable regulatory authorities and inspection bodies to ensure that nuclear material is not diverted. By "nuclear material" is meant here uranium, uranium depleted of the isotope 235 or fissile materials such as uranium-233, uranium-235, plutonium-239 and thorium. 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 spent nuclear 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 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 other activities are carried on that those indicated.

Programme

The requirements on safeguards for the different steps in the handling of spent fuel and sealed canisters are general and comprehensive. SKB works broadly with these questions and is responsive to the regulatory authority's recommendations. The overall requirements come from the IAEA, who sets the framework. The European Commission, Euratom, also imposes requirements on safeguards in keeping with the treaty signed by Sweden. This means that it is important that SKB's development of methods and equipment for handling, verification and logistics takes place in cooperation with the IAEA and Euratom. One step in this cooperation is the preparation of a draft BTC (Basic Technical Characteristics) document. The first versions of the draft BTC for Clink and for the Spent Fuel Repository have been submitted to Euratom for comment. Updates and development of the draft BTC will take place for a long time to come and the final version will not be adopted until trial operation.

Methodology and verification of safeguards are being planned for the coming RD&D period and will be coordinated with the aforementioned decay heat project. The programme for safeguards intends to accomplish the following specific goals for each part of the KBS-3 system:

Encapsulation

The spent fuel's content of nuclear 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 at the same time as the decay heat by calorimetric methods. To verify the calculated nuclear material content, it may be necessary to perform measurements of neutron radiation and gamma radiation on each fuel assembly. These two independent methods can provide sufficiently good information on how much nuclear material each fuel assembly contains (see also Section 11.7).

To guarantee the content of nuclear material in each canister, the fuel will, after it has undergone verifying measurements in the verification position in the encapsulation part of Clink, be positioned so that its identity cannot be mistaken. After handling, drying and emplacement of the fuel in the canister, the identity and position of the fuel is verified by optical equipment. The identity of the canister is verified at the same time so that there can be no doubt about which nuclear fuel has been placed in each canister. After sealing, the canister comprises the new smallest unit in the handling of the nuclear material. This step, also called re-batching, is extremely important in the handling chain, and it will be studied how handling, inspection, verification and accounting should be done to avoid any possibility of error.

SKB has initiated a project together with SSM, Euratom and the IAEA to incorporate safeguards in the design of the encapsulation part of Clink. Two main areas in need of technology development have been identified: verification of the fuel and verification of the canister. SKB has decided to carry out decay heat measurements and fuel verification. A joint project between Euratom, the US Department of Energy and SKB has been launched to determine whether a common measurement can be performed for verification of fuel. There is a need for similar studies for canister verification, and work is being pursued internationally in this area.

Transport

The canister is placed in a transport cask for transport to the Spent Fuel Repository. The canister transport cask will be provided with a seal and other equipment to ensure that the canister that is received in the final repository has not been tampered with and that no nuclear material has been diverted.

Final repository

The seal is checked at the final repository. The canister identity is verified in the final repository's receiving station when the canister is transloaded to the deposition machine to make sure it agrees with the consigned canister identity. The final repository's openings are under surveillance to make sure that no nuclear material can be diverted.

12 Technology development, canister

Since RD&D Programme 2010, SKB has submitted applications for the KBS-3 system. The canister's reference design is described in the production report for the canister (SKB 2010e). A need for continued technology development and testing was identified in the work with the production report and described in RD&D programme 2010. This work is under way and will be essentially finished and reported during 2014.

According to the reference design, the canister consists of a cylindrical container with a copper shell and a load-bearing nodular iron insert. Channel tubes of structural steel have been cast into the insert to fit the spent nuclear fuel. Two different designs of the insert have been developed where the size and number of the channels are adapted to receive fuel assemblies from pressurized-water or boiling-water reactors (PWR and BWR inserts).

Reference methods for manufacturing of components and welding of lids and bases have been chosen to fulfil the design parameters of the reference design with reproducibility and with a potential for industrialization. The copper tubes are manufactured by extrusion, lids and bases are forged and the insert is cast. Welding of copper base and sealing of the canister are done by means of friction stir welding (FSW). The development of friction stir welding has led to an automated and stable process.

Development of the canister is now in the design phase. Together with Posiva, SKB is planning further development work on:

- The reference design.
- Pierce and draw processing of copper tubes as an alternative reference method.
- The manufacturing process for the nodular iron insert.
- The manufacturing method for making copper lids and bottoms.

This chapter describes technology development for the canister with regard to dimensions, manufacturing, sealing, inspection and deposition.

12.1 Requirements and premises

RD&D Programme 2010 observed that work on specifying design premises for the canister had continued. The results of this work were presented in the production report for the canister (SKB 2010e, Chapter 2), which is included in the applications for the Spent Fuel Repository. The most important of the design premises reported there and currently valid for the canister are:

- The canister shall withstand an isostatic load of 45 megapascals, which is the sum of the maximum swelling pressure from the buffer and the maximum groundwater pressure.
- The copper shell shall remain intact after a five centimetre shear movement at a velocity of one metre per second. This applies for a buffer with the material properties of a calcium bentonite with a density of 2,050 kilograms per cubic metre (kg/m^3), for all positions and angles of the shearing fracture and for temperatures down to 0°C. The insert shall retain its load-bearing properties with respect to isostatic load.
- The copper canister shall have a nominal thickness of five centimetres, including at the welds.
- The properties of the spent fuel and the geometric configuration and material composition of the canister shall be such that criticality is avoided in the event water should enter the canister.
- The design premise stating that the canister shall withstand isostatic load also includes asymmetric loads due to uneven swelling of the bentonite. Temporary asymmetric loads can occur during the water saturation phase due to uneven saturation of the buffer or irregularities in the deposition hole. Irregularities in the deposition hole can also give rise to permanent asymmetric loads.

There are also some more indirect requirements on the composition and microstructure of the canister materials and the environment in the sealed canister. They are summarized as follows:

- There are indications that gamma radiation could cause embrittlement in the nodular iron insert if the copper content of the insert exceeds 0.05 percent.
- The copper material in the shell is verified for:
  – creep ductility: grain size < 800 micrometres, phosphorus content 30–100 ppm, sulphur content < 12 ppm,
  – brittleness: hydrogen content < 0.6 ppm.
- An oxygen content of some tens of ppm can be allowed in the copper shell.
- The quantity of nitric acid that can form in the insert shall be limited by replacing the atmosphere in the insert so that the argon content is > 90 percent. The permissible water quantity in the insert is set to 600 grams.
- The material composition for the nodular cast iron in the insert shall be: iron > 90 percent, carbon < 6 percent and silicon < 4 percent.
- To permit ultrasonic testing of copper components, the average grain size in the material shall be < 360 micrometres.

In addition there are requirements related to production and operation. Generally formulated, manufacturing and installation of the canister shall be based on proven or tested technology. Canisters with the specified properties must be possible to produce and install with high reliability.

The design premises for the canister will be revised as a part of the revision described in Section 10.2.1 "Design premises". The following design premises are planned to be revised:

- Permissible quantities of encapsulated gases and liquids, with a view to the risk of excess pressure and corrosion in the canister.
- Formulation and justification of the requirement on what isostatic load the canister must withstand with a view to future ice loads.
- The formulation of how the requirements for shear load on the canister should be verified to take into account uncertainties and spatial variation of defects in the insert and inhomogeneous distribution of the density of the buffer.
- A clearer description of the requirement on the dimensions of the copper shell with regard to corrosion protection that allows for tolerances in manufacturing.
- Maximum permissible temperatures in the canister materials during handling with a view to the impact on material properties.
- Permissible oxygen content of the copper.
- Acceptable surface defects on the canister's copper shell.

In no case does this mean that the conclusions in the SR-Site safety assessment (SKB 2011e) need to be revised other than that they must be even clearer on how fulfilment of the design premises can be verified.

12.2 Current situation and programme

Table 12-1 summarizes the regulatory authorities' comments regarding the development programme for the canister which SKB described in RD&D Programme 2010 and the current situation and programme for the coming period. More details on the current situation and programme regarding technology development for the canister are given in Sections 12.3–12.7.

Table 12-1. Summary of the regulatory authorities' comments on the account of technology development for the canister in RD&D Programme 2010, SKB's current situation, and SKB's programme.

Regulatory authorities' comments on RD&D Programme 2010	SKB's current situation	SKB's programme
Canister design – analysis of the canister		
Toughness and plastic properties of nodular iron need to be verified.	New tests have shown stable crack growth over 2.5 mm.	Continued verification of the plastic properties of nodular iron.
Analysis of uneven swelling pressure in the buffer and impact on the canister during shear.	Load case identified but not analyzed.	Analysis initiated.
Relationship between size of shear and risk of damaged canister.	Probabilistic calculations completed.	Evaluation and reporting of calculations.
Analysis of shear near the top and bottom of the canister.	Simulations done.	Analysis of effects in progress, will be reported in December 2013.
Importance of triaxiality for ductility.	Testing completed.	Evaluation of completed testing in progress.
Impact of radiation on fracture toughness.	Testing planned.	Carry out and evaluate testing of test bars.
Certain requirements on the mechanical properties of the canister are lacking.	Sensitivity analyses for BWR inserts done for the shear load case, which provides a basis for specifying requirements on mechanical properties.	Further calculations planned.
Residual stresses in the insert.	Cooperation with Posiva. Stresses on the level 30–60 MPa detected.	Surface measurements and measurement of internal stress distribution for BWR inserts completed. Measurements for PWR inserts in progress.
Manufacturing of inserts		
Further characterization of the mechanical properties of the insert.	Programme for characterization of stable crack growth, triaxiality and fracture toughness in progress.	Evaluation and continued characterization.
Nondestructive testing (NDT) of inserts		
Further development of technology for nondestructive testing for characterization of defects.	Ultrasonic technology has been further developed and tested on full scale on fabricated inserts.	Development of technology for surface testing. Continued work with development of technology for ultrasonic testing of the insert, including methodology for characterization and size determination of defects.
Verification of testing system by demonstrations on test blocks.	Initial trials with casting of test blocks in progress.	Further development of technology for production of relevant defects in test blocks.
Manufacturing of copper components		
Reason for bands with increased grain size.	Study carried out where a number of influential factors were identified.	Establishment of supplementary requirements on copper with regard to sound attenuation.
Nondestructive testing of copper components		
Explain the relationship between increased sound attenuation, average grain size and smallest detectable defect in nondestructive testing.	Investigations of detection capability in ultrasonic testing of copper tubes with different material structures and differing ultrasound attenuation in progress.	Preliminary POD (probability of detection) curves will be calculated for ultrasonic testing of copper with differing attenuation. The relationship between material structure and ultrasound attenuation will also be studied.
Seal welding		
Extent of joint line hooking in the radial direction.	Experimental study with variation of probe length and rotation direction.	Continuation of programme for increased understanding and minimization of the phenomenon.
Influence of weld defects on mechanical properties.	Investigation of defects and their influence on creep ductility. The influence of defects on the mechanical integrity of the canister is small.	Programme to minimize the occurrence of weld defects.
Nondestructive testing (NDT) of seal welds		
	Further developed technology for ultrasonic testing and radiography, plus implementation of a preliminary version of eddy-current testing of the weld surface.	Devise concept for how the testing methods that have been developed can be implemented in the encapsulation part of Clink.
Handling and deposition of canisters		
Study the feasibility of automatic detection of surface defects on the canister's copper shell before deposition.	Feasibility study done with laser scanning. Positive result.	Determine the smallest permissible defect and specify a system for scanning of the surface.

12.3 Canister design – analyses of the canister

Conclusions in RD&D 2010 and its review

SSM said in its review of RD&D Programme 2010 that there were fears regarding the toughness and plastic properties of the nodular iron insert. The Authority also wanted to see a verification of the importance of triaxiality for ductility under a combination of shear load and glaciation. Furthermore, it was deemed that the bending of the load case caused by uneven swelling pressure needed to be analyzed to determine permissible defect size.

SSM was of the opinion that fracture toughness ought to be verified for radiation-affected material. Shearing of the canister in a plane near the lid and bottom should also be analyzed to guarantee the integrity of the copper canister. The Authority also found that the importance of residual stresses for damage tolerance needs to be studied, since measurements indicate compressive stresses on a level of 100 megapascals near the surface of the insert.

SSM also pointed out (in the description of long-term safety) that certain requirements on the material properties of the canister are still lacking, along with geometric tolerances of importance for manufacturing. Handling loads have been analyzed to some extent (including lifting by the lid), but they are not included explicitly in the design premises. SSM further deemed that it is important in future FEM (Finite Element Method) analyses to describe the canister's entire load history in view of the fact that cold-worked copper has lower creep ductility than copper that has never undergone plastic deformation. SKB's reported calculation results where both primary and secondary creep are modelled also show that the impact of tolerance chains between the different parts of the canister should be further analyzed.

Current situation and programme

SKB's work with analyses of the canister was summarized in the production report for the canister (SKB 2010e) and in the underlying reports: design analysis report (Raiko et al. 2010), FEM calculations (Hernelind 2010) and damage tolerance calculations (Dillström and Bolinder 2010). The bentonite model used in the FEM calculations is based on laboratory experiments (Börgesson et al. 2010, Dueck et al. 2010) and has been verified against an older shear test (Börgesson and Hernelind 2010).

A number of different analyses and investigations for the insert are under way or planned. They will serve as a basis for the updated design analysis presented in the PSAR for the Spent Fuel Repository. The most important areas are as follows:

- Deterministic calculations for a canister with a BWR insert and with a PWR insert have yielded critical defect sizes in the insert. The results show that the acceptable defect depth for a semi-elliptical surface defect is 4.5 millimetres for the BWR insert. The acceptable defect size for the PWR insert is of the same order of magnitude as for the BWR insert. Further calculations will be done for the PWR insert with material data obtained in a more recent casting of three PWR inserts. The analyses, together with fractographic studies, have shown that the greatest risk for the insert is crack growth under shear load when the surrounding bentonite has a high density. The cracks can be initiated from inclusions in the material if they have an unfavourable size and direction.
- SKB has carried out testing of PWR and BWR inserts manufactured by means of a refined casting process, and the results show that the resistance of the material to crack growth is retained to a crack length of at least 2.5 millimetres. The results obtained differ between different laboratories, and the reason for these differences is currently being investigated.
- SKB has conducted an experimental study to determine whether triaxial compressive loads affect the ductility of the material. Test bars with a waist to obtain plane deformation were tested together with test bars for pure shear. The test results are currently being evaluated.
- Different load cases on the canister caused by uneven swelling of the surrounding bentonite buffer will be investigated during the coming period.
- Studies are under way to investigate the effects of radiation on the fracture toughness of nodular iron, see Section 24.2.6 "Radiation effects".

- In order to learn more about residual stresses in BWR inserts, supplementary testing of the internal parts of the insert has been carried out together with Posiva. The hole drilling method has been used and residual stresses on the order of 30–60 megapascals have been found. An example of results is shown in Figure 12-1. The effect of these stresses is not judged to be of any importance for the canister's integrity.
- Probabilistic calculations for BWR inserts have been carried out for the shear load case, which, together with deterministic calculations, will serve as a basis for determining the requirements on the insert's mechanical properties. Work on evaluation and reporting of results is ongoing.
- Previously performed FEM simulations for the different loading cases for the insert were done without contact definitions between the insert's cassette and the surrounding nodular iron included. Sensitivity analyses will be performed to verify that this simplification is reasonable.
- Improved mechanical calculations will be carried out for the steel lid. The calculations will take into account the valve where inert gas is injected, the bolted joint that attaches the steel lid to the insert and the gasket between the insert and the lid. Input data for the calculations will be based on actual stress and strain curves for the material in the steel lid.

As far as the copper shell is concerned, calculations are being performed of the effect of strains (plastic strains and creep strains) for different conditions in the Spent Fuel Repository. The analyses have mainly been done for the weld area using FEM-based calculations.

Research is being conducted on the creep properties of the copper shell, see Section 24.2.3 "Deformation of copper canister under external pressure", which includes both testing and modelling.

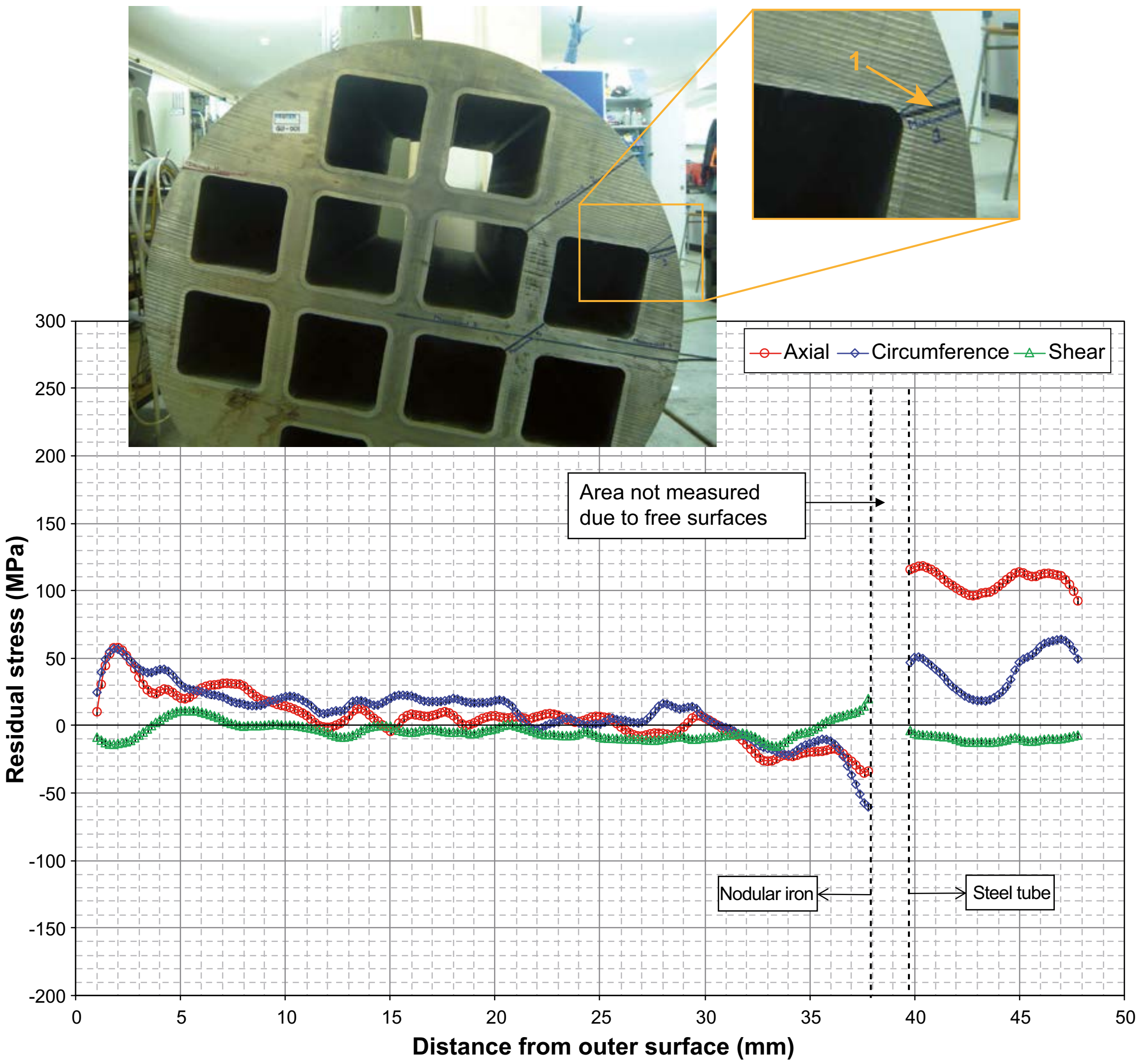


Figure 12-1. *Results of residual stress measurements along the line marked 1 in the above photo.*

Creep testing of test bars with three different typical defects is in progress. The defects – cylindrical, conical and spherical – are deemed to be representative of e.g. defects that occur during handling of the canisters when a lifting tool grips the copper lid. The results will be used to describe the size of permissible surface defects that can occur during handling.

12.4 Manufacturing and testing of inserts

12.4.1 Manufacturing of inserts

Conclusions in RD&D 2010 and its review

SSM concluded in its review of RD&D Programme 2010 that additional inserts need to be cast, preferably at several foundries, to increase the reliability of the material properties.

Current situation

The development work with insert manufacturing has been focused to achieve mechanical properties in the entire insert to conform to the stipulated requirements.

Areas with pore clusters have previously been detected during casting of inserts. Analyses of material from these areas indicate that the cause is related to the presence of moisture. Further development of the manufacturing process in order to prevent the presence of moisture, has resulted in inserts where pore clusters not has been observed.

The manufacturing of PWR inserts has been in focus since RD&D Programme 2010 due to the fact that the manufacturing of BWR inserts is considered to be developed. Simulations of the casting process have been carried out. SKB has used two foundries: one that uses top pouring and a sand mould and one that uses bottom pouring and a chill mould. Both methods are deemed to be usable in the future. In evaluation of manufactured inserts, it has been found that the foundry that uses top pouring and a sand mould has produced inserts with much better material properties due to better process metallurgy. Following this evaluation, three inserts have been cast at this foundry, and the results are currently being evaluated. More inserts will be cast during 2013. Top pouring with a sand mould is the process that has been used in the development of BWR inserts.

Development of manufacturing processes at two foundries will be needed in order to ensure a stable future supply of inserts. This work will be done closer to the start of operation of the Spent Fuel Repository. Posiva is pursuing similar development of the insert in a Finnish foundry and SKB is following this work.

The channel tubes have previously shown a tendency to buckle in the lower part of the insert due to high pressure and high temperature during casting. This phenomenon is avoided nowadays by filling the channel tubes with well-compacted sand before casting and by a minor modification of the design of the cassette of the PWR inserts.

Programme

Further development will be focused on learning more about casting parameters and the material's microstructure, as well as the relationship between these factors and mechanical properties such as fracture toughness.

Current efforts to update and verify the casting process will be completed. The work includes evaluation of previously cast PWR inserts, casting of additional inserts and continued evaluation and documentation.

12.4.2 Testing of inserts

Conclusions in RD&D 2010 and its review

SSM pointed out in its review of RD&D Programme 2010 that SKB should continue its work with nondestructive testing for characterization of defects. It was also pointed out that verification of the testing system needs to be done by means of practical demonstrations of test blocks with realistic defects. The technology for production of defects therefore also needs to be developed.

Current situation

RD&D Programme 2010 presented two techniques for ultrasonic testing: one based on phased array technology with a 0° inspection angle and one based on TRL (Transmitter Receiver Longitudinal) technology with a 70° inspection angle. These techniques have been further developed, and additional methods have been developed.

The phased array technique, which is used for inspection of the area from the insert's envelope surface down to a depth of 200 millimetres, has been improved with an array with a higher frequency (3.5 MHz). Furthermore, focusing of the sound has been optimized to obtain a higher sensitivity for detection of defects along the entire inspection depth. This is illustrated by sound field simulations with the modelling software Civa in Figure 12-2.

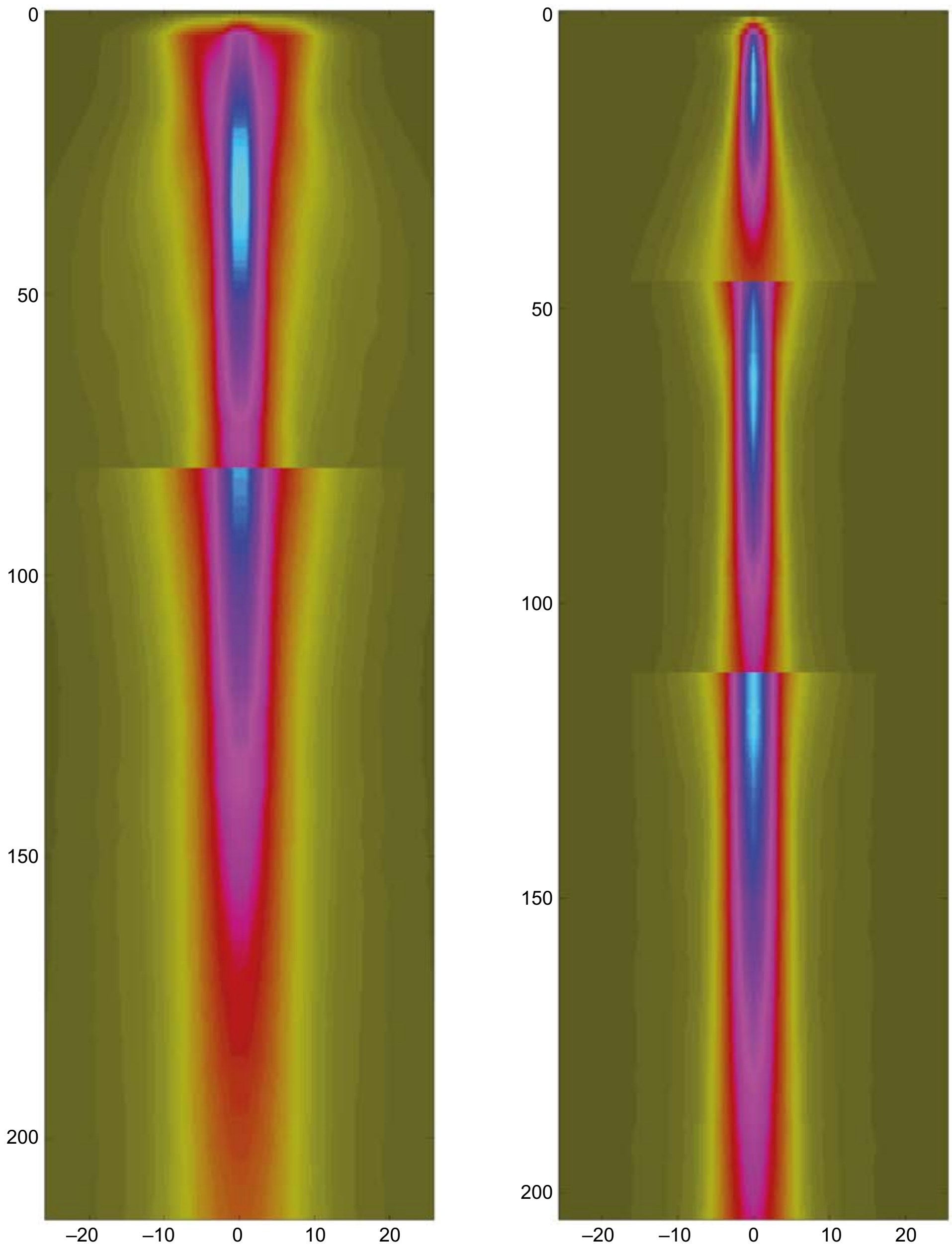


***Figure 12-2.** Results from ultrasonic modelling in Civa. The figures show, with a colour-coded signal amplitude, the width of the sound field in millimetres in the array probe's active direction (x axis) as a function of the inspection depth 0–200 millimetres (y axis). The left-hand figure shows original phased array technology (2 MHz) while the right-hand figure shows optimized technology (3.5 MHz). The optimized technology, which is divided into three focal depths, show a narrower sound field with a more uniform amplitude along the entire inspection depth compared with the original technology, which is divided into only two focal depths.*

Data from the TRL technique is complicated to evaluate, for instance due to the fact that several wave types are reflected from the walls of the channel tubes. A study was therefore initiated to determine whether inspection can instead be done with shear waves. The TRL method was initially modelled in the modelling software Civa, thereafter several different types of ultrasonic probes were modelled. Based on the modelling results, linear arrays were designed adapted to the radius of the insert with mechanical focusing in the passive direction. A comparison of modelling results from TRL and phased array technology is shown in Figures 12-3 and 12-4. Based on the modelling described above, the testing method has been developed for supplementary testing of the area from the envelope surface down to a depth of 50 millimetres. The work has involved several steps: manufacturing of linear arrays that nominally generate shear waves with a 70° incident angle, development of the fixture that holds the probes (see Figure 12-5), definition of focal depth and development of ultrasound settings and inspection instructions.

The technology for testing of the area between the channel tubes, which is the only area where the inspection differs between BWR and PWR inserts, has primarily been developed for inspection of PWR inserts. The reason for this is that this type of insert has been in focus for manufacturing trials recently. The development work has been done with the aid of ultrasonic modelling. The inspection is done using both pulse-echo and through-transmission technology. Figure 12-6 shows the test set-up and Figure 12-7 shows sound field simulations of these techniques.

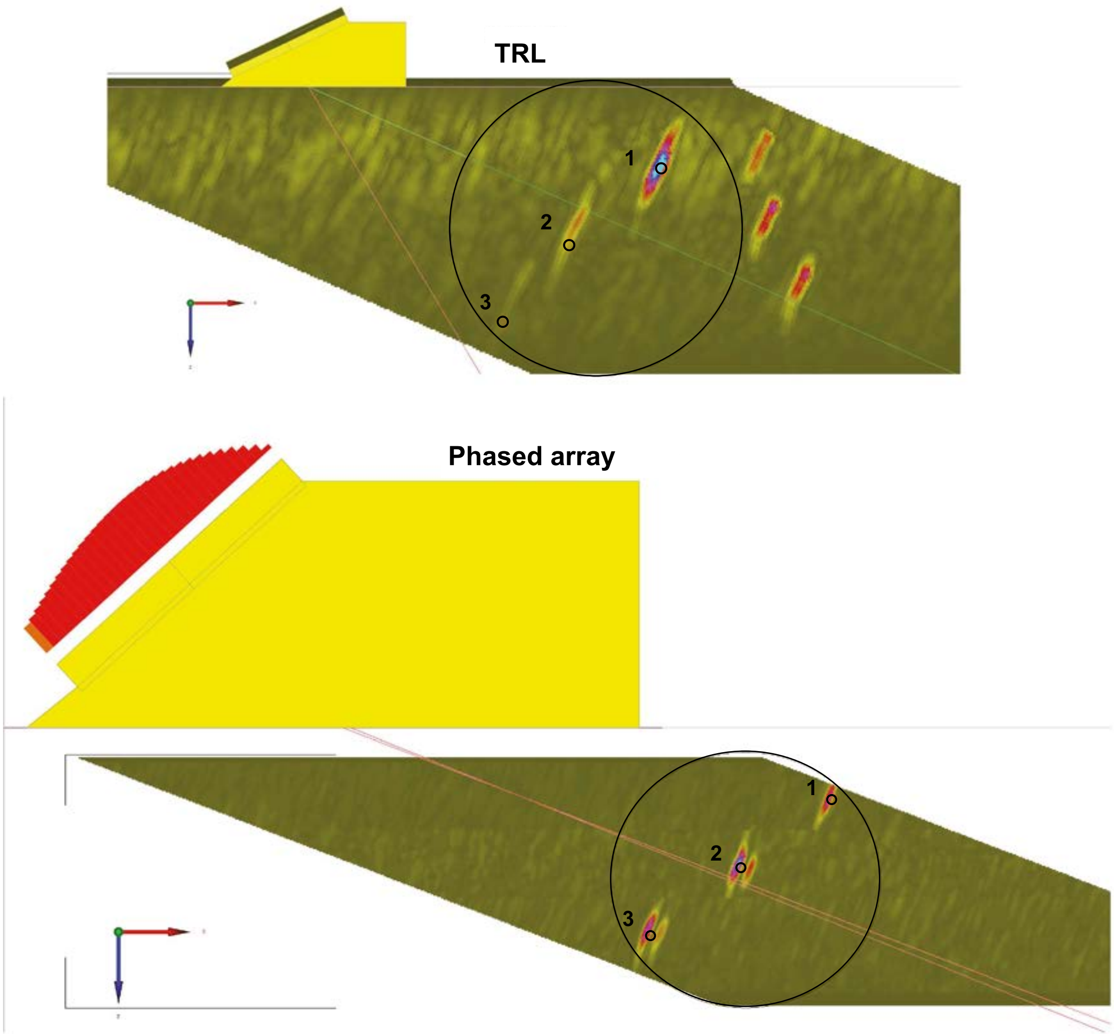


Figure 12-3. *Results from ultrasonic modelling in Civa of inspection of side-drilled holes at different depths (marked 1, 2 and 3). The top picture shows the TRL probe, while the bottom picture shows the results with the newly designed array probe. The modelling clearly shows improved results at greater depth (holes 2 and 3), and moreover the irrelevant indications visible at the right in the top figure have been eliminated by the use of shear waves.*

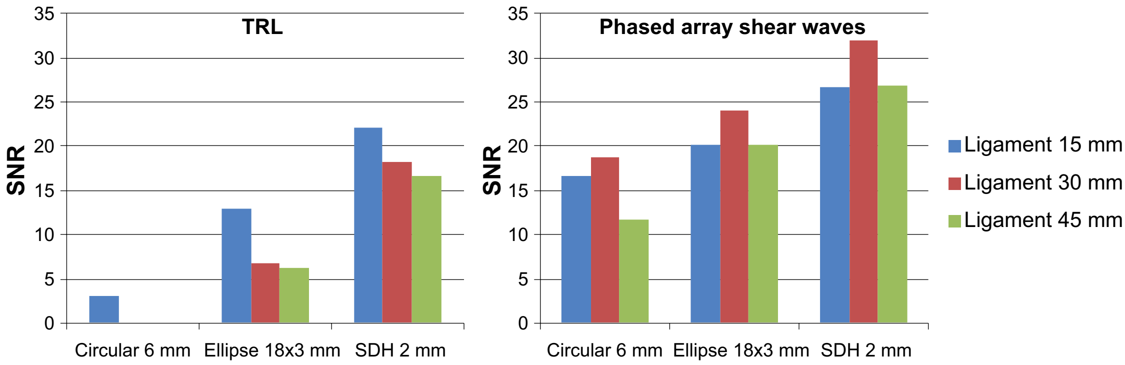


Figure 12-4. *Simulated signal-to-noise ratio (SNR) for different reflectors with phased array shear waves and TRL technology.*

Figure 12-5. *Fixture with four 32-element linear arrays (1.7 MHz) for ultrasonic testing of the outer five centimetres with approx. 70° incident angle with shear waves.*

In addition to the inspection of the main volume, which is done using the ultrasonic techniques described above, a feasibility study has also been done regarding the choice of technology for surface inspection. After initial tests with eddy-current technology and magnetic leakage flux measurement, a study was started to determine suitable parameters for eddy-current testing and test them in full scale. The results indicated that the material's magnetic/electrical properties were uniform, which is a prerequisite for employing the technique, and that defects with a distance to the envelope surface of up to about 0.5 millimetre can be detected.

All described testing methods and magnetic powder testing by the insert manufacturer have been used for testing of a number of inserts. The results show that indications of defects are rare.

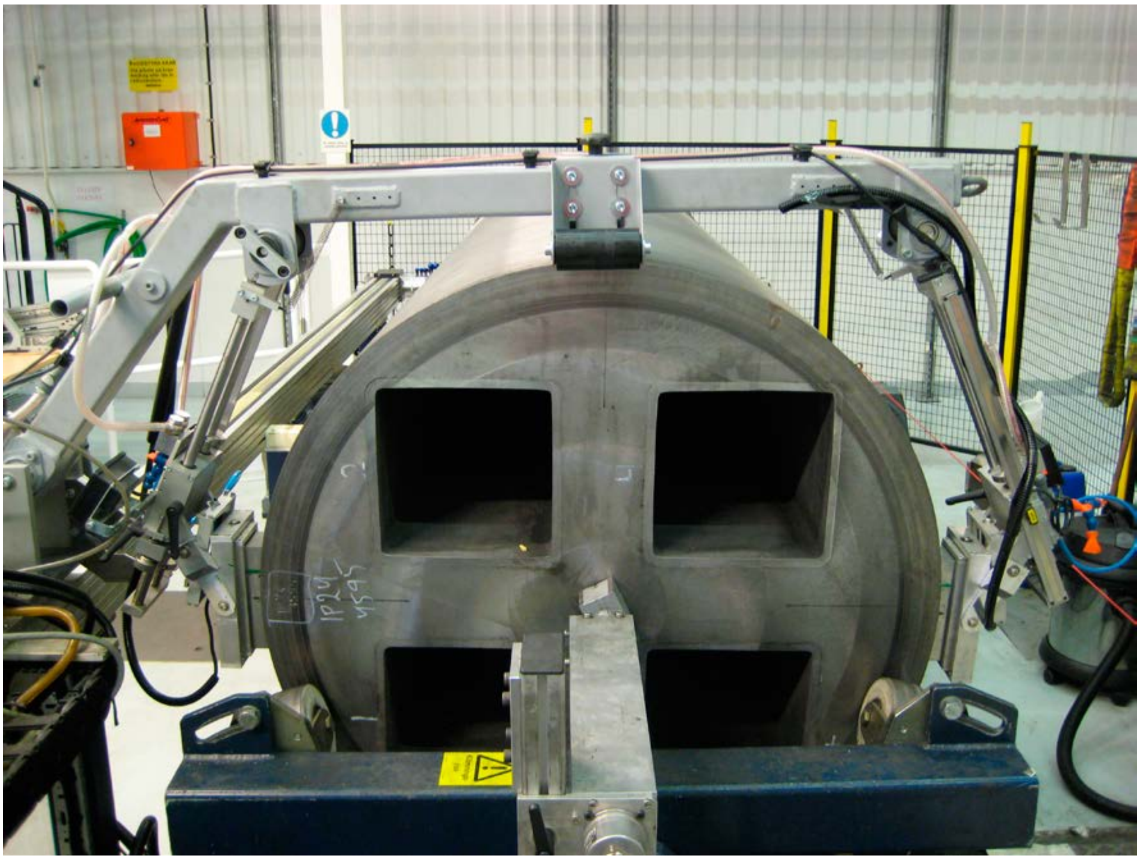


***Figure 12-6.** Ultrasonic testing of the area between the channel tubes in a PWR insert.*

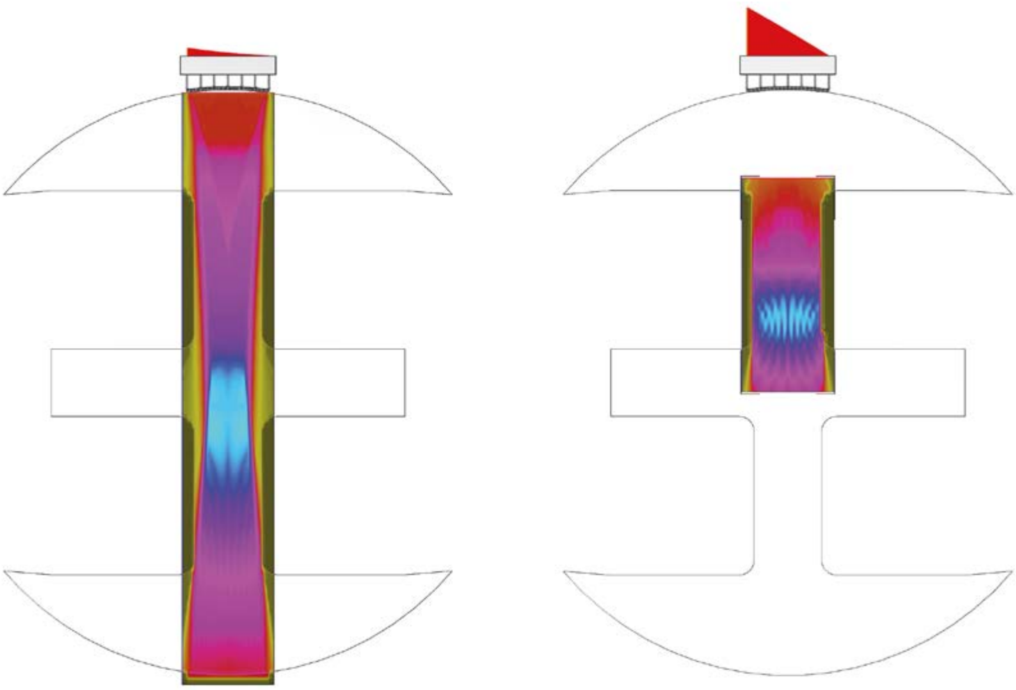


***Figure 12-7.** Results from ultrasonic modelling, presented with a colour-coded signal amplitude, of the ultrasonic technique for testing of the central parts in the PWR insert. The figure at the left shows the through-transmission technique where the focus is deeper than with the pulse echo technique at the right, which has been optimized for the area between the channel tubes down to the centre of the insert.*

By using the developed testing methods for inspection of full-size inserts, we have learned more about the occurrence of defects and how they can be detected by nondestructive testing. Examples of defects that have been detected are individual small pores and slag inclusions, pore clusters, and in some cases generally impure material with a high content of pores and slag. Some of these defect indications have also been investigated by computed tomography (CT scan) and destructive testing. An example of ultrasonic testing results and subsequent computed tomography and metallography can be seen in Figures 12-8 and 12-9.

To find out more about the characteristics of possible defects, a study has been initiated where small test objects (< 100 kg) with casting defects are manufactured in a controlled manner. Suitable casting parameters have been determined by simulations, and casting tools have been designed on the basis of the results obtained. A few test objects have been manufactured, and they are being evaluated during 2013. The objective is to determine whether this might be a suitable approach for making test blocks for future verification of the testing method.

As a step in developing reliable testing systems, studies have been conducted for the purpose of minimizing human factors in mechanized testing. Both theoretical studies and practical experiments have been conducted. They have included a study with adapted FMEA (Failure Mode & Effect Analysis) of both testing and evaluation of data. These studies show that the influence of human factors dominates in the evaluation of data. As a consequence, preliminary inspection instructions have been evaluated both theoretically and in practical experiments. The results of the evaluation have then served as a basis for the formulation of new and revised instructions.

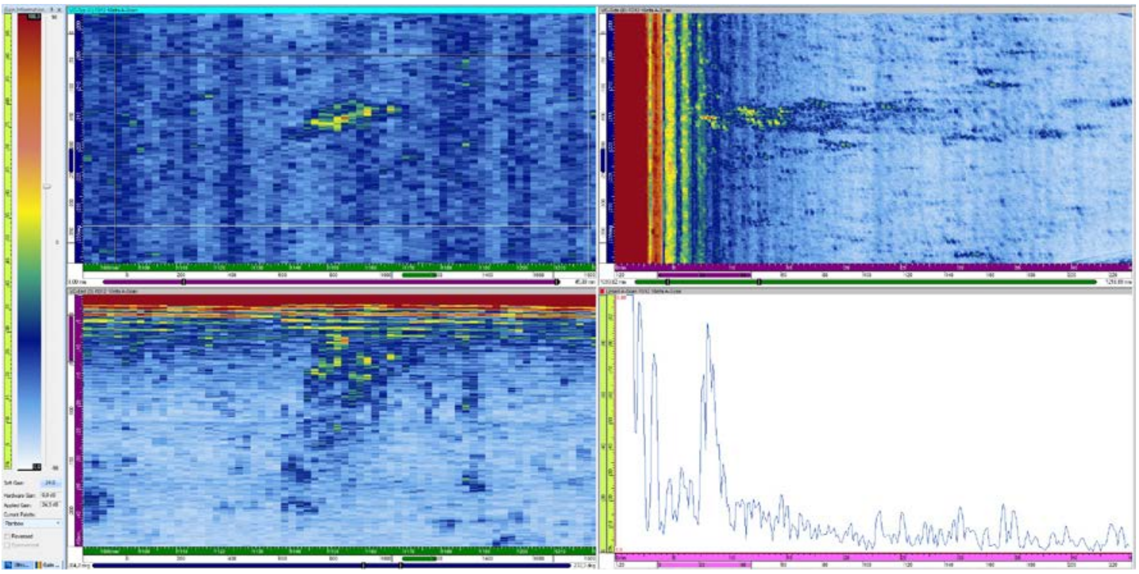


Figure 12-8. *Results from phased array ultrasonic testing of an area with porosity are shown in projections in three directions: a C-scan from the envelope surface (top left), a B-scan viewed from the end surface (top right) and a D-scan viewed along the circumference (bottom left). The maximum ultrasonic signal as a function of depth is shown in an A-scan (bottom right).*

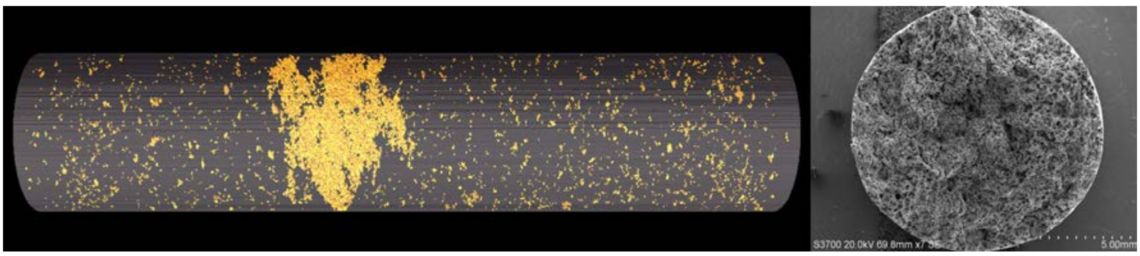


Figure 12-9. *Characterization of porosity indicated in Figure 12-8. CT scan (at left) and fractography of fracture surfaces (at right).*

In addition, an ongoing project for the purpose of developing methodology for determining the reliability of mechanized testing of the canister has continued. The project is being pursued in cooperation with Bundesanstalt für Materialforschung und prüfung in Berlin and the results will be reported in 2013.

Programme

A priority area during the coming period is developing technology for manufacturing of test objects with relevant defects in nodular iron in a controlled and repeatable manner. Based on these test objects, the work of developing testing methods will continue with a focus on characterization and sizing of defects. The work on the technology for characterization of defects will be pursued in close cooperation with the formulation of requirements on the insert. The testing technology will be developed so that detected defects can be characterized with respect to the properties that are important for ensuring the integrity of the insert. Moreover, further development of the technology for testing of the surfaces of the insert and ultrasonic testing of the bottom of the insert is planned.

12.5 Manufacturing and testing of copper components

12.5.1 Manufacturing of copper components

Conclusions in RD&D 2010 and its review

SSM pointed out in its review of RD&D Programme 2010 that the cause of the bands with increased grain size in the copper tubes needs to be investigated.

Current situation

SKB has chosen extrusion as the reference method for manufacturing of copper tubes. The method is robust and fabricated tubes meet the stipulated requirements.

The investigative work on bands of increased grain size and sound attenuation has been finished. Increased grain size reduces the detection capability of ultrasonic testing. The cause of the bands is deemed to be variations in friction, uneven heating and inexact centring in the extrusion process. SKB judges that the effect is possible to handle, see Section 12.5.2, but could result in a supplementary requirement concerning sound attenuation. Continued work will be done to minimize the bands with increased sound attenuation, since it is desirable to have a good margin to possible requirements for future serial production.

Heat treatment of forged copper lids is being studied, and the method has proved to be effective when it comes to removing the effects of cold deformation. The copper material exhibits no structural changes and uniform hardness, even in the internal parts, after the treatment. Hardness measurement has proved to be a suitable method for detecting the occurrence of cold work.

Programme

Pierce and draw processing will be further developed as a method for manufacturing copper tubes. The purpose is to be able to approve this method as an alternative reference method.

A heat treatment cycle will also be adopted for eliminating residual stresses in copper lids and bottoms. The forging process for manufacturing of lids and bases will also be further developed.

12.5.2 Testing of copper components

Conclusions in RD&D 2010 and its review

In its review of RD&D Programme 2010, SSM expressed the opinion that the relationship between increased sound attenuation, average grain size and smallest detectable defect in nondestructive testing of copper components needs to be clarified.

Current situation

In RD&D Programme 2010, SKB described a developed phased array ultrasonic testing technology with a ultrasonic frequency of 5 MHz. This inspection technique has been further developed with an array with a lower frequency (3.5 MHz) for the purpose of improving detectability in the testing of material with variations in grain size. Furthermore, focusing of the sound has been optimized to generate a narrow sound field, providing a higher detection capability along the entire inspection depth. Figure 12-10 shows sound field simulations of the two techniques, where it is clearly evident that the optimized technique has better focusing near the surface.

A study has been initiated to explore the question of how increased sound attenuation due to larger grain size influences the capability to detect defects. In this study, simple reflectors such as flat-bottom holes have been made in copper tubes with differing sound attenuation. In an initial step, the influence of the material's sound attenuation on the signal-to-noise ratio (SNR) has been studied for flat-bottom holes. The conclusion is that more attenuating material has a clear influence on the SNR in ultrasonic testing, which reduces its detection capability, see Figure 12-11.

A known type of defect in forged copper lids is forging laps. This is one of the main reasons why a method for surface testing of the copper components is needed. Different types of eddy-current technology have been studied, resulting in the purchase of new equipment for this, along with an array probe adapted for testing of copper.

Programme

The work of investigating the effect of varying sound attenuation on detection capability in ultrasonic testing will continue during the coming period. This will be done by calculating POD (Probability Of Detection) curves for ultrasonic testing of copper with differing attenuation. The relationship between material structure and ultrasound attenuation will also be studied.

The technique for testing of the surfaces of the copper components with array technique for eddy currents will be further developed. Moreover, ultrasonic technology for characterization and size determination of defects will be developed. In addition, a feasibility study will be launched for the purpose of identifying the need for testing of copper ingots and initiating the necessary technology development.

A feasibility study aimed at developing methodology for producing defects in copper in a controlled manner will also be started in 2013.

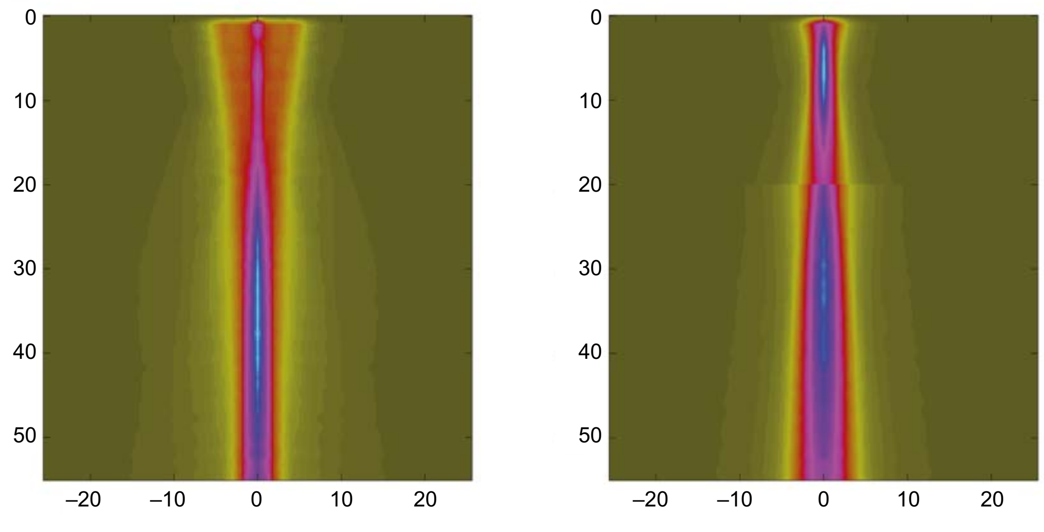


Figure 12-10. *Results from ultrasonic modelling in Civa. The figures show, with a colour-coded signal amplitude, the width of the sound field in millimetres in the array probe's active direction (x axis) as a function of the inspection depth 0–54 millimetres (y axis). The left-hand figure shows original phased array technology (5 MHz) while the right-hand figure shows optimized technology (3.5 MHz). The optimized technology, which is divided into two focal depths, results in a more uniform sound field along the entire inspection depth compared with the original technology with only one focal depth.*

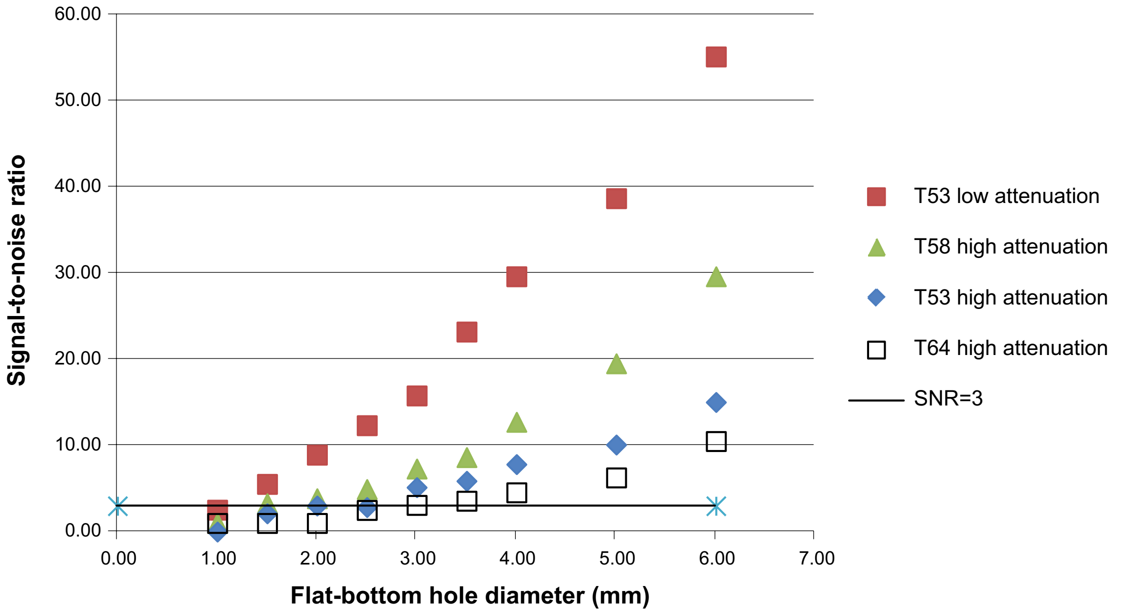


Figure 12-11. *Signal-to-noise ratio (SNR) for flat-bottom holes with different diameters at a depth of 50 millimetres in normal fine-grained extruded material (red) and different levels of higher-attenuation copper tubes (reduction of bottom echo amplitude by 10–18 dB) where the pierced and drawn tube T64 (white) exhibits the lowest SNR.*

12.6 Sealing and testing of the weld

12.6.1 Sealing

Current situation

Since RD&D Programme 2010, development of friction stir welding (FSW), illustrated by Figure 12-12, has focused on fully automating the procedure. A controller keeps the tool temperature between 790 and 910°C, since wormhole defects are formed below 790°C and there is a risk that the welding tool will break above 910°C. In order for the tool temperature to remain within this interval, the rotating welding tool must generate a varying power/heat input during the 45-minute-long welding cycle, since the thermal conditions change depending on heating and geometric parameters.

Results of completed welds on copper canisters show that the tool's rotational speed is best suited for controlling the tool temperature. This is a logical result of the fact that the heat generated by the tool is given by the product of the rotational speed and the motor torque required to achieve this rotational speed. The generated power/heat input therefore correlates well with the tool temperature (Cederqvist 2011), which means that the rotational speed can be used to control the tool temperature.

A cascade controller had been developed in June 2010 and verified in short welding cycles. The controller was then judged to be ready for verification in 360° welding cycles. The cascade controller controls both power input and probe temperature and has permitted repeatable welds where the tool temperature around the whole joint line varies by only ±10°C from the desired value. Figure 12-13 shows the results from the eight 360° welds that were done during the verification of the controller. Compared with the permitted process window at about ±60°C, this means that the cascade controller generates welds that have been produced within this process window with ample margin.

RD&D Programme 2010 described the development of the tool probe that has been done to increase the safety factor against fracture. It can be mentioned in this context that the two probes that were used in the verification of the cascade controller lasted for the four full welding cycles they were used for. In other words, the probe has a safety factor against fracture of at least 4 times.

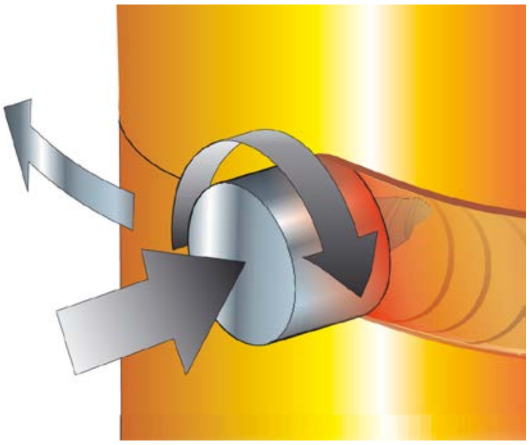


Figure 12-12. Illustration of friction stir welding.

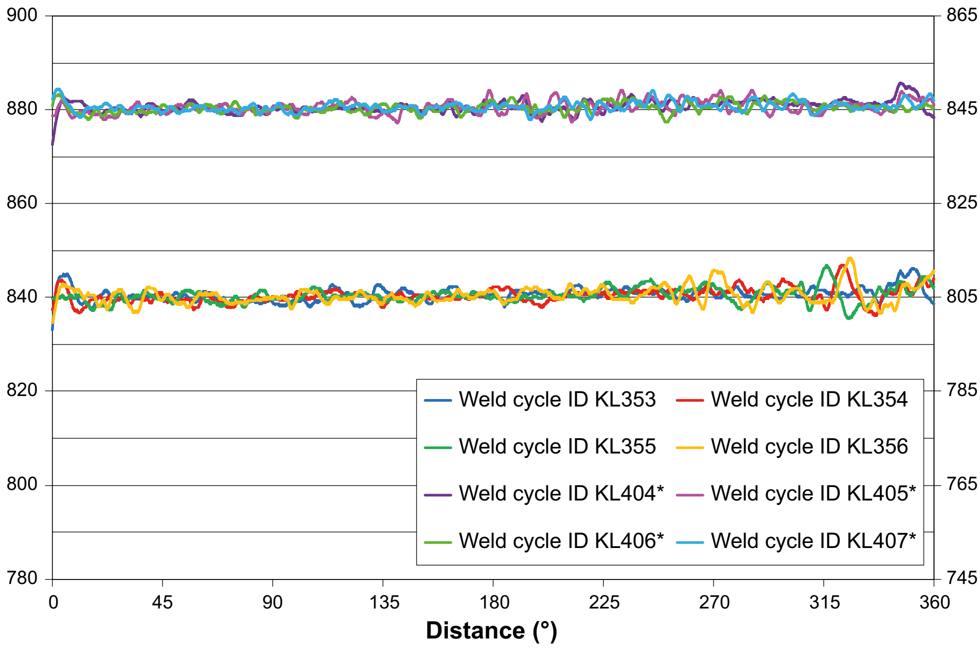


*Figure 12-13. Tool temperature during full (360°) weld cycle. Note that the desired value was either 840°C or 845°C for the eight verification welds. Values on the right-hand axis are indicated by *.*

With regard to joint line hooking (JLH), SSM, in its review of RD&D Programme 2010, called for an investigation of what factors affect the radial extent of this defect. SKB is currently conducting a study of this question. The parameters that influence joint line hooking are the length of the probe, the relative height of the probe relative to the joint line and the depth of the welding tool. The welding tool's depth cannot be varied much, about ±0.5 millimetre. This is because the convex shape of the tool shoulder counteracts changes in depth and the force that controls the depth is kept constant for the present time. Keeping the depth of the welding tool constant prevents the formation of flash and/or disturbance of the process. The length of the probe and its height relative to the joint line have been varied in welding trials. The length of the probe has been varied between 48 and 52 millimetres by one-millimetre increments, and the relative height of the probe has been varied ± 4 millimetres from the centred position.

Figure 12-14 shows results from welding with a probe length of 52 millimetres and a centred height position. In addition to a JLH of one millimetre at (b) in Figure 12-14, a part of the vertical joint between lid and tube contains oxide particles. This part of the joint is displaced upward during welding due to a material flow around the tip of the tool probe. This displacement is illustrated by the arrows at (c) in Figure 12-14. Joint line hooking reduces the canister's corrosion barrier, while the displaced joint does not cause any such reduction.

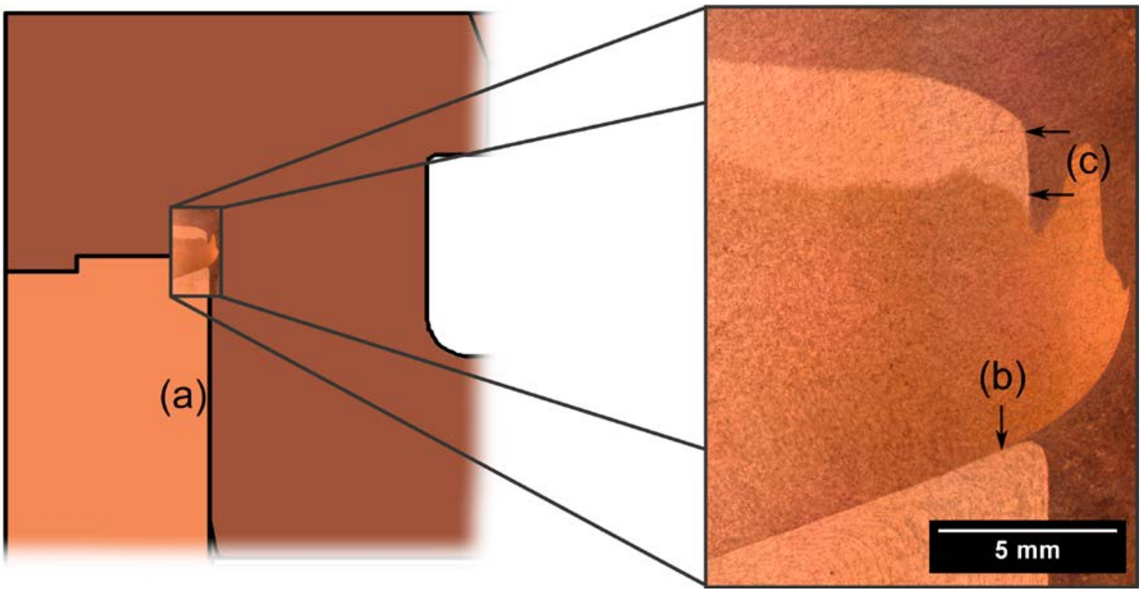


***Figure 12-14.** Joint line hooking (JLH) and displaced vertical lid-tube joint.*

The purpose of this study is to determine how joint line hooking and displacement of the lid-tube joint can be reduced or if possible eliminated, and to investigate how the size and number of the defects are dependent on weld depth and height position.

A research programme concerned with the presence of oxide particles in the weld has been under way since RD&D programme 2010. A number of welds, including four full circumferential weld cycles, have been carried out with a new gas shield around the weld tool and the whole joint line. Welding with this gas shield keeps much of the weld free of oxide inclusions, according to metallographic examinations. Oxide particles occur at certain locations, for example in the unstirred part of the vertical joint between lid and tube, see (a) in Figure 12-14. The areas of the weld where oxide particles mainly remain are below the start hole and also to some extent in the overlap area.

Programme

The initiated study concerning joint line hooking and displaced lid-tube joint that was described above will be finished during the coming period.

The ongoing research programme concerning oxide particles in the weld will also continue. At present, measurements are being made of oxidation at different oxygen concentrations and temperatures in the shielding gas argon. This information is combined with measurements of the temperature history at the joint surfaces to calculate the quantity of oxide formed in the weld performed with different oxygen concentrations in the shielding gas. This information will be used to derive a limit value for permissible oxygen concentration during welding. When the limit value has been set, the gas shield around the weld will be modified to verify that this limit value can be achieved. The produced welds will also undergo mechanical creep testing.

12.6.2 Testing of welds

Current situation

The technology for ultrasonic testing of welds has been further developed since RD&D programme 2010. An array with a lower frequency (3.5 MHz) is used now in order to reduce the effect that varying grain size in the copper lid might have on the detection capability of the inspection. In addition, the focusing of the sound and inspeciton angles have been optimized so that the entire weld volume can be tested with better focusing and with at least two angles.

In order to get as high and uniform detection capability as possible in the whole weld volume, the incident angle during radiographic inspection of the weld has been reduced from 35° to 8°. Furthermore, the past year's welds have been tested with eddy-current technology to acquire more knowledge of near-surface and surface-breaking defects.

Programme

During the coming RD&D period, further development work will be done on the eddy-current method for testing of the weld surface and the ultrasonic method for characterization and size determination of internal defects, as well as to enable the techniques to be implemented in the encapsulation part Clink. Furthermore, studies will be made of how the different testing methods (ultrasonic, radiographic and eddy-current testing) should be combined optimally in order to fulfil the detection goals of NDT.

12.6.3 Industrial-scale manufacturing of canisters

SKB's production system for canisters consists of a number of external suppliers who manufacture the canister components, i.e. inserts, copper tubes and copper lids and bases. SKB intends to build a canister factory where the canister components are assembled and the properties of the canisters are tested. The factory will have equipment for machining of copper components and inserts and equipment for nondestructive testing, friction stir welding and assembly. A feasibility study is under way to define further development work on the canister factory. SKB is collaborating with Posiva in the matter, since a joint canister factory could entail synergies.

SKB will be responsible for purchases of inserts and copper components as well as quality management of the entire production system. The work of quality management has been initiated and a feasibility study is being carried out during 2013.

12.7 Handling and deposition of canisters in the Nuclear Fuel Repository

Conclusions in RD&D 2010 and its review

In its review of RD&D Programme 2010, SSM stated the opinion that SKB should conduct an analysis of the transportation system's ability to handle return transport of defective canisters. The Authority was also of the opinion that SKB should study how a system for automatic detection of surface damage on the canister's copper shell before deposition should be designed.

Current situation and programme

Return transport of damaged canisters is a part of the retrieval process described in Section 10.3.2. Development of the return transport procedure is a part of development of the retrieval sequence for the Spent Fuel Repository. A study will be conducted of whether the retrieval process leads to changes in requirements regarding handling of the canister.

Detection of surface damage on the canister's copper shell is done at the transloading station in the Spent Fuel Repository. The same type of equipment will be used here as at the surface inspection station in the encapsulation part of Clink. Trials with laser scanning of the surface have been conducted with good results, and preliminary specification of equipment exists. Acceptance criteria for permissible surface damage will be established during the coming RD&D period, after which the work of specifying the equipment will be possible.

13 Technology development of buffer, backfill and closure

The technology development presented in this chapter includes fabrication, handling, installation and inspection of buffer, backfill and plugs for deposition tunnels. Development of the conceptual design of the closure is also included.

Section 13.2 “Current situation and programme” describes the status and overall planning. Sections 13.3 to 13.9 present the results of the work that has been done and plans for ongoing activities and projects.

Section 13.10 describes integration tests for installation of buffer, backfill and closure, and Section 13.11 describes experiments in the Äspö HRL.

13.1 Requirements and premises

13.1.1 Buffer

In RD&D Programme 2010, SKB reported that work was continuing on specifying design premises for the buffer. The results of this work were presented in the production report for the buffer (SKB 2010f, Chapter 2), which is included in the applications for the Spent Fuel Repository. The most important of the design premises reported there and now in effect for the buffer are:

- 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 kilograms per cubic metre (kg/m^3).
- The buffer’s dimensions shall be in accordance with SR-Can (SKB 2006b).
- The temperature in the buffer may not exceed 100°C.

In addition there are requirements related to production and operation. Generally formulated, fabrication and installation of 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.

The design premises for the buffer will be revised as a part of the revision described in Section 10.2.1 “Design premises”. In order to be able to verify that the requirements on the buffer are fulfilled, they will be limited to parameters that can be checked at installation of the buffer. The following design premises are planned to be revised:

- With regard to the dimensions of the buffer, the reference to SR-Can will be removed from the requirement. It will be replaced with minimum dimensions where the greatest permissible gaps between buffer and canister and between buffer and rock wall will also be taken into consideration.
- The requirement on buffer density is planned to be replaced by requirements on minimum and maximum quantity of buffer material and how these quantities may be distributed in a deposition hole at the time of deposition.

In no case does this mean that the conclusions in the SR-Site safety assessment (SKB 2011e) need to be revised other than that they must be even clearer about how verification that the design premises are met is to be done.

13.1.2 Backfill and plug

In RD&D Programme 2010, SKB reported that work was continuing on specifying design premises for the backfill and associated plug. The results of this work are presented in the application, in Chapter 2 of the production line report for backfill (SKB 2010g). The most important of the design premises reported there and now in effect for the backfill and plug are:

- The hydraulic conductivity of the backfill shall be lower than 10^{-10} metres per second (m/s).
- The swelling pressure of the backfill shall be greater than 0.1 megapascal (MPa).
- The packing and density of the backfill, both in an initially dry state and after complete water saturation, must be sufficient to ensure a compressibility that results in a minimum density in the saturated buffer in accordance with the stipulated conditions (i.e. 1,950 kg/m^3) with sufficient margin to allow for loss of backfill and uncertainties.
- The primary requirement on the plug is that it should resist the groundwater pressure and the swelling pressure exerted by the backfill in a deposition tunnel. The plug design should also prevent water transport out of the deposition tunnel.

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.

The design premises for the backfill will be revised as a part of the revision described in Section 10.2.1 "Design premises". The following design premises are planned to be revised:

- Since the inflows into the deposition tunnels can vary, the possibility of linking the backfilling method to the actual measured inflow pattern in the deposition tunnel. The inflows for which a particular method may be used shall thereby be specified for each backfilling solution. A similar change is planned for the requirements on inflows to a deposition tunnel, see Section 14.1.
- The required watertightness of the plug will be specified.

In no case does this mean that the conclusions in the SR-Site safety assessment need to be revised other than that they must be even clearer on how fulfilment of the design premises can be verified.

13.1.3 Closure

In RD&D Programme 2010, SKB reported that work was continuing on specifying design premises for the closure. The results of this work are presented in the application, in Chapter 2 of the production line report for closure (SKB 2010h). The most important of the design premises reported there and now in effect for the closure are:

- 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 in repository.
- Below the level of the top seal, the integrated effective connected hydraulic conductivity must be lower than 10^{-8} m/s. This applies to the backfill in tunnels, ramp and shafts and in the excavation-damaged zone (EDZ) surrounding them. This value need not be upheld in sections where e.g. the tunnel or ramp passes highly transmissive zones. There is no restriction on the hydraulic conductivity in the central area.
- Boreholes, drilled both from the surface and from openings in the repository, must be sealed. A preliminary assessment is that this can be achieved if the borehole seal has a hydraulic conductivity that is less than 10^{-8} m/s. Higher hydraulic conductivity is accepted in sections where the borehole passes zones with elevated transmissivity.

The design premises for the closure will be revised as a part of the revision described in Section 10.2.1 "Design premises". The following design premise are planned to be revised:

- Based on completed sensitivity analyses, the possibility of changing the requirement on maximum accepted hydraulic conductivity in sealed boreholes is being considered. The advantage of accepting higher hydraulic conductivity is that it permits solutions for borehole sealing where it may be simpler to demonstrate long-term durability.

13.2 Current situation and programme

Since RD&D Programme 2010, SKB has submitted applications for the Spent Fuel Repository. The reference design is described in the production line reports for buffer (SKB 2010g), backfill (SKB 2010g) and closure (SKB 2010h). The need for further development and tests described in RD&D programme 2010 was identified in the work with the production line reports. This work is now under way and will essentially be finished and reported during 2013.

It was observed in RD&D Programme 2010 that there are a number of development questions that span the whole buffer-backfill-closure system. The following work was identified:

- Dismantle 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 buffer installation, 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.

Figure 13-1 illustrates the production line for buffer (SKB 2010f).

The supporting material presented in the application described conceptual designs of buffer, backfill (including arched plug) and closure. Continued technology development has been pursued since then to start the work on system design for buffer, backfill and plugs in deposition tunnels according to the delivery control model, see Section 10.2.3 "Delivery control model".

The plug in the outer section of the Prototype Repository was dismantled at the end of 2010, and the backfill and the buffer and canisters in the two deposition holes were retrieved during 2011. The results of the field work and the laboratory programme for the outer section are being compiled and evaluated during 2013. The results will be reported at the end of 2013 in a technical main report with references to a number of background reports.

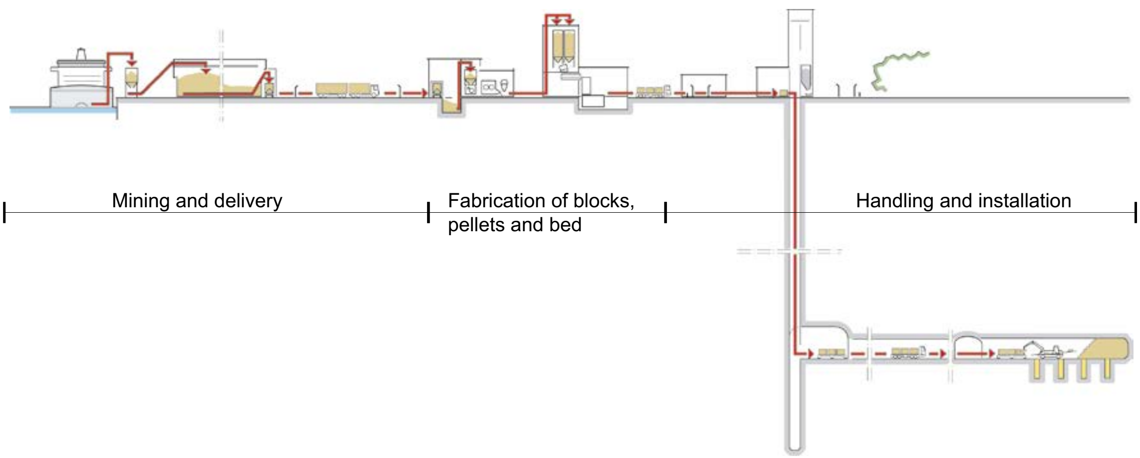


Figure 13-1. *Production line for buffer (SKB 2010f).*

The following preliminary conclusions can be drawn from the mining of the Prototype Repository's outer section and the analyses of backfill and buffer:

- When excavated, the backfill was completely water-saturated in all parts of the tunnel.
- No signs of piping or erosion of the backfill were observed during the excavation.
- As expected, the initial gaps between the canisters and the buffer blocks did not remain at the time of retrieval.
- Upward swelling of the buffer has occurred during water uptake, which has resulted in reduced density of the buffer in the upper parts of the deposition holes.
- The buffer in the two deposition holes was not water-saturated in all parts after eight years of natural wetting. Good contact between buffer and rock was observed during the mining.

Further development work has been pursued on the concept for closure of the final repository. Studies and development work have been done on alternative reference designs in order to evaluate the possibilities of optimizing the design of buffer and backfill from a production perspective without significantly affecting the long-term function of the repository.

Work was begun during the past RD&D period to address the material issues and analysis methods needed prior to the start of detailed design of the production building.

Continued development is being pursued of the reference design for backfilling of deposition tunnels. The focus will be on the inflow of water and on providing sufficient constraint to restrict the vertical expansion of the buffer.

During the coming RD&D period, SKB will commence the activities relating to quality control and inspection that are described in Section 10.2.2 "Quality control and inspection".

Integrated tests of installation of buffer and backfill are also planned during the period. In a subsequent step after the coming RD&D period, full-scale installation tests are planned for buffer, backfill and plug, see Section 13.10 "Integration tests".

13.2.1 Buffer

Conclusions in RD&D 2010 and its review

It was observed in RD&D Programme 2010 that 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.

It was further observed that it should be possible with conventional technology to further reduce the range of variation of density and material composition, which is deemed to be advantageous from a safety viewpoint, even if no formal requirements have been specified. According to SSM's review, SKB should prepare a quality programme for buffer manufacturing. The technology development programme should, besides compaction, include choice of material, verification of mineral composition, check of density, water ratio and homogeneity of blocks and rings. SSM further concluded that a relatively large number of full-scale blocks and rings need to be produced. The reason for this is to provide a statistically meaningful basis for determining whether homogeneous blocks and rings of the right density can be manufactured without unacceptable damage.

According to SSM's review, it is difficult to judge the feasibility of the new deposition method with a rubber sheet (SKB 2010f) until a sufficient number of full-scale tests have been completed. SSM was further of the opinion that SKB should specify whether the buffer protection will be used in all deposition holes or whether its use will restricted to cases where it is needed due to the water inflow into the deposition hole. Further, SKB should state whether the buffer protection can be used in deposition holes where spalling has occurred. SSM was positive to SKB's plans to further develop the components used during installation the buffer: the bottom slab, the buffer protection, the radiation shield and the cover plate. It was also encouraging that SKB intended to conduct trials in the Äspö

HRL where all components in a deposition hole are tested together. SSM agreed with Östhammar Municipality that a reference design should be developed with respect to other components (besides buffer) included in the deposition sequence.

Summary of current situation and programme

The production report for buffer (SKB 2010f) was compiled as a part of the application. Many of SSM's comments on RD&D programme 2010 were addressed in the report. In addition to the work described in the production line report, system design of the buffer has been initiated. It includes devising a better design of the bottom pad and the buffer protection.

In addition to the work of developing the reference design for the buffer, separate studies and activities have been conducted for an alternative design of the buffer. The conclusions from this work are presented under Section 13.6.1 "Installation of buffer".

During the coming RD&D period, work will be done on improving the reference design. At the same time, studies of alternative designs will continue. Based on this work, SKB will choose to proceed with the current reference design or switch to an alternative design. A system design will be carried out for the selected reference design. This will include full-scale tests underground.

Further development work within the buffer line has the following goals:

- Develop the design and the technology for installation of the buffer within the framework of the selected 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.
- Further develop the method and design the equipment for manufacturing (pressing), machining and handling buffer blocks so that this is ready when system design of the production building commences.

13.2.2 Backfilling

Conclusions in RD&D 2010 and its review

It was observed in RD&D Programme 2010 that development of the backfill had passed the concept phase and that the expected very low inflows of groundwater in Forsmark should permit a simple installation of the backfill.

The production line report for backfill (SKB 2010g) shows that the selected reference design for the backfill is technically feasible to achieve and that it fulfils the stipulated design premises.

In its review of RD&D Programme 2010, SSM was of the opinion that the requirement linked to the barrier function of restricting the upward swelling of the buffer in the deposition hole should be clarified. The absence of a quantitative requirement is an impediment to the formulation of design criteria and inspection during fabrication, handling and installation of the backfill.

SSM was positive to much of the work that had been done by SKB and Posiva. SSM thought that SKB needed to be open to further modifications of the concept for backfilling of deposition tunnels. The backfill concept has been modified several times when new premises have been introduced or when the concept has been analyzed more thoroughly.

SSM found that further efforts are required to show that the backfill in a deposition tunnel can be installed so that all requirements on long-term safety are fulfilled. SKB should moreover study the difficulties that can occur during installation of the backfill, such as changes in its density due to the irregular geometry of the deposition tunnel and piping caused by groundwater inflow.

Summary of current situation and programme

Further development and system design of the backfill within the selected reference design have been pursued since 2010. A concept has been devised for the handling process and equipment needed for installation and most of the prototype equipment has been designed. The size of the backfill blocks and the stacking pattern have been further developed compared to the reference design. The concept is based on the use of an industrial robot mounted on a mobile platform. With the aid of a specially designed lifting tool, the backfill blocks are moved and emplaced one by one according to a specific pattern. Preliminary results show that the chosen stacking pattern is so stable that the pellet bed on the tunnel floor does not have to be compacted.

A preliminary report has been published that presents defined requirements on backfill blocks, pellets, minimum block filling degree, tunnel geometries and installation. Furthermore, a preliminary quality programme for fabrication and installation of backfill is presented.

Both pellets and blocks have been manufactured on a large scale, providing experience for industrialization of the process.

A number of alternative methods have been identified for dealing with inflowing water during installation.

A trial with installation of backfill in the Äspö HRL is scheduled for early 2014. The purpose of this trial is to ensure that the envisaged method for backfilling and inspection works as intended. Based on the trial, the reference design will be evaluated and modified, if necessary. During the coming RD&D period, the system design of the backfilling process will be finalized. In addition to the standardized backfilling process for relatively dry tunnels, a special emphasis will be placed on different methods for handling water inflows. This will include further tests on a laboratory scale and possibly on a full scale. At the end of the RD&D period, an integrated installation test will be performed of buffer and backfill.

13.2.3 Plug for deposition tunnels

Conclusions in RD&D 2010 and its review

It was observed in RD&D Programme 2010 that development work remains to be done for the design of plugs. This includes what tightness criteria are needed to ensure controlled swelling of the backfill and to prevent "piping erosion" in buffer and backfill.

The continued development work for the plug was described as having the following goals:

- 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.

In its review of RD&D Programme 2010, SSM stated that SKB should continue working on the requirement specification and concept development for plugs for the deposition tunnel. According to SSM, certain additional measures will be needed to show that the requirement on the tightness of the plug can be satisfied. SSM thought that SKB should describe water transport through the plug in greater detail. The tightness of the interface between the plug and the tunnel wall should be better elucidated. Furthermore, SSM thought that SKB should study the geomechanical properties of the low-pH concrete. SKB's modelling work uses generic parameters that apply to general concrete materials, which may not agree with the parameters of low-pH concrete.

Summary of current situation and programme

The reference design of plugs for the deposition tunnels consists of a system that includes the end zone of the backfill, a filter section, material separators, a bentonite seal and outermost an cast-in-place concrete structure that mainly serves as a mechanical constraint (SKB 2010g).

In the plugs in the Spent Fuel Repository, the concrete part will be made of low-pH concrete instead of conventional concrete in order to avoid the possible negative effects of basic materials on the properties of the bentonite clay. For this purpose, a unique concrete mix design called B200 (Vogt et al. 2009) has been developed to meet the design requirements. This mix design alters the conditions for reinforcement, cooling and grouting compared to the use of standard concrete.

A report (Malm 2012) presents an evaluation of the plug design, with a focus on the concrete structure. The report cites studies and motivation for the choice of design. Another purpose of the report is to present studies and compilations of the loads that act on the plug system in order to shed light on the loadbearing capacity of the concrete plug. The primary function of the concrete plug is to act as a mechanical constraint, but strict requirements on watertightness must also be met, which affects the design of the concrete structure.

Two concept-distinguishing alternatives have been evaluated: a dome-shaped plug (see Figure 13-2) and a long, tapered friction plug. Numerical calculations have been carried out on both designs for relevant load cases. Different geometries have also been studied, as well as concrete structures with and without reinforcement. In summary, Malm (2012) points out the potential in using an unreinforced dome design together with the low-pH concrete B200 as a constraint against both the swelling clay and the groundwater pressure in the deposition tunnel. The advantage of making the concrete dome without any reinforcement is that it eliminates the risk of reinforcement corrosion and cracking related to the reinforcement due to shrinkage of the low-pH concrete. In addition, there are time savings and some cost savings in connection with installation.

When the outer section of the Prototype Repository was opened in 2011, four core samples were taken from the plug's slot, i.e. in the rock-concrete interface. The purpose was visual examination of the result of the contact grouting with silica sol that was done after the plug was cast in 2003. All samples showed good results of the grouting, which will be included in the final report from the Prototype Repository, see Section 13.11 "Prototype Repository". Contact grouting, which is done without cooling of the concrete dome, is needed to obtain an early seal function (before the bentonite seal is water-saturated) and to ensure that the concrete dome is firmly prestressed against the rock when cooling ceases. The planned technology for cooling and contact grouting in the reference design does not differ much from that used for the plugs in the Prototype Repository (Dahlström 2009).

A full-scale test of the updated plug design has been installed in the Äspö HRL. The test will be conducted during the coming RD&D period. This is described under Section 13.8 "Plugging of deposition tunnels".

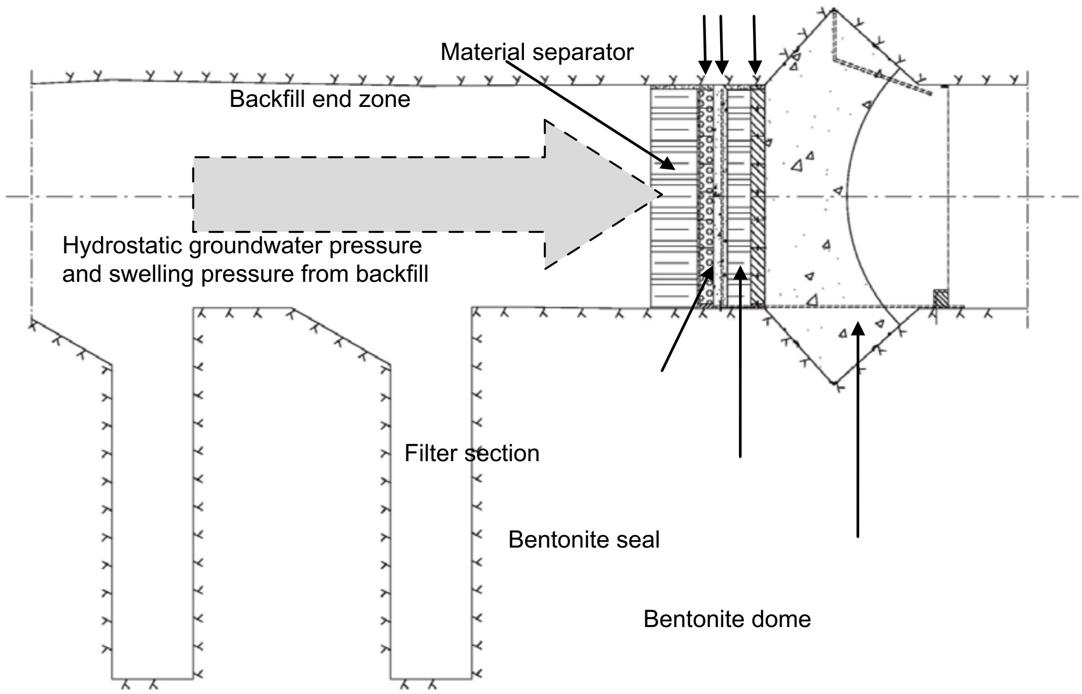


Figure 13-2. Reference design of dome-shaped plug for a deposition tunnel. The concrete dome is made of unreinforced low-pH concrete (B200).

13.2.4 Closure

Conclusions in RD&D 2010 and its review

SKB observed in RD&D Programme 2010 that, with the exception of certain boreholes, repository closure will not begin until all spent nuclear 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 backfill in deposition tunnels. This includes both material composition and geometric configuration. The requirements that need to be set for rock extraction 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 its review of RD&D Programme 2010, SSM took a positive view of the fact that SKB is planning further work on the development of the closure concept. However, SSM said that SKB should further investigate the feasibility of different closure concepts. Closure may be far in the future, but it is important for the Spent Fuel Repository as a whole and thereby for the requirements and technology development for other parts of the final repository. Moreover, SSM was of the opinion that SKB should study the long-term stability of the closure with respect to degradation of concrete and chemical erosion of bentonite caused by exposure to glacial meltwater.

SSM said that SKB should take into account the results from the site investigation and provide a better explanation for the requirements made on the closure. SKB should moreover clarify the role of the uppermost rock material, which extends down to 50 metres depth, in the closure.

Summary of current situation and programme

The application described a reference design where ramp and shafts were filled with blocks and pellets of bentonite up to 200 metres below ground level (SKB 2010h). This planning premise has since been changed, see Section 13.9 “Closure of Spent Fuel Repository”.

The reference design and design premises were evaluated in SR-Site (SKB 2011e). There it is concluded that the design premises that served as a basis for the reference design were adequate. It was further observed in SR-Site that sensitivity analyses where the hydraulic properties of the backfilled shafts and ramp are varied would be required to relax the design premises. It was further concluded that the reference design could presumably be simplified and still comply with the design premises. This means that there are further opportunities for a simplified design.

The reference design for the borehole seal was also evaluated in SR-Site. The evaluation showed that the impact of poorly sealed boreholes was very limited. It was further concluded in SR-Site that the design premises are adequate, but that the requirements are possibly too strict, since open boreholes without seals also appear to have a very limited impact on the groundwater flow in the final repository. This provided opportunities for simplifying the reference design.

Based on the conclusions in SR-Site, sensitivity analyses have been performed where the hydraulic conductivity in shafts and ramp as well as the EDZ have been varied (Luterkort et al. 2012). The level below the ground surface for the seal consisting of bentonite blocks and pellets was also varied.

The conclusion was that the requirements on sealing of both shafts and ramp can be relaxed and that the reference design of the closure can thereby be simplified. With this as a basis, SKB decided that no special consideration need be given to the permeability of the closure between ground level and down to 100 metres above the repository level.

No particular technology development is judged to be needed at present for the closure. In conjunction with the updating of the production line report for the closure, the placement of mechanical plugs etc will be described in greater detail. Their placement is dependent on the water inflow to the tunnels and the order in which the tunnels will be sealed. See also Section 13.9 “Closure of the final repository”.

13.3 Material studies, bentonite

Current situation

Since RD&D Programme 2010, three different shipments of backfill material have been investigated, two shipments of sodium bentonite from Kutch in India and one shipment of a natural calcium bentonite from Milos in Greece. A number of different material parameters have been investigated, which has led to greater knowledge of the material.

The content of montmorillonite in the backfill material must be sufficiently high for the backfill to satisfy the requirements on hydraulic conductivity and swelling pressure. How high the montmorillonite content should be to satisfy the requirements is currently being studied.

The strength of the backfill blocks is dependent on parameters such as material composition, water content, density and granule size distribution. It is important to have control of these parameters order to be able to produce blocks of good quality.

Programme

In order to ensure that methods and knowledge are available to specify, analyze and verify the properties of buffer and backfill material, SKB will carry out technology development in these areas during the next few years. Different suppliers' bentonite quality and their systems for guaranteeing this quality in their shipments will be reviewed. Both parameters of importance for long-term safety and parameters of importance for a robust production of buffer and backfill components will be examined. This includes looking at how properties such as granule size distribution and pressing technology influence the properties of the final product. An inventory will be made of bentonite suppliers. See also Section 25.4.4 "Bentonite composition" and 25.4.5 "Montmorillonite composition".

13.4 Production of buffer and backfill components

13.4.1 Production of buffer blocks and rings

Current situation

Since RD&D Programme 2010, approximately 50 blocks have been produced by uniaxial pressing, see Figure 13-3. These blocks have been produced in a new mould with a larger outside diameter than before. The blocks have then been machined.

Due to the limitations of existing presses, blocks with a maximum height of 500 millimetres can be pressed today. It is not possible today to compact rings according to the reference design where the blocks are 800 millimetres high. Work with modelling and scale tests is therefore being pursued to investigate the possibility of compacting rings according to the reference design. If it proves to be preferable for practical reasons to use blocks of a lower height, the reference design can be changed.

Work is under way to design the process for production of buffer blocks. Figure 13-4 shows a schematic illustration of the process as an example.

The possibility of producing completely cylindrical blocks that do not need to be machined has been studied and a schematic proposal for a press mould has been presented.

Programme

Small scale block pressing tests without machining of blocks are ongoing.

Development of isostatic pressing is being conducted by Posiva. They have estimated that they will be able to produce test blocks on a full scale in 2013. SKB and Posiva will then jointly study if there is any difference between buffer blocks manufactured by isostatic pressing and uniaxial pressing that could affect their function in the final repository. Proposals for design of the production plant will be prepared for both uniaxial pressing and isostatic pressing. They will serve as a basis for selection of a production method. Selection of the production method is planned to take place during 2014.

Figure 13-3. *Picture from manufacture of buffer rings.*

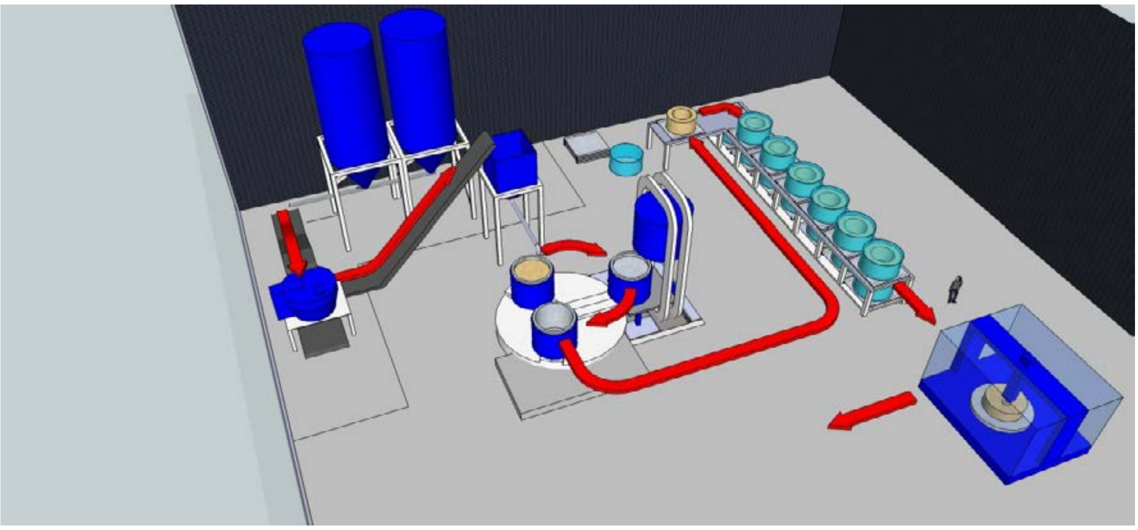


Figure 13-4. *Schematic illustration of manufacturing process for buffer blocks. The picture describes the flow of the bentonite from the left-hand corner in the sequence: from silo to mixer, transport and filling of press mould, transport to press and dispatch of finished block.*

The compaction properties of the bentonite material have been found to vary between shipments. The material (MX-80) that has been used during 2012 is much more easy to compact than bentonite from previous shipments and its bulk density has increased. SKB will investigate the reasons for this.

13.4.2 Production of backfill blocks

Current situation

A large number of blocks of the size that will be used for installation trials in the Äspö HRL have been manufactured. Different water ratios, compaction pressures and clays have been tested. Brittle blocks have been obtained in some tests with coarse-grained material. It will therefore be investigated how granule size distribution and other material parameters affect block quality.

Programme

So far the blocks have been manufactured manually, which is slow and limits the production rate. In order to enable larger quantities of blocks for the large-scale trials to be manufactured, the process must be automated. Automation makes it similar to the envisaged production process in a future production building. This provides experience that makes it possible to determine what problems might occur in an automated production line and what capacities can be anticipated. Production of the approximately 400 tonnes of backfill blocks that are planned for the full-scale backfilling trials will also yield information on what statistical variations in dimensions and density can be expected. A proposal has been prepared for the equipment needed to produce blocks and pellets for the final repository. This equipment iscommercially available and is deemed to have sufficient capacity. Reliability and the ability to satisfy requirements on tolerances and capacity will be important evaluation parameters for the design of the production plant.

13.4.3 Production of pellets for buffer and backfill

Current situation

Regarding the backfill, development work has been done to produce a pellet fill with good water absorption capacity. Extruded pellets have been found to work best for the backfill.

Considering the buffer, roller compacted pellets are used, but no special adaptation has been done for this application. Work is under way to investigate how the different types of pellets behave in the gap between rock and buffer during early water saturation.

In the Bentonite Laboratory, extruded pellets have been produced, see Figure 13-5. When it comes to production of roller compacted pellets, a small number of fabrication tests have been performed.

Programme

SKB plans to purchase a pellet press for the Bentonite Laboratory for the fabrication of pellets in large-scale trials. Work will be done during the coming RD&D period to design the buffer pellets.

Figure 13-5. *Manufacture of pellets in the Bentonite Laboratory on Äspö.*

13.5 Handling and transport of buffer and backfill

Transport of bentonite for buffer and backfill from mine to deposition hole and deposition tunnel is described in SKB (2010f, g). The equipment that requires special development efforts is described under Section 15.2 "Technical systems", "Current situation and programme".

Requirements for handling of the blocks and on the capacity of the transportation system will also be defined. SKB will use the results as a basis for compilation of a quality plan for buffer and backfill.

How long backfill components and buffer components can be stored without packaging under different conditions will need to be studied, as well as which packaging is the most suitable from a handling and storage viewpoint.

13.6 Installation of buffer and backfill

13.6.1 Installation of buffer

SKB is developing the installation method for the reference design of buffer, but is also pursuing development work for alternative installation methods. The results from the dismantling of the outer section of the Prototype Repository in the Äspö HRL will also provide a basis for improving the installation method.

Current situation

As a part of the development of the chosen reference design, tests are being conducted of important components related to the installation of the buffer.

According to the reference design, a buffer protection is used during the installation of the buffer. The protection is removed before the outer gap is filled with pellets and the tunnel over the deposition hole is backfilled. This means that certain deposition holes will stand with installed canister and buffer with a buffer protection for up to three months. During this time some redistribution of water may occur in the buffer blocks, which can lead to cracking of the blocks. Previous scale tests have been performed either with the expected temperature drop over the buffer blocks or with the expected temperature gradient. The goal of these investigations is to conduct tests with full-scale buffer blocks and with the expected temperature drop over the bentonite to study possible effects on the buffer.

After the buffer protection has been removed, the outer gap between buffer blocks and the walls of the deposition hole is filled with pellets. After that the buffer will absorb water from the surrounding rock. Previous investigations show that deformations (heave of the buffer blocks) occur under these conditions (Sandén and Börgesson 2010). The tests show that the deformations are dependent on the water inflow rate and the type of buffer blocks and pellets. The previous tests were performed with compacted pellets of MX-80 bentonite and with blocks that were uniaxially compacted. The goal of the new tests (see Figure 13-6) is to study heave of buffer blocks and pellets when other types of pellets are used (extruded pellets). Tests will also be performed with buffer blocks that are isostatically compacted. The tests will provide information on possible erosion and heave of the buffer blocks during the period when the outer gap is filled with pellets but the backfill above the deposition hole has not yet been installed.

Further development of bottom pad, buffer protection, drainage system and climate lid for installation of the buffer is under way. Equipment for buffer installation has been refined and tested in the Bentonite Laboratory, see Section 15.2.4 "Transportation system for buffer and backfill material".

Alternative solutions for design and installation of the buffer are being developed. This is being done to obtain a more industrialized design of the buffer without affecting the long-term function of the repository in a negative way. Focus areas have been installing buffer and canister without buffer protection and counteracting spalling. Final reporting of this work is currently in progress.

A large number of alternatives for design and installation have been identified. Non favourable alternatives were eliminated in an initial screening and the remaining ones are being further studied. In the preliminary conclusions from this work two favourable alternatives were identified.

Figure 13-6. *Experimental setup for test of buffer heave. The picture at left shows the whole test setup. The top right-hand picture shows bentonite block and pellets at start of the test, while the lower picture is from mining of the test.*

Programme

It will be studied if the design of the bottom pad can be improved to permit easier installation. This will include studying if it is possible to make the bottom pad entirely of copper and thereby avoid the use of concrete. The buffer protection and climate lid will be further studied to find a more industrialized solution. This will include full-scale tests in the Bentonite Laboratory.

In the development of alternative solutions, one alternative is to coat the blocks with a thin layer of a suitable material to prevent moisture absorption and drying-out. The second alternative is to control the humidity in the deposition holes by means of a dehumidifier. These two methods will be studied more closely and compared with the current reference design.

13.6.2 Installation of backfill in deposition tunnels

Current situation

A concept for installation of backfill has been chosen where installation of pellet bed and block stack takes place in short sections. For each section, a bed of pellets is laid out on the tunnel floor, after which backfill blocks are installed by means of an industrial robot. Finally, pellets are installed in the gaps between the block stack and the tunnel wall.

The requirement on installed density is being evaluated in the ongoing work with system design of the backfill. The requirement is affected by the amount of overbreak, the bentonite's montmorillonite content and the dry density of blocks and pellets.

An industrial robot for installation of backfill blocks and pellets has been purchased and block stacking tests are being conducted in the Äspö HRL. A test where the robot will be used for installation of backfill blocks and pellets is planned for early 2014. In further development of the reference design, an uncompacted pellet bed is being studied. In order to verify that the bed constitutes a stable base for block stacking, bed tests have been conducted where concrete blocks have been stacked on the pellet bed. The tests show that an uncompacted bed provides a stable base for block stacking, even above the deposition hole's bevel and with inflow of water to the tunnel (Johnsson 2011).

Water inflow during installation can cause problems. The main function of the pellet fill is to absorb inflowing water and in this way protect the blocks during installation. The pellets are installed as soon as possible after a section with blocks has been installed in the deposition tunnel. Full-scale pellet installation tests are under way to investigate the water-storing capacity of the pellets and estimate the time to water outflow at the backfill front during installation.

During installation of pellets in the gap between the block stack and the tunnel wall, pellets have been blown in with a shotcrete equipment, which has resulted in crushing of parts of the material and creation of dust.

Special methods may be needed to deal with the water inflow from the rock. Use of geotextiles to spread a point inflow of water over a larger area of pellets is being studied.

Optimization of pellet fill

Since RD&D Programme 2010, work on optimization of the properties of the pellet fill has been done, mainly with respect to water-storing capacity and erosion. Different materials and different types of pellets (extruded, compacted and granules) have been studied. Erosion properties and water-storing properties differ between both materials and pellet types.

Of the tested pellet types, extruded pellets with a diameter of six millimetres are recommended, since they had the best erosion properties and water-storing capacity (Andersson and Sandén 2012), see Figure 13-7.

In the current reference design, the pellets are blown into the gap between the block stack and the tunnel wall. This process creates finely divided material, which can affect the material's water-storing capacity negatively.

Installed density and geometric configuration of the tunnel – modelling

Sensitivity analyses have been conducted to refine the design of the backfill and the buffer. They have analyzed how geometry and mechanical properties of the buffer and the backfill affect the upward swelling of the buffer. The results of these studies will be reported in the autumn of 2013.

Figure 13-7. *Six-millimetre extruded pellets.*

Programme

Methods and equipment for installation of pellets in the gap between the block stack and the tunnel wall need to be further developed; among other things, alternative installation technique with a flexible screw feederwill be examined. The effects on the pellets during installation and alternative methods of installation will be further studied. Methods for water handling will also be further studied in continued development and testing. Among other things, temporary plugs that can be constructed to handle large water inflows will be studied.

The reference design of the backfill will be refined in a second step based on the results of current work and the results from the trial with the stacking robot in the Äspö HRL.

13.7 Quality programme, buffer and backfill

During the coming RD&D period, SKB will study and describe quality control, including inspection methods from mine to pressing of blocks and pellets. Quality control and inspection of the installation of buffer and backfill will be developed as a part of the continued development of buffer and backfill.

SKB will set up an inspection programme for buffer and backfill, which includes describing:

- Which inspections are to be carried out.
- Where in the process the inspections are to be carried out.
- How the inspections are to be carried out.
- Who should carry out the inspections (first, second or third party).

An inspection programme shall also be set up for the plug.

13.8 Plugging of deposition tunnels

Current situation

A full-scale test of the complete system for plugging of deposition tunnels is being conducted in the Äspö HRL, at a depth of 450 metres, see Figure 13-8. The experimental setup is included in a joint EU project. In the system design phase for the plug, theoretical and practical studies have been conducted in the field of rock and concrete engineering, along with material tests and modelling of concrete, bentonite and filter materials. Moreover, requirement specifications, risk analysis, technical descriptions and inspection programme are being updated for the plug. The goal of the continued development work is a finished system design for the plug before detailed design of the deposition area commences.

Full-scale test

A principal goal of the full-scale test is to monitor water leakage through the plug over time (at least 36 months). For this purpose a measurement system for leakage monitoring has been developed and the water will be dammed up within a tight atmosphere (plastic sheet) just downstream of the concrete dome and conducted by gravity to a suspended scale for on-line registration of the water flow. The experiment is artificially pressurized by water in the backfill and the filter behind the plug. The pressure is raised incrementally to seven megapascals for the actual tightness test and up to ten megapascals for a verifying strength test. The installation of the full-scale test is shown in Figure 13-9 and Figure 13-10.

The experiment is monitored by a total of about 100 sensors. More than half of the sensors measure the stresses, temperatures and movements in the concrete dome. The rest of the sensors monitor water pressure, total pressure, relative humidity (RH) and movements in the bentonite seal, the filter and the backfill transition zone.

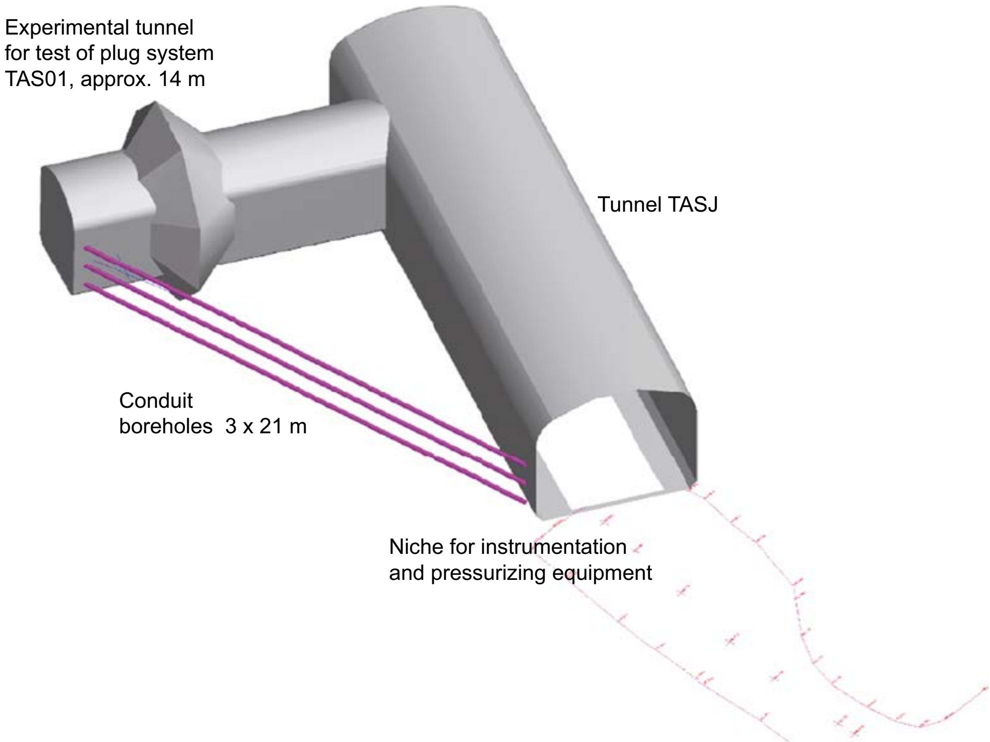


***Figure 13-8.** 3D view of the experimental area for the full-scale test of the complete plug system in the Äspö HRL at a depth of 450 metres. Sensor cabling and pressurizing tubes are run in three conduit boreholes to the instrument niche.*

***Figure 13-9.** Detail photo from the installation of the full-scale test of the complete plug system. The picture shows (from left): filter of Leca beams and crushed rock, drainage pipe (air evacuation), geotextile, bentonite seal with MX-80 clay (blocks and pellets) and concrete beams. All sensor cables are run in steel tubes on the pressure side of the experiment.*

Figure 13-10. *Installation of concrete beams that comprise the inner form wall for the concrete dome in a full-scale test of the complete plug system. A bentonite seal, crushed rock, filter and sensors are installed behind the concrete beams.*

The main pressurization starts after cooling and contact grouting have been done. Contact grouting is carried out when the concrete dome has hardened and finished shrinking (about 99 percent), roughly 90 days after casting. The whole test setup and the pressurizing programme have been preceded by extensive design, whose principal goal has been to reflect the actual conditions expected in the Spent Fuel Repository.

The results of the studies from the first part of system design and experience from the full-scale test will be reported at the end of 2014. Monitoring of the full-scale test will continue at least until 2016.

Rock excavation for plug slot

Plugs should be placed in areas with good and homogeneous rock, which means that no significant deformations can be expected in the margins of the concrete dome due to the compressive loads. The set of requirements for the slot's geometries and its possible need for flat pouring surfaces will be studied later. The purpose is to eliminate stress-related concrete problems from e.g. adhesion between concrete and rock and to be able to guarantee good conditions for contact grouting.

In the full-scale rock excavation done for the full-scale test in the Äspö HRL in 2012, the slot surfaces were made by wire sawing, see Figure 13-11 and Figure 13-12. As a consequence, the slot has a flat-edged shape (octagonal) instead of conical, equivalent to more than 8.5 metres in diameter. Sensitivity analyses have been done to verify that different geometries do not entail a problem for the loadbearing capacity of the concrete dome. Another aspect that is being studied practically is the impact of rock stresses at repository depth on the concrete plug.

Figure 13-11. Inspection of wire-sawn slot for the concrete dome in the full-scale experiment. Left side of experimental tunnel TAS01 (see Figure 13-8).

Figure 13-12. Wire-sawn slot for the concrete dome in the full-scale experiment. Right side of TAS01 (see Figure 13-8). Rock excavation was carried out in 2012.

Tests of low-pH-concrete

When it comes to the low-pH concrete B200 (Vogt et al. 2009), certain parameters are unknown, and some data can at present be classified as uncertain, mainly relating to long-term properties of the concrete material. It will take several years before these questions can be answered. The full-scale casting test that will be done in 2013 is deemed necessary in order to finally be able to verify assumptions and the results of numerical simulations.

An extensive laboratory programme is being conducted involving tests of the low-pH concrete's strength, creep, shrinkage and adhesion. A number of large castings with B200 were done at the Äspö HRL during 2012 to gain experience of the handling steps at the factory, impact during transport, execution of arrival inspections, pumping from concrete mixing truck to form, actual temperature evolution in the concrete, etc. A large number of test cubes and one large monolith of B200 were cast in conjunction with the full-scale test to permit future sampling and follow-up. The concrete mix design will be evaluated and possibly modified as a part of the continued development and testing of the plug.

Modelling and tests of filter and bentonite seal

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. The terminal backfill must reduce the swelling pressure from the backfill down to about two megapascals. One reason for this is that the concrete dome can thereby be shown to conform to the highest safety class for concrete structures. The terminal backfill can consist of either pellets alone or backfill blocks of a lower density.

A laboratory programme has been carried out to verify acceptable material types for the filter (natural and manufactured materials) and for the seal (blocks versus pellets). Further, analytical and numerical modelling has been done in the design work to determine e.g. suitable thicknesses of the material layers and predict what compressions and movements can be expected in the plug system. When it comes to filter materials, the laboratory experiments show that crushed rock of size 2–4 millimetres is the best filter material, but Leca blocks also work well from a filtration viewpoint. The filter in the full-scale test therefore includes both crushed rock (30 cm) and Leca beams (30 cm). With the support of the laboratory experiments, it is recommended that the bentonite seal behind the concrete dome consist of high-quality bentonite clay. This is similar to the buffer's bentonite material MX-80, but has properties that adjust the total swelling pressure for the plug system, counting some compression of pellets and filter, to about two megapascals. In the full-scale test, the plug's bentonite seal consists of MX-80 blocks (50 cm) surrounded by MX-80 pellets in the periphery against the rock.

In parallel with the full-scale test, a test programme is being carried out in a scale model that has been constructed in the laboratory environment. Several short-term tests were performed in the scale model as a part of the design and development of the pressurizing and wetting process for the full-scale test.

Programme

The results of ongoing laboratory tests and the full-scale test will serve as a basis for assessing the development need for the plug in the longer term. Material types and material delimiters, detailed dimensions of components, anchors and connections to the rock, mixing, handling and inspections of concrete and contact grouting are examples of areas that may be able to be further optimized.

According to the plan, the full-scale test will be interrupted after at least three years' data monitoring, i.e. no earlier than 2016. Prior to dismantling, a detailed plan will be made to specify the scope of the sampling and inspection that will be done during retrieval of the plug. It may be of interest to verify sensor data by taking samples in the bentonite clay and possibly also in the concrete dome if interesting events have been noted during the test period. It is further possible to investigate whether the material compressions are as expected and make a detailed study of the interfaces against the rock, whether the contact grouting has had the expected result and whether the clay has sealed as intended.

Further development of the concrete for the deposition tunnel end plug is focused on three themes:

- Rational and cost-effective production of unreinforced barrier structures made with low-pH concrete. The theme includes experience from the full-scale test and data on the properties of the young concrete.
- Developed safety philosophy for unreinforced concrete structures made with low-pH concrete related to the impact of long-term loads. The theme includes shrinkage and creep tests and operating trials on the full-scale test, including associated material tests of monolith casting. Depending on the results, future loading tests may be performed on domes on a small scale combined with more material tests.
- Unreinforced barrier structures of low-pH concrete as temporary structures for 100 years – long-term properties, durability and maintenance needs, if any. The theme includes aspects such as concrete shrinkage and creep related to creep in the rock and the temperature changes that occur with time. Further development of the plug in accordance with these aspects is linked to previously performed experiments. The theme also includes such issues as effects of concrete degradation (this part is being coordinated with other low-pH concrete development within SKB).

During the period 2013–2015 SKB is participating in an EU project called "Full Scale Demonstration of Plugs and Seals" (DOPAS) together with 14 organizations (Posiva, Andra, Nagra, Rawra and others). SKB's full-scale test is included in the collaborative project as one of five large-scale experiments. SKB is also the coordinator of the work of formulating design premises and other requirements for plugs.

It is fruitful for SKB to follow international efforts to identify alternative methods for plugging deposition tunnels, since rock properties and hydrological conditions vary in the Spent Fuel Repository. Even though the plug is designed for the toughest of the design conditions, a generic plug design is not necessarily always the best solution. For example, the plug could be built without filter and seal for use in very dry deposition tunnels. SKB intends to follow, and to some extent participate in, Posiva's full-scale plug test in Onkalo during 2013–2015. The experiment is one of the five large-scale experiments in the EU project.

The interface against the backfill needs to be further studied. Depending on the position of the plug, the backfill stack will connect in different ways. At least one full-scale installation test needs to be executed with backfill and plug in one sequence.

Equipment such as a drilling machine, a drilling platform and a wire saw or similar is needed to excavate the rock for the concrete dome's slot. Standard equipment are used for this equipment, but special development may be necessary in the future to find a production-appropriate solution in relation to encountered rock stresses, etc.

The plugs are a possible future inspection point in the Spent Fuel Repository. Until the bentonite seal is fully water-saturated, the water flow through the plug should be recorded. A measurement system similar to the one used in the full-scale test can probably be developed for the task. When the bentonite seal is water-saturated, the drainage pipes from the filter can finally be plugged from the outside of the concrete dome.

As part of the coming technology development work, data will also be collected to determine whether SKB should build its own concrete mixing station or whether the concrete can be delivered from existing commercial concrete mixing stations.

13.9 Closure of Spent Fuel Repository

Current situation

Based on the conclusions in SR-Site, sensitivity analyses were performed where the hydraulic conductivity in shafts and ramp as well as the EDZ were varied (Luterkort et al. 2012). The level for the competent seal, which consists of blocks and pellets of bentonite, was also varied.

The modelling is based on the hydrogeological models that were developed for Forsmark within SR-Site.

In the hydrogeological groundwater flow model, two different climate situations have been studied: the temperate period shortly after closure of the repository, and glacial conditions when the ice front of an inland ice sheet is situated directly above the repository. The situation when the climate is temperate is believed to be the most favourable scenario, while the ice age represents the worst possible scenario.

The results of the modelling, compared with the results from SR-Site, show that the properties of the seal in the ramp, the shafts and the excavation-damaged zone, EDZ, have little impact on long-term safety.

The modelling shows that the few particles that may leave a deposition hole mainly reach the surface via deformation zones, and only to a limited extent via ramp and shafts. Similar results also apply for investigation boreholes. Only when the hydraulic conductivity is greater than 10^{-6} m/s can some effect be seen on ramp and shafts. Nor is the penetration of oxygen down to the repository significantly affected (Luterkort et al. 2012).

The analyses and assessments that have been done show that the reference design for the closure can be simplified and that the seal can be made more cost-effective without compromising long-term safety (Luterkort et al. 2012). This makes it possible to simplify the design so that the ramp is filled with swelling bentonite clay in the form of blocks and pellets (or only pellets) from repository depth and 100 metres upward. Then the ramp is filled with crushed rock or similar material up to 50 metres below ground level. The last 50 metres up to ground level are filled with stone blocks of varying size. The fill is then injected with concrete grout. The shafts are filled with crushed rock that has been optimized for low hydraulic conductivity, from repository level to 50 metres below ground level. The last metres are sealed in the same way as the ramp.

This alternative is judged to satisfy the criteria for long-term safety. It is also a cost-effective alternative that requires less material transport, which in turn means less environmental impact. Installation of crushed rock instead of swelling bentonite clay in the shafts is judged to be more robust and to entail lower risks in production and installation of the seal. Further studies and tests must be done to verify this alternative.

Based on the analyses performed, SKB has decided to change the level for competent sealing from 100 metres below ground level to 100 metres above repository level. No other changes have been made in the reference design.

Furthermore, the results of analyses and evaluations show that the design premises for sealing of investigation boreholes can be simplified. The following is proposed as a new design premise: The resulting hydraulic conductivity over the length of the borehole shall be lower than 10^{-6} m/s (Luterkort et al. 2012).

Programme

The results of the above analyses do not provide a basis for proposing a new reference design for sealing of investigation boreholes. Alternatives that have been studied include filling investigation boreholes with crushed rock that has been optimized to provide low hydraulic conductivity. These alternatives are not considered to be as technologically mature and proven as the current reference design. Further studies and tests are needed before the reference design can be changed.

13.10 Integration tests

Components, methods and equipment will be tested in steps. This will be done as a part of the iterative development of the reference design of buffer, backfill and plug which includes development of installation equipment and demonstration and verification of the installation process. SKB has gone through a number of cycles in this incremental development process, from the tests in the Stripa Mine and further to tests in the Äspö HRL and the Bentonite Laboratory. The reference designs described in the application constitute the basis for coming refinement of the design and development, production adaptation and implementation of the installation process. Installation of buffer, backfill and plug will first be tested separately. This may lead to improvement or more detailed description of the design, which in turn can provide a better description of the requirements stated for rock excavation, installation equipment and other barriers. In the subsequent step, integrated installation tests of buffer and backfill will be carried outwith further refined equipment and with new specifications for rock excavation. Integration tests have the following main purposes:

- Integrate different equipment and technical systems.
- Test the installation of the barriers to provide input to further development.
- Provide a basis for verification that the installation results in an initial state in accordance with stipulated requirements. This includes methods for manufacturing, installation and inspection.

After this, an evaluation is made of whether the design of buffer, backfill and plug can be further improved before a test that also includes the plug is performed.

13.11 Prototype Repository

The biggest large-scale test in the Äspö HRL, the Prototype Repository (Börgesson et al. 2002, Johannesson et al. 2004) consists of a total of six full-scale deposition holes with canisters and buffer in a deposition tunnel that has been backfilled with a mixture of bentonite (30 percent by weight) and crushed rock (70 percent by weight). 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 are compared with model-calculated predictions. The Prototype Repository, which was installed during 2001 and 2003, is divided into two independent sections, each sealed with an arched concrete plug, see Figure 13-13.

SKB is also participating – together with Posiva, Ciemat and Nagra – in a prototype test with heaters according to a Spanish reference design. This prototype test has been conducted since 1997 in Nagra's test facility in Grimsel, Switzerland. The outer section was mined in 2002 and the inner section is planned to be mined in 2015.

Current situation

Dismantling of the outer section

The plug in the outer section of the Prototype Repository was breached at the end of 2010, and the backfill and the buffer and canisters in the two deposition holes were retrieved during 2011. Approximately 9,700 samples were taken of the bentonite material to determine density and water ratio in the Äspö HRL's geolaboratory. In addition, a large number of samples were taken for laboratory purposes where hydromechanical, mineralogical/chemical and microbiological properties were studied during 2012. Rock investigations, sensor inspections, canister examiniations and modelling were also performed.

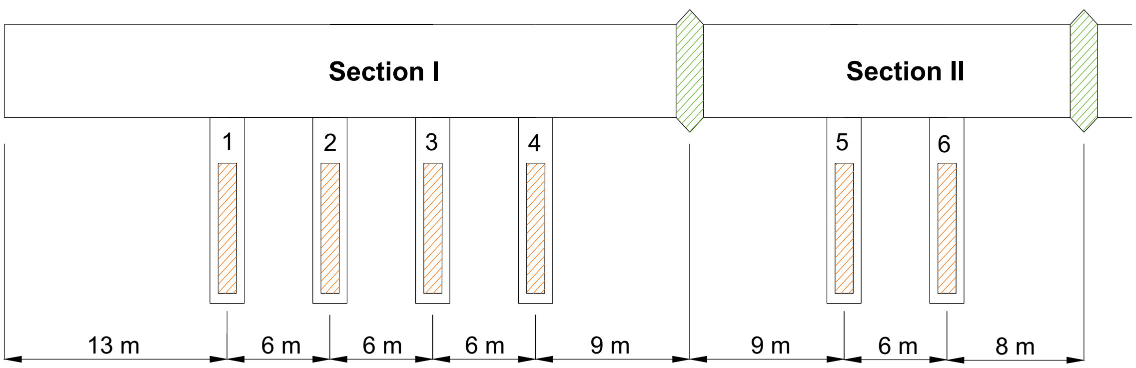


Figure 13-13. *Schematic illustration of the Prototype Repository at the Äspö HRL. The outer section (II) has been excavated, whereby buffer and canisters in the two full-scale deposition holes (nos. 5 and 6) have been retrieved.*

Results

The results of the field work and the laboratory programme for the outer section are being compiled and evaluated during 2013. The results will be reported at the end of 2013 in a technical main report with support of a number of new public background reports.

The following preliminary conclusions can be drawn from the excavation of the Prototype Repository and the analyses of backfill and buffer:

- When excavated, the backfill was completely water-saturated in all parts of the tunnel.
- No signs of piping or erosion of the backfill were observed during the excavation.
- The variations in dry density at installation of the backfill, due to the installation method used, could still be observed after more than eight years of natural wetting. As expected, the initial gap between the canisters and the buffer blocks did not remain at the time of retrieval.
- Upward swelling of the buffer has occurred during water uptake, resulting in reduced density of the buffer in the upper part of the deposition holes.
- Variations in water ratio and dry density were observed between different parts of the buffer, probably caused by variations in the water inflow to the two deposition holes.
- The buffer in the two deposition holes was not water-saturated in all parts after eight years of natural wetting. Good contact between buffer and rock was observed during the excavation. An account will be given of density distribution and degree of water saturation in the reporting of the retrieval.
- Measurements of hydraulic conductivity and swelling pressure for buffer material taken from the deposition holes have been compared with similar results for reference material. These investigations preliminarily indicate small changes in hydraulic conductivity and swelling pressure in the buffer material after eight years' exposure in the field.
- The mechanical properties of the buffer material have been studied by means of both triaxial tests and uniaxial compression tests. In these tests as well, results from field-exposed material have been compared with results from tests on the reference material. The comparisons indicate small differences in mechanical properties between field-exposed material and reference material.
- Studies of the rock around the deposition holes indicate that no damage has occurred to the rock, despite the fact that ticking sounds were noted in the rock via monitoring by Acoustic Emission (AE) during almost the entire operational phase of the outer section. These recorded ticking sounds in the rock have low energy content and are probably caused by microcracking (on a mineral grain scale). The results verify earlier studies of how the properties of the rock are affected in the vicinity of a heated deposition hole.

Functional principles of measurement instruments

A general conclusion of the continuous data follow-up is that the measurement systems seem to work well.

Temperature, total pressure, pore water pressure and water content are measured in the Prototype Repository's backfill and buffer (Collin and Börgesson 2001).

The temperature in the Prototype Repository is measured with the following three methods:

1. Thermocouples.
2. Fibre optics.
3. Resistivity.

Total pressure and pore water pressure are measured in the Prototype Repository with the following two methods:

1. Vibrating wire.
2. Piezoresistive sensors.

The water content in backfill and buffer are measured with the following three methods in the Prototype Repository:

1. Capacitive sensors.
2. Psychrometer.
3. Resistivity measurements.

Of a total of 363 installed sensors (with the exception of water pressure gauges in the rock) in the remaining inner section (Section 1), 261 were out of order (Goudarzi 2012). Many of them (64) are RH (relative humidity) gauges, which stop working at water saturation.

Of the 394 sensors in the outer section (with the exception of water pressure gauges in the rock), 142 were out of order at retrieval. Moreover, some psychrometers (21) located in the backfill had stopped giving relevant values due to the fact that the backfill was water-saturated.

Collection of sensors was not prioritized when the outer section of the Prototype Repository was dismantled, since continuous progress in the sampling was judged to be more important. The sensors that could nevertheless be recovered and found not to have been damaged by the excavator, drilling equipment etc have been collected to check the reliability of the data. The results will be presented in the final report in 2013.

Some sensors in the Prototype Repository stopped working during the wetting phase due to water leakage into the sensor. A possible cause for this leakage is mechanical damage to sensors and tubes during installation and wetting.

Corrosion does not appear to have occurred on sensors and tubes since titanium was used in most cases. Nor has any corrosion been observed on the thermocouples, which are made of cupronickel.

The cables to the heaters in the canisters were found on dismantling of the outer section to have been damaged mechanically as a result of swelling of the buffer. Further, the cable used does not seem to have withstood the environment that prevailed near the canister, with high temperature and high humidity.

Big problems have been observed with sensors installed in the rock for measurement of stress and strain around the Prototype Repository. In principle, no measurements of this type are reliable.

Programme

Section I of the Prototype Repository will be in operation for another eight years or so, according to the original plans. The time for mining was warranted by the fact that the results will be used as supporting material for an application to commission the Spent Fuel Repository.

The results of the dismantling of the outer section, together with experience from ongoing development projects, will serve as a basis for planning of sampling and laboratory programmes for dismantling of the inner section.

14 Technology development, rock

The chapter on technology development for the rock line includes detailed characterization, design, construction and maintenance of the Spent Fuel Repository's underground openings. The development work spans over a wide field and concerns methods for investigations and modelling. It also includes rock construction, including sealing and rock support measures, as well as development of special equipment with a focus on the rock conditions prevailing in Forsmark.

The development activities are aimed at meeting the requirements on the repository's underground openings with respect to long-term safety, occupational safety and efficiency, as well as the current reference design. Specifications for execution of investigations, design and construction are being written to permit verification of requirements and documentation of the initial state of the rock. Detailed characterization is an integrated part of the design methodology and the construction process (rock excavation, sealing and rock support).

A parallel activity is documentation of executed 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).

14.1 Requirements and premises

In RD&D Programme 2010, SKB reported that work was continuing on specifying design premises for the underground openings. The results of this work were presented in the rock line report (SKB 2010i, Chapter 2), which is included in the applications for the Spent Fuel Repository. The most important of the design premises reported there and now in effect for the underground openings are:

- The deposition holes must be located more than 100 metres from deformation zones with a trace length at the ground surface of more than three kilometres.
- As far as reasonably possible, the deposition holes must be chosen so that they are not exposed to greater shear than what the canister is designed for. To achieve this, EFPC shall be applied in the choice of deposition positions, see Section 14.4.3 "Detailed characterization, Large fractures".
- 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} square metres per second (m^2/s).
- Excavation damage should be limited and may not result in a connected effective transmissivity higher than 10^{-8} m^2/s along at least 20–30 metres of the deposition tunnel and averaged over the tunnel floor.
- Below the level of the top seal, the integrated effective connected hydraulic conductivity must be lower than 10^{-8} metres per second (m/s). This applies to the backfill in tunnels, ramp and shafts and in the excavation-damaged zone (EDZ) surrounding them. This value need not be upheld in sections where e.g. the tunnel or ramp passes highly transmissive zones. There is no restriction on the hydraulic conductivity in the central area.
- In order that buffer and backfill can fulfil their barrier functions, requirements are made on the design of deposition holes and deposition tunnels, 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 for ramp and shafts.

The design premises for the underground openings will be revised as a part of the revision described in Section 10.2.1 “Design premises”. The following design premises are planned to be revised:

- The requirements on how deposition holes are to be located with respect distance to deformation zones for major earthquakes may be revised based on results from current research, see Section 26.8 “Reactivation – movement along existing fractures”.
- The requirement that EFPC (see Section 14.4.3) shall be applied to the selection of deposition positions is planned to be modified. Deposition holes shall, as far as is possible, be located so that they are not intersected by fractures with a potential for displacements that can lead to shear failure of the canister. EFPC is a tool for identifying such fractures, but it can be replaced or complemented by other tools. For example, a deposition hole can be accepted even if it doesn’t fulfil EFPC. This can be done if it can be shown by means of other geological characterization methods that the intersecting fracture is smaller than the smallest fracture size with a potential, according to current earthquake analysis, for shear of less than five centimetres.
- The requirements on permissible inflows in deposition holes to ensure that not too much buffer material is lost due to piping and erosion will be revised. This will be done based on the results of current research regarding the piping/erosion process, see Section 25.5.8 “Piping/erosion”.
- Hydraulically based criteria for avoiding deposition holes with high Darcy flows are planned to be determined and need to be developed. It will probably be a combination of maximum acceptable inflow rate, rejection of holes with documented grouting and rejection of holes that are intersected by deterministically modelled conductive fractures on a tunnel scale.
- The requirement on inflow to deposition tunnels in conjunction with installation and water saturation of the backfill is planned to be revised and linked to what installation methods can be used. If there is not an approved installation method for the specific inflows prevailing in a specific tunnel, this tunnel may not be used as a deposition tunnel.
- Only material which, in contact with water, generates a pH lower than 11 is accepted below the level for the top seal. All types and quantities of material brought into the final repository’s underground part must be recorded.

14.2 Current situation and programme

Conclusions in RD&D 2010 and its review

In RD&D Programme 2010, 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, SSM stated the following:

- SSM takes a positive view of the fact that SKB is continuing to work to develop the design methodology including the Observational Method. However, this imposes tough demands on the requirement specification for construction and monitoring of compliance with these requirements. The Authority would like to emphasize how important it is that the requirements imposed by long-term safety and environmental impact on production should be a natural part of the design and construction work. In its continued work, SKB should ensure that the processes are designed so that this is achieved. SSM also mentions the following as aspects worth considering: MTO (human factors), the difficulty of judging the impact of different outcomes on long-term safety, and the fact that certain work operations cannot be undone, for example rock excavation that does not comply with requirements on the excavation-damaged zone.
- SSM thinks that SKB should further develop and clarify how Eurocode EN 1997-1:2004 (SIS 2005) will be implemented in the design work and during production.
- SSM is of the opinion that SKB should clarify its plans for development of the detailed characterization programme with more precisely specified times for different types of investigations related to different purposes and requirements.
- SSM would like to know SKB’s plans for monitoring of relevant parameters.
- SSM takes a positive view of the fact that SKB will revise its methodology for interpretation of hydraulic properties of flowing structures. In this context, SKB should clarify the requirements on the effectiveness of the tests and their ability to determine flow dimensions and boundary effects. In connection with this revision or in some other suitable context, SKB should also clarify what is

meant by "actual transmissivity" and "connected effective transmissivity" in the requirement specification for the rock as a barrier and how the parameters relate to different measurement methods.

- SSM considers precision in the placement of the facility in the rock mass and in relation to geological units to be important. The Authority believes that it can be advantageous to have a closer link between geodesy (tunnel coordinates) and photogrammetric results from the geological mapping.
- SSM noted once again the absence of information on methods and instruments for determining the strength and deformation properties of the rock mass and the deformation zones. This was originally noted by SKI in the review of RD&D programme 2007. The Authority would like SKB to present plans for this or explain why it is not mentioned. SSM takes a positive view of SKB's initiated study of a local seismic network in Forsmark and considers it important that a good background picture be established of microseismicity before construction of the repository begins. The Authority further considers that SKB should evaluate the possibility of utilizing microseismic data to gain a better understanding of the large-scale deformation behaviour of the rock mass.
- SSM considers the methodology for detection of long fractures to be important and that SKB should fulfil the described plans. The Authority also considers it important that the possibility of detection with associated uncertainty description be linked to the design premises in an appropriate manner.
- SSM would like to know whether SKB believes that it has sufficiently good knowledge concerning the impact of the grout on the hydrogeochemical conditions and whether further research and development is planned. Similarly, the Authority would like a description of the effect of rock support measures (rock bolts, wire mesh and shotcrete) on long-term safety.
- SSM believes it is important that SKB can show that the requirements made by the design premises on the EDZ can be met in tunnelling under production conditions in Forsmark.
- SKB should describe the expected nature of the EDZ in the floor.

Summary of current situation and programme

Since RD&D Programme 2010 was presented, SKB has drawn up a framework programme for detailed characterization, submitted licence applications and extended the Äspö HRL. The extension, which was finished in January 2013, has made it possible for the first time to integrate detailed characterization, design including application of the Observational Method, and rock works where quality requirements on execution have been applied in production. The new tunnels permit testing of methods for detailed characterization. The test results will show how well the requirements made on tunnelling have managed to minimize the extent of the EDZ, especially in the floor.

The mining of the outer section of the Prototype Repository in the Äspö HRL has substantiated previous rock mechanics research, see Section 13.11 "Prototype Repository".

The main tasks and goals of the technology development programme for rock, which is described in Section 14.3 to 14.5, can be summarized as follows:

- Further develop the methodology for underground design and the application of the Observational Method. This pertains mainly to strategies for detailed adaptation of deposition areas and coordination of detailed characterization and building production.
- 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 buildout of the deposition areas.
- Further develop production methods adapted to the requirements made regarding rock excavation, stability and tightness. The goal is primarily to be able to specify performance requirements in the construction documents for accesses as well as methods to verify that the requirements are satisfied. The performance requirements and their verification will be further developed to satisfy the unique requirements that apply regarding the rock below the level for the top seal.
- Ensure that approved and sufficiently duty-proven engineering materials are available in time for construction of the accesses and later for construction of the repository's deposition areas.
- Develop inspection plans, including format and procedures for documentation and for as-built plans, so that they have been evaluated and finalized when the rock excavation works are begun. The programme for this will be presented in conjunction with coming safety analysis reports.
- Demonstrate strategies and methods for an integrated and quality-assured mode of working from investigation to finished tunnel, see Figure 14-1.

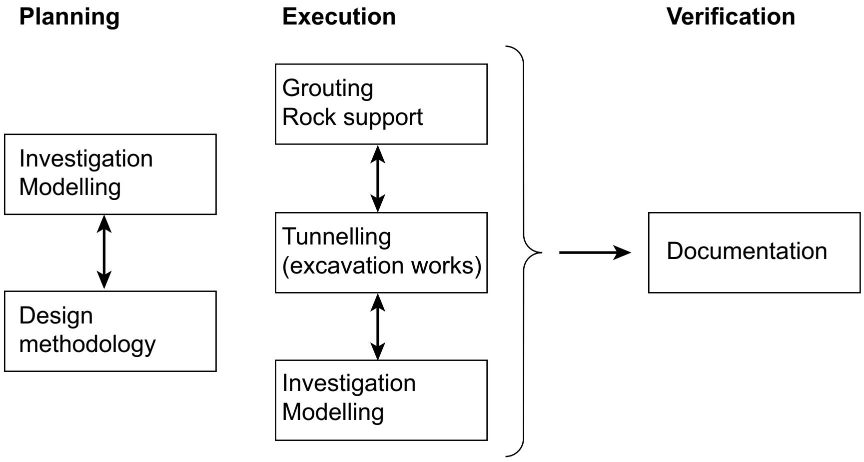


Figure 14-1. *Overview of the main activities needed for integrated and quality-assured construction of the final repository.*

The timetables for technology development for all activites related to tunnelling differ somewhat from other technology development. This is because essential parts of the technology related to tunneling must be ready for use 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.

14.3 Methodology for underground design

Underground design includes structural design and site adaptation of the Spent Fuel Repository. Requirements and premises are given in Section 14.1 and are based on SSM's regulations, among other things. Additional requirements are expected to come in the form of environmental conditions associated with future licences. Premises for underground design are also given by site conditions in Forsmark and the final repository's layout requirements (location, dimensions, design lifetime etc). Requirements on verification of structures are given in Eurocodes, see "Programme" in Section 14.3. Since the design serves as a basis for the construction of the facility, specifications in execution are also included to satisfy stipulated requirements. Integration with detailed characterization that identifies site conditions is fundamental for site adaptation of the underground openings and for the choice of technical solutions adapted to the variability of the rock.

Current situation

Since RD&D Programme 2010, SKB has kept track of and participated in industry-wide work concerning the structural design of tunnels and rock caverns. The Royal Swedish Academy of Engineering Sciences has pursued a project for implementation of Eurocodes in geotechnical design called IEG (Implementeringskommission för Europastandarder inom Geoteknik), which has brought together a number of actors in Sweden. Some of the work is specifically aimed at underground construction. The documents that are presented are advisory. In connection with the merger of the Swedish Road Administration and the Swedish Rail Administration to the Swedish Transport Administration, the authority revised and updated its guidelines for structural design of tunnels (Trafikverket 2011), which provides good support in the continued planning work for detailed design of the Spent Fuel Repository.

RD&D Programme 2010 described SKB's strategy in applying the Observational Method. 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. Possible and acceptable behaviour must be established for the structure in question. By "behaviour" was originally meant the strength and stability of the structure.

In the case of the final repository, the concept has been broadened to include adaptation to the design premises, including requirements regarding long-term safety and performance requirements that relate to other barriers. This applies, for example to maximum permitted in-leakage for installation of the backfill. IEG has an application document about the Observational Method (IEG 2010), which, according to a quotation from this document, draws the conclusion:

"Application of the Observational Method can relatively easily be integrated in modern geotechnical design. What is specific with the Observational Method is that technical solutions and inspection programmes should be prepared in a way that makes it possible to handle the uncertainties in the behaviour of the geotechnical structure. The behaviour of the geotechnical structure can relate to two categories of uncertainties, either the behaviour of the ground or the behaviour of the geotechnical structure, and define which limit states or combinations of limit states need to be verified for the design situation in question. Depending on the nature of the uncertainties, technical solutions should be designed that are adapted to possible deviations from expected conditions. Tendering and construction documents must be designed so that they formally show how the uncertainties in the behaviour of the geotechnical structure that has been identified in the design stage will be handled via inspection programmes, criteria and prepared measures. The Client's and the Contractor's site organizations bear responsibility for ensuring that the application of the Observational Method is integrated in the construction process in an effective and quality-assured manner."

Within the framework of the extension of the Äspö HRL, the Observational Method has been applied in two respects:

- Control of the grouting work.
- Site adaptation of a number of niches from new tunnels to reduce the need for grouting.

In the case of one of the experimental niches that was to be built, it was desired to have as high a groundwater pressure as possible. The key requirement on the grouting works was to ensure that the groundwater level was drawed down less than 50 metres. The primary uncertainty in the design of the grouting work was linked to acceptable water inflow to keep from exceeding the groundwater lowering limit. The Hydro Monitoring System (HMS) on Äspö permitted effective observation to ensure that the groundwater lowering limit was not exceeded. It was decided that the principles of the Observational Method should be used to design the grouting process and verify its efficiency vis-à-vis the stipulated requirements.

The Observational Method consists of the three fundamental concepts prediction, monitoring/observation and corrective action.

Planning for the application of the Observational Method in a project is done in the design stage when all relevant information (data and models) is used to make predictions and establish a monitoring programme. Corrective action plans are prepared to deal with cases where uncertainties exist regarding site conditions that affect e.g. firmness, stability or tightness. This guides the application of the model in the execution phase so that stipulated requirements can be fulfilled.

When the method was applied to the extension of the Äspö HRL, it was noted that the greatest uncertainty lay in designing grouting to meet tightness requirements on some of the planned experimental areas. The focus in the design stage was therefore on making predictions of the hydrogeological properties of the rock and the efficacy of the grouting based on existing models and new borehole information. Based on assessment of possible deviations from expected conditions and results, acceptable corrective actions and technical solutions were determined. In the project this included:

1. Input data consist of borehole data, e.g. measured inflows during drilling of the pilot holes for the parallel tunnels.
2. In-leakage to the tunnel with and without grouting is assessed and the sealing effect is estimated. Conservative assumptions should lead to an assessment of the worst case.
3. The relationship between inflows from pilot holes and groundwater lowering (drawdown) in nearby holes included in the Äspö HMS is determined and the borehole sections that are most affected are selected as monitoring points during tunnelling.
4. Decisions on corrective action in the contracting works are based on acceptable drawdown levels according to the HMS and/or inflows from holes. Acceptance levels should be evaluated during tunnelling within the first grouting fan. Continuous follow-up may then lead to adjustment of the acceptance levels.

The hydrogeological properties of the rock and the intended function of the grouting were then verified in the execution phase based on the results of measurements and observations. Prepared measures, for example changes in grouting methodology, were implemented when warranted. The observational parameters in the various steps in the grouting cycle for tunnelling are summarized in Table 14-1.

A predictive model of water-conducting structures had been constructed based on the investigation results. The tunnels were continuously mapped and the model of water-conducting structures was continuously updated. Based on this detailed characterization and modelling, the experimental niches were site-adapted with the ambition of minimizing the grouting need.

Programme

Application of the Observational Method to grouting and site adaptation strategies in conjunction with the extension of the Äspö HRL will be included as an illustrative example in the strategy description of the Observational Method. Fundamental ingredients of this strategy description are detailed process descriptions that demonstrate the implementation of characterization with design, as well as execution of the tunnelling works with flexibility within the framework of predetermined measures in keeping with the overall purposes of the Observational Method.

With the support of experience from design for and extension of the Äspö HRL in 2012, a process survey of the design process and its verification has been commenced. An important principle is adaptation wherever possible to terminology and control in Eurocodes, particularly:

- EN 1990 (SIS 2010), mainly the Sections 2 "Requirements" and 3.2 "Design situations".
- EN 1997-1:2005 (SIS 2005) regarding application of the Sections 2.4 "Geotechnical design by calculation", 2.5 "Design by prescriptive measures" (including empirical methods) and 2.7 "Observational method".
- Furthermore, SKB will prepare proposals for design procedures for the Spent Fuel Repository with respect to resistance to earthquakes during operation, since EN 1998-4:2006 (SIS 2006) and its national annexes are not relevant for nuclear facilities.

Table 14-1. Overview of observational parameters and corrective actions used in applying the Observational Method to grouting in the Äspö HRL's 2012 extension project.

Step in tunnelling cycle	Observational parameter	Decision based on observation
Probe holes	Flow/hole (measured one hole at a time).	Grouting strategy (one or two fans).
	HMS response.	Important information as early indication of expected responses.
Grouting	Pressure, flow, volumes, times per hole. Type of grout.	RTGC* – post-analysis to assess grout spread and evaluate stop criteria.
	HMS response.	Is recovery of the pressure acceptable? Can blast hole drilling start?
Drilling of blast holes	Water in blast holes.	Report to activity leader.
	HMS response.	Is drawdown according to HMS monitoring within the acceptance criterion so that charging – blasting can be done?
	Photo documentation of working face.	Documentation of location of flowing boreholes. Where are corrective actions needed?
Blasting – mucking – scaling	HMS response.	Is drawdown according to HMS monitoring still within the acceptance criterion?
Tunnel mapping	Water-bearing fractures, location for injection grout.	Assess efficacy of executed grouting.
Measure in-leakage at weir	Weekly reading (Monday morning) to monitor total in-leakage.	Assess sealing effect. Need for improvement?
	HMS response.	Assess sealing effect. Need for improvement?

* RTGC = Real Time Grouting Control, see Section 14.5.1 "Grouting".

Based on the process survey of the design process and verification of structures, a preliminary quality plan for design will be prepared for the next step in the design of the Spent Fuel Repository (detailed design). Preparing and implementing quality plans is a task for all production lines for KBS-3, but the Rock Line will be first to come out with quality plans for control of planning and construction of the Spent Fuel Repository's accesses. The quality plan for design will be integrated with detailed characterization, providing valuable control of e.g. site adaptation of the Spent Fuel Repository. The quality plan must be completely ready prior to procurement of contractors for the rock works and fully implemented in relevant parts for construction of the Spent Fuel Repository.

14.4 Detailed characterization

A detailed characterization programme will be devised as a further development of the framework programme for detailed characterization presented previously (SKB 2010j). Further, methods and tools for detailed characterization during the construction and operation of the final repository will be refined. The updated design premises will be taken into account in this work. Further development of work processes will be an important component, especially in connection with the establishment of quality management and control during the construction and operation of the Spent Fuel Repository.

During the period up until start of construction, a detailed investigation programme will be prepared for construction of the final repository's accesses and central area that will include investigations, development of geoscientific models, data management and quality assurance of data and models. Further, a preliminary programme will be prepared describing strategies and methodology for verification of requirements concerning post-closure safety and how requirements on acceptance of deposition holes are to be realized in practice.

14.4.1 Overview and strategies

Current situation

By "detailed characterization" is meant the investigations (including monitoring), analyses and modelling that are carried out in conjunction with the construction and operation of the Spent Fuel Repository. According to SKB's framework programme for detailed characterization, the purposes of detailed characterization are:

- To provide a basis for adaptation of the repository to the site-specific conditions in order to meet the design premises, e.g. with respect to long-term safety.
- To provide a basis for engineering-related decisions concerning e.g. grouting and rock support measures.
- To update site models, which will in turn serve as a basis for safety assessments regarding long-term safety.

The framework programme (SKB 2010j) was presented in close conjunction with RD&D Programme 2010 and served as a basis for SKB's applications for licences to construct a final repository for spent nuclear fuel. The framework programme gave an integrated account of strategies and methods for the execution of detailed characterization taking into account site-specific conditions and questions in Forsmark. It also included a general plan for further development of methods and tools for investigations, modelling and data management.

Further development of the methodology for detailed characterization is divided into two phases. In the initial development phase, the focus is on methodology for construction of the repository, i.e. the repository's accesses (ramp and shafts) and central area. In a later development phase, the work will be focused on methodology for the final repository's operational phase with successive preparation of deposition areas at repository level.

Programme

In accordance with the framework programme's planning, a project is to be carried out for further development of methods, tools and programme for investigations and modelling prior to start of construction. The project's target outcome is that requirements regarding site characterization and design premises pertaining to requirements on the rock shall be verifiable, that the facility can be built with acceptable impact on the surrounding environment, and that updated safety assessments can be performed.

The objectives of the project are to:

1. Further develop methods and tools for detailed characterization with associated modelling. This is being done with a focus on those parts that will be used during construction of accesses and central area. The word "tools" refers here in a broad sense to measurement instruments as well as the data systems and codes that are used for visualization and modelling, with associated methodology.
2. Further develop the programme for detailed site investigations for construction of accesses and central area describing investigations and modelling with associated data management, quality assurance and reporting.
3. Devise and test strategies and methodology for verification of requirements on post-closure safety (design premises). Initiate further development of methods and tools needed for detailed characterization to be used in deposition areas.

The development work is focused on the tools that will be used during construction of accesses and central area (points 1 and 2 above) and development for deposition areas that is expected to take a long time and therefore needs to be started early (point 3 above). An example of this is the methodology for identification and characterization of large fractures. Other development for detailed characterization in deposition areas will mainly be carried out in a subsequent development project.

After a thorough planning stage, development of investigation and modelling methodology as well as of instruments and data systems has now been initiated. The development need, which is ultimately based on the need for information, has been determined following analysis of existing methodology. The development work spans a broad field and is grouped in technology areas and disciplines, some of which are described in Section 14.4.2 "Modelling and investigations". Technology development will result in finished methods and tools with associated documentation in the form of manuals and method descriptions. An important component is the transfer knowledge and implement the results of this in the overall work with the Spent Fuel Repository.

Further development of the detailed characterization programme presented in SKB (2010j) is based on:

- Obtained new knowledge.
- Results of ongoing and performed development work.
- Experiences from completed site investigation and modelling work.

Strategies, methods and tools will be compiled into a programme for detailed site investigations that describes how site characterization can be executed. The programme should enable verification of the design premises regarding requirements on the rock, enabling construction of the facility with acceptable environmental impact, and enabling performance of updated safety assessments. A programme report will be written presenting an updated programme for detailed site investigations for construction of the Spent Fuel Repository. It will serve as a basis both for detailed design and for writing up the document on safety during construction of the final repository (see 10.3.1). Furthermore, strategies for investigations, including monitoring, and modelling in deposition areas will be presented. This concerns the evolving construction of the deposition area includning acceptance of canister positions as well as documentation of initial states serving as a basis for the PSAR (see 10.3.1).

An important task for the detailed site investigations is, during the successive buildout of a deposition area, to contribute characterization results that will serve as a basis for decisions regarding the location and realization of deposition tunnels and deposition holes. Verification that the design premises have been fulfilled is an important part of this. In the current work, design premises related to deposition holes (large fractures and water inflows) will primarily be dealt with. Other design premises, as well as further studies of large fractures, will be dealt with in the subsequent development work, which will focus on the detailed site investigation methodology for the deposition areas.

Large fractures include minor deformation zones and individual fractures of large extent which could, in connection with seismic events, entail canister damage due to shear. Methods for identifying and characterizing large fractures will be studied by field investigations in the Äspö HRL, whereby the usefulness of the investigation and modelling methods will be analyzed and evaluated, see Section 14.4.4 "Detailed characterization", "Large fractures". The results of this work will provide a basis for decisions on necessary further development and ideas on how to improve the formulation of design premises with associated criteria.

Detailed site investigations require controlled data flows, quality assurance of data and models and clearly defined information and decision processes during construction and operation of the final repository. Planning for how detailed site investigations will be quality-assured during the construction and operating phases constitutes part of the planned quality assurance work , see Section 10.2.2. This work includes further development of work processes, preparation of procedures and instructions for the methods and tools used for detailed site investigations as well as procedures for documentation and presentation of the results of the detailed site investigations. Experience from e.g. SKB's completed surface-based site investigations and the expansion of the Äspö HRL comprises as an important basis for this.

14.4.2 Investigations and modelling

Current situation

Following the publication of the RD&D Programme 2010, planning has been under way for detailed site investigations with progressive prioritization and focusing. Requirement specifications have been prepared for each development object shedding light on both the end product and the need for methodology development. These specifications have subsequently served as a basis for the preparation of specified development plans.

Two projects concerning rock stress measurements have been carried out during the past three-year period. One project has developed a borehole method called SLITS for determination of the orientation of the rock stresses (Hakami 2011). The method is based on inducing thermal stresses in a borehole so that spalling occurs and the orientation of the spalling can be correlated with the orientation of the maximum horizontal stress. The latter is of importance for e.g. optimizing the orientation of deposition tunnels. The second project (Hakala et al. 2013) has developed an LVDT (Linear Variable Differential Transducer) cell for rock stress measurement in tunnels. The instrument is installed in a number of pilot holes with a diameter of 120 millimetres in a tunnel section and the latter which are overcored with a drill bit with a diameter of 200 millimetres. Inverse modelling of the in situ stress field is based on strain data collected from the LVDT sensors, which measure in four different directions, a detailed laser-scanned model of the tunnel section where the measurement has been performed, and the modulus of elasticity measured on solid cores from the pilot holes.

Verification measurements with an LVDT cell in the Äspö HRL at the 450-metre level, where the stress field is well-characterized, show that the results of the measurements with the LVDT cell agree with previous results from overcoring measurements performed in the Äspö HRL, see Figure 14-2, as well as previous results and established stress model (Christiansson and Janson 2003).

According to the RD&D Programme 2013, SKB has developed a customized tunnel mapping system, since no existing commercial system could meet SKB's needs. The mapping system is based on modules for photogrammetry and digital mapping in the field. After each blast round, photography is carried out to generate a basis for geological mapping of the tunnel face in three dimensions. Following successful system tests in the Äspö HRL, the mapping system was used during the construction of new experimental tunnels at the 420-metre level in the Äspö HRL. The results were used there to build and successively update geological models and predictions. When the tunnel face had advanced further, the mapping system was mobilized once again for mapping of tunnel walls and ceiling once the tunnel section had been declared safe. Constructed models of the tunnel in three dimensions, with or without structures, can be imported to visualization systems and CAD systems where further processing and analysis can be done, for example checking of blasted rock volumes (based on measured tunnel contour) relative to the intended contour of the design.

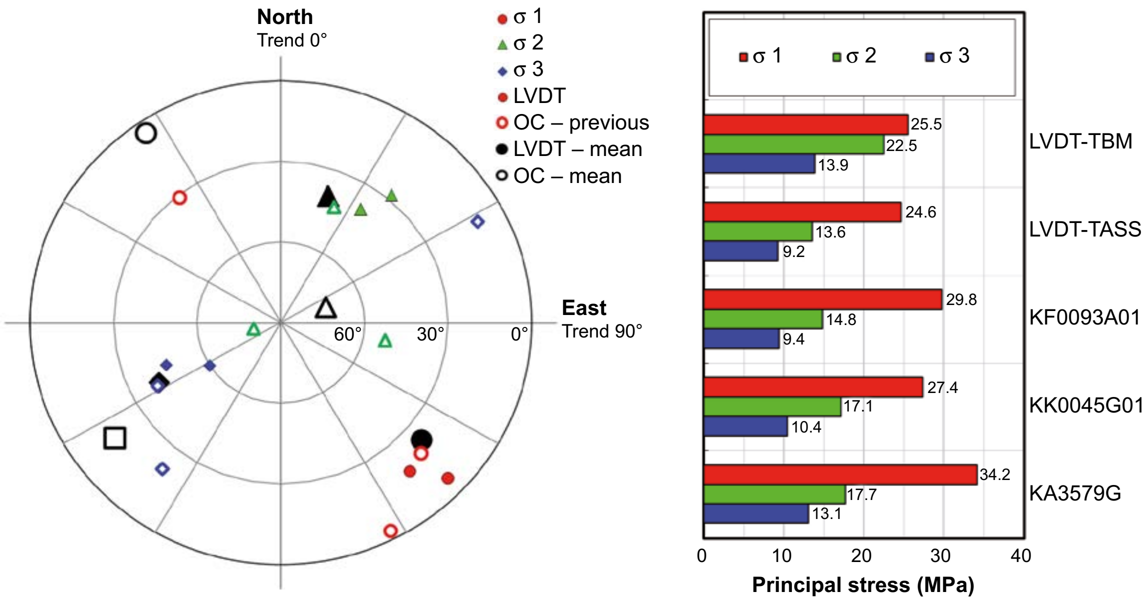


Figure 14-2. *Comparison of results from LVDT with results from the conventional overcoring method. The stereo plot (left) shows the orientation of the principal stresses, while the histogram (right) shows the stress magnitude for different measurements. OC = overcoring method. Magnitudes of previous results with the overcoring method at the 450-m level at the Äspö HRL are presented in the lowermost three groups of bars. LVDT-TBM is from a section in a full-face bored tunnel while LVDT-TASS is from a tunnel excavated using drill-and-blast.*

Since 2010, SKB has participated in MoDeRn (Monitoring Developments for Safe Repository Operation and Staged Closure), a four-year project (2009–2013) within the European Union's 7th Framework Programme. The project has dealt with questions concerning monitoring and how monitoring can contribute to long-term safety strategies and technical designs of final repositories in three different geological environments (clay, salt and crystalline bedrock), as well as to the public's understanding of and confidence in final disposal of radioactive waste in underground facilities.

The overall goal of the project was to define and document the approaches and methods of the different participating countries as well as the standpoints of various stakeholders (regulatory authorities and private citizens) for the purpose of providing references and support to the national monitoring programmes. The project has included:

- The goals and strategies of monitoring, plus guidelines for the design of a monitoring programme that takes into account relevant technical and societal contexts, applicable legislation in each country and demands of stakeholders (including regulatory authorities and private citizens).
- Development and demonstration of innovative technology.
- Documentation of case studies that illustrate goals and strategies, which parameters need to be monitored, and how measurement error and unexpected events can be detected or prevented .
- Sociological studies to create better understanding for the viewpoints of private citizens and other stakeholders.

At present, discussions are under way regarding a joint follow-up project with a focus on how monitoring data should be used as input data for modelling and analyses of post-closure safety and as support for the decision process during construction and operation of the final repository.

Programme

Modelling during the construction phase will have different recipients and users with different requirements at different stages, including design, building production, site understanding and post-closure safety. Design and building production have a need for smaller-scale predictions and models, which means that modified and in some cases new modelling methodology needs to be developed. Furthermore, methodology needs to be updated for site descriptive modelling, with a special emphasis on use of smaller-scale models as a basis. Development of integrated modelling methodology is underscored, where the principal integration with respect to design and construction takes place within the disciplines of geology and hydrogeology. Specified needs, requirements and criteria linked to long-term safety are provided by the nuclear fuel programme, which updates the design premises to measurable requirements in relation to the results of SR-Site, see Section 10.2.1 "Design premises".

Modelling during construction of the accesses and central area takes place in a transient and disturbed environment. This is an important point of departure for the further development of modelling methodology for all disciplines and requires efficient data collection from monitoring in relation to the measured baseline situation (see section "Monitoring" below). Modelled and measured changes are related to the baseline, which thereby comprises an important prerequisite for calibration of quantitative models.

Discrete fracture network models (DFNs) comprise an important components in the description of the rock and have links to a number of the disciplines. Development of DFNs is carried out as part of the research for assessment of long-term safety, see Section 26.24.1, made in close interaction with the development of other tools for detailed site investigations.

Development of methods and tools for investigations that produce primary data for modelling is being conducted so that tested and approved methods and instruments will be ready when investigations are to be executed during construction and operation of the Spent Fuel Repository. The development work may involve refinement of an existing method, acquisition of a new method or instrument, or upgrading of a method description, which is mainly concerned with adaptation for underground use.

Current and planned discipline-specific development and modelling methodology are described below, along with associated development of methods and instruments.

Geology, geophysics, geodesy

Further development of single-hole interpretation (SHI) of borehole data and initial development of an application to underground openings according to the same principles, called single-tunnel interpretation (STI), are important parts of the modelling methodology. An essential component in the development work is to incorporate hydrogeological information in the description of the geological framework using SHI and STI.

Within methods/instruments and data systems, further development is being carried out of the tunnel mapping system and the Boremap system for mapping of drill cores. Moreover, relevant borehole geophysical methods are being adapted for underground use. An experimental installation (analogue for a borehole) is being built on the ground surface at Äspö. This installation has a well-established small-scale deviation which reflects the strict requirements of KBS-3H (horizontal deposition), see Chapter 16. The surface facility permits tests, comparisons and verifications to be made of different measurement methods and instruments for deviation measurement and is expected to enhance confidence in them.

There are a number of geometric requirements on the Spent Fuel Repository's underground openings. In order for the buffer in the deposition holes and the backfill in the deposition tunnels to be emplaced correctly and fulfil the design premises, the dimensions of the openings must be kept within specified tolerances. This imposes requirements on accuracy of equipment and methods employed in the measurement of the geometry of the openings. Methodology for such measurements will be developed and tested before commissioning tests are carried out prior to the Spent Fuel Repository's operating phase.

Rock mechanics

Development within rock mechanical modelling includes methodology for modelling of the properties of the rock mass with greater integration of geological and hydrogeological information. Other development of modelling methodology is taking place within research for assessment of long-term safety, see Section 26.24 "Geosphere, Modelling".

Further studies have been made regarding the dependence of microstructures on dominant structures in the bedrock (Nicksiar and Martin 2012, Lim et al. 2012). This development pertains to application of the new generation of rock mechanical numerical codes developed in response to the needs of the mining industry. The purpose of the development work is to describe the composition of the rock mass from micro- to macro-scale. Rock failure can be initiated in the weakest link, whether it be intact rock, fractures or deformation zones. An initial calculation case based on the well-documented Äspö Pillar Stability Experiment (Andersson 2007) shows good agreement between observation and modelling (Lan et al. 2013).

In the continued development of methods and instruments, further adaptation is being done of the SLITS and LVDT methods for in situ determination of the orientation and magnitude of the stress field. In addition, method development is being done for evaluation of the rock stress situation based on measurements of the convergence of the tunnel contour.

Hydrogeology

Development within hydrogeological modelling includes a number of components and is distinguished by strengthened integration with geology as well as hydrogeochemistry, transport properties and surface systems. This integration will be further strengthened both with regard to descriptive (conceptual) and quantifying modelling. Development of methodology for modelling of discrete fracture networks will be conducted in a fully integrated context for geology and hydrogeology as part of the research for assessment of long-term safety, see Section 26.24.1. Underground investigations are expected to provide a better basis for describing the size distribution of the fractures and for describing the distribution of transmissivity in individual fracture planes. An improved description of the heterogeneity of the fracturehydraulic properties also provides a better basis for describing the rock for the purpose of grouting, see Section 14.5.1.

More refined models will be required for describing and understanding the sometimes complex processes near underground openings. Results from inflow measurements and hydraulic tests for a deposition tunnel or for a deposition hole, in combination with modelling, are expected to provide a better basis for verification that the associated design premises are fulfilled. Modelling for the purpose of arriving at practically useful inflow criteria has been initiated, see Section 26.4.3. Improved methods and description of the extent and properties of the EDZ will be obtained from current studies at the Äspö HRL, see Sections 26.9 and 26.4.2. Development work is being done to include chemical reactions in existing flow simulation codes s. These tools can, for example, be used to study the effects of cementitious solutions on the buffer and the interplay between rock and buffer.

The development of modelling methodology is being conducted with concurrent adaptation of equipment and methodology for execution and interpretation of hydraulic tests, including flow logging for application underground in low-permeability rock. This also includes simplified tests in probe boreholes as a basis for improved conceptual description of flowing structures, including the distribution of transmissivity in individual fracture planes as described in DFNs. Furthermore, methodology and equipment for hydrogeological monitoring including measurement of inflows to tunnels will be refined on the basis of experience from the Äspö HRL.

Hydrogeochemistry

Development within hydrogeochemical modelling includes further development of explorative data analysis. This is being done with the support of statistical methods and development of methodology for monitoring of short-term impact on the hydrogeochemistry of a disturbed system. Great emphasis is placed on improved integration of modelling of hydrogeochemistry and hydrogeology in the construction of hydrogeochemical site descriptive models, see also Sections 26.11 and 26.15. Further development of methods and instruments includes measurement with a measurement cell online, and sampling of dissolved gases, matrix fluid and groundwater in packed-off sections during drilling.

Transport properties

Modelling strategy and methodology for site descriptive transport modelling is being updated based on the actual execution and experiences from SDM-Site (Crawford 2008, SKB 2010k). This is being done in close collaboration with other research, internal and external, linked to transport and retention, see Sections 26.12–26.16. Methodology for modelling of the transport properties of the rock does not need to be fully developed at the start of construction. Methodology for modelling of the situation in the vicinity of deposition areas has therefore been deferred to the following development project with a focus on detailed site investigation methods for deposition areas.

Surface systems

Development of modelling methodology for surface systems is mainly linked to strengthened integration with the hydrogeology of the rock in order to establish an integrated description of groundwater flow that covers the whole hydrological cycle. Modelling of disturbances in the surface system during buildout is an important component. It includes quantification and identification of causes, and analyses of measures needed to prevent or reduce undesirable changes. This is closely connected to development of methods for water sampling and follow-up in the field of landscape evolution and natural values.

Drilling

For exploratory drilling, methods and tools used in core drilling are being industrialized to allow execution with a uniform and high quality. Certain auxiliary equipment for core drilling underground will be developed. The equipment should permit short mobilization times and create favourable conditions for drilling up to 300-metre-long cored boreholes with high quality requirements on execution, geometry, data collection and data delivery.

In exploratory drilling, the inflow of groundwater may need to be reduced, and/or surrounding rock in sub-horizontal or horizontal cored boreholes may need to be stabilized. This can be done by means of mechanical stabilization and (selective) grouting so that core drilling can be continued and the planned investigations in completed boreholes can be carried out. Equipment must be adapted and developed for application underground.

Monitoring

The baseline describes natural conditions on the site before they are affected by construction of the repository. An extensive baseline has been established that includes measurements of hydrogeological state variables (groundwater pressure/groundwater levels in soil and rock), hydrological parameters (groundwater flows in watercourses, lake levels), as well as hydrogeochemical, meteorological and oceanographic parameters. Monitoring of changes in relation to baseline comprises a part of the programme for detailed site investigations where existing monitoring systems for meteorology, surface water and groundwater will be used. Certain borehole installations may need to be modified in order to enable measurement of drawdown near the facility. When underground openings are accessible, the monitoring will also include the inflow of groundwater to different tunnel sections. Improvements of traditional technology will be required in part for this purpose. The system for collection of monitoring data will also be further developed, see also Section 14.4.4.

Establishment of a local seismic network is planned prior to the construction of the Spent Fuel Repository in Forsmark. This will allow monitoring of earthquakes of considerably lower magnitude than those that can be registered within the national network. Both natural earthquakes and earthquakes caused by the construction work will be monitored by this microseismic network. The latter category mainly includes earthquakes caused by blasting, but also stress-induced seismic events caused by rock excavation or temperature-induced earthquakes caused by the disposed fuel canisters. The seismic network is expected to provide detailed knowledge of the natural seismicity in the Forsmark area and contribute to greater detail in the structural geological site model and to a better understanding of the stress field in Forsmark. Furthermore, the network is planned to be included in the system for preventive safety work and efficient progress of the construction works. SKB is currently studying the scope and magnitude of the local signal noise so that instruments of adequate sensitivity can be procured and installed. The network is expected to be in full operation a few years before the start of construction.

14.4.3 Large fractures

Current situation

Design premises associated with large fractures pertain to potential planes of movement that are so large and have such properties that they could, if a seismic event occurs, damage a canister due to shear. There is currently no available methodology for determining the size of these fractures with certainty. The FPI criteria for the identification of potentially large fractures in the positioning of a deposition hole (Munier 2010) are therefore still applicable, see also Section 26.8. FPI stands for Full Perimeter Intersection, i.e. a fracture or fracture zone that can be traced around the entire tunnel perimeter. The FPI criteria comprise FPC (Full Perimeter Criterion) and EFPC (expanded FPC, which refers to fractures that intersect at least five deposition holes but not the deposition tunnel), whose implications are illustrated by Figure 14-3.

Identification of a fracture or fracture zone that can be traced around the full tunnel perimeter (FPI) is in practice purely a mapping task. The applicable criterion, FPC, is therefore judged to be verifiable with great certainty, regardless of excavation method. EFPC is more difficult to verify, since the fractures cannot be identified in conjunction with tunnel mapping. It therefore presumes an integrated detailed site investigation methodology with application of both direct measurement methods and indirect methods (such as geophysics and hydraulic cross-hole measurements) in tunnels and pilot holes.

FPC, however, excludes all fractures larger than the tunnel diameter. Since the geometric size must be on the order of ten times greater in order for a critical shear to occur, the criterion is conservative in the sense that even acceptable deposition positions may be excluded. There is therefore reason to try to refine the methodology for identification and characterization of the large fractures so that their impact on deposition can to a greater extent be based on actual properties and characteristics and to a lesser extent on the FPI criterion. The strategy for this methodology was presented in general terms in SKB (2010j) and involves the use of volume-scanning investigation methods such as geophysics (seismic, radar and resistivity methods) and hydraulic interference data. Experience from SKB's collaboration with Posiva concerning testing and evaluation of methods for identifying large fractures in Onkalo is taken into account.

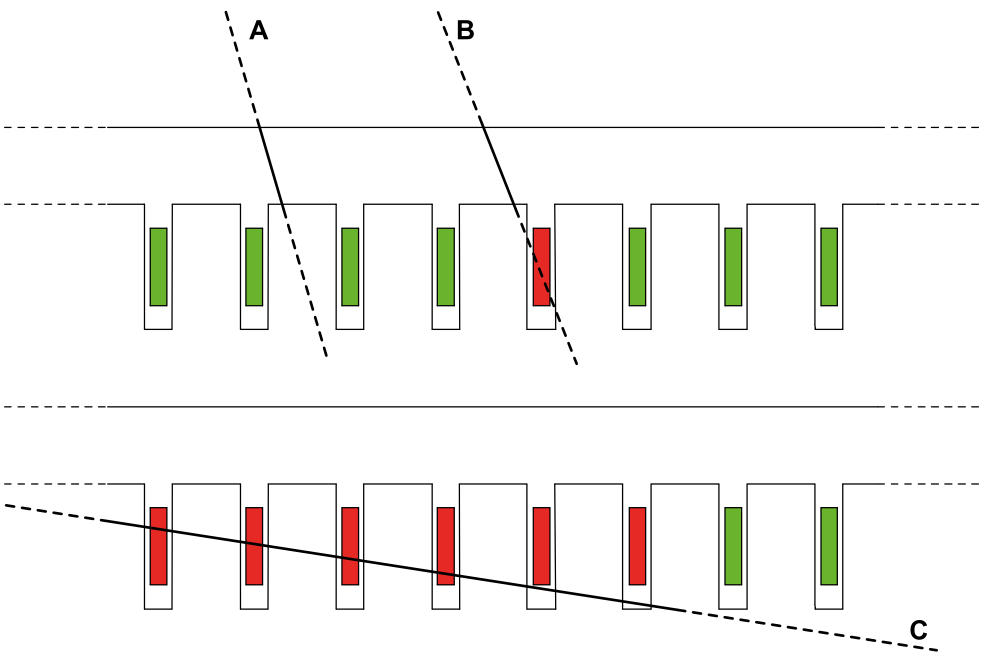


Figure 14-3. *Illustration of the implications of the FPI criteria. A and B are two FPI objects, of which A does not affect deposition, whereas B prevents deposition. C intersects at least five deposition holes, which, according to EFPC, means that they may not be used for deposition (from SKB 2010j).*

Programme

Investigations to identify and characterize large fractures will be conducted in two parallel tunnels as well as in two new cored boreholes in connection with the new underground openings in the Äspö HRL. Tunnels and cored boreholes will be spaced at a distance of about 40 metres, which corresponds to the typical distance between two deposition tunnels. The purpose is to test, evaluate and refine investigation and modelling methods for identification and characterization of large fractures as described above.

The first stage of the planned work includes measurements in the tunnel system using radar, seismic and resistivity methods. The results of these measurements are then integrated, along with the results of hydraulic interference tests in pilot boreholes for the tunnels, with the established structural model of the rock volume in question. Tunnel mapping provides data on fractures or fracture zones that can be traced around the full tunnel perimeter. The results of hydraulic tests in probe boreholes performed during tunnelling are expected to constitute a complementary body of data on the distribution of transmissivity in individual fracture planes. They can therefore be used to reduce the risk of hydraulically misjudging a potentially large fracture section (see also the section on hydrogeology above).

In a subsequent test campaign, the two aforementioned cored boreholes will be drilled from one of the investigated tunnels. Investigations of a similar kind as those done in the tunnels will be conducted in these boreholes, along with integrated modelling with a focus on large fractures. The two test campaigns comprise an important part of the investigation sequence included in the aforementioned strategy. In a concluding step, a hole is drilled through the rock volume to test hypotheses regarding the geometry and size of modelled structures.

The programme concerning large fractures is primarily focused on the size of the fracture planes and the underlying properties that can serve as indicators of size. To some extent, it will also be possible to study the hydraulic and mechanical properties of the structures in order to better assess the risk that a movement will actually occur in large fractures.

Site-specific conditions determine how the methodology for selection and verification of deposition positions will be applied in the Spent Fuel Repository. Further development efforts will therefore mainly be done in conjunction with the integration tests planned to be done in the Äspö HRL and in Forsmark. The ultimate goal is that the identification of large fractures around deposition positions will be based to a greater extent on actual properties and to a lesser extent on the FPI criterion.

14.4.4 Computer systems for detailed characterization

Current situation

Functional computer systems are a prerequisite for efficient and quality-assured execution of detailed site investigations. Different systems and programmes for data collection, modelling, visualization and presentation will be used. The need is determined primarily by the investigation and modelling methods to be used. Based on this, requirements made on computer systems during construction of the final repository have been identified, and based on this needs for new or improved functions have in turn been identified. This has in turn led to a plan that includes both establishment of new computer systems and further development of existing ones.

Development and adaptation of computer systems is done taking into account SKB's general requirements regarding development of computer systems, procedures for information storage and requirements on information security. When selecting computer systems, commercially available systems should be chosen above own-developed systems if they are deemed to adequately meet the functional requirements.

The development activities that are planned to meet the needs of detailed site investigations are summarized below.

Programme

SKB's new tunnel mapping system (see Section 14.4.2) will be further developed, based in part on the experience gained when applied to the expansion of the Äspö HRL. Further development will also be done of the existing system for borehole mapping, Boremap.

Computer systems for investigations and monitoring will be modernized and refined. An important aspect of groundwater monitoring during construction of the final repository tunnels is that observations that influence the construction process and have a bearing on decisions must reach the interested party within an acceptable time. This requires both efficient data management and suitable visualization techniques for different kinds of primary data with simultaneous presentation of tunnel geometries and relevant interpretation of geological conditions.

A model database that supports uniform, traceable and searchable management of existing and future models and model components will be created. The current primary database, Sicada, will be further developed to meet the requirements of detailed site investigations.

Software and an associated database for the geographic information systems (GIS) will be augmented and an interface will be created for online publication of GIS data.

The current modelling environment for geological modelling, rock visualization system (RVS), does not fully meet the needs of the detailed site investigations . A study is currently under way to determine whether the program needs to be replaced or whether it should be improved and a supplementary system added. Present-day model codes for groundwater flow modelling (see 26.4.1 “Surface hydrology and near-surface hydrogeology” and 26.4.2 “Hydrogeology in the deep rock”) will be retained and refined.

There is at present no system for impact assessment that includes processing, presentation and interpretation of data from hydrogeological, hydrological, hydrochemical and ecological monitoring. The system must be able to be used to identify and quantify changes in key parameters in the surface system. It must also be able to be used to distinguish anthropogenically induced variations and trends – due to construction and operation of the Spent Fuel Repository – from natural variations and trends, including those that stem from other anthropogenic influences. Existing commercially available software will be evaluated relative to identified system requirements. It will then be decided whether the need can be met by acquiring available software or whether own software needs to be developed.

Development of interfaces has been initiated with regard to software for modelling of hydrogeochemical reactions. This pertains to the coupling of the DarcyTools and PFloTran codes, which are intended to be used for reactive transport modelling with associated three-dimensional visualization. Existing software for mass balance analysis, multivariate data analysis and mixing modelling will be adapted to new technical requirements.

A number of commercial tools are currently used for modelling of rock mechanics. These programs can probably be retained, but must be developed to integrate the different tools and automate the work.

14.5 Execution methods and construction materials

14.5.1 Grouting

Current situation

RD&D Programme 2010 described the fine sealing project, where an 80-metre-long tunnel, called the TASS tunnel, was built in the Äspö HRL at a depth of 450 metres. 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. Two different grouting materials were used: a cementitious grout devised by SKB and Posiva with a pH lower than 11.0 in connection with leaching from hardened grout, and silica sol.

RD&D Programme 2010 explained that the grouting technology SKB needs must:

- Meet the requirements related to long-term safety.
- Cope with the specific sealing scenarios that follow from the hydrogeological conditions on the site.
- Meet the requirements on quality-assured and efficient execution.

For some time now, SKB has both followed and contributed to research on grouting methods. This includes characterization, materials and their behaviour during grouting (penetration, filtration, etc). Recent years’ research has led to a deeper understanding of the grouting process and how the injection grout spreads in the rock. A methodology has been developed called Real Time Grouting Control (RTGC) that permits control of the grouting process in real time and setting of stop criteria based on the spread of the grout in the rock. Grouting of a borehole can be considered to be finished when the grout has created a zone of the desired size around the tunnel. The stop criterion may be that the flow diminishes below a certain level, that a predetermined quantity of grout has been pumped into the hole, or that a certain amount of time has passed. The latter criterion is used for verification of dry holes. The methodology makes it possible to optimize the process and not continue grouting for longer than necessary. It also makes it possible to control hydraulic widening or propagation of fractures in the bedrock during the grouting process. A current example based on grouting in the City Tunnel beneath Stockholm was reported by Tsuji et al. (2012) and Stille (2012). This has been analyzed and today's grouting knowledge has been compiled from characterization to acceptance of the finished job, see Figure 14-4. Some key issues were raised that must be taken into account in the planning and execution of grouting:

- Characterization of the hydraulic properties of the rock and how the hydraulic system reacts to the tunnelling works is an uncertainty that must be monitored during tunnelling.
- The properties of the grout obtained in laboratory are not always the same as during grouting. Small variations in the properties of additives or in the mixing procedure can have a great impact on the end result. Rigorous quality procedures and documentation of material used and the mixing procedure are important. A Ph. D. thesis indicates that ultrasonic measurement of the flow to grouting packers could contribute to increased quality control of the rheological properties of the injection grout (Rahman 2013).
- Existing models for internal erosion of the grout and backflow to the tunnel are conservative, so outflow of grout to the tunnel must be monitored carefully during the grouting work.
- Real Time Grouting Control should be used to verify and, if necessary, update predictions of fracture apertures and grout spread. This is one of the uncertainties due to probable heterogeneities in the hydraulically connected fracture network.

Experience from the fine sealing project served as a basis for design of the grouting works prior to extension of the Äspö HRL, see Section 14.3. In that project, a requirement was set on a maximum head drawdown of 50 metres of on maximum in-leakage . Procedures were established so that monitoring of tightness became an integrated part of construction monitoring, where measurement of in-leakage was also included, see Section 14.3. By integration with the designer's responsibility for the grouting strategy and a construction inspection with many opportunities for observation in the tunnelling cycle, the requirement on maximum drawdown was satisfied, see Figure 14-5. Methods and procedures for inspection of the grouting work and control to meet design requirements worked well in the contracting work in this project.

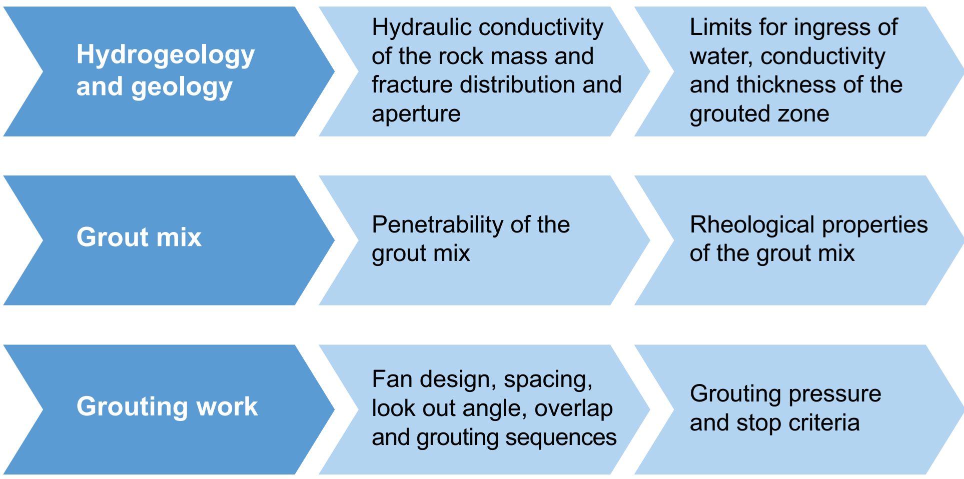


***Figure 14-4.** Overview of the most important aspects in the design and execution of grouting (Stille 2012).*

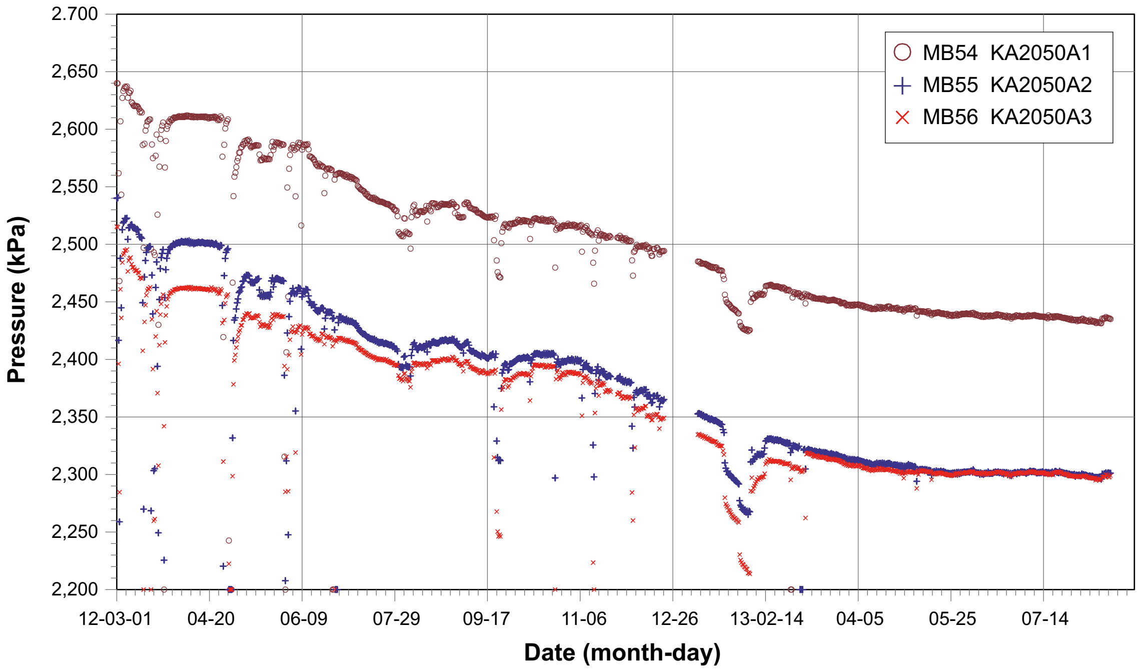


Figure 14-5. *Drawdown during the construction period (March through December 2012) recorded in the Äspö HMS in observation hole KA2050A. The three curves represent three packered-off measurement sections in the borehole, which is drilled from the level above the extension area. The upper curve is the measurement section nearest the extension area at the 400-metre level. The disturbances in the curves are caused by construction activities, mainly drilling for grouting. The total drawdown is about 15 metres.*

Programme

Method and strategies for the grouting work and its quality assurance need to be further developed and modified to meet the various requirements and situations that can be expected during construction of the Spent Fuel Repository.

SKB intends to deepen the study of documented grouting results from the extension of the Äspö HRL with post-analysis of grouting data to improve the strategy for controlling the grouting work for the Spent Fuel Repository with Real Time Grouting Control. An important component in this is more efficient software in the grouting equipment's logger. SKB has collaborated with other actors in the rock construction business on this matter, which has stimulated leading manufacturers of grouting equipment to update the software in the logger equipment to provide the operator with better information on the grouting process.

14.5.2 Rock excavation

Current situation

RD&D Programme 2010 explained that SKB had chosen drill-and-blast as the rock excavation method for the Spent Fuel Repository. The greatest advantages of drill-and-blast are high flexibility, mature technology and comparatively low cost, not just for the rock excavation work itself but also for subsequent backfilling work. The method can easily be adjusted to different rock conditions by adaptation of tunnel shape and blasting design to prevailing requirements and site conditions.

Experience from rock excavation for the fine sealing tunnel in the Äspö HRL (Karlzén and Johansson 2010) was applied to the extension of the Äspö HRL. The results obtained were reformulated in technical requirements for tendering specifications for procurement of the rock construction works. This also included the prescribed blasting plan (drilling and charging plans) for the different tunnel areas. A proposed drilling plan is normally requested when a drilling contract is procured. However, SKB was influenced by positive experience from previous tunnelling in the Äspö HRL regarding contour control and minimization of blast damage. In summary, the requirements in the tendering specifications included:

- Requirements on contour control, which in turn set requirements on the drilling equipment.
- Requirements on charge concentrations. In contrast to previous tunnelling where cartridge explosives had been specified in the contour to ensure the prescribed charge concentration, string emulsion was now allowed in the contour as well, since this is the most common procedure in normal tunnelling. This created the necessary conditions for testing how well requirements on restrictions of the EDZ can be achieved in more industrialized tunnel production.
- Requirement on precision in detonators.
- Requirements on the logger systems on the drilling and charging equipment.
- Requirements on the contractor's supervision of his works.
- Documentation requirements.
- Contractor's management system.

Prior to the start of construction, the Contractor's inspection plans and checklists for the tunnelling works were examined to ensure procedures for supervision and its documentation.

The tunnelling was mostly carried out during 2012 and was finished with rock support and fitting-out during the first 4 months of 2013. Contour control largely lies within stipulated requirements, except in part of the right-hand wall, see Figure 14-6.

Programme

Experience from the rock excavation works done under more conventional production conditions in conjunction with the extension of the Äspö HRL is positive with respect to the requirements made by the backfill on the tunnel contour. The principles of how inspection was carried out against specified requirements are illustrated in Figure 14-7. As-built inspection included both the contractor's inspection plans and checklists, as well as SKB's verification that these procedures were followed and that the requisite documentation was handed in. Inspection of tunnelling results could be done after the grouting works and after blasting, where the finished contour could be observed. With the support of the photogrammetric aid for tunnel mapping (see Section 14.4.2), geometrical control of the tunnel contour could also be done in less than 24 hours after blasting, which permitted continuous feedback to the Contractor regarding how well he was complying with the contour requirements, see Figure 14-7.

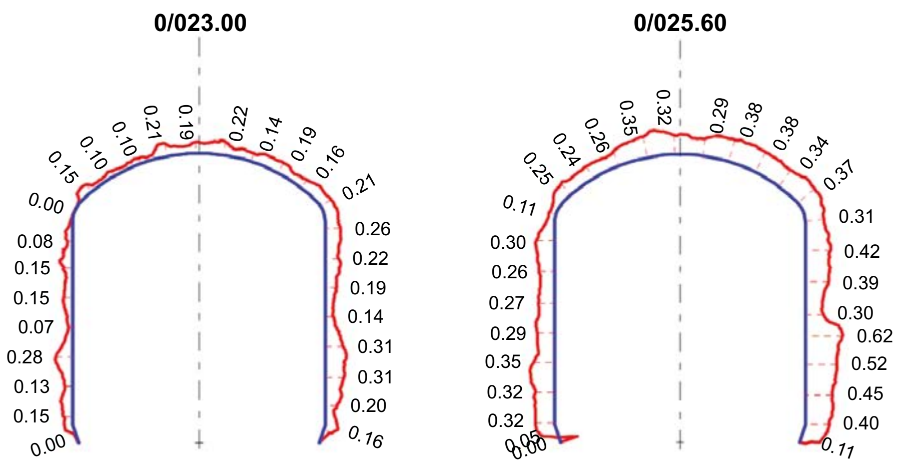


Figure 14-6. *Example of verification of actual tunnel area against specified requirements. Blue contour = theoretical section. Red contour = as-built contour. The illustration at the left is at the beginning of a round, at the right at the end of the same round. Dimensions indicate overbreak in metres. Misdirected drilling in the right-hand wall leads to local overbreak.*

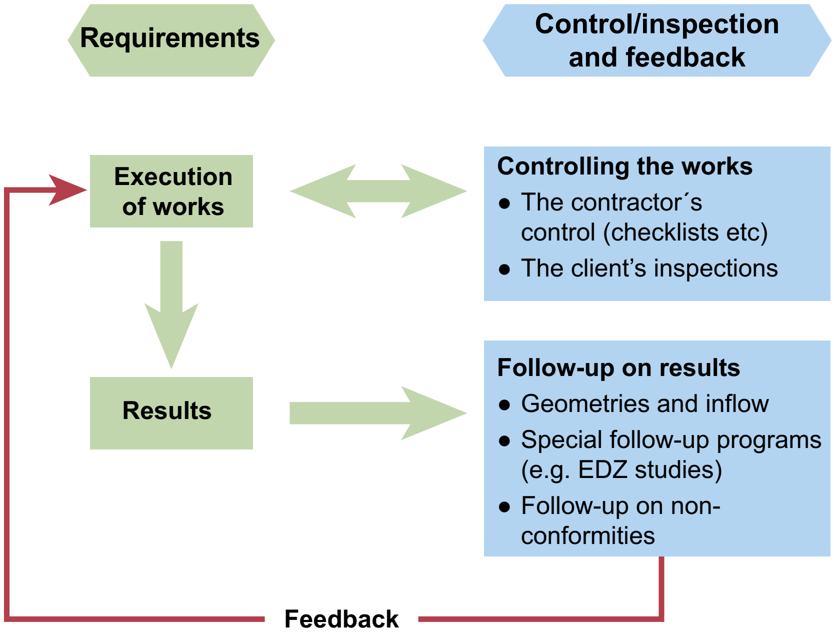


Figure 14-7. *The main principle for quality assurance of the tunnel works in conjunction with the extension of the Äspö HRL.*

Inspection of the extent of the EDZ, especially in the floor, cannot be done until the blasting work is finished, so a separate project has been initiated. The most important components of the project pertain to:

- Requirement formulations.
- Strategies and inspection methods to minimize the EDZ.
- Requirements on tools to verify the extent and hydraulic properties of the EDZ.

The goal of the project is to define and develop requirements, strategies and methods needed for design and procurement of rock construction contracts and construction of the Spent Fuel Repository. This includes excavation works to the extent needed to plan and control the rock works in connection with the construction of the Spent Fuel Repository, and to verify the initial state with respect to the extent of the EDZ.

The requirements on execution and inspection, as well as strategies for quality assurance, were defined as described above in conjunction with design, procurement and execution of the rock construction contracts for the extension of the Äspö HRL. An approximately 50-metre-long niche with the same dimensions as a deposition tunnel will be investigated with respect to blast damage in the floor.

The proposed strategy for verification of the extent of the EDZ is based on the following principles and methods:

- Verification that the execution of blasting complies with requirements on drilling, charging and detonation with documentation and analysis of deviations.
- Geometrical control of the tunnel contour to ensure that tolerances in contour control are complied with.
- Follow-up using geophysical methods on the tunnel floor after blasting. Mainly ground-penetrating radar, but possibly also geoelectric methods will be used.
- Mapping of the tunnel floor for supplementary visual assessment of the extent of the blast damage.
- Construction of a model with the tunnel floor's topography, fracturing and estimated extent of the EDZ.
- Hydraulic characterization of the EDZ in short boreholes (1–2 metres) by means of injection tests.
- Modelling of the transmissivity of the EDZ.

During 2012, SKB carried out pilot tests of methods and equipment in a niche in the Äspö HRL. Special packer seals were then devised for characterization of the hydraulic properties of the EDZ. Since the EDZ is believed to be shallow, the design of the packer seals was aimed at being able to measure directly beneath the tunnel floor. A borehole extension was devised in the form of a plate with tubes that is bolted to a flat-ground surface around the borehole (diameter 150 millimetres), see Figure 14-8. Several variants of packer seals were devised, with one or two measurement sections. The logger equipment is mounted on a small cart together with the gas cylinder for pressurizing the packer seals so that it can easily be moved around on an uneven blasted floor, see Figure 14-9. The test in the niche in the Äspö HRL was a Site Acceptance Test of the equipment (see Section 15.1.2 "General about production of machines and technical systems") and was not intended to characterize the hydraulic properties of the EDZ. Specified requirements on the equipment have been met, including:

- The equipment shall be designed for absolute pressures of up to about 4,000 kilopascals (kPa).
- There shall be a straddle-packer system for the active hole and a single-packer system for the observation hole. In the active hole, it shall be possible to perform measurements in sections of 100 and 200 millimetres, and between a single packer and the bottom of the borehole.
- The equipment shall be able to make sectional determinations of transmissivity in the interval $5 \cdot 10^{-10}$ to $5 \cdot 10^{-7}$ m^2/s. This means that if the pressure disturbance is 500 kPa, the flow interval will be 1 to 1,500 millilitres per minute.
- The equipment shall also be able to measure outflows before the start of injection.

Measurement and analysis of the extent and hydraulic properties of the EDZ in a niche from the Äspö HRL's extension will be carried out during 2013 and serve as a basis for the assessment of whether further development of requirements and control of the excavation works will be needed.

SKB is monitoring technology development with regard to other methods than just drill-and-blast. Rapid development of technology has occurred in the area of mechanized rock excavation, for example by tunnel boring machines (TBMs). This development has in part been driven by the needs of the mining industry, so that today's TBM technology today permits greater flexibility with regard to curve radius etc, while at the same time the rate of advance is increasing due to stronger engines. Diamond wire sawing has increased in complex civil engineering projects. SKB intends to evaluate how these methods could be used to obtain a smoother floor in deposition tunnels.

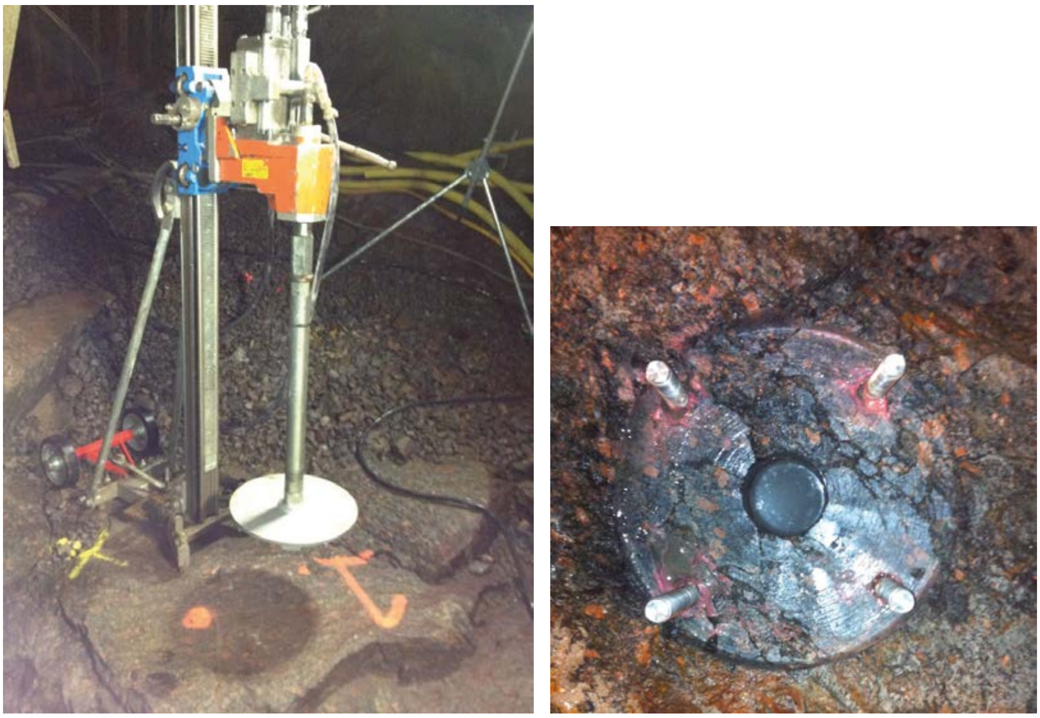


***Figure 14-8.** Left: Diamond wheel for flat grinding of the floor around a borehole. Right: Flat-ground floor around the borehole before the borehole extension is bolted to the floor. Note also the blast-induced damage fractures.*

Figure 14-9. Test of equipment for characterization of the hydraulic properties of the EDZ. The borehole extension (A) for water injection into the active hole can be seen at the left. The injection packer (B) is anchored to the borehole extension at the level of the rock surface. A packer system installed in an observation hole can be seen in the middle (C). The cart with injection system and data collection system can be seen at the right.

14.5.3 Rock support

Current situation

Rock support may be needed in the repository tunnels for the sake of occupational safety. The conventional rock support method in tunnels in Scandinavia is a combination of bolting and shotcreting. Bolting can be done selectively to secure individual blocks, or systematically to create an arch effect. Shotcreting can secure small blocks in good rock conditions, either together with bolting or by itself. Bolts fitted with washers interact with the shotcrete. There is a conflict with requirements on backfilling of the deposition tunnels (see Section 13.1.2), and systematic use of shotcrete in tunnels top seal.

When the fine sealing tunnel, TASS, was completed, some rock support was also required to ensure occupational safety for future experiments in the tunnel. Ittner (2011) mapped the tunnels and analyzed different instability situations. Three different situations required rock support:

1. Local risk of stress-induced spalling in projecting rock, caused by irregularities in the blasted contour.
2. Potentially loose blocks formed by natural or blast-induced fractures.
3. Partially loosened rock at the end of a blast round where the explosive effect is influenced not only by the bottom charge (primer), but also by the geometric variation in the tunnel contour caused by the look-out angle created during blast hole drilling.

The situation described in points 1 and 3 above requires surface support, provided by shotcrete or wire mesh. This is used frequently in areas with good rock quality in the Äspö HRL. The situation described in point 2 is addressed by bolting. Ittner (2011) proposed limited rock support measures, mainly in parts of the tunnel roof in the TASS tunnel. The greater part of the roof and wall surfaces are included in the systematic rock maintenance programme in the Äspö HRL with recurrent inspection and scaling of loose blocks.

Programme

The rock support in the TASS tunnel in the Äspö HRL, which was carried out in 2011, is part of the continuous rock maintenance programme and will be monitored as a basis for decisions on measures in the Spent Fuel Repository's deposition tunnels.

The rock conditions in Forsmark may require more systematic rock support due to different rock and stress conditions. SKB intends to monitor the development of mechanical equipment for application of wire mesh, where LKAB is conducting technology development for the needs of the Kiruna Mine. Collaboration with Posiva is also planned in this occupational safety question.

14.5.4 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.

Current situation

A negative experience in connection with the grouting works for the extension of the Äspö HRL was the long hardening time for the low-pH injection grout that was also used in 2007 in the fine sealing project in the TASS tunnel.

A recent study (for later publication) shows that the Spent Fuel Repository's barrier function is not affected even if ordinary cement is used down to a depth of at least 200 metres. Ordinary cement is therefore now planned to be used in the construction of the repository's accesses down to this level.

Programme

SKB has installed 20 rock bolts with low-pH grout in the experimental niche. The bolts will be monitored over a ten-year period. An aspect of particular interest is to study 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 finding a uniform method for measuring the pH of cement products.

SKB needs to improve the mix design for low-pH grouting material with acceptable accelerators. Additives included in the current mix design for low-pH concrete are being phased out of the market by the manufacturer. This is a common trend in the construction industry when new chemical products are introduced, for example in the area of cement and concrete. SKB needs to develop a more long-term strategy for maintaining useful grout mix designs with good penetration and acceptable hardening times. They must also have an acceptable composition from the viewpoint of long-term safety.

The programme also includes continued observation of the experiments in the Äspö HRL with rock bolts and shotcrete as well as corrosion tests on rebar. The first samples will be taken during the coming three-year period.

SKB is participating in an industry-wide project concerning performance requirements on underground facilities with respect to the chemical environment. The overall goal of the project, which is headed by Chalmers University of Technology, is to further develop standards to meet performance requirements on underground facilities with respect to groundwater composition and underground atmosphere.

15 Technical systems

This chapter covers technology development of machines, equipment and processes for handling of canisters, buffer and backfill material, etc used in the operation of the Spent Fuel Repository and Clink.

15.1 Requirements and premises

15.1.1 Overall planning and control

The design of the Spent Fuel Repository is currently in the system design phase. This will later pass into the detailed design phase, after which the construction phase begins once the necessary licences have been obtained to start construction.

System design is carried out as integrated design that includes the whole Spent Fuel Repository and is basically a revision and detailing of layout D, which was included in the 2011 applications. During the detailed design phase, a division is made between the different parts of the facility with respect to when the supporting material is needed for the part in question prior to the start of construction. Parts that come early in the construction phase are preparation of the construction site plus construction of ramp and skip shaft and the auxiliary functions this requires.

The times for this are given in the Nuclear Fuel Programme's main timetable, which was presented in general terms in Section 2.3 "Plan of action for spent nuclear fuel".

During the system design phase, extensive logistics studies are done to analyze bottlenecks in the activity and possible problem areas. The logistics studies are an interactive process where different solutions for a given function are tested and then incorporated into the design work.

During the system design phase, the foundation is also laid for the Spent Fuel Repository's future overall production system, in which different planning and logistics programs are important tools.

15.1.2 General about production of machines and technical systems

Many of the machines, vehicles and technical systems etc that will be used in the Spent Fuel Repository and Clink during future operation for transport of canisters, buffer and backfill material etc can be regarded as specific for SKB. Specific for SKB are machines and technical systems not currently available to purchase on the open market and therefore in need of being developed. They must have high operational availability and be able to function together with other machines and technical systems, and they must be user-friendly for the personnel.

Development and industrialization of the machines and technical systems required will follow a technology development model that proceeds incrementally through different phases, see Figure 15-1. Requiring machines and technical systems to be developed in accordance with the phases in the technology development model increases the probability that they will meet requirements on function, reliability and availability.

By "operational experience" is meant how well a system have performed once it has been put into use. Both the experience of the personnel and experience from maintenance databases will be used to broaden operational experience.

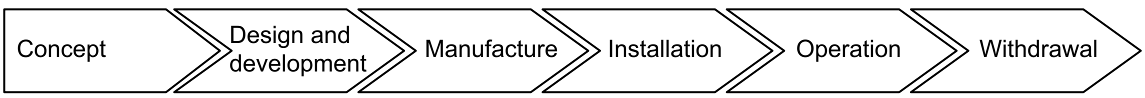


Figure 15-1. *General phases in the development of machines, systems and equipment.*

Concept phase

In the concept phase, a design is made from a concept based on the needs and experience that are available either internally or externally. The choice of conceptual solution for the technical system shall be made in consideration of functional system requirements, high operational safety, good maintainability and low lifetime cost. The latter refers to the costs which the technical solution generates throughout its technical lifetime with respect to operational and maintenance aspects. A personal safety analysis, risk analysis and risk assessment are carried out in parallel with the technical analysis. This is done to ensure low safety risks and technical risks in connection with the operation and maintenance of machines and technical systems.

Design and development phase

The design and development phase entails making a choice between two main alternatives for design and development. The preferred alternative is to procure the service from an external supplier. If this is not possible or economically defensible, design and development are carried out in-house. The contents of this phase differ depending on which of these alternatives is chosen.

If an external supplier is chosen, a procurement process must be carried out. This entails a formal process where it is important that the tender specifications are clearly defined and that any uncertainties are limited. It is important that requirements will be focused on function and not design of the technical solution.

The alternative of in-house development entails a more detailed development process. In some cases it may also entail greater flexibility, which can be an advantage if the set of requirements changes during the course of the project.

Manufacturing phase

The manufacture of technical systems can be procured from a supplier both during the design and development phase and during the manufacturing phase. The manufacturing process may therefore look different depending on which way is chosen after concluded design and development.

At the end of the manufacturing phase, a Factory Acceptance Test is performed. This is an acceptance test of the system before delivery to the intended operating environment.

Change management entails specifications once being approved and validated and subsequently changed must undergo renewed review, risk assessment and validation. Instead of redoing the entire validation, risk assessment and review work, a procedure has been developed to handle this. When the change management process is launched and specifications are changed after an approved acceptance test, this procedure takes over. The change management procedure continues until the process, product or technical system is withdrawn. The procedure also applies in the concept phase and in the design and development phase, but only for the specifications that have been approved, in other words requirement specifications on a higher level.

Installation phase

The purpose of the installation phase is to verify and validate the intended performance of the technical system in its operating environment in a Site Acceptance Test.

The following is required in for this:

- The technical system must have been approved once the outstanding issues after the acceptance test have been resolved.
- Training must be offered in the use of the system.
- Documentation such as instructions and technical documentation must be available.

After the installation phase, the technical system is approved for operation.

Operating phase

Documentation of how well the technical system performs shall be done from the day it is put into operation for the first time. Follow-up of operating experience shall be used to:

- Support verification and validation of the operational safety characteristics during the warranty period.
- Provide support for updating of plans for preventive maintenance.
- Provide support for design improvements.
- Transfer experience to other technology development projects.

Follow-up of operational experience shall as a minimum include information on:

- Faulty function.
- Corresponding fault message in vehicles and subsystems.
- Faulty component.
- Cause of fault.
- Consequence.

Relevant operational experience shall be reported in the maintenance system and its database.

The procedure for change management ensures that the risk assessment for the technical system is kept up-to-date and that all changes are documented via configuration management.

Withdrawal phase

By "withdrawal" is meant here withdrawal of a machine or a technical system in order to compile experience from operation for development of the next system, to be able to recover and recycle the material in the product, and to enable recycling or destruction to be done in a safe and appropriate manner.

In order for the withdrawal phase of technical systems to be carried out safely, all risks associated with withdrawal must be determined and evaluated. Withdrawal of the technical system can be done in parallel with regular operation or during a scheduled outage. Good planning is required in both cases.

15.1.3 Development of production systems

In order to create an overview and also to document the need for machines and equipment etc for the Spent Fuel Repository, a database has been established where machines and equipment have been arranged in system groups in accordance with practice in the nuclear power industry. System group 200 includes equipment for reception, handling, storage and rock work while system group 400 includes transportation systems with boats, terminal vehicles and different types of transport casks and containers. The greatest need for extensive technology development exists within systems groups 200 and 400.

The inventory work for the construction of the database has been under way for many years. Identification of the need for machines and equipment connected to the system groups 200 and 400 comes mainly from the work with production lines. This is exemplified in Figure 15-2 with the production lines for canister and buffer. One of the objectives has been to try first to find standardized equipment that is available on the market, second to investigate whether it is possible to modify standard components, and last to develop products in-house. Machines and technical systems shall be built as far as possible from standard components. The total number of products, when their development should be initiated and concluded and the costs for this are also documented in the appropriate database.

A production system for the Spent Fuel Repository is required in order to carry out these activities efficiently and reliably. The production system in turn contains a number of subsystems that can monitor and report the status of the facility and the facility's resources, information systems, product status and automated functions. Figure 15-3 gives an example of this.

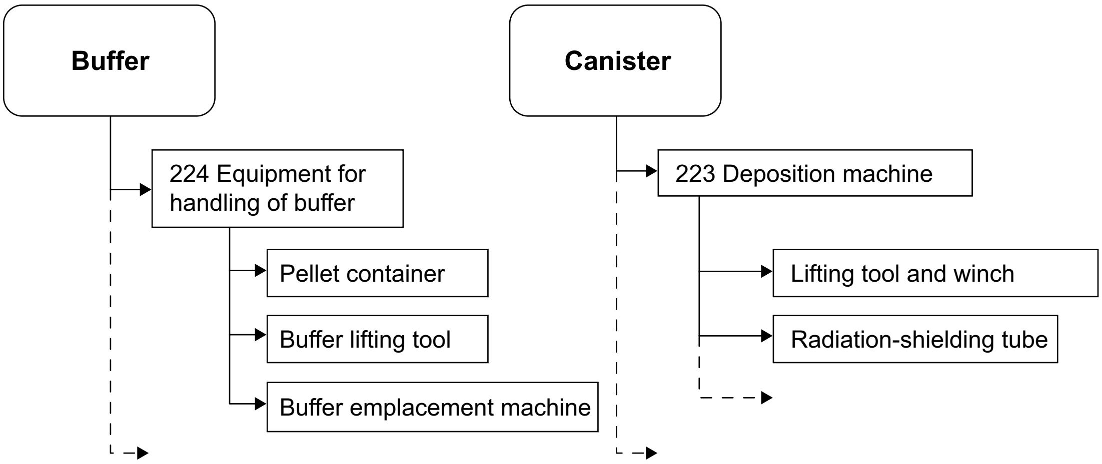


Figure 15-2. *Example of machines and equipment identified in the production lines for buffer and canister.*

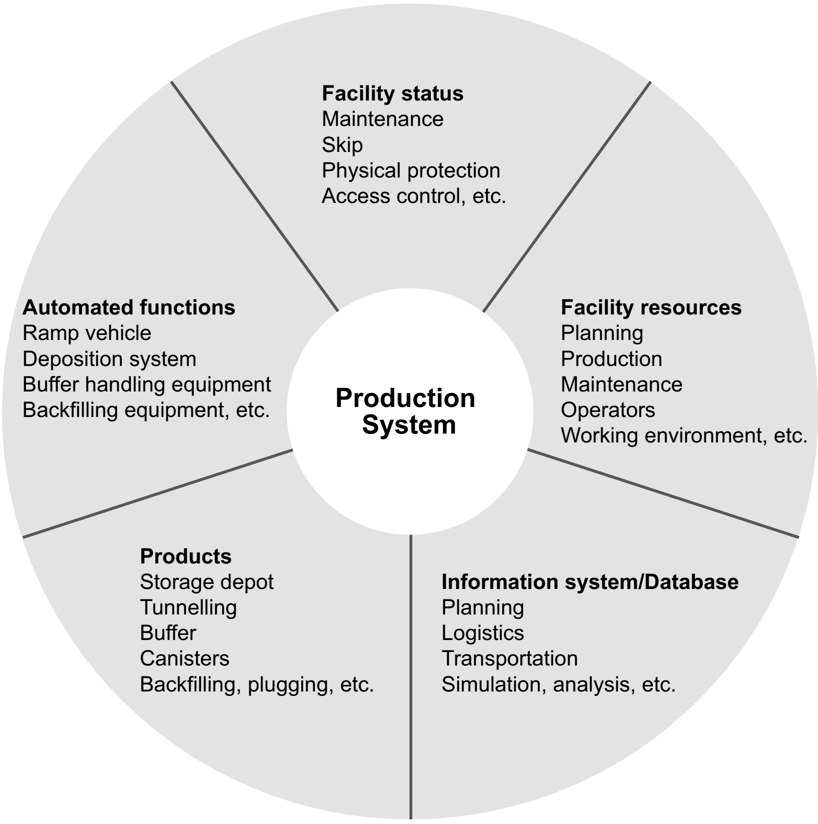


Figure 15-3. *The production system with auxiliary systems, facility parts and functions is a suitable starting point for what will be needed for the Spent Fuel Repository.*

The total number of products, with a focus on machines and equipment, is greater than only the products that require extensive technology development. The total number of products, when development of them should be initiated and concluded and the costs for this are described in the database. Figure 15-4 illustrates in general terms what the database contains for a selection of products in system group 200 and shows future activities. The timetable for the execution of these activities will of course by influenced by the main timetable for the Nuclear Fuel Project, see Section 2.3 "Plan of action for spent nuclear fuel." The database includes the phases concept, design and development, fabrication, installation, operation and withdrawal, and these phases are presented with different colour codes in the database and Figure 15-4.

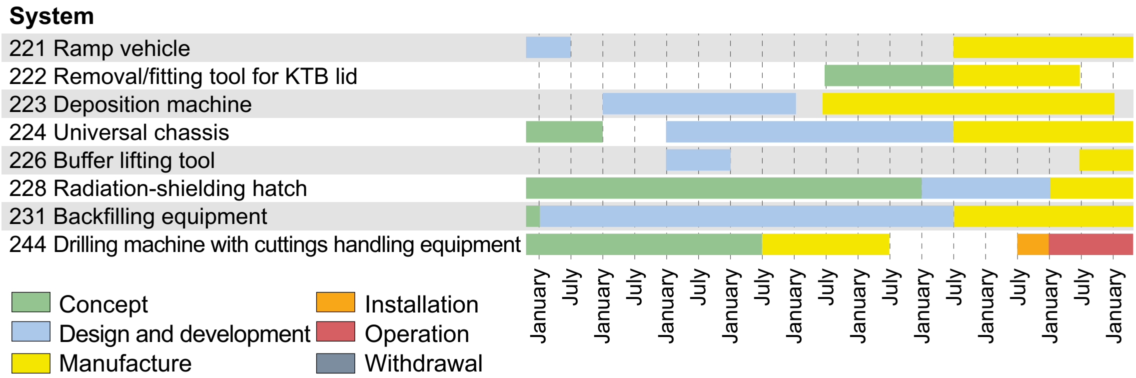


Figure 15-4. *Presentation of the contents of the database for machines and equipment, which also schematically illustrates the different development phases.*

15.1.4 Strategies and activities to achieve established goals

Development of SKB-specific systems proceeds incrementally over many years, which means that technology development needs to start long before the equipment has to be in place. Nor is it possible to pursue technology development for all systems and equipment at once. In establishing a timetable, besides taking common milestones into account it is also necessary to even out the resource need over time to make it possible to have an appropriate organization with a reasonable size for technology development.

Prototypes are needed in some cases for verification and validation of the systems. Parts of today's technology development relating to products in the KBS-3 system are therefore concerned with development and design of prototypes. A choice must be made between whether the prototype will be developed for its ultimate purpose or only be used to a limited extent, for example to show that a certain function can be performed in the desired manner. One of the purposes of producing a prototype is to obtain and test a realization in order to reduce the risk in the production of the final product. The produced prototype is evaluated and the results are considered when formulating requirements and specifying the final product.

The aim of product and process development is to include early in the development process not only aspects related to nuclear safety, but also aspects related to personal safety, operational safety and maintainability. In other words, to influence the design while it is inexpensive to introduce changes. The planning is based on SKB-relevant parts of standards from the USA.

If the produced prototype meets all specified requirements, it can be industrialized and implemented. In conjunction with this, the prototype must be analyzed with respect to e.g. lifetime cost, functional reliability and maintainability. In many cases, these analyses lead to the conclusion that it is cost-effective to restart the development of the final product. The experience gained from the prototype makes it possible to develop the final product with less risk and facilitates many decisions. Much of the analysis work can also be used in the work of developing the final product.

Using the methodology described above is an important part in realizing the industrialization of the product and its deployment in the relevant process. The work method permits continuous verification that a technical system under development fulfils the specified requirements. An acceptance test in the supplier's environment is then performed at delivery. The purpose of the test is to ensure that the requirements made on the design are met by the manufactured system.

In the installation phase, an acceptance test is performed in the operating environment, along with the trial operation that is required to verify and validate that the technical system fulfils the specified requirements. The purpose of the test is to verify and validate the integration of the technical system with other systems and its functionality in the actual production environment, including solutions for maintenance.

It is important to describe when the different phases will be executed for the particular machine and equipment. This makes it easier to envisage industrialization and deployment of those machines and equipment in the KBS-3 system that, for instance, require special adaptation, as well as what the coming development phases contain.

15.2 Products and processes

Most of the technology development needs to be completed prior to establishment of the deposition sequence. In cases where prototypes are needed, they must be available and their function must have been verified. Important activities during the coming period are to:

- Continue logistics studies.
- Carry out the design phase according to the technology development model for buffer handling equipment, backfilling equipment and the transportation system for buffer and backfill.
- Commence development of the production control system.
- Prepare and commence integration tests.

Integration tests of buffer, backfill and arched plug require that prototypes are available to transport and install buffer and backfill. The prototypes shall be produced in accordance with the functional and regulatory requirements that exist for the type of product. They must be CE-marked if this is deemed necessary, which is more or less a formality if technology development is pursued according to the technology development model.

Sections 15.2.1 to 15.2.7 comprise brief descriptions of the technology development projects that are under way and that need to be carried out.

Prior to the preparation of the PSAR and the report on safety during construction of the final repository (Suus), see Section 10.3.1, ongoing technology development projects must deliver data for updating of production reports and system descriptions, mainly for the 200 systems and some of the 400 systems.

For updating of system descriptions and production reports for canister, buffer and backfill, results are also needed from ongoing projects concerning the deposition machine, the transportation system for buffer and backfill and buffer handling equipment, including the ramp vehicles.

Intensified contacts with machine suppliers are needed to establish the suppliers' knowledge and interest regarding the future needs for, in particular, SKB-specific machines and systems in the Spent Fuel Repository. Manufacture and design of the final versions of SKB-specific machines and systems shall be carried out by external suppliers in collaboration with SKB, after completed prototype development. The system descriptions are prepared in collaboration with suppliers.

Synchronization of timetables between the different production lines and design of the Spent Fuel Repository must be coordinated with the development of machines and equipment. Overall production systems must be developed for the different processes. These production systems shall describe how the different machines and equipment (the products) are to function within the different production lines (the processes).

Detailed design can be initiated when the design and development phase for equipment for handling and transport of canister, buffer and backfill is concluded. Inspection programmes must also be available, along with equipment for inspection of buffer and backfill installation.

Planned full-scale tests of buffer and backfill must also be taken into consideration so that prototype equipment is available and can be used in these tests, see Section 13.10 "Integration tests".

15.2.1 Logistics studies

Current situation

The need for logistics studies was described in RD&D Programme 2010 and pertained to logistics studies for the entire Spent Fuel Repository. A small demonstration project was carried out during 2010 regarding logistics studies for the Spent Fuel Repository. The purpose of the project was to evaluate procedures and software for logistics simulations. The simulations have yielded valuable information on logistics, which has led to initiation of another logistics study during 2012 focused on skip building, skip hall and transloading hall pending information from the final repository's ongoing system design phase.

The basic model from the demonstration project will also be expanded to include four deposition tunnels in order to show conditions during normal operation as well as with rock haulage. This means that activities such as buffer installation, deposition of canisters and backfilling, and construction of the arched plug are included in the logistics studies.

A planning tool was also acquired in 2011 and has been adapted to the Spent Fuel Repository according to the logistics model from the demonstration project and the machine control system. All data used in the logistics studies will gradually be entered into the planning tool's database. The database will also be used for the machine control system and the production system.

Programme

The results from the logistics studies will be used as a basis for the system design phase and subsequent detailed design. The purpose of the logistics studies is to gather data for different types of decisions during system and detailed design of the Spent Fuel Repository. The project will also be used to support decisions regarding when processes, products and technology should be developed and how this should be done. This is achieved via simulation of models for parts or all of the activities for operation of the Spent Fuel Repository. Rock excavation during the operating phase and shipments of canisters from the harbour in Forsmark and bentonite material from Hargshamn are also included in this. These studies will also be utilized for time and resource planning for the future operation of the Spent Fuel Repository.

15.2.2 Ramp vehicle

Current situation

A vehicle commercially available on the market for heavy transport was delivered to Äspö in the autumn of 2010. During the autumn of 2012, initial tests of the vehicle with a payload of about 100 tonnes started. The payload consisted of a converted waste transport container from SFR that can be loaded to a desired total weight to simulate transport of a future canister transport cask with canister. Figure 15-5 shows the vehicle with payload during the initial tests in the Äspö HRL.

Figure 15-5. *Photo of vehicle with modified waste transport container to simulate future shipments of canister transport casks.*

Programme

The purpose of the ramp vehicle tests is to demonstrate whether a heavy transport vehicle commercially available on the market can be used as a ramp vehicle in the Spent Fuel Repository. The performance, safety, reliability and manoeuvrability of the vehicle are some of the parameters that will be tested and validated. The project also includes compiling more detailed information for the updating of the PSAR. This is done for the purpose of improving and adding greater detail to the system description for the ramp vehicle. The vehicle will also be equipped with a navigation and positioning system in the same way as the deposition machine.

15.2.3 Deposition machine

Current situation

During the past period, a long series of fully automatic depositions has been carried with the deposition machine. The tests have been carried out according to a testing programme and any deviations and disturbances have been documented. Very stable conditions were obtained at the end of the test series, so the tests were concluded after about 200 completed deposition cycles. It was concluded during the tests that it will be necessary to modify the grapple and replace equipment such as limit switches. Experience from the tests will be taken into account in the continued work with the deposition machine.

Programme

Work on further development of the deposition machine will commence during the coming period. This applies in particular to development of the canister grapple, winch and radiation-shielding tube. This work will also include further development of the radiation-shielding hatch, since it interacts with the deposition machine.

15.2.4 Transportation system for buffer and backfill material

Current situation

The work with the transportation system for buffer and backfill material has largely gone according to the programme. It has provided support for tests of backfilling and for system design of the Spent Fuel Repository.

Programme

A project is under way to develop methods, equipment and vehicles required for transport of buffer and backfill material in the Spent Fuel Repository. The project is planned to proceed during the period 2010–2015. The goal is to develop a functional transport and handling system for buffer and backfill material and to some extent verify this with prototypes.

The project also includes gathering supplementary data for documents pertaining to the PSAR, reference activity, reference design, facility description for the Spent Fuel Repository and production line reports for the purpose of improving and increasing the degree of detail in them.

Development will take place in modules to enable separate verifications to be done of the modules and to create a system that can be quickly modified by replacing modules. Another advantage is that working with modules facilitates development, procurement, maintenance competence and spare parts keeping.

The total transportation system for buffer and backfill material is illustrated schematically in Figure 15-6. One important module is a universal chassis with driven and individually steerable wheels and with a wheelbase that permits safe straddling of the deposition holes. Other modules, such as equipment for lifting and emplacement of buffer blocks and rings in the deposition holes, can be mounted on this chassis. The idea is that a robot for emplacement of backfill blocks as well as feeding equipment for pallets and pellet tanks can also be mounted on this chassis. The system also includes a transport vehicle for pallets and pellet tanks as well as a transloading table.

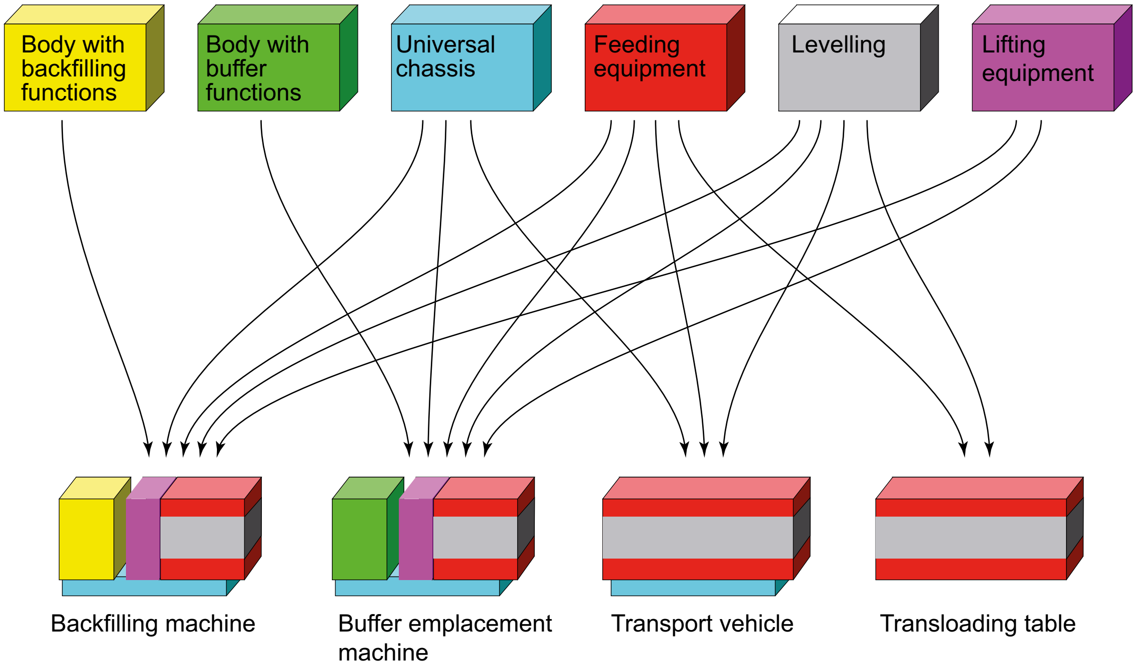


Figure 15-6. *Illustration of the transportation system's different modules. The figure schematically illustrates the composition of the items of equipment and which modules each one contains.*

The function of the transloading table is to permit rational transfer of full and empty pallets and pellet tanks to and from the transport vehicle in the production building, the skip building and the skip hall on the repository level. Handling of the individual pallets and pellet tanks is done by a conventional forklift.

The backfilling machine, the buffer emplacement machine and the transport vehicle will also be equipped with navigation and positioning equipment.

The backfilling machine consists of five modules: universal chassis, robot with tool for handling and emplacement of backfill blocks, lifting equipment, feeding equipment and levelling equipment. In addition to this there is navigation and positioning equipment. Tests of these types of equipment began during 2012 with an industrial robot and a tool developed for handling of backfill blocks. The blocks are fetched by the robot from pallets developed to be a part of the future transportation system. A prototype of the feeding equipment for pallets has also been developed and will be tested in the Äspö HRL. Figure 15-7 shows the equipment used in the tests of installation of backfill blocks by robot in the Bentonite Laboratory and with the use of prototypes of the modules included in the transportation system for buffer and backfill material.

In the tests performed in the autumn of 2012, the backfilling equipment was mounted on a frame that can be moved by the ramp transport vehicle.

The backfilling equipment also includes a unit for placing pellets on the floor in the deposition tunnel and between the backfill blocks and the rock. This equipment will either be a separate unit or be included as a part of the equipment on which the robot is installed.

The machine for emplacing the buffer consists of four modules: universal chassis, buffer handling equipment, feeding equipment and levelling equipment. In addition to this there is navigation and positioning equipment.

The transport vehicle for pallets for buffer and backfill blocks and pellet tanks consists of three modules: universal chassis and equipment for feeding of pallets on two levels. The working name of this equipment is pallet truck. In addition there is levelling equipment as well as navigation and positioning equipment.

The transloading table consists of two modules for feeding of pallets and pellet tanks on two levels which is mounted on a fixed frame. The transloading table also has levelling equipment.

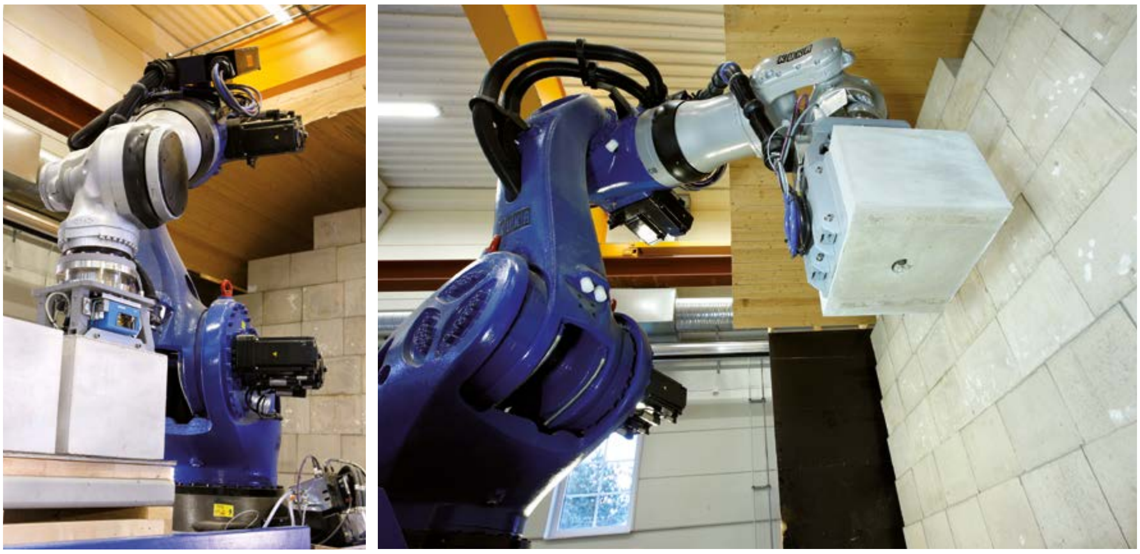


Figure 15-7. Installation of backfill blocks by robot in the Bentonite Laboratory. The picture at the left shows the robot fetching a block from the pallet, while the picture at the right shows emplacement of the blocks.

15.2.5 System for navigation, machine and production control

Complex machines such as the deposition machine and the buffer and backfilling machine require complex onboard computer systems for them to execute the tasks they are intended for in a safe and controlled manner. The suppliers of standard machines are unable to supply all or parts of these systems today. Since there is almost certainly not one supplier who can manufacture all types of SKB-specific systems, it is SKB's strategy to conduct in-house development of the systems that have been identified as common to these machines and technical systems. Systems currently considered for in-house technology development are:

- Navigation system.
- Machine control system.
- Production control system.

Current situation

Navigation system

At the present time there are several suppliers of systems for fully automated vehicles that work underground. None of them has a suitable chassis or a sufficiently well-adapted computer and sensor system to be able to execute more complex tasks such as backfilling or deposition of canisters. SKB has been working for several years on development of the navigation system that is expected to be found in the Spent Fuel Repository.

SKB, together with an external company, has further developed a navigation system of the type that exists in mining vehicles on the market for the deposition machine. The navigation system was implemented in around 2009, and extensive tests have been conducted in the Äspö HRL showing that the desired function and availability can be achieved.

Machine control system

During the period 2009 to 2012, the deposition machine has been tested from manual operation by a driver to use in a fully automated mode. In the fully automated tests, a tool has been used that has a limitation in that only one vehicle can be controlled at a time. The commercial systems often use systems that control not just one but several machines at the same time within the same area of application, so this will be used in the future.

There are currently several suppliers of these systems/tools, but none has the ability to control the planned SKB-specific systems. SKB has been conducting technology development on this problem since 2011. A system with the ability to control several SKB-specific machines is under construction where experience from the tool for control of one vehicle and different machine control systems is being utilized.

The machine control system contains logic that handles functions such as:

- Choice of transport assignments from the list of assignments and breakdown of assignments into driving orders.
- Choice of suitable available machine.
- Choice of shortest route for the particular driving order taking into account available automation areas and any blocked zones.
- Reservation of route segments for the chosen machine according to availability. Critical route sections and transport assignment priority.
- Issuance of driving orders to machines.
- Follow-up of driving order and release of passed route segments.
- Driving order extension, which entails issuance of a new driving order while the machine is still carrying out another order. This is required to achieve smooth operation.
- Reporting to the production control system.

The machine control system is a central part of the logical separation of the different activities in the Spent Fuel Repository in order to permit safe and effective facility and operating activities in the chosen layout of the central area and the disposal area.

Programme

Navigation system

The remaining technology development for the navigation system is further development of the software platform for data management and control.

Machine control system

Tests of the machine control system are planned to begin in 2013 and continue during 2014.

15.2.6 Production control system

The production control system is a tool that is used to plan, schedule and supervise daily production. At present it is not possible to create an environment where all modules in the system can be developed, since parts of the activities that will be carried out in the Spent Fuel Repository are not in place in the Äspö HRL. In order to be able to conduct a coordination test in which most of the steps will be evaluated, the ambition is to develop this environment before the start of operation of the Spent Fuel Repository.

15.2.7 Production building for buffer and backfill material

The equipment that is needed in the production building consists almost entirely of standard machines. The concept phase, according to the technology development model, needs to be carried out to obtain data as a basis for requirement specifications and choice of equipment. This is done within the framework of the buffer and backfill lines.

16 Horizontal deposition – KBS-3H

The KBS-3 method permits canisters to be emplaced either vertically (KBS-3V) or horizontally (KBS-3-3H). Vertical deposition is the reference design, but SKB, in collaboration with Posiva, is studying horizontal deposition to see if it can constitute an alternative to vertical deposition. An evaluation of KBS-3H will be carried out after 2016, and based on this a decision will be made as to whether SKB should proceed with KBS-3H.

16.1 Design of a KBS-3H repository

KBS-3H entails that long horizontal deposition drifts are bored directly from the Spent Fuel Repository's main tunnels, see Figure 16-1. A row of so-called supercontainers is deposited in these deposition drifts. A supercontainer consists of a canister surrounded by bentonite buffer and held together by a perforated outer metal shell. Distance blocks of bentonite clay are placed between the supercontainers, partly to prevent water flow along the deposition drift (hydraulic separation of the supercontainers), and partly so that the temperature in the buffer will not get too high (thermal separation of the supercontainers). The deposition drifts are up to 300 metres long and are divided into two sections by compartment plugs. A drift end plug is installed in the opening of the deposition drift. Both the compartment plug and the drift end plug have lead-throughs that permit artificial watering and venting of the deposition hole. The artificial watering makes the bentonite swell out towards the rock wall, even under dry conditions in the rock.

The rock volume that needs to be excavated for a KBS-3H repository is smaller than for vertical deposition, which also means that smaller volumes need to be backfilled. The facilities on the operations area above ground and the central area and the accesses underground are marginally affected if horizontal deposition is done instead of vertical deposition.

16.2 Current situation and programme

Conclusions in RD&D 2010 and its review

In RD&D Programme 2010, SKB noted which development issues are important in order for KBS-3H to achieve a technical level equivalent to that of KBS-3V. The work of updating and establishing the geometric criteria for the deposition drift, choice of drilling technology and finding suitable equipment for measuring and verifying that the pilot holes meet the requirements was mentioned.

In its review of RD&D Programme 2010, SSM asserted that a switch to KBS-3H requires that a sufficiently exhaustive account of the alternative design be available for review. This applies, for example, to the assessment of long-term safety. At the time in question, SSM did not take a stand on SKB's plans for development of KBS-3H.

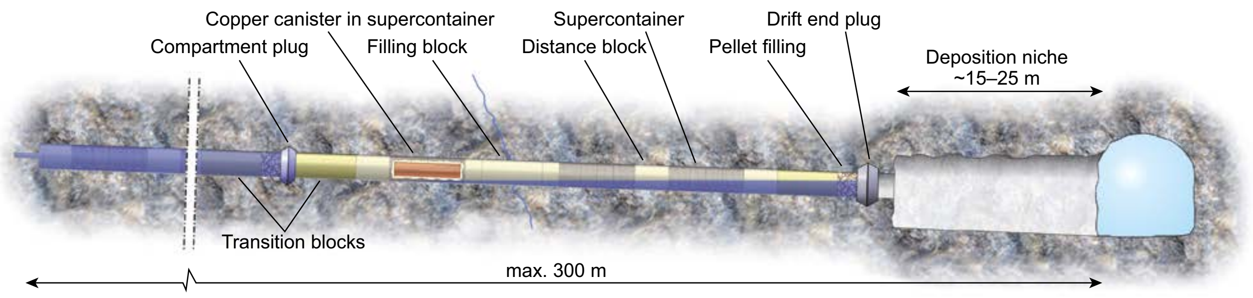


***Figure 16-1.** Schematic illustration of a KBS-3H repository. The deposition drift is divided into two parts by a compartment plug, and the parts are artificially filled with water after the components have been installed. The figure above shows the filling process for the outer section.*

Current situation

Since RD&D Programme 2010, SKB has, in collaboration with Posiva, completed the project phase KBS-3H Complementary Studies. The project phase focused on the function of the buffer and interactions between metal and bentonite. Titanium has been chosen as a reference material for plugs, supercontainer shell and other small support components. The choice of titanium instead of steel largely eliminates problems with interactions between metal and bentonite as well as production of hydrogen gas. Copper, the material the canister is made of, was also a possible alternative for the supercontainer shell, but titanium has better mechanical properties for this application. The requirements on the deposition drifts have been established (SKB 2012) and a testing programme has been initiated at the Äspö HRL to choose the drilling method and a measurement method for verification of the pilot holes, see Section 16.3.

After completion of the project phase, the judgement is that a concept based on the DAWE design (Drainage, Artificial Watering and air Evacuation) using titanium has good prospects of fulfilling the requirements made with respect to long-term safety. The design is also judged to be feasible with respect to construction, manufacture and installation.

Based on these results, SKB and Posiva decided in 2011 to initiate the next stage. The goal of this stage is to compile data and raise design and system understanding for KBS-3H to a level that permits a PSAR to be prepared. This will ultimately permit a comparison to be made between KBS-3V and KBS-3H.

Programme

Main activities during the coming three-year period are to:

- Evaluate chemical erosion and earthquake-induced shear along fractures for KBS-3H.
- Prepare production line reports, specific for KBS-3H, and update other design-specific documentation.
- Verify components on full scale by demonstration.
- Perform an evaluation of the long-term safety of a hypothetical KBS-3H repository in Forsmark.

After the current project phase, KBS-3H will be evaluated and a decision made on whether to continue development and implementation. SKB assumes that if KBS-3H were considered as an alternative, SKB would apply for this, which would lead to consequences for timetables and licensing.

16.3 Demonstration in the Äspö HRL

Current situation

A full-scale demonstration of the KBS-3H technology, called the Multi Purpose Test (MPT), see Figure 16-2, is being conducted to verify the implementation of the DAWE design and the function of the components when they integrate with each other (subsystem test). The test includes manufacture of full-scale components, assembly and deposition of a supercontainer and distance blocks, plus installation of a compartment plug with associated filling components. The deposition drift and the components are instrumented so that the initial course of events can be monitored. After monitoring, the test will be dismantled so that sampling and analysis of the bentonite can be carried out.

Previous studies of excavation of deposition drifts indicate that directional core drilling has the greatest chance of succeeding. The straightness of the cored borehole must however be verified with a device for deviation measurement, see the programme for geology, geophysics and geodesy in Section 14.4.2 "Modelling and investigations". It must therefore be ensured that the deviation measurement equipment is capable of measuring with the required precision.

In order to create more repository-like conditions for the tests, a new niche for KBS-3H was excavated in conjunction with the extension of the Äspö HRL at a depth of about 400 metres.

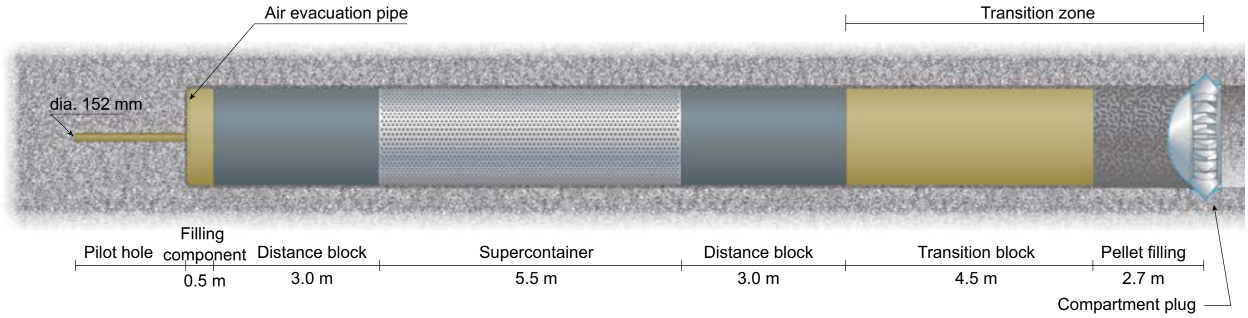


Figure 16-2. *Schematic illustration of the Multi Purpose Test (MPT) and its components.*

Programme

The current work with the Multi Purpose Test will continue with the goal of demonstrating:

- Full-scale manufacturing, handling and transport of a supercontainer with bentonite buffer.
- The technique for deposition of a supercontainer and buffer components of bentonite.
- The DAWE design with full-scale artificial watering of the buffer in the supercontainer and distance blocks.
- The function of the compartment plugs in connection with artificial watering.
- A methodology for installing sensors and cabling in the KBS-3H components in preparation for possible future system tests.

Current work to evaluate drilling techniques and deviation measurement equipment as well as establishment of a new test site at the Äspö HRL will continue.

16.4 Long-term safety

Current situation

A preliminary assessment of long-term safety has previously been carried out for a KBS-3H repository with data from Olkiluoto. The work was presented in the report (Smith et al. 2008). The conclusion of the safety assessment is that KBS-3H is a potential design for a repository on that site, in terms of long-term safety and for the conditions prevailing at Olkiluoto.

Programme

Since RD&D Programme 2010, SR-Site has been concluded for KBS-3V. Experience from this work indicates that earthquake-induced shear along fractures and chemical erosion are urgent issues to evaluate for a KBS-3H repository, since they could differ from the results for KBS-3V. SKB and Posiva therefore plan to evaluate the effects of chemical erosion and earthquake-induced shear for KBS-3H by means of scoping calculations early in the current project phase.

Provided that the aforementioned studies yield positive results, SKB and Posiva will continue working with the design premises for KBS-3H. They constitute the basis for the production line reports, which are important in the continued evaluation of long-term safety.

The evaluation will be carried out incrementally, starting with a system evaluation of the long-term safety of a deposition drift installed in accordance with the DAWE design. After this, the plan is to carry out a site-specific evaluation of the long-term safety of a KBS-3H repository in Forsmark. The evaluation will be based on the results of SR-Site, with supplementary analyses of the parts that are specific for KBS-3H. The final purpose is to be able to compare the safety of the two repository designs and, at a later stage, to be able to 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

SKB's research programme in natural science is being pursued with a view to the need to assess safety in connection with the final disposal of radioactive waste and spent nuclear fuel. This means that the research programme is designed to cover the needs of both the Spent Fuel Repository and SFR, and eventually SFL as well.

The aims of the research are to:

1. find solutions to identified problems that affect safety at the final repositories or that reduce uncertainty in the assessment of the repositories' long-term safety,
2. follow scientific progress, in part to identify the consequences of new findings for SKB's activities, in part to benefit from new knowledge discovered by other organizations,
3. maintain and develop the competence necessary to carry out assessment of long-term safety and inform regulatory authorities, reviewers and the public in a scientific manner.

The research covers both general issues that are shared by the different repository systems and issues related to a specific repository system. 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 in recent years mainly been conducted within the framework of the SR-Site safety assessment (SKB 2011e). The focus of this research is on processes that affect the engineered and natural barriers in the repository concept or affect the consequences of a possible release of radionuclides. Table 17-1 provides an overview of the processes and the scope of the efforts that are planned during the coming three-year period, see Section 17.2. An account of the detailed research programmes for the different disciplines is given in Chapters 18–27.

Research issues that are specifically related to the low- and intermediate-level waste that is disposed of in SFR are dealt with in Chapter 20, while research issues related to the waste in the future SFL are dealt with in Chapter 21. Issues that pertain to concrete barriers are common to SFR and SFL and are dealt with in Chapter 22. The processes that are being studied for the safety assessment for the extension of SFR are compiled in Table 17-2, see Section 17.2.

A summary description of the different research areas and the research initiatives during the coming RD&D period is provided here below. In addition, a summary is given of the connection between the research programme and the different projects being conducted at the Äspö Hard Rock Laboratory (HRL), along with an overview of the research activities at Nova FoU (Nova Research and Development) in Oskarshamn and within the framework of the pan-European "Implementing Geological Disposal Technology Platform".

17.1 Summary – research areas

The purpose of the final repositories is to protect man and the environment from ionizing radiation. The final repository for spent nuclear fuel contains the waste until its radioactivity has decayed to a harmless level, while SFR and SFL release radioactivity on a scale that does not harm man or the environment. The repositories' protective barriers interact to achieve the overall purpose. Conditions on the ground surface also influence what effect radionuclides would have on people and the environment if they reach the ground surface.

The current level of scientific understanding of issues relevant to long-term safety is quite advanced. This is the fruit of decades of research in the Swedish programme and other national programmes, as well as in joint international projects. Research on the evolution and post-closure safety of the Spent Fuel Repository has led to an understanding of such key processes as copper corrosion, canister shearing and other potential causes of canister failure, as well as of important processes related to radionuclide retardation. In SR-Site, SKB was able to show that it is possible to build a final repository for spent nuclear fuel that meets SSM's requirements on long-term safety.

Research on processes of importance to safety continues, even though the prevailing view is that existing knowledge is sufficient to verify long-term safety. The need for continued research applies particularly to factors which, according to SR-Site, contribute to risk and where the premises for the assessment can be improved by further research. The need for research is usually identified in the safety assessment work. It is therefore probable that new research needs will be identified in future safety assessments. Continued research on processes of importance for long-term safety is also necessary in order to keep the state-of-the-art up-to-date and for competence development.

As far as short-lived low- and intermediate-level waste is concerned, SKB is in the midst of an evaluation phase for SFR. Continued research for this waste will largely be guided by the results of SR-PSU. The work for the long-lived low- and intermediate-level waste is in the concept stage. Efforts that specifically concern the long-lived waste will be progressively concretized, even though quite a few processes will probably resemble those in SFR and can therefore be co-studied.

Much of the research within established geoscientific disciplines is being conducted as Ph. D. projects, mainly at KTH, Chalmers University of Technology, Stockholm University, Uppsala University, Umeå University and the Swedish University of Agricultural Sciences. Stockholm University and Chalmers University of Technology also have associate professors from SKB. This gives SKB an opportunity to keep up with the latest research, influence its direction and assure competence in important fields.

Safety assessment

The safety assessment is the instrument that is used by both SKB and SSM to determine whether a repository for radioactive waste and spent nuclear fuel complies with the regulatory requirements for such a facility. The assessment methodology has been developed over a long time, in parallel with the development of the KBS-3 system. In recent years, this development has largely been conducted and reported in conjunction with safety assessments, the most recent being SR-Site, the assessment underlying the application for a licence to build a final repository for spent nuclear fuel in Forsmark.

The development needs during the coming RD&D period pertain to supplementary development of methods for sensitivity analyses and radionuclide transport, as well as further development of quality assurance of SKB's safety assessments. In addition, preparedness exists to deal with any methodology-related viewpoints that may be expressed in conjunction with SSM's ongoing review of SR-Site.

Climate evolution

Both the spent nuclear fuel and the low- and intermediate-level waste will be disposed of deep down in the bedrock. The rock offers a stable environment that makes it possible to describe what will happen even over the very long term. At the same time, SKB must show that the repositories comply with the regulatory authorities' safety requirements even in the most extreme situations that can occur on the ground surface. In the case of the Spent Fuel Repository, SKB must be able to predict how the climate will affect safety during periods of up to 1 million years.

Climate research is an overarching theme of SKB's research program. It touches upon and affects several other research areas. Climate-related processes such as the growth of ice sheets and permafrost can affect sea levels, groundwater flows, stresses in the Earth's crust and, not least, living conditions at the ground surface.

Unlike in much other climate research, SKB's focus is not only on global warming or exactly when the next ice age will occur. Instead, it is of greater importance to identify and describe a number of possible future climate evolutions which together cover the uncertainty that exists surrounding the evolution of the climate in these very long time perspectives. These climate cases constitute examples of possible climate evolutions, as well as of extreme limiting cases for the analysis of different functions of the repository. Furthermore, the needs of the different repositories differ somewhat due to the fact that they are located at different depths, use barriers of different materials and have analysis periods of different lengths.

In the case of the Spent Fuel Repository, attention is focused on processes that could affect the buffer and the canister up to a million years in the future. Future efforts will mainly be aimed at obtaining more information on historic variations of the ice sheets. The purpose is to be able to set limits on how thick future ice sheets could become.

Other questions also have to do with what happens to the repository during an ice age and what the hydrogeological conditions look like beneath an ice sheet. To improve knowledge concerning these matters, SKB, together with the sister organizations in Finland and Canada, has pursued a major research project on Greenland called the Greenland Analogue Project. The project, which has been going on since 2009 and is now in its final phase, has yielded valuable knowledge of benefit to several of SKB's research areas, in particular hydrogeology and hydrogeochemistry. This information is also useful when SKB looks at the behaviour of future inland ice sheets above a final repository.

The expected climate evolution during the next ten thousand years is of greater importance for the assessment of the long-term safety of SFR. Here it is important to get an idea of, for example, when permafrost might form at the earliest and how deep it could reach during the assessment period.

Within the climate programme, SKB is also working with scenarios that describe future climates dominated by global warming in order to be able to assess what these climates would mean for the safety of the different repositories. The work includes aspects of both long- and short-term safety. The latter case includes gathering data on possible sea level increases as a basis for the geotechnical design and construction of the repository.

Short-lived low- and intermediate-level waste

The short-lived low- and intermediate-level waste is highly varied and complex in nature. The properties of the waste at closure comprise what is defined as the initial state and are dependent on the waste packages the waste is contained in, how the waste has been conditioned and possible grouting of the waste packages. The environment in the repository during the operating period will also affect the properties of the waste and thereby the initial state. In order to achieve the initial states of the waste defined in the safety assessments for SFR, the contents of the different waste types must conform to acceptance criteria and type descriptions.

When the repository is closed and becomes water-saturated, the waste will interact with the penetrating groundwater and various processes will be initiated. They will affect the chemical and physical properties of the waste and thereby the release of radionuclides. A report dealing with the processes in the waste in SFR has been produced within SR-PSU and serves as a basis for the assessment of long-term safety. Efforts have also been made to better estimate the total material and radionuclide inventory at closure of SFR, which now also includes, in addition to operational waste, waste from dismantling and demolition. Continued research on short-lived low- and intermediate-level waste will largely be guided by the results of the safety assessment being done in SR-PSU.

Long-lived low- and intermediate-level waste

Even though SFL is in the concept phase, a repository with a retarding safety function instead of a containing safety function is being planned. In this way, many of the components and materials used in SFR will probably be found in SFL as well. Furthermore, groundwater will seep in after closure and interact with the waste. As a result, the processes that are expected to occur in SFL will for the most part resemble those that occur in SFR, which in turn means that much of the research can be coordinated with equivalent programmes for the short-lived waste. Some differences between the repositories are, however, their depth and their material and radionuclide inventory. Specific research needs for long-lived low- and intermediate-level waste will be identified as the design of SFL and its inventory is concretized.

Concrete barriers

Concrete barriers exist in SFR and will probably also exist in SFL. The main function of the concrete barriers is to retard releases of radionuclides. This is achieved because the concrete acts as a sorption surface for the radionuclides and because the concrete structures reduce the flow of water through the rock vaults and thereby the waste. These functions are best achieved by a fracture-free concrete barrier Much of the research being done on concrete barriers is therefore aimed at developing recipes (mix designs) and methods for concrete casting and increasing our understanding of degradation of the concrete.

Fuel

As long as the copper canister containing the fuel in the Spent Fuel Repository is intact, no radioactive substances can escape from the repository. If a hole should occur in a canister, the properties of the fuel are crucial for determining if and when radionuclides can escape into the rock. One of the most important properties of the fuel in this context is that it generally dissolves very slowly in the water present in the repository environment. This retards any dispersion of radioactive substances. Fuel dissolution is therefore one of the crucial topics for SKB's continued fuel-related research. For example, research activities are planned to gather data on fuel dissolution under repository-like conditions and clarify the mechanisms of the different processes that contribute to fuel dissolution.

During their time in the reactor, some of the radionuclides have segregated to the surface of the fuel pellets. They are therefore more loosely bound to the fuel. In the event of a canister breach, these substances dissolve faster than the fuel matrix and determine the initial pulse of radionuclides that can be transported to the ground surface. There are indications that high-burnup fuel contains a larger fraction of quickly released radionuclides, which has occasioned further research into the properties of high-burnup fuels. Research is also being conducted to investigate new types of fuels, for example fuels to which chromium has been added to optimize the fuel's performance in the reactor. Investigations of these new fuel types are largely being undertaken in the form of joint international projects.

Canister

The copper canister is the isolating barrier in the KBS-3 repository. It is supposed to ensure that no radionuclides escape into the rock. In the reference scenario in SR-Site, the canister remains intact throughout the analysis period. There are two factors that could cause breaches in a canister: mechanical loads and copper corrosion. Central questions related to mechanical loads therefore have to do with how large isostatic loads the canister can tolerate and how large shear movements in the rock the canister can withstand. This is also described in Chapter 12, "Technology development, canister". The creep properties of copper are an important research area in that respect.

The biggest research initiatives are being done on copper corrosion to see which processes can affect the canister. There are, for example, phenomena whose underlying processes need to be better understood. Experiments are in particular being conducted on copper in oxygen-free water where hydrogen gas is generated in quantities much greater than predicted by thermodynamic data.

At the same time, efforts to reduce uncertainties surrounding processes that can affect the long-term durability of the canister are continuing. An example is work on sulphide corrosion, which in SR-Site was the corrosion process that made the greatest contribution to risk and is thereby also included as a topic of further research. Stress corrosion cracking is another complex process on which further research is needed.

Buffer and backfill

The canister in the Spent Fuel Repository is surrounded by a protective buffer of bentonite clay. The function of the clay is to limit the groundwater flow into the canister. If a canister is breached, the buffer will also retard the transport of radioactive particles out into the rock.

As long as the buffer is intact, there are no processes that could cause a breach in the canister. In conjunction with the safety assessment work, it has emerged that the buffer might under certain conditions dissolve and be carried out into fractures in the rock, a phenomenon known as buffer erosion. This could happen if large quantities of meltwater of low calcium content from an ice sheet penetrate down to the repository. The biggest research efforts therefore have to do with clarifying under what conditions the clay is stable, which is being done in an extensive experimental programme both in SKB's own projects and in large joint projects within the EU.

Efforts are also being made to develop a programme for clay characterization. Progress has been made here thanks to experiments with alternative buffer materials that are being conducted in the Äspö HRL. Progress has also been made in achieving a better understanding of the mechanisms behind changes in the clay, i.e. homogenization. New models have been developed to better describe this process in the clay, and development efforts in the area are continuing.

Bentonite clay will also be used to backfill the deposition tunnels in the Spent Fuel Repository. The research being conducted on the buffer also covers the needs that arise in connection with the development work for the backfill.

The properties of the clay that enable it to limit water flow and retard the transport of radionuclides are also utilized in the disposal of low- and intermediate-level waste. Research on buffer-related issues in the Spent Fuel Repository is largely applicable to conditions in SFR. Some efforts are also aimed at gaining a better understanding of how bentonite clay and concrete interact. This could be applicable to final disposal of the long-lived waste as well. The programme for this waste is still in the concept stage, which is why no concrete research issues have yet been formulated.

Geosphere

Final disposal of radioactive material will take place down in the bedrock, a stable environment where conditions change very little compared to what takes place on the ground surface. Based on the knowledge that the rock has been stable for millions of years, predictions can be made regarding conditions in the rock for a long time to come. Research on the geosphere embraces four disciplines: geology, hydrogeology, hydrogeochemistry and transport properties of the rock.

Future research in these areas will be aimed at broadening the knowledge base concerning rock conditions of great importance for the outcome of the safety assessments. SR-Site clearly shows which properties and processes are of the greatest importance for the Spent Fuel Repository. The aim is to obtain greater knowledge of how the properties vary in the rock volume and as a function of rock type. SKB must also improve the modelling tools used to describe and predict the processes that take place in the rock over a very long time.

Future repository construction will yield more data on the rock and its variations. This will enable the picture of the spatial variation of the rock properties to become more detailed in the future. In the meantime, investigation and sampling methods in the Äspö HRL are being developed, mainly within the detailed characterization project.

Research efforts in the discipline of geology will focus on gaining a better understanding of spalling caused by stresses in the rock and by high temperature, methodology for identification of large fractures, further studies of glacially induced faults, seismic measurements to support earthquake modelling and general increased knowledge regarding seismicity in Swedish bedrock. Discrete fracture network models, which serve as a basis for analysis of rock movements and solute transport in fractures, is a major field of research with a bearing on hydrogeology, geology and rock mechanics.

Data on the hydrogeology of surface systems is mainly being gathered in the investigations on Greenland. The goal is to be able to describe how the flow changes and varies during a period dominated by permafrost. The results are being used, for example, in the safety assessment for the extension of SFR. Hydrogeological research on deep groundwater systems involves integrating data and models with other disciplines (geochemistry, rock mechanics and transport) and maintaining and improving the codes that are used for flow and transport calculations. Flow conditions during a glaciation with extensive permafrost have consequences for the situation at repository depth. Investigation data from the project on Greenland are being used both to understand the flow processes and optimize the modelling tools.

Research in the field of geochemistry continues to focus on reactions between water and rock and effects of movements of the water in the fracture system. Efforts are being focused on models where geochemical conditions are integrated with hydrogeological conditions and the transport properties of the rock. Microbial processes are also an increasingly important topic. Examples are the importance of acetogens (acetate-generating microorganisms), the interaction between microbes and viruses in the rock, biofilms on fracture surfaces and microbial processes in the presence of hydrogen, methane and sulphide.

The programme examining how solutes are transported in groundwater includes studies of flow-related transport resistance, channelling (width and frequency of channels) and diffusion in stagnant zones. There are also plans to improve the K_d concept for element-specific distribution coefficients where the hydrogeochemical conditions in fractures are taken into account. The purpose is to be able to use the newfound knowledge in future safety assessments. International research is being pursued within the SKB Task Force for Groundwater Flow and Transport of Solutes. Modelling tasks are being carried out by several groups from different organizations using data from experiments and investigations at the Äspö HRL or other hard rock laboratories.

Surface ecosystems

Research and development in the area of surface ecosystems is mainly aimed at gaining a better understanding of the processes that influence transport and accumulation of radionuclides in the surface systems and at developing the methodology for calculating and assessing the radiological risk to human health and the environment. There will be a great emphasis on questions and uncertainties that have been identified in the work with the safety assessments for the Spent Fuel Repository and the extension of SFR, as well as in conjunction with the review of licence applications for the Spent Fuel Repository. Questions identified during preparations for the assessment of the future repository for long-lived radioactive waste, SFL, will also be investigated.

Studies conducted so far show that if radionuclides from a repository should get out into the rock and be transported up to the ground surface, they will eventually end up in low points in the landscape. If this occurs to terrestrial environments, the receiving area will in all probability be a wetland or agricultural land. An important area for future research efforts in the field of surface ecosystems is therefore processes that control transport and accumulation of radionuclides in wetlands and agricultural lands. It is also important to gain a better understanding of processes that control the transport of radionuclides from terrestrial areas to lakes and watercourses. Examples of processes where studies are planned are retention and biological uptake. In the programme for biogeochemistry in the surface system, collected data will be further processed to gain a better understanding of and improve the description of retention and biological uptake on different spatial scales.

The coastal area in Forsmark is changing over time as a function of climate variation and shoreline displacement, which has consequences for the possible exposure of man and the environment. In order to assess the consequences if radionuclides should escape from a repository, knowledge is needed of both landscape evolution and how processes change when the climate changes. Activities aimed at refining the description of the landscape and its evolution are therefore planned within the programme for surface ecosystems. In order to gain a better understanding of how processes will change when the climate in Forsmark gets colder, SKB will continue the current efforts to describe periglacial environments on Greenland.

When it comes to refinement of calculation methodology, efforts are planned to replace or supplement the use of uncertain concentration factors with mechanistic models of retention and biological uptake in order to reduce the uncertainties. SKB will also continue its development of the methodology for dose calculations based on newfound knowledge from the biosphere programme and from applications of the methodology to completed and ongoing safety assessments.

SKB is keeping track of international research by actively participating in a number of joint international fora concerned with radiological safety for man and the environment, for example IAEA programmes and EU projects, as well as associations such as BIOPROTA and the International Union of Radioecology (IUR).

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 is being pursued within the research areas fuel (Chapter 23), canister (Chapter 24), and buffer and backfill (Chapter 25). Research relevant for SFR and the SR-PSU safety assessment is being pursued within the areas short-lived low- and intermediate-level waste (Chapter 20) and concrete barriers (Chapter 22). Research on long-lived low- and intermediate-level waste (SFL) is described in Chapter 21.

Spent Fuel Repository

The SR-Site safety assessment (SKB 2011e) describes the long-term safety of the Spent Fuel Repository and bounds the importance of identified uncertainties. The research programme is continuing so that SKB can gain further knowledge and reduce remaining uncertainties. In this way, it should be possible to make more realistic assessments of the Nuclear Fuel Repository's safety margin in future safety assessments. The greatest uncertainties (knowledge gaps) concern corrosion of copper canister and the long-term function of the bentonite clay. Knowledge of the rock will increase as construction of the repository proceeds. This will take place via observations and measurements that provide a more detailed picture of the variation of different properties in the rock, such as permeability, stresses and groundwater composition.

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 research aimed at defining the source term for nuclide transport calculations. Some of these processes are bound to the initial state, for example type of fuel and burnup. Information on this can be found in Chapter 11 "Technology development, fuel handling" and Chapter 23 "Fuel".

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 loadings. Knowledge concerning the initial state of the canister is presented in Chapter 12 "Technology development, canister" and Chapter 24 "Canister".

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 the ones that occur in the buffer, and research on the backfill is therefore presented jointly with the buffer in Chapter 25 "Buffer and backfill".

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 an extension of the facility. The processes that occur in this type of waste are specific for the waste type, and the research programme for issues related to long-term safety 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 described in Chapter 22. 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 25.

SFL

The research and safety assessment work for SFL is in the process of being built up and will be further developed based on the concept study being done where different strategies for final disposal of long-lived waste are being evaluated. The study shows that SFL will be based on geological disposal, which is why much of the research related to the Spent Fuel Repository and SFR is also relevant to the research that will be needed for a future safety assessment for SFL. Research initiatives for long-lived low- and intermediate-level waste are also presented in Chapter 21.

Processes

Research initiatives for the different processes dealt with in the assessment of long-term safety are graded on a three-point scale as major, moderate or minor. These processes may be re-examined in the light of new results with regard to both their significance (definition) and the research effort required in the future (scope). The contents of each RD&D programme and the viewpoints received from the Swedish Radiation Safety Authority in particular influence the definition of the processes.

The grading of the processes is shown in Table 17-1 compared with the grading in RD&D programmes 2010 and 2007 (i.e. the time period during which the period SR-Site and SR-Can safety assessment projects have taken place). As evident from the figure, the assessed scope of the initiatives needed for most of the processes has not changed. Among those that have changed, some have been clarified, while others have been added.

Table 17-1. Processes in fuel, canister, buffer and backfill, as well as geosphere. The colour coding in the table shows which research initiatives are deemed necessary during the coming three-year period and what this assessment looked like in RD&D Programme 2007 and 2010.

Type of process	Fuel	Canister	Buffer and backfill	Geosphere
	2007 2010 2013	2007 2010 2013	2007 2010 2013	2007 2010 2013
Radiation-related processes	Radioactive decay 23.2.2			
		Radiation attenuation/heat generation 24.2.1	Radiation attenuation/heat generation 25.5.2	
	Induced fission – criticality 23.2.3			
Thermal processes	Heat transport 23.2.1	Heat transport 24.2.1	Heat transport 25.5.3	Heat transport 26.3
			Freezing 25.5.4	
Hydraulic processes	Water and gas transport in canister cavities, boiling/condensation 23.2.1		Water transport under unsaturated conditions 25.5.5	Groundwater flow 26.4
			Water transport under saturated conditions 25.5.6	Gas flow/dissolution 26.5
			Gas transport/dissolution 25.5.7	
			Piping/erosion 25.5.8	
Mechanical processes	Thermal expansion/cladding failure 23.2.1	Deformation of insert 24.2.2	Swelling 25.5.9	Movements in intact rock 26.6
		Deformation of copper canister under external pressure 24.2.3		Reactivation – movement along existing fractures 26.8
		Deformation from internal corrosion products 24.2.5		Fracturing 26.9
				Time-dependent deformations 26.10
		Thermal expansion 24.2.4		Thermal movement 26.7
		Radiation effects 24.2.6		

Major initiatives | Moderate initiatives | Minor initiatives/monitoring during coming three-year period | The process is not described in this RD&D programme

Type of process	Fuel	Canister	Buffer and backfill	Geosphere
	2007 2010 2013	2007 2010 2013	2007 2010 2013	2007 2010 2013
Chemical processes	Advection and diffusion 23.2.1	Corrosion of insert 24.2.1	Advection 25.5.10	Advection/mixing – hydrogeochemistry 26.11
	Residual gas radiolysis/oxygen formation 23.2.1	Galvanic corrosion 24.2.1	Diffusion 25.5.11	Diffusion – groundwater chemistry 26.13
	Water radiolysis 23.2.4	Stress corrosion cracking of insert 24.2.7	Osmosis 25.5.12	Reactions with the rock – hydrogeochemistry 26.15
	Metal corrosion 23.2.5		Ion exchange/sorption 25.5.13	
	Fuel dissolution 23.2.6	Corrosion of copper canister 24.2.8	Montmorillonite transformation 25.5.14	Microbial processes 26.17
	Dissolution gap inventory 23.2.1	Stress corrosion cracking of copper canister 24.2.9	Iron/bentonite 25.5.15	Decomposition of inorganic engineering material 26.18
			Copper/bentonite 25.5.16	
	Speciation of radionuclides, colloid formation 23.2.7	Earth currents – stray current corrosion 24.2.10	Dissolution/precipitation of impurities 25.5.17	Colloid formation – colloids in groundwater 26.19
	Helium production 23.2.8	Precipitation of salt on canister surface 24.2.11	Cementation 25.5.18	Gas formation/dissolution 26.21
	Transformation of fuel matrix 23.2.9		Colloid release/erosion 25.5.19	Methane ice formation 26.22
			Radiation-induced montmorillonite transformation 25.5.20	Salt exclusion 26.23
			Radiolysis of pore water 25.5.1	
			Microbial processes 25.5.21	
Integration/ modelling				DFN 26.24.1
			THM evolution, unsaturated conditions 25.5.5	THM evolution 26.24.2
				Hydrogeochemical evolution 26.4.3
Radionuclide transport			Advection 25.5.10	Advection/mixing 26.11
			Diffusion 25.5.11	Diffusion 26.14
			Ion exchange/sorption 25.5.13	Sorption 26.16
				Colloid transport 26.20
			Radionuclide transport in the buffer 25.5.22	Radionuclide transport in the geosphere 26.24.4

Major initiatives

Moderate initiatives

Minor initiatives/monitoring during coming three-year period

The process is not described in this RD&D programme

Table 17-2. Processes in short-lived low- and intermediate-level waste and barriers for this (for initiatives within the geosphere see Table 17-1).

Type of activity	Waste	Concrete barriers
Initial state	Geometry	Geometry
	Radiation intensity	Temperature
	Temperature	Hydrovariables
	Hydrovariables	Mechanical stresses
	Mechanical stresses	Material composition
	Total radionuclide inventory	Water composition
	Material composition	Gas variables
	Water composition	
	Gas variables	
Radiation-related processes	Radioactive decay 20.2.2	
	Radiation attenuation and heat generation, 20.2.3	
	Radiation-induced degradation of organic matter, 20.2.4	
	Water radiolysis, 20.2.5	
Thermal processes	Heat transport, 20.2.6	Heat transport, 22.2.2
	Phase change/freezing, 20.2.7	Phase change/freezing, 22.2.23
Hydraulic processes	Water uptake and transport under unsaturated conditions, 20.2.8	Water uptake and transport under unsaturated conditions, 22.2.4
	Water transport under saturated conditions, 20.2.9	Water transport under saturated conditions, 22.2.5
		Gas transport and water solubility, 22.2.6
Mechanical processes		Pressure from swelling waste, 22.2.7
		Pressure from bentonite, 22.2.8
	Fracturing, 20.2.10	Fracturing, 22.2.9
		Fall of ground, 22.2.10
Chemical processes	Advective transport of solutes, 20.2.11	Advection and mixing, 22.2.11
	Diffusive transport of solutes, 20.2.12	Diffusion, 22.2.12
	Sorption, 20.2.13	
	Colloid formation and colloid transport, 20.2.14	Colloid transport and filtration, 22.2.14
	Dissolution, precipitation and recrystallization, 20.2.15	Dissolution, precipitation and recrystallization, 22.2.15
	Chemical degradation of organic compounds, 20.2.16	Pore water speciation and concrete interactions, 22.2.16
	Water uptake/swelling, 20.2.17	
	Microbial processes, 20.2.18	Microbial processes, 22.2.17
	Metal corrosion, 20.2.19	Metal corrosion, 22.2.18
	Gas formation and gas transport, 20.2.20	Gas production, 22.2.19
Radionuclide transport	Speciation of radionuclides, 20.2.21	Speciation of radionuclides, Section 22.2.20
	Radionuclide transport in the aqueous phase, 20.2.22	Radionuclide transport in the aqueous phase, Section 22.2.21
	Radionuclide transport in the gas phase, 20.2.23	Radionuclide transport in the gas phase, Section 22.2.22

The scope of the initiatives is identified within SR-PSU (spring 2014) Minor initiatives/monitoring during coming three-year period

17.3 Other methods

SKB is following the development of other methods besides the KBS-3 method for management and disposal of spent nuclear fuel. These other methods are Partitioning and Transmutation (P&T, Section 28.1) and disposal in deep boreholes (Section 28.2).

17.4 Research in the Äspö HRL, Nova FoU and European cooperation

17.4.1 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. The new projects being planned in the Äspö HRL are mainly focused on technology development and research occasioned thereby. Only a few pure basic research projects are planned.

Table 17-3. Overview of projects at the Äspö HRL with a link to research on long-term safety. For the location of the projects in the Äspö HRL, see Figure 1-12.

Project	Process	Section
MiniCan	Deformation due to internal corrosion products	24.2.5
	Corrosion of copper canister	24.2.8
LOT	Montmorillonite transformation	25.5.14
	Copper-bentonite interaction	25.5.16
	Corrosion of copper canister	24.2.8
Alternative buffer materials	Diffusion	25.5.11
	Ion exchange/sorption	25.5.13
	Montmorillonite transformation	25.5.14
	Iron-bentonite interaction	25.5.15
Lasgit	Gas transport/dissolution	25.5.7
TBT	Water transport under unsaturated conditions	25.5.5
	Iron/bentonite	25.5.15
Canister Retrieval Test (CRT)	Water transport under unsaturated conditions	25.5.5
	Water transport under saturated conditions	25.5.6
	Swelling	25.5.9
Apse (Pillar Stability Experiment)	Thermal movement	26.7
Prototype Repository	Microbial processes	25.5.21
	Water transport under unsaturated conditions	25.5.5
	Thermal movement	26.7
	Heat transport (geosphere)	26.3
	Fracturing	26.9
TRUE	Integrated modelling – radionuclide transport	26.24.4
LTDE-SD	Diffusion – radionuclide transport	26.14
Microbial Projects	Microbial processes	26.17
SWIW tests	Reactions with the rock – sorption of radionuclides	26.16
	Integrated modelling – radionuclide transport	26.24.4

Interest in using 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 pursued in the Äspö HRL. The foreign organizations are participating both in the experimental work and in the modelling work within the task forces. There are task forces for modelling of transport of solutes (Äspö Modelling Task Force on Groundwater Flow and Transport of Solutes) and for modelling of processes in the clay (Äspö Modelling Task Force on Engineered Barrier Systems). The work in each task force is carried out in the form of projects, usually linked to experiments in the Äspö HRL. In this way, SKB gets access to results from several different modelling task forces, which in turn get access to relevant data. The task forces gather about twice a year to report results and discuss matters of common interest.

The following organizations are participating in the Äspö Modelling Task Force on Groundwater Flow and Transport of solutes:

Posiva Oy, Finland

BMWi, Bundesministerium für Wirtschaft und Technologie, Germany

CRIEPI, Central Research Institute of (the) Electric Power Industry, Japan

KAERI, Korea Atomic Energy Research Institute, South Korea

JAEA, Japan Atomic Energy Agency, Japan.

The following organizations are participating in the Äspö Modelling Task Force on Engineered Barrier Systems:

Posiva Oy, Finland

BMWi, Bundesministerium für Wirtschaft und Technologie, Germany

CRIEPI, Central Research Institute of (the) Electric Power Industry, Japan

NWMO, Nuclear Waste Management Organization, Canada

RAWRA, Radioactive Waste Repository Authority, Czech Republic

NAGRA, Nationale Genossenschaft für die Lagerung radioaktiver Abfälle, Switzerland

NDA, Nuclear Decommissioning Authority, UK.

SKB International handles the contracts with the foreign participants and offers different degrees of involvement in the specific wishes of the organizations. Participation in the work of the task forces will from now on be administered by SKB, and joint costs will be shared among the participants.

17.4.2 Broadening of research at Nova R&D

Nova FoU (Nova R&D) is a research and development platform under the umbrella of Nova Utbildning, FoU och Affärsutveckling (Nova Training, R&D and Business Development). The platform, which is a collaboration between SKB and Oskarshamn Municipality, is a centre of excellence where several universities, including Linnaeus University, and other educational institutions and companies, from both 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. SKB and Oskarshamn Municipality run Nova FoU together via a steering committee with two representatives each from SKB and the municipality and one representative each from KTH and Linnaeus University. Research projects that are accepted at the platform are externally funded without any link to SKB's activities. Results can nevertheless be of importance to SKB. SKB and Oskarshamn Municipality jointly fund the management of Nova FoU.

By agreement, Nova FoU has access to SKB's laboratories (the Äspö HRL, the Bentonite Laboratory and the Canister Laboratory), as well as to SKB's data and expertise. The research and development projects are conducted in these facilities by different universities and companies. 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 many 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. A complete list of the projects being pursued within Nova FoU can be found at www.novafou.se. Examples of projects being pursued by different universities and educational institutions are:

Linnaeus University: Groundwater composition, transport of hazardous substances, climate change, management of contaminated sediments (Ph. D. and post-doctoral studies).

Royal Institute of Technology: Water management, interaction between land and water (Ph. D. studies).

Chalmers University of Technology: Groundwater changes during tunnel construction (Ph. D. studies).

Swedish University of Agricultural Sciences: Residual heat from industry for greenhouse cultivation.

Swedish Museum of Natural History in Stockholm: Microbes as fossils.

University of Göttingen: Effects of microbes in groundwater (Ph. D. and post-doctoral studies).

University of Queensland: Interpretation of measurement data in boreholes (Ph. D. studies).

Companies: Technology development for monitoring people and machines in a tunnel environment, corrosion of rock bolts (business development).

Master's programme in nuclear engineering and final disposal: Conducted in cooperation between KTH, Linnaeus University, the EU and Nova Utbildning, FoU och Affärsutveckling. The Äspö HRL is used as a classroom for studies in geological disposal technology.

National geosphere laboratory

The Swedish Research Council has granted Stockholm University, as the main applicant, a planning grant for studying and planning how the Äspö HRL and SKB's databases can be utilized as a research infrastructure available for research outside of SKB's areas of interest. The project, which is supported by a large number of universities and educational institutions in Sweden, is called NGL, National Geosphere Laboratory. It can be seen as a broadening of the activities in Nova FoU. The aim of current planning is to apply for funds to build up and run a National Geosphere Laboratory for a period of 5–15 years.

17.4.3 Platform for European cooperation

Eleven organizations from eleven European countries have created a platform for closer European cooperation on nuclear waste disposal. The purpose of the platform, which is called the Implementing Geological Disposal Technology Platform (IGD-TP) and is supported by the European Commission, is to give support to those waste programmes that are closest to a realization. This will later benefit all the other programmes (IGD-TP 2011, 2012). The IGD-TP identifies and prioritizes research and technology development initiatives that are necessary and time-critical for ensuring that the first geological repositories in Europe will be in operation by 2025. The IGD-TP has no financial resources at its disposal, but has an indirect influence on how the EU's research funding is allocated within the area.

The IGD-TP supports a number of prioritized areas, ranging from large joint EU-funded technology development projects, such as DOPAS (plugs in repository tunnels), and research projects, such BELBAR (bentonite stability and buffer erosion), to joint task forces and expert networks. SKB's reason for being active on the platform is that it provides a common platform for scientific cooperation and communication throughout Europe.

The following organizations participate in IGD-TP:

ONDRAF, Belgian Agency for Radioactive Waste and Enriched Fissile Materials, Belgium

RAWRA, Radioactive Waste Repository Authority, Czech Republic

Andra, Agence Nationale pour la gestion des déchets radioactifs, France

BMWi, Bundesministerium für Wirtschaft und Technologie, Germany

PURAM, Public Agency for Radioactive Waste Management, Hungary

Posiva Oy, Finland

NAGRA, Nationale Genossenschaft für die Lagerung radioaktiver Abfälle, Switzerland

ENRESA, Empresa Nacional de Residuos Radiactivos S. A., Spain

SKB, Svensk Kärnbränslehantering AB, Sweden

COVRA, Centrale Organisatie Voor Radioactief Afval, Netherlands

NDA, Nuclear Decommissioning Authority, UK.

18 Safety assessment

18.1 Introduction

The safety assessment is the instrument that is used by both SKB and SSM to determine whether a repository for radioactive waste complies with the regulatory requirements on long-term safety for 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 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. Safety assessments and recurrent safety analysis reports will need to be done up until the closure of the three repositories.

In 2011, SKB finished the work with the safety assessment for the applications for licences to build a final repository for spent nuclear fuel in Forsmark. At present, a safety assessment is being carried out in support of the applications for an extension of the existing repository for short-lived radioactive waste in Forsmark (SFR).

All essential development of methodology for safety assessments has taken and is taking place within the framework of the two projects where the aforementioned assessments are or were being conducted. Complete accounts of methods and applications will be given in the reports from the projects. This RD&D report gives an account of ongoing methodology development for the Spent Fuel Repository in Section 18.2 and of methodology-related issues for the safety assessment for SFR in Section 18.3.

The term "safety case" is normally used internationally for the whole body of evidence and arguments concerning long-term safety included in e.g. an application for a licence to construct a repository, while the term "safety assessment" often refers to the specific analyses and calculations used to investigate compliance with relevant safety-related limit values, see for example NEA (2004). The content of SKB's safety assessments is determined by, among other things, the requirements on content in SSM's regulations. As a result, the implications of the term "safety assessment" in SKB's use are similar to the implications of the term "safety case".

18.2 Methodology for assessment of the long-term safety of the Spent Fuel Repository

A ten-step methodology was used in the SR-Can safety assessment (SKB 2006b). The methodology was reviewed by SKI and SSI within the framework of the consultations with SKB during the site investigation phase. The authorities found at that time that the methodology agreed for the most part with applicable regulatory requirements, but also indicated a number of points where it needed to be improved. These viewpoints were heeded in the SR-Site safety assessment (SKB 2011e), which serves as a basis for SKB's application for a licence to build the Spent Fuel Repository. SR-Site is currently (autumn 2013) being reviewed by SSM, and the viewpoints on the safety assessment methodology that emerge from the review will be taken into account in the assessment of post-closure safety for the preliminary safety analysis report (PSAR) that is required for an application to start construction of the Spent Fuel Repository. Alternatively, SSM may specify supplementary requirements on the methodology already within the ongoing licensing process, and in that case SKB will address the requirements before the work with the PSAR is begun.

Conclusions in RD&D 2010 and its review

SSM also offered viewpoints on SKB's safety assessment methodology in its review of RD&D programme 2010. These viewpoints partially overlapped those in the review of SR-Can and have then been dealt with within SR-Site. In the review, SSM expressed the following viewpoints regarding safety assessment of the Spent Fuel Repository (*Translation from original Swedish*):

- In SSM's judgement, in the light of the experience from SR-Site, SKB should be at the forefront of safety assessment and should then be in a good position to contribute to international development within the field.
- However, in SSM's judgement, SKB needs to modernize certain central areas in the safety assessment, for example methods for uncertainty and sensitivity analysis and tools for calculations linked to radionuclide transport.
- Maintenance of a long-term capacity in a specialized and competence-demanding area such as safety assessment is associated with a number of difficulties. One example is dependence on a few individuals who possess knowledge and experience that are difficult to replace. SKB may therefore need to devote extra attention in the future to educational activities and knowledge management in the area of long-term safety.
- There may still be a need to develop the methodology for quality assurance of safety assessment with development of control documents and instructions, quality audits, experience feedback, etc. It is a great challenge to get the large number of individuals who are needed to meet competence needs within many different disciplines to work in a consistent and structured manner. SSM intends to return to this question in connection with the review of SR-Site.

The development initiatives surrounding safety assessment methodology that are being pursued or are planned in these areas are described in the following section.

Newfound knowledge since RD&D 2010

As far as methods for uncertainty and sensitivity analysis are concerned, development of tools for variance-based sensitivity analysis is under way. This powerful method for ascertaining which uncertain input data parameters have the greatest impact on uncertainties in a probabilistic calculation result is suitable for problems that can be handled with a quick calculation model, since the method requires a large number of model realizations to be carried out. Since SKB has previously developed fast analytical calculation models for radionuclide transport (Hedin 2002), the necessary conditions exist for using variance-based sensitivity analysis on the results of these models. After further simplifications, the calculation model can be made so fast that the variance-based sensitivity analysis can be carried further than is normal within other areas. Methodology for this has been developed and described in general terms (Hedin 2013a), and an initial application has also be reported (Hedin 2013b). Development is continuing, and it should be possible to apply the method if needed to calculation results from both numerical and analytical models for radionuclide transport in the PSAR.

A simple method for graphic presentation of sensitivities for uncertain input data has been applied to probabilistic calculation results taken from SR-Site (Hedin 2013b). The method is called a cobweb plot and can most easily be explained using an example. Figure 18-1 shows a cobweb plot for a calculation of the summed radionuclide release from the far-field converted to total dose. On the left-hand vertical axis, the dose result in each realization is represented by a point that indicates the percentile to which the result corresponds. Only the realizations that yield the 10 percent highest (red points) and the 10 percent lowest (black points) dose results are shown in the figure. The next vertical axis shows the time for canister failure in each realization, also as a percentile. For each realization, the point on the dose axis is connected to the point on the axis for canister failure time by a line. The pattern in the figure then clearly shows that high doses are preferably associated with early canister failures and vice versa. The next axis shows in a similar manner fuel dissolution rate, and it can be seen here that high doses are, as expected, associated with high fuel dissolution rates and vice versa. The method is simple to both use and understand and requires no method or code development, since the program that has been used is freely available. To the extent that new radionuclide transport calculations are required for the PSAR for the Spent Fuel Repository, SKB intends to use this method to visualize the sensitivity of the probabilistic calculation results to uncertainties in input data.

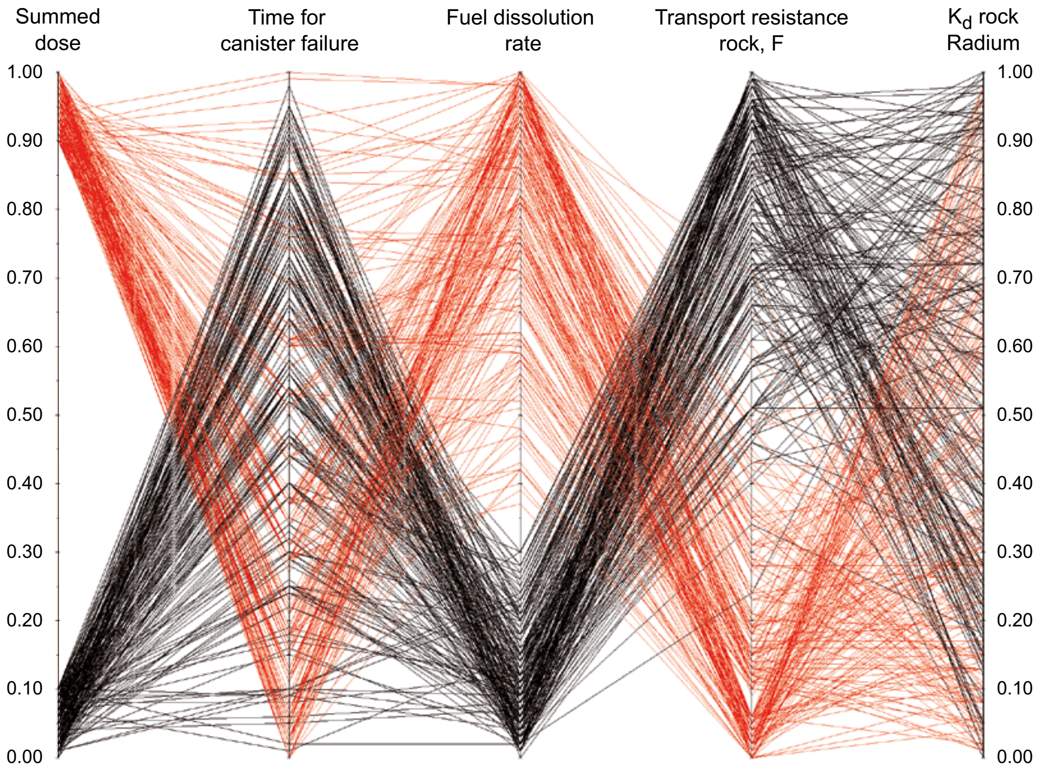


Figure 18-1. *Cobweb plot to visualize sensitivities of probabilistically calculated releases from the far-field (converted to total doses) to uncertainties in input data. See the text for further explanation.*

SKB judges today that the tools for calculations linked to radionuclide transport are appropriate for the Spent Fuel Repository. Since the review of RD&D Programme 2010, the radionuclide transport model MARFA has been further developed (see Section 26.12 "Advection/dispersion – radionuclide transport") and is deemed suitable today to serve as a quality-assured tool for safety assessment of the Spent Fuel Repository.

Programme

Within the safety assessment for the extension of SFR, new methods are being tried to link radionuclide transport in the geosphere and biosphere and dose calculations in the biosphere, see Section 18.3. The outcome of these efforts will be taken into account when determining how to handle any updates of the calculations in SR-Site that may be needed for the PSAR for the Spent Fuel Repository.

SKB shares SSM's view that training and knowledge management in the area of long-term safety are of urgent importance. Since SR-Site, SKB has strengthened its capacity for assessment of long-term safety and will, as before, train new employees primarily by having them participate in safety assessment projects.

SKB considers that there is still a need to develop the methodology for quality assurance of the safety assessment. SKB deems that procedures for e.g. project control and report handling (particularly issue-specific and quality review) work well. The company-wide procedures SKB has devised and is constantly developing are used in these cases. As far as procedures for quality assurance of data are concerned, there is a need for improvement, since the procedures that were used in SR-Site were perceived to be complicated and unnecessarily laborious. New procedures are being tried in the safety assessment for the extension of SFR, see Section 18.3. They will be evaluated and used to develop improved procedures for the PSAR for the Spent Fuel Repository. There is also a need for more developed and standardized procedures for quality assurance of calculation models and calculations. SKB intends in this case to benefit from experience gained in branches of industry where many demanding calculations are performed. In its review of RD&D Programme 2010, SSM has also given notice that they intend to bring up the matter of quality assurance when the Authority's review of SR-Site is conducted, and SKB is therefore prepared to deal with any viewpoints that may be offered.

18.3 Assessment of long-term safety of SFR

Assessment of the long-term safety of SFR and the planned extension is being pursued within the SR-PSU project. Methodology and reports produced within the project correspond for the most part to those produced in the most recent safety assessment for the Spent Fuel Repository, SR-Site.

Conclusions in RD&D 2010 and its review

RD&D Programme 2010 presented the plans for SR-PSU, which were based on experience and review comments from the previous safety assessments, SR-Can (SKB 2006b) and SAR-08 (SKB 2008a). SSM had no objections to the fundamental elements of SKB's safety assessment methodology. At the same time, they noted that the account in RD&D Programme 2010 was cursory and that no new further development or significant insights had been included. One explanation for this was that lessons learned from the work with SR-Site had not yet been formulated. SSM particularly mentioned the need for further work in the safety assessment for low- and intermediate-level waste. As SKB pointed out themselves, considerable work remained to bring the account to the level prescribed by SKB's own methodology. In SSM's judgement, in the light of the experience from SR-Site, SKB should be at the forefront in the field of safety assessment and should then be in a good position to contribute to international development within the field.

Programme

A number of reports will be produced in the safety assessment work for SFR and its extension. They will be included in the application for extension of SFR and are being prepared in accordance with the project's timetable. Additional research and development needs may be identified during the work. They will be presented in the project's main report. Besides producing this main report, the work with the safety assessment includes a number of different activities which are primarily presented in report form. All features, events and processes process (FEPs) of importance for the repository system are summarized in an FEP report. Since large parts of the repository system and its surroundings are the same as for the Spent Fuel Repository, the FEP report will, as far as possible, be based on work that has already been done within SR-Can and SR-Site.

A necessary part of a safety assessment is presenting the current understanding of the processes that can take place in the repository system and its surroundings. This process understanding embraces the identified features, events and processes and is based on the research that is being conducted inside and outside SKB. In the most recent safety assessments for the Spent Fuel Repository, SR-Can and SR-Site, the process reports were based on a predefined template. A similar template is being used for the safety assessment for an extended SFR, and the process reports will have a similar structure. As a part of the safety assessment, process reports will be prepared for the following topics:

- Geosphere.
- Waste.
- Barrier system.
- Climate.
- Biosphere and surface systems.

Data used in SR-PSU will be presented and qualified in a data report. It will follow a similar structure as the data report for SR-Site. Input data included in the assessment will be summarized in an input data report. These reports are a part of the project's QA documentation.

All models for calculation of radionuclide transport used in SAR-08 have been implemented in a new code. Further, a more detailed sensitivity analysis than the one performed within SAR-08 has been carried out. Extensive development in the area of biosphere modelling was pursued in SR-Site and has now continued in SR-PSU. This includes collection of site-specific data and development of models and modelling tools. The models used in SR-PSU will, where applicable, be based on the results obtained within SR-Site. In a similar manner, the results of SR-PSU will, where possible, be used for future safety assessments of the Spent Fuel Repository.

Within the extension project for SFR, work is being pursued on repository design in cooperation between design, safety assessment and technology development. The long-term properties of the repository components are taken into account in choice of technology and geotechnical design. This includes development of the engineered barriers such as bentonite and concrete structures, as well as development of tunnel backfilling and plugging.

The work of preparing descriptions of the initial state is also being pursued in cooperation with other parts of the project organization. Instead of the production reports that comprised a part of the application for the Spent Fuel Repository, a report describing the initial state will be prepared for SR-PSU.

SR-PSU includes preparation of descriptions of the repository's reference evolution. This work is based to a great extent on similar work in SR-Site, but certain repository-specific modifications are necessary. For example, the description of the future climate evolution will be handled differently. This is due in part to the fact that the time period analyzed differs between a repository for short-lived waste and a repository for spent nuclear fuel or long-lived waste. The analysis period for SR-PSU is at most 100,000 years, while the corresponding period for long-lived waste and spent nuclear fuel is 1,000,000 years. The relative importance of different issues is also different, for example the time when a future glaciation/permafrost occurs is of crucial importance for a repository for short-lived waste, while in the case of the Spent Fuel Repository it is more important to analyze how thick a future ice sheet may get, or how deep future permafrost may reach. The future evolution of surface systems/biosphere is also based on the models used in SR-Site. However, the evolution of the barrier systems will differ in safety assessment for an extended SFR compared with the Spent Fuel Repository, partly because the waste and the barrier system are different.

The safety functions used in SR-PSU are based on those used in SAR-08. The safety functions will be used to construct scenarios in the same way as in SAR-08, but also to express requirements and preferences regarding the long-term function of the repository.

19 Climate evolution

In the time perspective in which safety is being studied for SKB's different final repositories (Spent Fuel Repository, SFR and SFL), i.e. hundreds of thousands of years, the climate in Scandinavia has varied very much in the past. It has alternated between warm interglacial conditions similar to those 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 may have a great impact. An example is the growth of ice sheets and permafrost and changes in sea level. These processes in turn affect e.g. groundwater flow, groundwater chemistry and stresses in the Earth's crust, which are of importance for repository performance. The climate at the ground surface also affects the evolution of the biosphere and the landscape. The biosphere in turn has a great influence on human activities, which must be able to proceed in the landscape without man being affected by nearness to the repositories.

The overall purpose of the climate research programme at SKB is to furnish the safety assessments with scientifically substantiated scenarios for the future evolution of the climate, which in turn serve as a basis for the assessments of long-term safety. Important aspects of this work are to i) improve process understanding where important knowledge gaps exist, ii) validate models used in the climate research programme (permafrost, ice sheet and isostatic models) and iii) collect and validate input data to these model simulations. Another important purpose of the climate research programme is to furnish other disciplines in SKB's safety assessments with climatological input data and boundary conditions for different types of calculations.

19.1 Climate scenarios in SKB's safety assessments

As has been noted in previous RD&D programmes, it is not possible to predict future climate in the long time perspective (100,000 years and more) that are covered by SKB's safety assessments. However, it is possible to construct scientifically substantiated scenarios for various conceivable future climate developments, based 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 climate in Scandinavia may vary, even in very long time perspectives. Within these bounds, certain characteristic climate domains (temperate, periglacial and glacial climate domains) can be identified that are of importance for repository performance. As before, SKB is therefore focusing its research efforts in the climate field on identifying and understanding conditions and processes within these climate domains. If the safety requirements are fulfilled, given the different conceivable climate domains and during transitions between them, it is less important in the assessment of long-term safety to take into account the actual future evolution of the climate in time and space to as great a degree. However, the expected climate evolution during the next ten thousand years is of greater importance for the assessment of long-term safety for SFR, see below.

SKB's approach in handling the great uncertainty inherent in present-day knowledge concerning future climate evolution involves identifying, describing and analyzing the effects of a broad spectrum of conceivable future climate evolutions. They range from warm climates dominated by pronounced global warming to climates dominated by glacial ice age conditions. One of the climate cases therefore comprises a future evolution based on a repetition of conditions reconstructed from the most recent glacial cycle, including the Weichselian glaciation. Since this is the glacial cycle we know most about, this climate case provides important process knowledge concerning how the different climate-related processes (ice sheet, permafrost, shoreline displacement) work and interact.

Besides being used in the assessment of long-term safety, this climate evolution also serves as a suitable scientific starting point for an extended analysis of the impact of the climate on the different repositories. Based on this reference evolution, as well as on SKB's knowledge regarding possible climate variations and the safety and performance of the repository, a number of other different climate evolutions are identified in a structured manner. These cases cover all conceivable situations where climate-related processes could potentially have a greater impact on repository performance than in the case based on a repetition of last glacial cycle conditions. Also for these other climate cases, 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, shoreline displacement and climate vary. In describing possible future climate variations, it is important to also consider cases with anthropogenic influence on the climate.

Näslund et al. (2013) explain how climate and climate-related processes are handled in safety assessments of SKB's different types of repositories. One of the main conclusions is that the climate scenarios in a given safety assessment need to be adapted to the specific questions that arise from the repository concept in question. This means that the set of conceivable and relevant climate scenarios may vary, depending on what repository concept and waste type is being analyzed. An example of this is the SR-Site safety assessment for the Spent Fuel Repository and the SR-PSU safety assessment for the extension of SFR. Due to different repository depths, different lives of the radionuclide inventory and other factors, the set of future climate scenarios need to differ to some extent between the two safety assessments (Näslund et al. 2013).

The climate evolutions that are constructed for SKB's safety assessments should not be regarded as predictions of future climates. Instead, they offer relevant examples and bounding cases for the future climate evolution, where climate-related processes are described in a realistic and integrated way in a 100,000-year or longer time perspective.

The climate evolutions that describe cases with human impact on climate, through an increased greenhouse effect, include a global rise in sea levels caused by increased melting of ice sheets and glaciers 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 ice sheet during the Weichselian glaciation, whose results were described in RD&D Programme 2007, is used in turn to study shoreline displacement (see Section 19.4), permafrost (see Section 19.5) and the occurrence of glacially induced earthquakes (see Section 26.8).

In order to gain a better understanding of future climate evolution and variability, SKB has also conducted studies of climate variations, as such, during the past six years. They have included studies of conceivable future climates from climate modelling, as well as studies of historical climate variations from geological climate archives and compilations of climatological information for selected periods of the Weichselian glaciation, see Section 19.6.

With the current approach and safety assessment methodology, the identified climate domains – and their occurrence in time and space in the selected climate cases – serve as basis for SKB's safety assessments. The climate programme is being carried out in close cooperation with the research programmes for hydrogeology, geochemistry, surface ecosystems, thermo-hydro-mechanical processes, buffer and canister, see Figure 19-1.

In its climate programme, SKB describes firstly a reference evolution that includes a repetition of conditions reconstructed for the last glacial cycle, and secondly alternative future climate evolutions based on specific issues of interest for particular safety assessments and repository concepts. The following alternative evolutions are studied:

- A colder and drier climate than in the reference evolution, which results in deeper permafrost and longer periods with the periglacial climate domain, see Section 19.5. Several alternative evolutions are treated. One of these includes the earliest expected occurrence of permafrost, of particular relevance in the assessment of long-term safety for the SFR repository concept.
- A colder climate with higher precipitation leading to a thicker ice sheet, or the existence of an ice sheet for longer periods of time than in the reconstruction of the last glacial cycle, see below in this section. Geological 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 19.4 and 19.6. Several alternative evolutions are treated.

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.

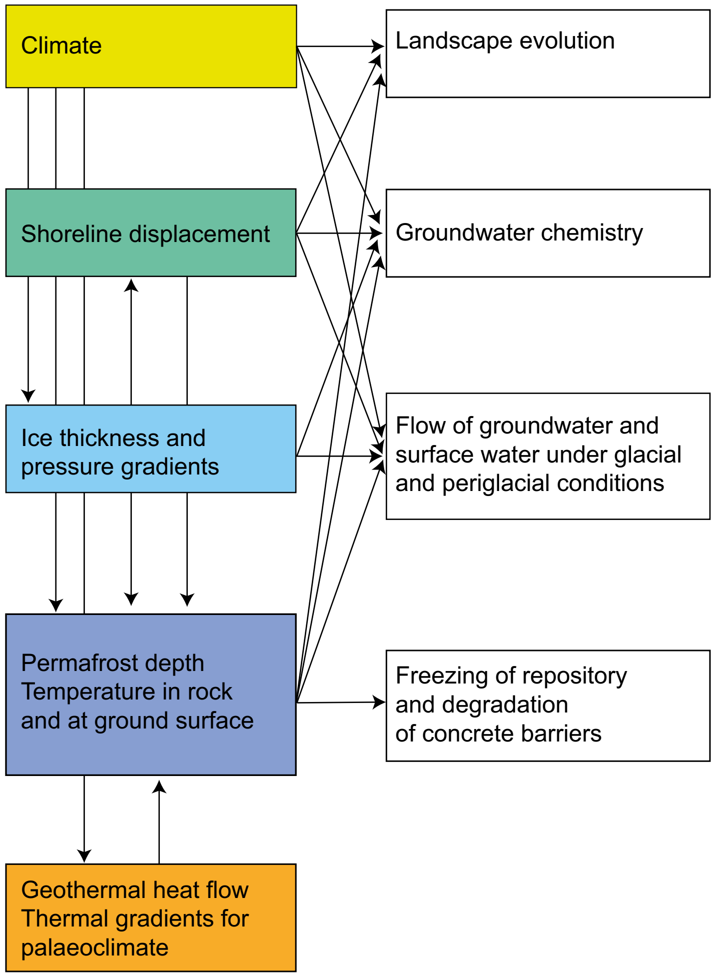


Figure 19-1. *Examples of how different parts of the climate programme are linked to other parts of a safety assessment. The example is from the SR-PSU safety assessment for the planned extension of SFR. Coloured boxes show areas studied in the climate programme. White boxes show other activities in SR-PSU where data from the climate programme are used. The arrows indicate main data flows.*

19.2 Ice sheet dynamics and glacial hydrology

The glacial climate domain prevails in areas covered by glaciers or ice sheets. The most important research areas for SKB for this climate domain are ice sheet dynamics, glacial hydrology and the Weichselian glacial history.

During the past 800,000 years, a typical glacial cycle had a duration of around 100,000 years. At the time of maximum ice sheet extent during the glacial periods, all of Sweden was covered with ice. However, the average extent of the ice sheets over Scandinavia during the Quaternary Period was considerably smaller than that. The Forsmark region was probably ice-free during most of these 800,000 years.

The thermal and hydrological 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 the section "Hydrology of ice sheets" below) and how a groundwater of glacial origin affects, for example, the properties of a buffer clay, see Chapter 24 "Buffer and backfill". When an ice sheet advances and retreats, 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 26.8 "Reactivation – movement along existing fractures". The presence of an ice sheet also affects the thermal, hydraulic, and mechanical (THM) processes in the rock, see Section 26.4.2 "Hydrogeology in the deep rock".

The numerical reconstruction of the ice sheet during the Weichselian (Figure 19-2), which was described in RD&D programme 2007, is still being used in SKB's reconstruction of conditions during the last glacial cycle. The methodology that described how extreme climates within the glacial domain are studied, for example concerning maximum ice thicknesses in Forsmark, is also still relevant in the climate programme. The results showed that the maximum expected ice thickness above Forsmark in the reconstruction of the Weichselian is around 2,900 metres, whereas the greatest ice thickness over the site during the past two million years (during the Saalian glaciation) is estimated at around 3,400 metres.

As reported in RD&D programme 2010, the reconstructed extent of the ice sheet during the Weichselian has been used as input data in model simulations of shoreline displacement, permafrost and stress changes in the Earth's crust. The latter are of interest in studies of the stability of faults and the occurrence of glacially induced earthquakes, see Section 26.8 "Reactivation – movement along existing fractures".

Conclusions in RD&D 2010 and its review

SSM noted in its review of RD&D Programme 2010 that SKB had responded to the comments expressed by the Authority in the review of RD&D programme 2007. SSM took a positive view of the Greenland Analogue Project (GAP) and the efforts to gain a better conceptual understanding of hydrology and hydrogeology under glacial conditions. SSM said that the study planned on transferring knowledge from Greenland to Fennoscandian conditions was important so that the information will be relevant to the safety assessments for the existing and planned repositories. The consequences for the model uncertainties should also be analyzed in the safety assessments.

In its review of RD&D programme 2010, just like in its review of RD&D Programme 2007, SSM commented on the lack of an explanation of how the described initiatives are linked to the plans for studies of the interaction between mechanical and hydraulic processes, which are for example described in the section "Groundwater flow and integrated modelling – thermo-hydro-mechanical evolution".

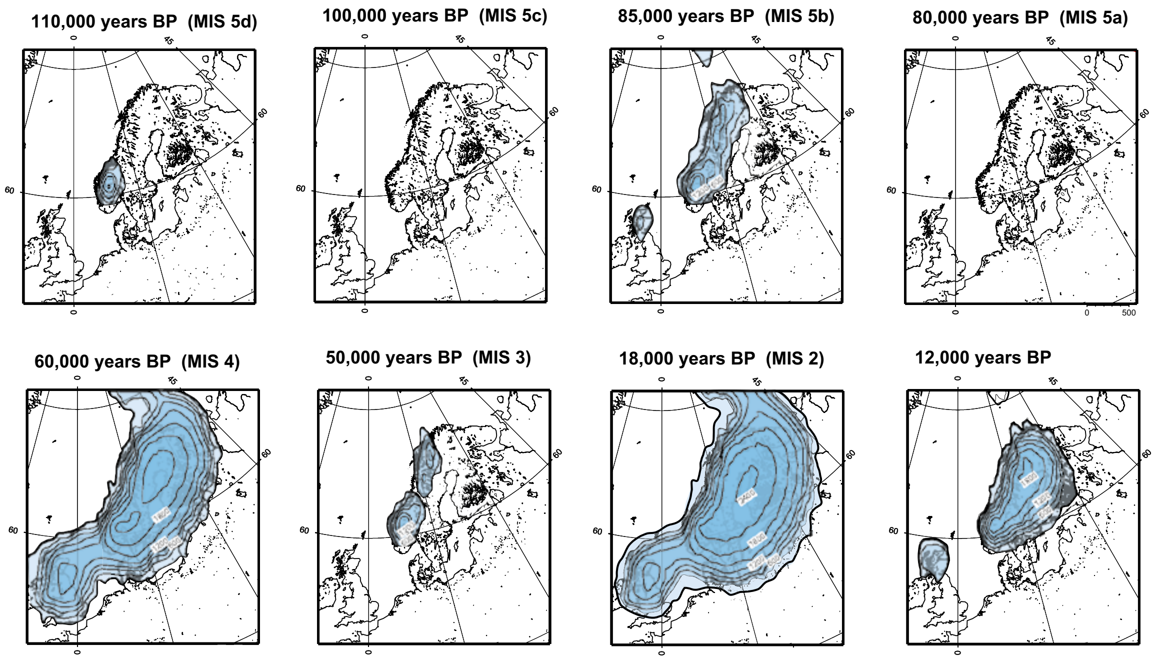


Figure 19-2. *Examples of ice sheet configurations and ice surface elevation (with a 300 metre contour interval) in the reconstruction of the Weichselian glaciation used in SKB's safety assessment work. It is worth noting that the extent of the ice sheet during the long period called Marine Isotope Stage 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 area around Forsmark 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 (starting around 70,000 years ago) until the end of MIS 2 (12,000 years ago), information from SKB's climate programme and from other research has shown that a long ice-free period prevailed during parts of MIS 3, i.e. in the middle of the last ice age.*

SSM found it laudable that SKB had studied weathering and erosion, so-called denudation (Olvmo 2010). However, the Authority was of the opinion that further analyses may be justified, depending on how SKB evaluates the effects of more extensive glacial erosion in the coming safety assessments. The Authority also considered that SKB should shed light on possible uncertainties in estimated values of glacial erosion.

Newfound knowledge since RD&D 2010

As noted by SSM, there are links between SKB's climate programme and the programme for thermal, hydraulic, and mechanical (THM) processes, as well as the programme for hydrogeology, which are described in Chapter 26 "Geosphere". These links can be exemplified by the fact that information from SKB's climate programme concerning ice sheets (this section) and permafrost (Section 19.5) in SR-Site was used to set up boundary conditions for THM studies of hydraulic jacking (Lönnqvist and Hökmark 2010). The links between SKB's climate programme and the programme for hydrogeology can be exemplified by the fact that results from the studies of the properties of the ice sheet and the glacial history during the Weichselian in SR-Site were used to set boundary conditions for hydrogeological simulations of groundwater flow under glacial conditions (Jaquet et al. 2010, Vidstrand et al. 2010a, b, 2012, Selroos et al. 2012). In a similar manner, results from the permafrost studies conducted within SKB's climate programme (Section 19.5) were used to set boundary conditions and conditions for simulations of groundwater and soil water flow under periglacial conditions (Bosson et al. 2010, Vidstrand et al. 2010a, b). Additional links between disciplines, this time between climate and surface ecosystems/landscape evolution, are exemplified in Lindborg et al. (2010, 2012, 2013).

Glacial geological information

In order to get a better substantiated and more detailed picture of conditions during the last glacial cycle, further studies have been undertaken of the glacial history of the Weichselian. 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 glacial, see Section 19.6.

RD&D Programme 2010 presented research that contributed information which revised the classic picture of the Weichselian glaciation. Previously, a very long uninterrupted period of glaciation over Sweden has been assumed, from sometime during Marine Isotope Stage 4 (MIS 4) during the middle of the Weichselian (starting about 70,000 years ago) up to the latest deglaciation at the end of the MIS 2 period (around 12,000 years ago). The new picture shows a more dynamic ice sheet, and thereby inferring also a more dynamic climate, with ice-free conditions in large parts of Scandinavia and Sweden during parts of MIS 3, i.e. in the middle of the ice age. The results agree with the ice sheet evolution used in SKB's reference evolution.

After the study of conditions during the Mid-Weichselian, MIS 3, similar studies were initiated for the early part of the Weichselian where palaeoclimate (see Section 19.6) and the ice sheet extent are being studied for the interstadials MIS 5c–e with the aid of macrofossils, Chironomidae, pollen and vegetation. These studies have been under way over the past three-year period, and the results are planned to be reported in 2013.

The results of the studies help to reduce the uncertainties in the Weichselian glacial history. They have contributed important information on different parts of the Weichselian glacial cycle see for example the summary of the glacial history of Scandinavia in Lundqvist (2011, Figure III.3). As a result, the safety assessments have better input data for the reference evolution based on the repetition of the Weichselian glaciation, and a better picture of the variability in ice sheet extent that can occur during a glaciation.

Knowledge regarding the configuration and evolution of the ice sheet at the time of the last glacial maximum about 20,000 years ago, as well as regarding the subsequent deglaciation of Scandinavia, is already relatively good.

Numerical ice sheet simulation

The reconstruction of the ice sheet that is used in SKB's work (Figure 19-2) was obtained from the University of Maine Ice Sheet Model (UMISM). This ice sheet reconstruction has been evaluated in two studies: one at Uppsala University (Schmidt et al. 2010) and one done in collaboration between the University of Calgary and Uppsala University. Both studies use Glacial Isostatic Adjustment (GIA) models to analyze the isostatic response to a given ice load history. In this work, SKB's reconstruction of the evolution of the Weichselian ice sheet has been compared with ice sheet reconstructions done with the models ANU (Lambeck et al. 2010) and ICE-5G (Peltier 2004). The latter two models are based on reconstructions of isostasy/sea-level variations, whereas the UMISM is based on ice sheet thermodynamics. Part of the study aims to see how well the isostatic response from the different ice sheet reconstructions agree with today's isostatic uplift of the Earth's crust in Scandinavia as observed in GPS studies. Preliminary results from the comparison are published in Schmidt et al. (2010). The results show clear differences between the different ice sheet reconstructions (see Figure 19-3), but also that all three reconstructions can recreate the general pattern for the isostatic uplift of the Earth's crust, depending on the choice of Earth parameters in the GIA model. Results from these studies have also been presented at a number of scientific conferences (Lund et al. 2011a, b, 2012, Schmidt et al. 2010).

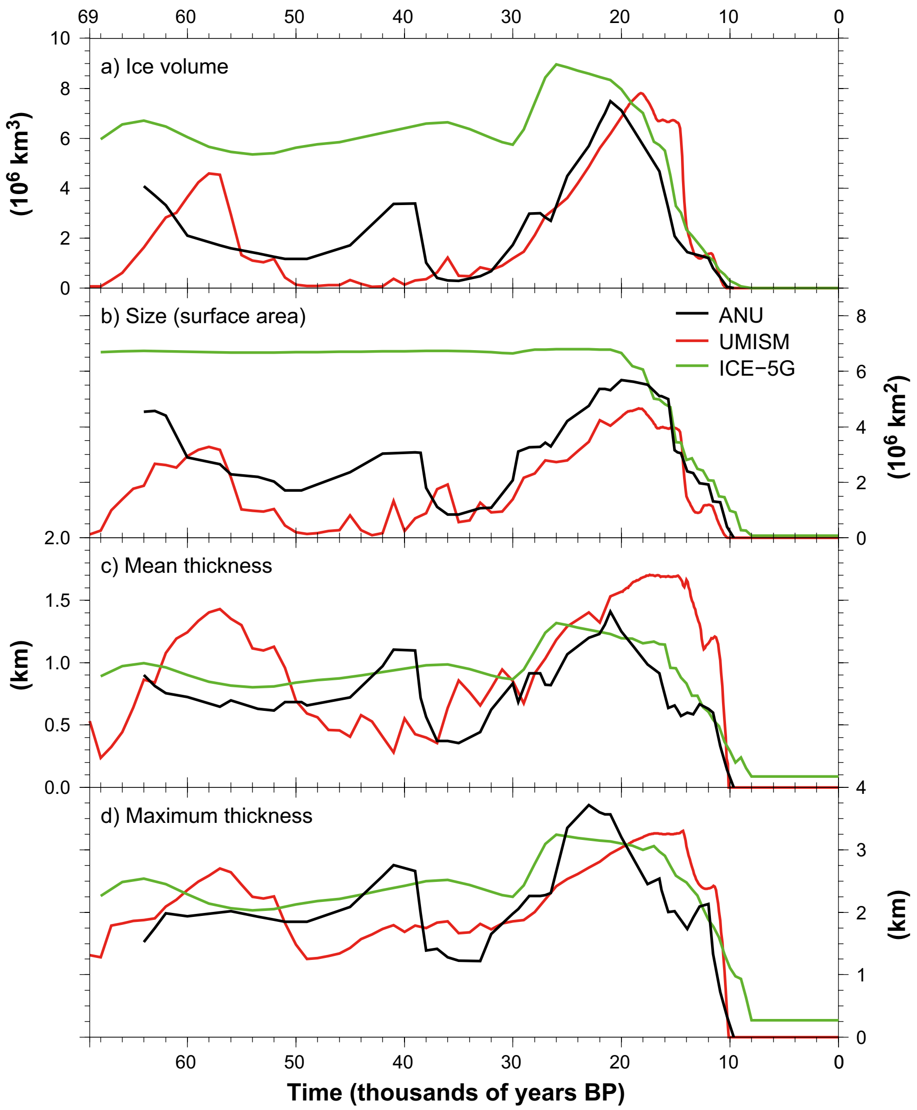


Figure 19-3. *Example of comparison of three reconstructions of ice sheet evolution during the Weichselian glaciation: UMISM (SKB 2010w), ANU (Lambeck et al. 2010) and ICE-5G (Peltier 2004). The comparison shows both similarities and differences between the reconstructions and comprises part of an analysis of the ice sheet reconstruction (UMISM) that is being used in SKB's safety assessment work.*

Ice sheet hydrology

The theoretical knowledge (up to 2007) of how water flows in and beneath a glacier or ice sheet, and how this knowledge is applied in model simulations, has previously been compiled and reported in Jansson et al. (2007). Jansson (2010) summarizes the knowledge up until 2010 concerning the hydrology of ice sheets gathered via observations from Greenland and Antarctica, i.e. before results from the GAP project was available. Many of the studies described in the report fit into one of the categories i) well-documented processes from small glaciers are verified on an ice sheet scale, or ii) processes specific for ice sheets. These categories constitute the basis for the literature study.

The compilations show that glacial-hydrological field observations have for the most part been done on valley glaciers and not on continental ice sheets which is more relevant to SKB's safety assessment work. Knowledge concerning glacial-hydrological processes in ice sheets has been scarce, and the lack of process understanding has made it difficult in many cases to conceptualize the processes in today's large-scale ice sheet models. To address these issues, SKB, together with its sister organizations Posiva in Finland and NWMO (Nuclear Waste Management Organization) in Canada, has since 2008 conducted extensive research within the GAP project on western Greenland: the aim is to study hydrological processes at an existing ice sheet, see Section 19.7. This project has contributed, and will contribute further during the coming years, a large quantity of new and unique information on the glacial-hydrological systems of ice sheets. The purpose of the project is to describe the hydrological system both on the upper ice sheet surface, inside the ice, at the interface between ice and bed, and in the rock beneath the ice, see further Section 19.7.

Within the same investigation area on western Greenland, SKB has since 2010 also been conducting a study (Greenland Analogue Surface Project, GRASP), whose purpose is to i) study the surface hydrology in the periglacial environment (collaboration with Stockholm University), see Section 26.4.1 "Surface hydrology and hydrogeology", and ii) study how solutes are distributed and transported in this environment (collaboration with Umeå University), see Sections 27.7 "Effects of long-term variations" and 27.8 "Landscape evolution and deposits". The overall purpose of these studies is to show how these processes may act in the Forsmark area during future periods with a periglacial (cold and dry) climate with permafrost.

Programme

The studies of palaeoclimate during different periods of the Weichselian glaciation by analysis of sediments from the Sokli site in northern Finland will continue and in some cases be concluded during the coming RD&D period, see Section 19.6. These studies will, as during the past RD&D period, continue to contribute information on the extent of the ice sheet during the Weichselian period.

Numerical ice sheet simulation

The study comparing SKB's ice sheet reconstruction with the models ANU and ICE-5G (see above) will continue to be conducted and the results reported. The collaboration between the University of Calgary and Uppsala University regarding the evaluation of the UMISM reconstruction of the Weichselian glaciation (Benchmark of UMISM for Northern Europe with 1D Laterally Homogeneous and 3D Spherical Finite-Element Models – Comparison to Sea-Level, GPS and GRACE data in Europe) will also be concluded.

The results of both studies will be used to discuss the uncertainty in the reconstruction of the Weichselian glaciation (Figure 19-2), as well as to further improve the descriptions of the glacial history of Scandinavia.

In addition to SKB's own studies and other published results regarding maximum ice thicknesses during the Saalian glaciation (reported in SKB 2010w, Section 5.4), SKB plans to conduct a supplementary model study on ice thicknesses above Forsmark during the Saalian glaciation (i.e. the biggest glaciation that occurred during the Quaternary Period). One possibility here is to use a combination of ice sheet simulation and climate simulation for all or selected parts of this glacial period. The planned study is primarily relevant for the Spent Fuel Repository and SFL.

Hydrology of ice sheets

The scientific discipline that describes the hydrology around and beneath ice sheets is making rapid progress, via both the GAP project (see Section 19.7) and many other research groups' studies of the evolution of the Greenland ice sheet in a warmer future climate.

The current state of knowledge concerning ice sheet dynamics and glacial hydrology, which was compiled and reported in SKB (2006c), will be updated in conjunction with the final reporting of the GAP project.

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. SKB intends to continue working within this area by studying how glacial hydrology can and should be conceptualized in hydrogeological studies, see Sections 19.7 and 26.4.2 "Hydrogeology in the deep rock". The GAP is important in this context. The results will provide a better understanding of the hydrological and geochemical conditions in and around an 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 is conducting a study concerning ice sheet modelling, with a focus on basal conditions and basal hydrology in ice sheet models. Data gathered from the study area for the GAP (meteorological information for meltwater production on the surface, ice movement data, ice thickness, basal temperature, hydrological conditions, etc) are being used in this study (Numerical Modelling to Further Understand Thawed versus Frozen Conditions at the Base of the Ice Sheet) for simulations of the Greenland ice sheet in the study area for the GAP.

In conjunction with the GAP, SKB is also conducting a study with the specific purpose of transferring the glacial-hydrological knowledge obtained from the GAP to Scandinavian conditions, including sub-projects that specifically deal with the area around Forsmark. This study (Water Routing Theory through Ice Sheets Based on Greenland Field Data and its Application to the Fennoscandian Ice Sheet) is being conducted as a Ph. D. project at Stockholm University. The dissertation defence is planned for the end of 2014.

In conjunction with the GAP, SKB is also co-funding the Ph. D. projects "Modelling the Hydrology in an Area of the Greenland Ice Sheet, Kangerlussuaq, West Greenland with Mike-She," conducted at the University of Copenhagen, and the study "Characterizing Subglacial Conditions and Processes of a Land Terminating Section of the Greenland Ice Sheet Using Geophysical Methods," conducted at Uppsala University. The dissertation defences for both of these studies are planned for 2014.

The GRASP project will also continue during the coming RD&D period (see Section 26.4.1 "Surface hydrology and hydrogeology", 27.7 "Effects of long-term variations" and 27.8 "Landscape evolution and deposits".

19.3 Denudation

Conclusions in RD&D 2010 and its review

In its review of RD&D Programme 2010, SSM wanted SKB to shed further light on possible uncertainties in the values for glacial erosion.

SKB should also elucidate the importance of the fact that the Forsmark area was probably not affected by more than six glaciations during the Quaternary Period for the generalization of the Quaternary Period with respect to erosion.

Studies of future erosion at Forsmark might be warranted, depending on how SKB values the effects of major glacial erosion in the coming safety assessments.

Newfound knowledge since RD&D 2010

In 2010, SKB concluded a project aimed at describing and quantifying denudation (weathering and erosion) of the ground surface over long time spans (a hundred thousand and a million years) in the Forsmark and Oskarshamn regions. The method that was used was studies of the long-term morphological evolution of the bedrock surface.

The rate at which the bedrock in Sweden's coastal areas was worn down due to erosion and weathering during the Pliocene-Pleistocene (a five-million-year-long period) was in general low, less than 100 metres per million years (Riis 1996). This value only applies to the *general* lowering of the bedrock surface and therefore does not include the expected higher erosion rates that have occurred along e.g. valleys and bedrock fracture zones. Olvmo (2010) showed that the site of the Spent Fuel Repository and SFR in Forsmark is situated within the intact parts of the sub-Cambrian peneplain, with very low relief. As a result, future non-glacial denudation is expected to be very low on a time perspective of both a hundred thousand years and a million years (up to five metres per million years in the Forsmark area). Future glacial erosion is also estimated to be relatively low in the Forsmark area – on the order of one to two metres per glacial cycle. This low value of glacial erosion is a result of the fact that the site locally above the repositories is not located in a local depression or a major fracture zone, but rather in an intact area with a low tendency to erode. However, in other locations, with greater relief and occurrence of major fracture systems, glacial erosion is expected to be potentially much higher. Examples of such areas are found at the closest 25 kilometres southeast of Forsmark, where it is estimated that future glacial erosion rates in topographical depressions could be more than ten metres per hundred thousand years (Olvmo 2010).

Programme

To further clarify the uncertainty in the estimated values of glacial erosion in the Forsmark area, SKB plans to conduct a supplementary study of denudation with a focus on glacial erosion. The planned study is GIS-based and includes a statistical description and analysis of glacial overdeepenings (geometry and location) with the aid of topographical bedrock data from several study areas on Earth. The areas will be chosen as to cover both currently glaciated regions (for example the study area for the GAP on Greenland) and previously glaciated regions (e.g. Forsmark).

In addition, SKB plans to keep track of the current scientific literature in the field of denudation and glacial erosion and include this information in future safety assessments.

19.4 Isostasy, eustasy and shoreline displacement

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, including both climates that are colder than today and all conceivable climates in Sweden dominated by global warming. Within the temperate climate domain, shoreline displacement is the most important climate-related process of importance for a final repository in Forsmark, regardless of repository concept. Shoreline displacement is the net result of isostatic changes (for example present-day uplift of the crust) and eustatic changes (changes in sea level).

In SR-Site, the description of a warmer future climate, caused by an increased greenhouse effect, was broadened compared with previous safety assessments. Two cases with different durations of the initial temperate period and with different climate impacts were included. The first was the Global warming climate case, equivalent to the UN Intergovernmental Panel on Climate Change's (IPCC) medium-emissions scenario A1B, and the second case was the Extended global warming climate case, equivalent to IPCC's high-emissions scenario A2. In the first case, a temperate climate domain prevails during the first approximately 60,000 years at Forsmark. After that, the climate becomes gradually colder, with at first short, but then increasingly longer, periglacial periods of permafrost. The first period of glacial conditions occurs at the end of this approximately 100,000 year long period. The second case is completely dominated by global warming. In this case, the temperate climate domain prevails during the entire coming 100,000 year period.

For studies specifically concerning the climate in SKB's scenarios for global warming, see Section 19.6 "Climate and climate variations".

Conclusions in RD&D 2010 and its review

In its review of RD&D Programme 2010, SSM took a positive view of the fact that SKB will consider new knowledge regarding possible future sea level rise. SSM further found it important that SKB should follow through on its plans to assess shoreline changes at Forsmark in the longer time perspective that is relevant for the final repository for spent nuclear fuel as well.

Newfound knowledge since RD&D 2010

Based on the issues that are relevant to the SR-PSU safety assessment, the climate cases from SR-Site have been revised and supplemented. The two climate cases with global warming from SR-Site, Global warming and Extended global warming, have been updated in the light of scientific literature published in recent years. Since future changes in sea level are of great importance for society, a great deal of international research is being done in this area, and a great deal of information on future changes in sea level has been published in the scientific literature since RD&D programme 2010. The following literature reviews have been done for these two climate cases in areas related to sea level change, and the results will be reported within the framework of SR-PSU:

- Updated knowledge concerning possible sea level rise due to thermal expansion and melting of glaciers and inland ice sheets. The study covers both sea level rise up to 2100 and the magnitude and time of maximum sea levels after full thermal expansion.
- The potential for changes in ocean circulation in the North Atlantic due to global warming, including estimated cooling of the climate in Scandinavia, for evaluation of the risk that global warming could lead to a regionally colder climate in Forsmark. The conclusion of the study is that a future change in ocean circulation in the North Atlantic cannot be ruled out, and that such a change could potentially lower the temperature in Scandinavia. However, even assuming low future carbon dioxide emissions, this change in ocean circulation could not lead to annual mean temperatures below 0°C in Forsmark due to the ongoing process of global warming.

SKB is addressing the problems that sea level rise could cause during the construction work on both the Spent Fuel Repository and the extension of SFR. The study describing future changes in the sea level at Forsmark up until 2100 for the safety assessment for the Spent Fuel Repository (Brydsten et al. 2009) has been updated for SR-PSU based on literature studies. The study 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 compilation has been updated and now includes more references relating to global sea level rise. The results show, like before, that the maximum short-duration (during storms) water level in 2100, in a scenario with maximum global sea level rise, is around plus three metres for Forsmark. The compilation further shows that the uncertainty in estimates of how much the sea level could potentially rise up to 2100 is still very large. In this context it is important to note once again that this scientific area is in a very intensive phase, and revisions of these numbers are to be expected.

The GIA simulations used to describe shoreline displacement in SR-Can were used for SR-Site as well. However, they were supplemented with new information from published scientific literature, particularly with respect to future sea level rise, as well as with information from 3D GIA simulations, which also included lateral variations in the thickness of the Earth's crust, see SKB (2010w). These GIA simulations are also being used in SR-PSU together with the updated information on possible future sea level rise.

Programme

SKB's reconstruction of the glaciation history of the last glacial cycle is being evaluated via the isostatic response to ice sheet load history in simulations with different GIA models, as described in Section 19.2 "Ice sheet dynamics and glacial hydrology". These studies will continue and be concluded during the coming RD&D period. Besides providing information that is useful in discussions of the Weichselian glacial history, the results will also provide important information concerning assumptions that must be made in GIA simulations of isostatic changes in Scandinavia.

The scientific literature concerning future sea level change (causes, mechanisms and consequences) will continue to grow in the years to come. SKB will therefore continue to follow research in this area and use relevant results for assessment of the impact that changes in the shoreline at Forsmark would have on the Spent Fuel Repository and on SFR, both in shorter (up to year 2100) and longer (several tens of thousands of years) time perspectives.

19.5 Permafrost

Aggradation (growth) and degradation (melting) of permafrost are the most important climate-related process for a final repository within the periglacial climate domain, regardless of waste and repository concept. The periglacial climate domain prevails for about one-third of the last glacial cycle in SKB's reconstruction of the Weichselian glacial cycle (SKB 2010w). The occurrence of permafrost greatly affects the groundwater's flow pattern. Groundwater composition may also be affected by salt exclusion.

Conclusions in RD&D 2010 and its review

In its review of RD&D Programme 2010, SSM took a positive view of the fact that SKB had conducted further site-specific studies on permafrost growth and its influence on hydrological and geochemical changes. These studies were not yet published, so SSM could not judge whether all the Authority's viewpoints on RD&D Programme 2007 had been addressed in these studies and whether further initiatives might be called for. Similarly, it was difficult for SSM to judge whether an increased understanding of the palaeotemperature in Forsmark was called for, since the results of the study that was done (Sundberg et al. 2009) needed to be considered in the context where they were used. SSM considered it important that SKB should make use of the findings of the GAP for studies of the importance of the permafrost for hydrogeochemistry and hydrology.

Newfound knowledge since RD&D 2010

The results of the detailed studies of permafrost growth in the Forsmark area that were described in RD&D programme 2010, conducted within SR-Site, have been published (Hartikainen et al. 2010, SKB 2010w, Sections 3.4 and 5.5). The studies were conducted in a 2D model along a 15-kilometre-long and 10-kilometre-deep profile that runs in the regional topography's gradient, through the planned Spent Fuel Repository and through SFR. Detailed data from the site investigation programme – for example on surface topography, thickness and composition of the soil cover, topographic wetness index (including vegetation cover type), shoreline displacement and future sea level change, locations of rock domains and fracture zones, and bedrock thermal, chemical and hydraulic properties – were used as input data in the simulations. The purpose of the study was to provide a realistic picture of permafrost growth in the Forsmark area, and the extent to which it affects repository performance. The results were used in SR-Site to set up conditions regarding permafrost in hydrogeological studies of groundwater flow (Vidstrand et al. 2010a, b) and soil water flow (Bosson et al. 2010) under periglacial conditions. In connection with this, the effect of the presence of potential future taliks (unfrozen areas in soil with permafrost) in the Forsmark area was also studied (Bosson et al. 2010, Hartikainen et al. 2010).

The results of the permafrost simulations that were done for SR-Site show that salt exclusion in connection with the formation of permafrost only occurs to a limited degree in Forsmark, see SKB (2010w, Section 5.5.3) and Hartikainen et al. (2010, Section 5.1 and Chapter 5). If a very sharp reduction of the air temperature is assumed, the process can be demonstrated more clearly but is still not extensive. The reason why salt exclusion is not a significant process is that the groundwater in the upper parts of the rock has a low salinity. No great impact on the groundwater's chemical composition is therefore expected during future periods with a periglacial climate and permafrost in Forsmark.

The results from the permafrost study (Hartikainen et al. 2010, SKB 2010w) demonstrated how the site-specific process of permafrost aggradation and degradation takes place 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 the prevailing conditions at the ground surface (temperature, vegetation and snow cover) are the principal controlling parameters for the spatial and temporal extent of the permafrost in Forsmark, whereas the properties of the rock and the groundwater are of secondary importance only.

In addition to simulations with constant air temperature, SKB's reference evolution was also analyzed, along with supplementary climate cases that were more favourable for permafrost growth. Extensive sensitivity studies of input data were carried out with the permafrost model, including on the temperature curve that is used for these time-transient simulations. The broad spectrum of simulations, including the sensitivity studies, show in detail what the uncertainty in input data looks like and its consequences for the results (Hartikainen et al. 2010, SKB 2010w). The results of the permafrost simulations were used in SR-Site's safety assessment scenario for buffer freezing. The main conclusion was that freezing of the buffer around the canisters can be ruled out with good margin, even if all uncertainties in the permafrost simulations are set to the most favourable number for permafrost growth and pessimistically added to each other (SKB 2011e).

In the climate cases where permafrost occurs within the period when high decay heat from the Spent Fuel Repository heats up the bedrock, i.e. during the first tens of thousands of years, this heat results in a permafrost depth that is several tens of metres shallower than in the cases where no heat comes from the repository (Hartikainen et al. 2010). As the heat gradually declines, so does this effect, and the permafrost reaches deeper. No such effect exists in the case of SFR.

Even if permafrost is not included as a dedicated sub-project in the GAP, the GAP's research programme is planned so that the project contributes important information in this area as well, see Section 19.7. 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, the salinity in the groundwater beneath the permafrost exemplify what the geochemical composition of the groundwater may look like in crystalline bedrock under periglacial conditions in the vicinity of an ice sheet.

Several of the approaches and methods used in the climate programme for the safety assessment of the Nuclear Fuel Repository can also be used for safety assessments of the extension of SFR. However, as mentioned above, 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. Within the framework of the SR-PSU safety assessment, a study is being done of how this influences the selection and description of the climate evolutions that are relevant to the assessment of the long-term safety long-term safety of SFR, see Näslund et al. (2013).

In addition to the permafrost simulations that were done for the Spent Fuel Repository in SR-Site, which also included the site of SFR, permafrost simulations adapted to the special conditions and issues that apply for SFR have been done within SR-PSU. The study "The Potential for Cold Climate Conditions and Permafrost in Forsmark in the next 60,000 Years" aimed at analyzing the possibility of permafrost in Forsmark during the next 60,000 years considering known variations in insolation and a range of possible atmospheric carbon dioxide concentrations. There was a special focus on the question of whether the climate in Forsmark could become sufficiently cold to create permafrost during periods of low insolation, which will occur in about 17,000 and 54,000 years. The future climate was simulated for these periods using a simplified climate model, LOVECLIM (Driesschaert et al. 2007) and a state-of-the-art climate model, CCSM4 (Gent et al. 2011). In order to provide input data for the analysis of the possibility of permafrost in Forsmark, both equilibrium simulations (where forcing conditions such as atmospheric greenhouse gas concentrations and insolation were held constant) and transient simulations (where the spatial and seasonal distribution of insolation varied in time as expected based on future variations in the astronomic parameters) were done.

The equilibrium simulations were carried out for the times 17,000 years and 54,000 years after present, with atmospheric carbon dioxide concentrations in the range 180 to 440 ppmv (parts per million, volume). The simulated air temperature in Forsmarkthat gave the coldest climate was then used as a input for the same permafrost model as was used for SR-Site (Hartikainen et al. 2010, SKB 2010w). Based on the sensitivity studies that were done in Hartikainen et al. (2010), a dry climate and a dry ground surface above SFR were here assumed in the permafrost simulations, since these conditions are most favourable for permafrost growth. Furthermore, a number of new sensitivity experiments were done with the permafrost model where the air temperature was reduced compared to the temperature simulated with the climate model. The results of the climate simulations show, as expected, that the climate may be colder during the future periods around 17,000 and 54,000 years after present. The conclusions of the study were based on a comparison of the results in question with other studies of future climate where an evaluation of the uncertainties in the results was of central importance. All uncertainties in the simulated climate were assumed to influence the results towards a maximally cold climate.

When the uncertainties in methodology and in the current state of knowledge concerning future climate evolution are included, the conclusions of the study is that permafrost could form in the area around SFR in both 17,000 and 54,000 years, provided that the atmospheric carbon dioxide concentration has declined to relatively low levels. However, this does not automatically mean that SFR repository structures will freeze, since for example concrete has a lower freezing temperature than water. The results will be used in the SR-PSU safety assessment, together with other information, for analysis of when the first freezing of the concrete structures in SFR could occur. An example of results from the permafrost simulations are shown in Figure 19-4.

Programme

Within SR-PSU, studies are being conducted that address the special issues and conditions that are relevant to SFR. One important question is how early permafrost, and associated freezing of SFR, could occur in Forsmark in the future. This and related questions are being investigated by modelling of the future climate, modelling of permafrost, and palaeoclimatological studies of how the climate has varied during selected portions of the Weichselian glaciation.

The numerical permafrost model that was used in the SR-Site (Hartikainen et al. 2010, SKB 2010w) and SR-PSU (Brandefelt et al. 2013) safety assessments are planned to be further evaluated by application of the model to the study area for the GAP project on western Greenland. On this site, proglacial permafrost occurs down to a depth of 300–350 metres, see Section 19.7. The same 2D versions of the model are planned to be used in this study as were used in the respective safety assessments. 3D simulations are also planned to be done in this study. In addition to the fact that the results can be used for evaluation of the permafrost model, they will also contribute to a better understanding of permafrost and taliks in the study area for the GAP and their importance for hydrology and groundwater composition, results that will be applied to coming analyses of the Forsmark site.

Different repository concepts, with different types of barriers, have different sensitivities to degradation by freezing. For example, concrete barriers suffer worse physical degradation than bentonite barriers if they where to freeze. Moreover, different repository depths provide different preconditions for freezing. Within the framework of the planned safety evaluation for the SFL repository, research initiatives may be undertaken regarding the link between potential freezing of disposal structures and choice of repository concept and repository site.

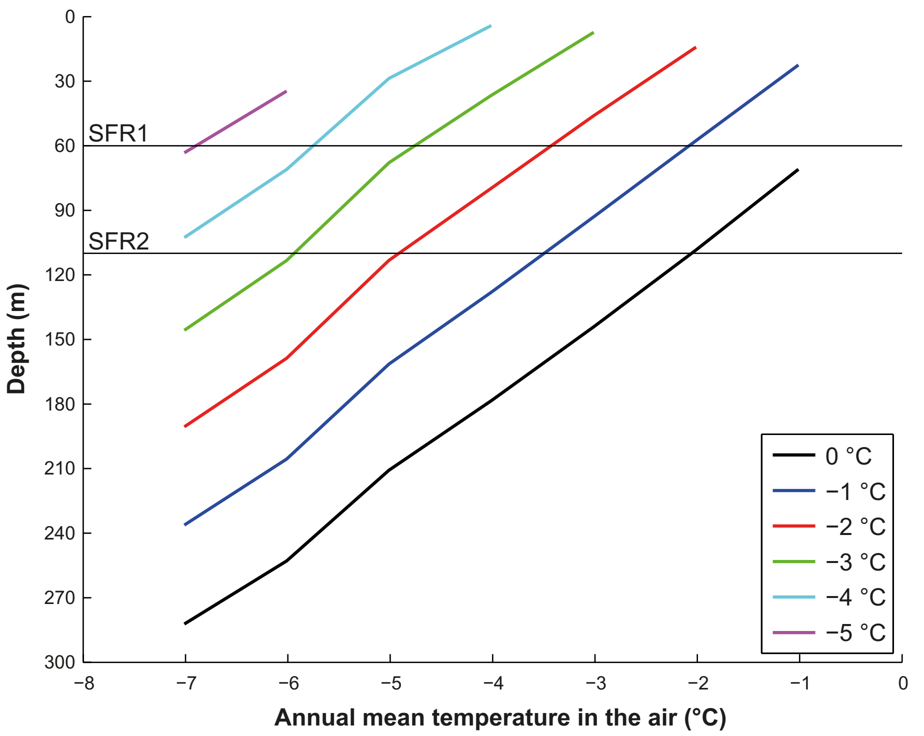


Figure 19-4. *Example of results from permafrost simulations done within SR-PSU. The figure shows simulated depth of the 0, –1, –2, –3, –4 och –5°C isotherms at SFR for different annual mean air temperatures. The depth of the 0°C isotherm corresponds to the permafrost depth, while the –3°C isotherm corresponds to the freezing depth for concrete structures in SFR if the freezing criterion in the safety assessment is set to –3°C. The depths of the existing SFR (SFR1), about 60 metres, and of the planned extension (SFR2), about 110 metres, are shown by horizontal black lines.*

The uncertainty in the temperature curve that was used for e.g. simulations of permafrost in SR-Site was analyzed and described in detail in that safety assessment (SKB 2010w, Appendix 1). The effect of this uncertainty was included and analyzed in the permafrost simulations (Hartikainen et al. 2010, SKB 2010w) and the analysis of buffer freezing (SKB 2011e, Section 12.3). To achieve a better understanding of the palaeotemperatures that have prevailed at Forsmark and to shed further light on the uncertainty in the temperature curve that was used in SR-Site (which is a reconstruction of the last glacial cycle), a study is planned where historical climate information will be extracted from borehole data on temperature and thermal parameters in the rock at Forsmark, Laxemar and Lake Vättern, see also Section 19.6. The planned study is a continuation of the study presented in Sundberg et al. (2009). A locale in Åre in Jämtland may also be included in the study, in cooperation with the drilling project called "The Collisional Orogeny in the Scandinavian Caledonides (COSC)," which comprises a part of the Swedish Deep Drilling Program (SDDP). COSC is planned to be carried out in cooperation with the International Continental Scientific Drilling Program (ICDP). If the COSC project is realized, SKB plans to participate in the part "Geothermal Measurements in Deep Boreholes as Input to an Understanding of Past Climate Evolution."

19.6 Climate and climate variations

In addition to the studies of climate-related processes such as ice sheets, permafrost and shoreline displacement described above, the climate programme also includes studies of the climate as such. The purpose is to provide more complete information and examples of what the climate may be like at the transitions between the different climate domains and to provide a better basis for the construction of climate scenarios that describe conceivable future climate evolutions. For this purpose, SKB uses both natural climate archives and climate models.

Conclusions in RD&D 2010 and its review

The Geological Survey of Sweden, SGU, thought that the results obtained from the investigations in Sokli, Finland were interesting and that it was positive that SKB planned to continue the investigations. However, the new results concerning climatic conditions during MIS 3 were somewhat contradictory in comparison with results from the continent for the the same time, The uncertainty of the sediment datings should be evaluated, and it is also important to continue the investigations in Sokli.

SSM noted that SKB had fulfilled the plans in RD&D Programme 2007 and was, like the Geological Survey of Sweden, positive to SKB's plans to continue studies that make use of the information from the sediment core from Sokli. SSM also thought that SKB should investigate the questions stemming from a comparison with conditions on the continent during MIS 3, see Section 19.2.

Newfound knowledge since RD&D 2010

The palaeoclimatological study of sediment cores from Sokli that was described in RD&D programme 2010 has continued. The site from which the sediments have been taken is unique for Scandinavia in that the oldest sediments are around 130,000 years old and that continuous sedimentation has occurred on the site during very long periods up to the present. An extensive reconstruction of conditions for the site for a period around 50,000 years ago (MIS 3) has been carried out based on an analysis of a large body of proxy data, see RD&D programme 2010. The results have, together with those of other studies, revised the classic picture of the glacial history of Scandinavia during the Mid-Weichselian. The study has also shown that the climate during parts of the last ice age (Middle Weichselian) was very warm, with mean summer temperatures as warm as today.

Since 2010 the study has been extended to include similar analyses for two interstadials during the early Weichselian (MIS 5e and 5c) and for parts of the Eemian interglacial, i.e. the warm period before the beginning of the Weichselian glacial period (Climate Dynamics during the Eemian Interglacial (MIS 5e) in the North Atlantic Region Inferred from a Unique Sediment Sequence at Sokli). In order to further verify that the methodology that was used to quantify climate variations works, similar studies are now also being conducted on the Holocene sequence of sediment cores from Sokli (Weichselian–Holocene Climate Variability and Environmental Change in Scandinavia Based on the Sokli Sediment Sequence). The study is primarily being carried out at Stockholm University. Two ongoing Ph. D. projects are linked to the study, see section "Programme" below.

Two literature studies have, within SR-PSU, been carried out to shed light on how conditions and climate may change at the transition from a warm interglacial climate, such as the current warm period, to cold glacial conditions. One study describes the transition from the Eemian period to the Weichselian glaciation, including the wide variations in climate and glaciation that occurred during the early and middle part of the Weichselian. The study also provides a comparison (as requested by SSM) of the climate as it has been interpreted at Sokli with information for corresponding periods from other locales on the European continent. In the study, information from Sokli is compared with information from the locales La Grande Pile in France, Oerel in Germany, Horoszki Duże in Poland and Lake Yamozero in northwestern Russia.

Comparisons are also made with other terrestrial climate archives from central and northern Europe, as well as with marine archives from the North Atlantic and with ice core data from the Greenland ice sheet. The study provides an updated view of conditions and climate evolution during the last interglacial-glacial cycle in the temperate and boreal regions of Europe. The results show that the environment and the climate during the Early Weichselian (MIS 5) were distinctly different than during the Middle and Late Weichselian (MIS 4–MIS 2). MIS 5 was characterized by three intervals with forest vegetation, in both temperate and boreal Europe. These periods were about 10,000 years long and correspond roughly to MIS 5e, 5c and 5a. These warm periods were interrupted by two shorter cold and dry periods (MIS 5d and 5b), with glaciation in the Fennocandian mountains. Superimposed on these long-scale variations, climate variations od shorter duration (thousand-year cyclicity) occur during the Early Weichselian. In contrast to the early part of the Weichselian, the Middle Weichselian (MIS 4–MIS 2) is dominated by open vegetation. During the two periods of extensive ice cover in northern Europe (MIS 4 and 2), a continental climate prevailed with very dry conditions in Eastern Europe. Warmer conditions with summer temperatures close to those of today frequently prevailed during the relatively moist period MIS 3.

The other literature review that has been carried out within SR-PSU illustrate how climate and conditions vary during a transition from warm interglacial conditions to glacial conditions, also describing the transition from the Eemian to the Weichselian (MIS 5d-a, the period from 122,000 to 70,000 years ago), see Wohlfarth (2013). The study summarizes historical climate information from terrestrial, marine and glacial archives to describe and discuss the magnitude, duration and cyclicity of the climate changes that constitute the highly variable transition from a warm interstadial to a cold glacial climate. The study also discusses the mechanisms that cause the transition to a glacial climate, based on information from ice cores and marine and terrestrial climate archives.

The studies at Sokli – and the literature reviews for a large number of other terrestrial, marine and glacial climate archives – contribute important palaeoclimatological information that enable us to better understand the limits within which the climate can vary naturally during a glacial cycle. The results are important because they also show how climate and environment change at the transition from a warm interglacial climate to a colder glacial climate, and the fact that the climate during the Weichselian glacial period was characterized by great temporal and amplitudinal variability, greater than had previously been assumed.

In addition to the extensive publications from the Sokli study that were reported in RD&D programme 2010, results from the studies at Sokli have been published over the past three years in the following scientific articles: Helmens and Engels (2010), Engels et al. (2010), van Meerbeeck et al. (2011), Helmens et al. (2012), Salonen et al. (2013), Shala et al. (2013). The study has also been presented at a large number of scientific conferences during this period.

Information from climate models

Further results from the previously reported study of conceivable climates in a 100,000-year perspective (Kjellström et al. 2009, Brandefelt and Otto-Bliesner 2009, RD&D programme 2010) have been published as scientific articles (Strandberg et al. 2010, Brandefelt et al. 2011). In order to analyze the potential for a future cold climate without an ice sheet at Forsmark, and possible resulting permafrost formation, simulations with climate models have been done for the next 60,000 years (Brandefelt et al. 2013), see also Section 19.4 above. The study is being done within the framework of SR-PSU for the purpose of analyzing the earliest time at which permafrost may form at Forsmark, and in extension whether the temperatures that then prevail in the rock could cause the concrete barriers in SFR to freeze.

The lowest insolation during the next 100,000 years will occur in 54,000 years (see for example Berger and Loutre 2002). The study therefore covers only the next 60,000 years and not the entire SR-PSU safety assessment period of 100,000 years. The simplified climate model LOVECLIM (Driesschaert et al. 2007) was used to study the importance of the future variation in insolation and atmospheric carbon dioxide. Further, two simulations were done with the state-of-the-art climate model CCSM4 (Gent et al. 2011) to exemplify the uncertainty in the results linked to differences between different climate models. In order to provide input data for the analysis of the possibility of permafrost in Forsmark, both equilibrium simulations (where forcing conditions such as atmospheric greenhouse gas concentrations and insolation were held constant) and transient simulations (where the spatial and seasonal distribution of insolation varied in time as expected based on future variations in the astronomic parameters) were done. The equilibrium simulations were carried out for the times 17,000 years and 54,000 years after present and with atmospheric carbon dioxide concentrations in the range 180 to 440 ppmv.

Figure 19-5 exemplifies the annual mean two metres air temperature in the Forsmark area in these simulations. As expected, the results show that insolation and carbon dioxide have a marked influence on future air temperatures. The results further show, in agreement with other studies in the scientific literature, that the difference between the two climate models is significant. The uncertainty in the simulated annual mean temperature due to differences between different climate models, the uncertainty in the atmospheric concentration of other greenhouse gases than carbon dioxide, and the uncertainty in the future evolution of glaciers and ice sheets were estimated based on the scientific literature. Since the main purpose of the study was to see whether permafrost could form at Forsmark, all uncertainties in the simulated climate were assumed to influence the results towards a maximally cold climate.

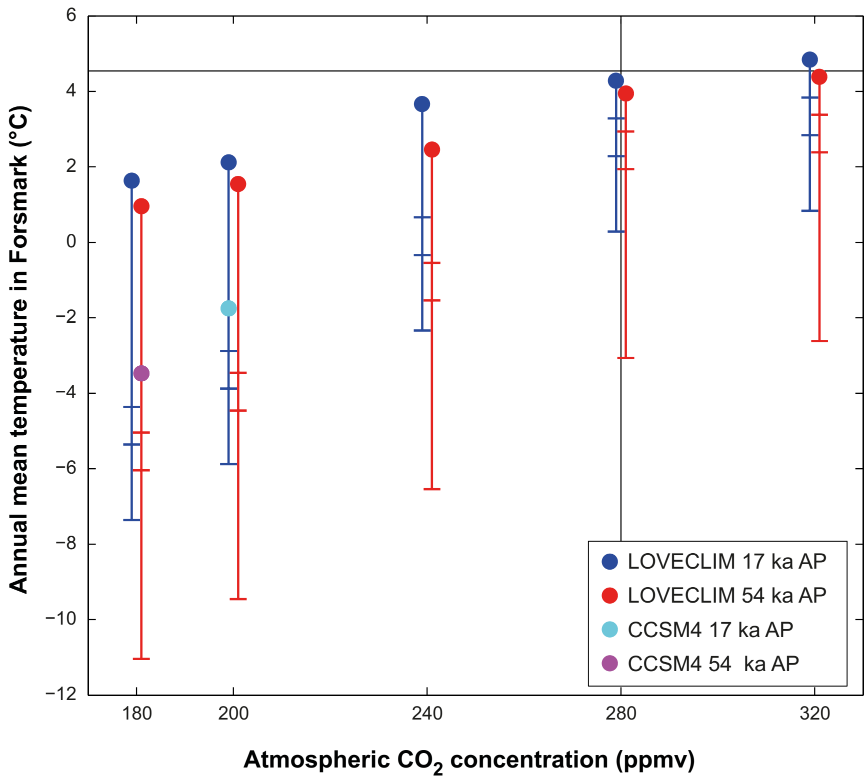


Figure 19-5. *Example of results of climate model simulations done for the SR-PSU safety assessment. The results are from a study where the possibility of a cold climate and permafrost growth in central Sweden is analyzed in relation to future variations in insolation and atmospheric carbon dioxide concentrations (Brandefelt et al. 2013). The figure shows annual mean air temperature in Forsmark in simulations of future periods with low insolation during the summer (17,000 and 54,000 years after present). The observed annual mean temperature (+4.5°C) and the preindustrial atmospheric carbon dioxide concentration (280 ppmv) during the period 1751–1850 are also shown with black lines as a reference. The uncertainty in the simulated temperature is shown in the figure (blue and red vertical lines) under the assumption that all uncertainties interact to give a colder temperature in Forsmark than the one simulated with LOVECLIM. Uncertainty due to differences between different climate models (top part of the line), uncertainty in the atmospheric concentration of other greenhouse gases than carbon dioxide (middle part of the line) and uncertainty in the future extent of glaciers and ice sheets (lower part of the line) are reported separately.*

The results of the climate simulations were used, together with the estimated maximum uncertainty in these results, as input data to site-specific simulations of permafrost in order to study whether the climate during the future periods with low insolation could generate permafrost in Forsmark, see Section 19.5.

The impact which global warming may have on groundwater chemistry is handled in the safety assessments for the Spent Fuel Repository, SFR and SFL.

The studies of geological climate archives, as well as the use of different climate models, have contributed essential information to the description of possible change and variability in the climate, which comprises an important part of SKB's safety assessments. Among other things, the studies have contributed to a better understanding of the more extreme climate evolutions that are analyzed, as well as to investigating the realism of these climate cases.

Programme

SKB has previously studied the climate during specific types of climate domains, which have been described in detail for Forsmark with the aid of climate simulations of periods of periglacial, glacial and temperate climate with global warming (Kjellström et al. 2009). These simulations were done with constant forcing and boundary conditions. SKB plans to start a supplementary study aimed at studying in detail *transitions* between different climate domains, by means of both geological archives and climate modelling. The study will analyze the period from the last glacial maximum, about 21,000 years ago, through the deglaciation, to the point in time when the current warm Holocene period was established about 10,000 years ago. The period spans a range from full glacial conditions via a warmer climate, with cold setbacks such as during the Younger Dryas, to temperate conditions during the Holocene. This is, in contrast to e.g. MIS 3, a period where a great deal of information is available. This study of transitions between climate domains has already been initiated via a Ph. D. study at Stockholm University, "Quantifying Rapid Climate Transition 20–10 ka BP Using Paleo Proxies". The Ph. D. study will be finished in 2016. Most of the study of climate transitions, including literature reviews and climate modelling, is planned to start during 2013. The study is planned as a collaboration between KTH (Royal Institute of Technology), Stockholm University (Department of Geological Sciences and Department of Physical Geography and Quaternary Geology) and SMHI (Rossby Centre).

SKB is also planning a follow-up of the study reported in Sundberg et al. (2009), where borehole temperatures from Forsmark and Laxemar were analyzed with respect to e.g. historical climate records. The follow-up study will perform a more detailed analysis and include data from a two-kilometre-deep borehole underneath Lake Vättern. Temperature logging of this borehole was carried out during October 2012. A future borehole in the Åre area may also be included in the study, in cooperation with the COSC project, see also Section 19.5. The analyses are expected to provide quantitative historical climate information (temperature) on different time scales. The results are planned to be used for further evaluation of the uncertainty in the palaeotemperature curve for the Weichselian glacial period which SKB has used for e.g. modelling of permafrost for the SR-Site safety assessment.

The ongoing studies of the climate during the early Weichselian based on the sediment cores from Sokli will continue and be concluded during the coming RD&D period. As a part of the study of the sediment cores from Sokli, SKB is carrying out two Ph. D. projects which will continue during the period. The primary purpose of the first Ph. D. project ("Holocene Climate Variability and Environmental Change in Scandinavia Based on the Sokli Sedimentary Sequence") is to investigate climate variations during the Holocene with the same methodology that has previously been used for the early and middle parts of the Weichselian (the periods MIS 3, MIS 5c and MIS 5e). The study is being done in part to evaluate the method, and in part to obtain new climate information as support for the climate scenarios that are used in SKB's safety assessments. The study is being done at the Department of Physical Geography and Quaternary Geology at Stockholm University and is planned to run until the end of 2013. The second Ph. D. project is studying the climate during the preceding interglacial, the Eemian ("Climate Dynamics during the Eemian Interglacial (MIS 5e) in the North Atlantic Region Inferred from a Unique Sediment Sequence at Sokli"). The dissertation defence is planned for the end of 2015.

19.7 Greenland Analogue Project

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 in Finland and the Nuclear Waste Management Organization (NWMO) in Canada, 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 the Spent Fuel Repository, SFR and SFL.

The field investigations for the project are 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:

- How deep down into the bedrock can glacial meltwater penetrate?
- What is the chemical composition of the meltwater if and when it reaches repository depth (around 500 metres)?
- Where beneath the ice sheet are meltwater and groundwater generated?
- How large a fraction of the oxygenated meltwater reaches repository depth?
- What is the pressure situation beneath the ice sheet?
- Does the talik that is being studied in the area act as a discharge point 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 processesduring 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 GAP consists of three sub-projects (A–C) working towards specific objectives. Sub-project A is studying the subglacial hydrology of the ice sheet and groundwater recharge indirectly by means of glaciological and geophysical investigations. A network of continuous GPS stations and automatic weather stations has been established on the ice sheet to measure ice movement and to provide data for calculations of meltwater production. Furthermore, radar and seismic surveys are made to study the properties and thermal conditions of the base of the ice sheet.

The purpose of sub-project B is to make direct observations of subglacial hydraulic conditions (including spatial and temporal variations in water pressure) and groundwater formation by means of hot water drilling through the ice sheet were pressure gauges are installed at the base of the ice sheet.

Sub-project C is studying the hydrogeochemistry and hydrogeology in the area outside and beneath the ice sheet by means of bedrock drilling and monitoring programmes. A borehole down to the equivalent of repository depth has been drilled and instrumented. Information on permafrost in the area is also being obtained from sub-project C.

Figure 19-6 is a schematic drawing illustrating the areas in which the different sub-projects within the GAP are conducting investigations. Table 19-1 lists the various types of data being gathered in the GAP. These data are being used to create an integrated understanding of the hydrological and hydrogeochemical system and are also used in ice sheet and groundwater modelling being done within, but also in parallel with, the GAP.

Table 19-1. Data and information obtained within the GAP.

Sub-projects	Information/data
A	Ice sheet velocity (GPS), presence of meltwater at the base of the ice sheet, ice depth, basal 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 temperature (ice drilling), borehole images (ice drilling), ice sheet 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 groundwater and surface water, hydrogeological data (pressure, temperature and conductivity data), microbial data, temperature profiles in boreholes, mineralogical results and petrophysical information

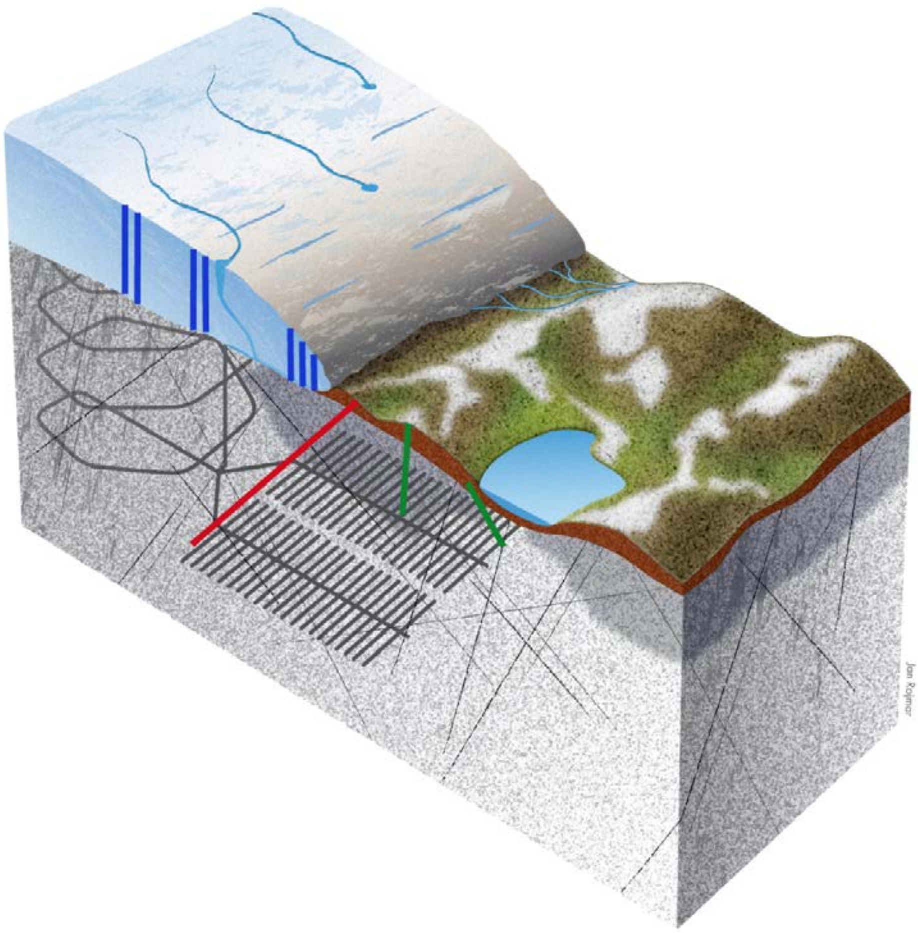


Figure 19-6. *Schematic illustration of part of the field investigation area on Greenland. The shaded dark grey areas are permafrost. Beneath the proglacial lake is a talik, i.e. an area of unfrozen ground in the permafrost. An exchange between surface water and groundwater can take place in a talik. A hypothetical final repository has been juxtaposed on the figure – in this case the layout of the Spent Fuel Repository in Forsmark has been used. There is a network of GPS stations and automatic weather stations on the ice sheet for continuous data collection for sub-project A. Via the GPS and weather station network, information is obtained on the ice sheet movement of the ice and surface melting. The blue vertical lines show where sub-project B has drilled through the ice. Green lines show the cored borehole that has been drilled by sub-project C to investigate a talik in the area and the thickness of the permafrost. The red line shows the projection of the deep borehole that was drilled by sub-project C and reaches down to repository-like depth.*

There is continuous cooperation between the three sub-projects in terms of 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, hydrogeology and geochemistry within the study area. A number of disciplines within SKB's research programme derive direct or indirect benefit from the results produced by the GAP:

1) Climate and hydrogeology: description of the hydrogeological conditions in and around an ice sheet (Section 19.2 "Ice sheet and glacial hydrology" and 26.4.2 "Hydrogeology in the deep rock").
2) Geochemistry: hydrogeochemical conditions in connection with glaciation/permafrost (Section 26.15 "Reactions with the rock – groundwater chemistry" and 26.23 "Salt exclusion").
3) Biosphere: the evolution of the biosphere in a periglacial environment (Section 27.7 "Effects of long-term variations").

Besides SKB, Posiva and the NWMO, the following organizations and universities are also participating in the GAP:

GEUS (Geological Survey of Denmark and Greenland)	Terralogica (Sweden)
GTK (Geological Survey of Finland)	University of Wyoming (USA)
In2EarthModelling (Switzerland)	University of Montana (USA)
TerraSolve (Sweden)	Aberystwyth University (UK)
Bergab (Sweden)	Waterloo University (Canada)
Geosigma (Sweden)	Stockholm University (Sweden)
Hydroresearch (Sweden)	Uppsala University (Sweden)

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 2010 and its review

In its review of RD&D Programme 2010, SSM took a positive view of SKB's Greenland Analogue Project and the efforts to gain a better conceptual understanding of hydrology and hydrogeology under glacial conditions. The Authority considered it important that the results be appropriately interpreted in relation to conditions for existing and planned final repositories in Forsmark. Section 19.2 describes a process study of ice sheet hydrology that focuses on direct transfer of knowledge obtained from the GAP to Fennoscandian conditions.

Newfound knowledge since RD&D 2010

Extensive investigations in the field area on Greenland have been carried out since RD&D programme 2010. With the aid of core drilling and high-resolution temperature profiling, knowledge has been obtained on the depth of the permafrost in the area, which turns out to have a thickness of 300–350 metres in the areas near the ice sheet margin. The results from temperature monitoring in the boreholes show that there are active taliks in front of the ice in the investigation area, and geothermal modelling indicates that so-called through taliks (taliks that extend through the entire permafrost cover, in this case 300 metres) can form in crystalline bedrock in less than 500 years.

In 2011 a hole was cored immediately adjacent to the ice sheet margin down to a depth of 640 metres. The borehole goes through 350 metres of permafrost, and the end of the borehole is located in the bedrock beneath the ice sheet margin. The borehole is instrumented with a system that permits water sampling at a depth interval of 550–580 metres and continuous monitoring of pressure, temperature and electrical conductivity. It is also possible to log the temperature along the borehole and thereby obtain data on permafrost depth as well as information on water flows in the bedrock. Water sampling was carried out in three sections in September 2011. As expected, however, these water samples still contained high concentrations of flushing water from the drilling. The preliminary results suggest that the water at this depth has a composition equivalent to that of glacial meltwater.

Investigations of fracture-filling minerals from drill cores to study the redox front in this area show that there is a clear change in redox potential where the transition from oxidizing to reducing conditions was indicated at a depth of between 40 and 60 metres. These results agree well with results from investigations of the redox front in the rock in Laxemar (Drake et al. 2009) and on Äspö (Landström et al. 2001).

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. A large part of the basal topography of the ice in the field area has been examined with ground-penetrating radar. Radar surveys also give an indication of where warm-based versus cold-based conditions prevail, which is an indirect indication of where groundwater can form.

Furthermore, both active and passive seismic surveys have been conducted to study the composition and properties of the subglacial material and how water is transported from the top surface of the ice sheet, through the ice and down to its base. By combining radar data with the results from the surface investigations on the ice sheet, and with information from the seismic studies, indirect information is obtained on how the subglacial hydrological system works in the area. It also gives an indication of what material the subglacial interface consists of and how thick it is.

Some 30-odd holes have been drilled by means of hot water drilling through the ice sheet in a transect that extends from the ice sheet margin in about 40 kilometres towards the middle of the ice sheet. The ice holes have been instrumented with pressure sensors, strain gauges and temperature gauges. Different types of hydrological tests have been performed in the boreholes, such as injection tests and pumping tests, to study the basal hydrological system. Samples have also been taken of subglacial water in a number of boreholes. Preliminary results show that silicate weathering is a dominant process beneath the ice sheet in the investigation area. Analyses of the chemical composition of subglacial water show that the subglacial drainage system is efficient and that it can transport large quantities of subglacial water quickly.

The ice holes have been drilled in areas with different types of basal conditions, and the ice thickness where the holes have been drilled has varied from 100 to 850 metres. Holes have been drilled near the ice sheet margin to study basal conditions in the marginal zone. A number of holes have been drilled in deep valleys where drainage is expected to be very good, and a number have been drilled on high plateaus where drainage is not expected to be as good as in the valleys. All holes that have reached the base of the ice sheet have exhibited warm-based conditions. Furthermore, field investigations have shown that the ice sheet has not been cold-based at the margin. The basal pressures from the boreholes show that an efficient subglacial drainage system with meltwater is active from the ice sheet margin and at least 30 kilometres in towards the middle of the ice sheet. Preliminary results further indicate that these tunnels and channels are pressurized in the wintertime as well and are thus not at atmospheric pressure, as assumed in previous models. The preliminary results further indicate that the horizontal pressure gradients are small except at the ice sheet margin and that the basal pressure over time lies close to the ice overburden pressure in this area.

Ice modelling with data from these ice boreholes and field investigations show that it is not only geothermal flow that is important in the formation of basal meltwater, but that frictional heat from basal sliding and advective heat caused by ice movement are also important processes.

Preliminary results from the indirect and direct observations of the subglacial hydrological system show that the appearance and properties of the system exhibit a clear spatial variation. Work is now being pursued within the project to present a conceptual model of the basal hydrological system, based on these new and unique observations from an existing inland ice sheet.

It further remains to analyze data to find out how large a portion of the basal layer is covered with water, i.e. how large a portion of the basal layer constitutes a potential infiltration area.

Project workshops have been held in Turku, 2011, and in Stockholm, 2012. The focus was then on discussing ice sheet modelling, hydrogeological modelling, the results of the field investigation programme, and the integrated conceptual understanding of the system that has been studied within the GAP. Results from the GAP and related studies have been published in reports, presented at a number of conferences and published as articles in scientific journals (Aaltonen et al. 2010, Bartholomew et al. 2010, Booth et al. 2012, Brinkerhoff et al. 2011, Engström et al. 2012, Follin et al. 2011, Harper et al. 2011, Jaquet et al. 2010, Johansson et al. 2013, Pöllönen et al. 2012, SKB 2010n, van As et al. 2012).

Programme

Reconnaissance observations were conducted in the field area around Kangerlussuaq in 2005, after which the GAP was initiated in 2008 with initialfield investigations. The large-scale field investigations in the GAP were concluded in the autumn of 2012. The project is now in a reporting phase where the results of the 2008–2012 investigations are being evaluated and interpreted. Final reporting of the results from the GAP is expected to be finished at the end of 2014. Relevant results from the project will be included in ongoing and upcoming safety assessments.

Even though the large-scale field investigations in the GAP are finished, SKB is planning to maintain the monitoring systems in the boreholes, as well as the networks of weather and GPS stations, during the period 2013–2016. It will be easier to quantify the variability and uncertainty in the results with longer time series and more data.

20 Short-lived low- and intermediate-level waste

This chapter describes SKB’s natural science research on the properties of the waste in the final repository for short-lived low- and intermediate-level waste. The work is being conducted in part within the safety assessment in the SFR Extension Project, SR-PSU. Further research needs are identified in the safety assessment included in the application under the Nuclear Activities Act for an extended SFR.

The research SKB plans to carry out in order to gain a better understanding of how the performance of the engineered concrete barriers changes is described in Chapter 22 “Concrete barriers”. The research and development being pursued by the nuclear power plants regarding waste to be disposed of in SFR is presented in Part II “Low- and intermediate-level waste”.

Conclusions in RD&D 2010 and its review

RD&D Programme 2010 initially described how SKB’s natural science research on the properties of waste in the final repository for short-lived low- and intermediate-level waste is conducted and how it employs a process-based methodology.

Newfound knowledge since RD&D 2010

A report that will be published within SR-PSU that deals with processes related to the waste packages (the conditioned and unconditioned waste and its packaging) that have been identified as relevant to the long-term safety of SFR. The systematics of this process report is based on the systematics used previously by SKB in the safety assessments for spent nuclear fuel (SKB 2006d, e, f). The processes described in the report are based on the features, events and processes (FEPs) that have been considered to be of importance for SFR. This report is included as a supporting document for the application under the Nuclear Activities Act for the extension of SFR.

Programme

One of the tasks of a safety assessment is to identify and prioritize remaining questions. The conclusions in SR-PSU will have a great influence on the research programme during the coming RD&D period. The research needs that so far have been identified are described under the relevant variable and process.

20.1 Initial state in the waste

The initial state of SFR is defined as the state of the repository at closure. In conjunction with closure, the repository’s drainage pumping will cease and the repository will fill with water.

The properties of the waste at deposition are determined by how the waste producers condition the waste and by the properties of the waste containers. This is in turn regulated by the waste acceptance criteria and the type descriptions. The waste acceptance criteria represent requirements that must be met by the waste in order for it to be deposited in SFR. The properties of the waste at deposition are guaranteed by the verification described in Section 4.3.1 “Operation of the facility”.

The properties of the waste at closure are also dependent on the environment in SFR and its grouting. The operating and deposition procedures used at SFR, for example grouting procedures, thereby determine the properties of the waste at closure.

Conclusions in RD&D 2010 and its review

In its review of RD&D Programme 2010, SSM commented that it is the properties of the waste at the time of deposition that are of interest for the ability to control what subsequently happens with the waste in the final repository. For this reason, the waste must meet certain requirements in order to be deposited. SKB will need to define and present such requirements in the form of waste acceptance criteria.

Programme

SR-PSU assumes certain properties of the waste, which entails that equivalent requirements are made on the waste. A review of the waste acceptance criteria is also done to ensure that they agree with the assumed properties.

20.1.1 Variables

The initial state is the starting point for an assessment of long-term safety and is described by the initial values of a number of variables, see Table 20-1. The variables characterize the waste packages in a suitable manner for the safety assessment. The description applies not only to the waste itself, but also the waste containers and the cavities (voids) that exist in and between the waste packages.

20.1.2 Geometry

SFR contains waste and waste containers of different types with different geometries, which are specified in the type descriptions. Cavities exist between the waste packages as well as between the barriers and the waste packages. The cavities are in some cases backfilled. Some types of waste packages contain waste in different forms, for example scrap or different types of conditioned ion exchange resins. If a waste package is not completly filled with waste, it may also contain filler material and cavities.

In the case of certain types of waste, the proportion of cavities is a waste acceptance criterion intended to ensure that swelling waste does not exert too much pressure on the barriers.

In the case of corroding material, the total surface area is an important parameter, along with the ratio between surface area and volume.

For porous materials, geometry parameters include porosity and pore geometry. The porosity of concrete containers is determined initially by quantity of aggregate, cement type and water/cement ratio. The initial porosity of cement-conditioned waste depends on what recipe has been used.

In the case of certain waste containers and waste types, geometry parameters also include fracture distribution and fracture geometry. The quantity of fractures and the geometry of the fractures that exist initially in concrete containers and cement-conditioned waste are affected when the volume changes during hardening as well as when the cement dries out and is re-wetted.

Table 20-1. Variables for description of the initial state.

Variable	Definition	Section
Geometry	Volumes and dimensions for waste containers, waste, voids in the waste and voids between the waste and the waste containers.	20.1.2
	Porosity and pore geometry of waste and waste containers.	
	Fracture distribution and fracture geometry in the waste and waste containers.	
Radiation intensity	Intensity of alpha, beta, gamma and neutron radiation.	20.1.3
Temperature	Temperature in the waste and waste containers.	20.1.4
Hydrovariables	Water saturation, water pressure and water flows in waste containers, waste, voids, pores and fractures.	20.1.5
Mechanical stresses	Mechanical stresses in waste and waste containers.	20.1.6
Total radionuclide inventory	Total occurrence of radionuclides in different parts of the waste. Type, quantities, chemical and physical form.	20.1.7
Material composition	Quantity, composition, chemical form and surface-area-to-volume ratio of material in waste and waste containers.	20.1.8
Water composition	Composition of water including radionuclides.	20.1.9
	Eh, pH, ionic strength, dissolved species, dissolved gases and concentration of colloids and particles in waste containers, waste, voids, pores and fractures.	
Gas variables	Quantities, composition, volume, pressures and flows.	20.1.10

Conclusions in RD&D 2010 and its review

SKB stated in RD&D Programme 2010 that the impact of cavities on hydrovariables and long-term safety would be studied and that backfilling would be studied with the aid of improved radionuclide transport models.

SSM expressed doubts regarding the claim that the additional expansion volume in the silo would be sufficient or available to take up all volume expansion in the waste. An update of the current state of knowledge for this process would therefore be desirable.

Newfound knowledge since RD&D 2010

Models have been developed that describe the existing SFR and the extension with a refined geometric representation of repository parts, barriers and waste, see Section 20.1.5 "Hydrovariables".

Calculations have been carried out where swelling of bitumenized waste has been compared with available void volume, see Section 22.2.7 "Pressure from swelling waste".

Programme

No questions have been identified today that require additional research. Knowledge gaps regarding the initial state of the variable are identified in the safety assessment included in the application under the Nuclear Activities Act for an extension of SFR.

20.1.3 Radiation intensity

Radiation intensity is dependent on the radionuclide inventory and the density and geometry of waste and waste containers. The waste that is deposited in SFR has low radiation intensity initially and it declines thereafter. Radiation intensity has been deemed to have the greatest impact on the bitumen-solidified waste with the highest activity. The impact of radiation intensity has been deemed to be negligible for this waste as well (Pettersson and Elert 2001).

Conclusions in RD&D 2010 and its review

SKB thought that additional calculations may be needed as a result of changes in the radionuclide inventory, for example if it is decided to dispose of reactor pressure vessels in SFR. If it turns out that waste with high radiation intensity from dismantling and demolition is to be disposed of, additional calculations may be needed.

Newfound knowledge since RD&D 2010

The impact on the evolution of the repository of the radiation intensity from the waste from dismantling and demolition that SKB plans to dispose of in SFR has been deemed negligible. SKB is following developments within the discipline.

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. When the repository is water-saturated the corrosion rate will increase, but this will only lead to a negligible increase in temperature. The radiation intensity of the waste deposited in SFR is so low that temperature changes due to radioactive decay are negligible. SKB will continue following developments within the discipline.

20.1.5 Hydrovariables

The hydrovariables for the waste include water saturation, water pressure and water flows in waste containers, waste, voids, pores and fractures.

The boundary conditions that control the water flow through the waste are given by the surrounding pressure field in the model. This field is dependent on the repository's layout and location in the rock, the hydrogeological properties of the rock, the properties of the engineered barriers and the external boundary conditions that apply for the flow model. The local flow pattern is also affected by the size, shape and material properties of the waste and its containers.

There is no flow of water through the waste during operation since the repository is drained by pumping. When the drainage pumping ceases the repository will become water-filled. Water saturation in the waste after closure is low. Some water may, however, be present initially due to seepage via 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 corrosion rate of the reinforcement.

Newfound knowledge since RD&D 2010

Method and model development for calculations of water flows on repository scale has been carried out in the software Comsol Multiphysics, as a part of SR-PSU. Models describe the existing SFR and its extension with a refined geometric representation of repository parts, barriers and waste compared with SAR-08, see Figure 20-1. Calculations have quantified water flows in the repository. The size and direction of the flows through the waste influence the transport of radionuclides out of the repository.

Research needs pertaining to the hydraulic properties of the waste may be identified in SR-PSU.

20.1.6 Mechanical stresses

Mechanical stresses may exist initially in the waste due to the weight itself and expansion/contraction of the waste and the material in the waste containers. The waste and its containers may also be subjected to external mechanical forces, which serve as a basis for the acceptance criteria for the mechanical properties of certain waste packages. The waste containers are stacked in such a manner that they do not collapse under their own weight and/or due to overcasting. The waste containers and their embedding grout may have a supporting function for surrounding barriers during the water saturation phase. In the case of waste packages embedded in grout, the ability to absorb mechanical stresses is an acceptance criterion for the waste.

SKB will continue following developments within the discipline.

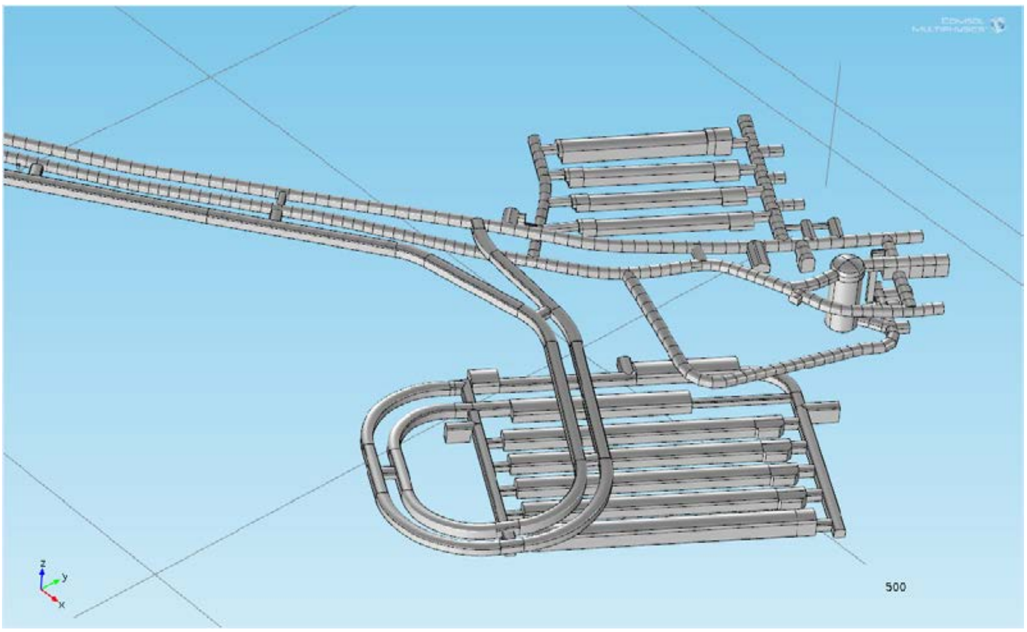


***Figure 20-1.** Geometric description of the existing SFR and the extension, which is included in a flow model for the repository's near-field.*

20.1.7 Total radionuclide inventory

At closure, SFR will contain radionuclides from operational waste, waste from dismantling and demolition of the NPPs and smaller quantities of waste from industry and research. SKB calculates and keeps track of changes in the inventory to ensure that current radiation protection criteria are not exceeded. The radionuclide content is determined by measurements and calculations of annual generated activity at the NPPs and by correlation between difficult-to-measure nuclides and the key nuclides cobalt-60, cesium-137 and plutonium-239/240.

Conclusions in RD&D 2010 and its review

In RD&D Programme 2010, SKB described the work of determining the radionuclide inventory in SFR and intended to carry out a general inventory of the uncertainties in measurement and calculation methods. The work of reducing uncertainties in the determination of the radionuclide inventory is something which SSM encouraged, and in particular the development of alternatives to the methods that are based on correlations between difficult- and easy-to-measure radionuclides. SKB also intended to improve methods for activity determination of the radionuclides where the total activity added to the waste is determined on the basis of measurements of reactor water and storage pools. Furthermore, SKB wanted to study how to best take into account decay of cobalt-60 for waste that is interim-stored in tanks for an extended period prior to gamma measurement.

SSM took a positive view of the work being done by SKB together with the NPPs with regard to the presence and properties of carbon-14 in the waste. SKB should make sure that the studies that are done and the conclusions that are drawn are in agreement with international opinion in the area.

SSM further considered that a better estimate of the uncertainty in the radionuclide determination is needed for the coming review of the application under the Nuclear Activities Act for the extension of SFR.

Newfound knowledge since RD&D 2010

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 radionuclides such as C-14 (SKBdoc 1339709). The inventory of radioactivity and materials expected to be deposited in SFR is being calculated in SR-PSU, both for operational waste and for waste from dismantling and demolition. Within the framework of this work, SKB has re-examined and reconsidered how uncertainties are estimated and applied.

The calculation method for determining the quantity of transuranics and strontium-90 delivered to the waste from the internal cooling circuit at the receiving station in Clab has been reviewed and modified to ensure greater accuracy.

A survey of available information on interim-stored ion exchange resins is currently taking place.

SKB is studying and compiling each individual NPP's radionuclide inventory that will be deposited in SFR as a consequence of the dismantling and demolition of the reactors. In the decommissioning studies, unit-specific calculations have been done of the radionuclide composition in different water systems and on system surfaces.

The measurements done by KTH of emissions of carbon-14 activity during bitumen conditioning in FKA's facilities show that about 40 percent of the hydrogen carbonate ($H^{14}CO_3^-$) content and 90 percent of the formic acid ($H^{14}CO_2H$) is emitted during drying. The formic acid is broken down catalytically to carbon monoxide (^{14}CO) by corrosion products in the ion exchanger at 140°C. If drying takes place at 180°C, nearly 80 percent of the hydrogen carbonate is emitted. Some organic carbon-14 is not emitted during drying or stripping with sulphuric acid. It probably consists of radiolytically formed $R^{14}CO_2H$ groups where R is the ion exchanger's polymer skeleton.

Programme

Further measurements in addition to those reported in Magnusson et al. (2007) are needed to obtain a more reliable body of statistics for deposited carbon-14 activity. As part of the attempt to obtain a better body of statistics, a project is under way where condensate filter demineralizer from all BWRs and operational ion exchange resins from all PWRs are being analyzed. This will provide additional information on deposited carbon-14 activity in SFR, in both organic and inorganic form.

The carbon-14 activity that is present in induced materials, such as RPVs, differs in composition from that sorbed on ion exchange resins. Measurements and experiments may be necessary to determine this composition.

SKB has requested that Studsvik Nuclear AB should inventory the presence of carbon-14 at its facilities. SKB requests information on how much carbon-14 activity has been received, and in what form, at the company's facilities since 1991 and how it has been treated. The purpose of these investigations is to determine how much carbon-14 activity from non-nuclear facilities has been deposited in SFR or incinerated or interim-stored at Studsvik Nuclear AB's facilities.

SKB is participating in a joint international project on carbon-14 aimed at gaining a better understanding of carbon-14 in geological repositories. Applications have been made to the EU for research grants under the auspices of the EU's 7th Framework Programme for Research and Technological Development (FP7).

SKB is investigating the possibilities of further experimental measurement and determination of the presence of the difficult-to-measure radionuclides, mainly chlorine-36 and iodine-129, which can be determined by means of mass spectrometry.

The radionuclide composition that has been determined in the decommissioning studies in different water systems and on system surfaces will be further analyzed. The purpose is to improve or replace the methods that are used today to calculate the inventory of difficult-to-measure radionuclides by correlation with cesium-137.

20.1.8 Material composition

The largest volume of raw waste generated at the nuclear facilities consists of combustible solid waste. The volume remaining for deposition in SFR is greatly reduced by means of incineration, clearance or use of the near-surface repositories that are available at the power plant sites. The raw waste consists mainly of cellulose (paper, cotton and wood) and plastics (e.g. polystyrene, polyvinyl chloride, polyethylene and polypropylene).

Ion exchange resins, mechanical filter aids, evaporator concentrates and precipitation sludge are also deposited in SFR. Much of this waste is conditioned, in other words solidified in cement or bitumen.

A large portion of the waste volume in SFR consists of metals, mainly carbon steel and stainless steel. Scrap metal arises mainly during maintenance outages when equipment is discarded, modified or renovated. Plans for the extension of SFR call for whole RPVs from BWRs to be disposed of in SFR.

Other materials in the waste include mineral wool (used for insulation), concrete and brick. Various additional materials are also included in smaller quantities.

The ion exchange resins also contain residues (organic complexing agents) from the cleaning agents used in the decontamination of radioactive surfaces at the nuclear facilities.

Conclusions in RD&D 2010 and its review

In RD&D Programme 2010, SKB described the ongoing programme to quantify the amount of non-radioactive ions sorbed on ion exchange resins and thereby expected to be deposited in SFR. In the programme, SKB also described the work being done to reduce the quantity of organic complexing agents deposited in SFR.

In its review, SSM pointed out that it is the properties of the waste at the time of deposition that are of interest for being able to control what ultimately happens with the waste in the final repository. SSM judged that there may be organic material in the waste that needs to be reconsidered, for example certain filter aids.

Newfound knowledge since RD&D 2010

Estimates have been made of the quantity of waste from dismantling and demolition that may be deposited in SFR. This waste consists primarily of metal and concrete, as is evident from the reported material inventory determined in SR-PSU. This inventory serves as a basis for the further analyses that are needed to guarantee long-term safety. The inventory is based on the actual material quantities deposited as well as forecast quantities and estimates made in the decommissioning studies. The decommissioning studies are included as a part of the application under the Nuclear Activities Act for the extension of SFR.

SKB has inventoried the quantity of non-radioactive ions that could theoretically be deposited in SFR via ion exchange resins. The estimates are based on known compositions of the materials used at the nuclear power plants. Based on a number of known and assumed parameters, a corrosion rate can be calculated on system surfaces for the different reactor systems. It is pessimistically assumed that 100 percent of the corrosion products that are formed and dissolved in the water for the studied system are sorbed on the ion exchange resins. The study shows that large quantities of non-radioactive corrosion products would then be deposited in SFR.

The efforts described in RD&D Programme 2010 to develop a new cleaning agent have been concluded. In connection with this, SKB has tightened its guidelines for the use of complexing agents in activities that could generate waste that will be deposited in SFR.

Since the most recent safety analysis report, SAR-08 (SKB 2008b), SKB has inventoried and estimated the quantity of organic complexing agents, including degradation products of cellulose, that will be deposited in SFR. This inventory is included as input data in SR-PSU. Since the last account (Fanger et al. 2001), new information and knowledge regarding the degradation of cellulose has come to light (Glaus and Van Loon 2008). Similarly, the cleaning products used for decontamination have been replaced and/or reformulated with respect to organic complexing agents. The filter aid UP2 (a polyacrylonitrile-based polymer) has been evaluated with respect to how its degradation products affect the sorption of radionuclides, see Section 20.2.13 “Sorption”.

Programme

It is important for the assessment of long-term safety that the quantities and occurrence of materials deposited in SFR are known to such a degree that the impact of these materials can be evaluated. Further details regarding the composition of the materials in the waste may be required if this proves necessary in order to reduce the degree of pessimism in a safety assessment.

20.1.9 Water composition

The groundwater that enters the repository is affected by the materials in the repository. The composition of the water in the waste and the repository includes radionuclides and is dependent on the degree of degradation of the waste as well as the solubility of the radionuclides. Eh, pH, ionic strength, dissolved species and gases, the concentration of colloids and particles, as well as water content (including radionuclides and dissolved gases) in waste containers, waste, voids, pores and fractures, are relevant input parameters for the water composition. The composition of the water will also be affected by an initially very low water flow through the repository after closure (see further Section 22.2.4 “Water uptake and transport under unsaturated conditions” and 22.2.5 “Water transport under saturated conditions”). SFR 1 is covered today by the Baltic Sea, and the groundwater flowing into the repository is characterized as a saline groundwater. The reference composition of the saline groundwater that is assumed to infiltrate and fill up the repository will be included as input data for the application under the Nuclear Activities Act for the extension of SFR and will then be updated with regard to new site data (Nilsson et al. 2011).

Initially, the pore water in the waste will be affected by the concrete pore water with a pH higher than 13 (whose chemical composition is dependent on the mineral composition of the concrete and how the concrete has been exposed to groundwater). Pore water composition has an influence on the chemical processes in the repository, such as the tendency of the reinforcement to corrode. The water in the repository will become anoxic soon after closure (Duro et al. 2012a).

Newfound knowledge since RD&D 2010

Pore water composition is affected by chemical processes in the waste as well as by the composition of the penetrating groundwater.

The new thermodynamic modelling that has been done for different waste containers (included in the supporting material for the applications for the extension of SFR) has contributed to greater knowledge of potentially altered redox conditions at repository depth. Sensitivity analyses have provided insight on the relevance of improving transport parameters for different waste types as well. It is very important that different waste types are well-characterized to permit the calculation of chemical parameters and provide input to radionuclide transport calculations in the long-term safety assessment.

Programme

Further development of near-field models will be done and these will if possible be linked to chemical reactions in the waste.

20.1.10 Gas variables

The gas variables that will affect the waste are the quantity of generated gas and its composition. How large a volume the gases occupy is dependent on their partial pressure and temperature, but also their solubility, degree of saturation in water as well as the magnitude and direction of the gas flows. See also Section 20.2.9 "Water transport under saturated conditions".

SKB does not at present have a programme for research on gas composition and gas flows in the initial state of the short-lived waste. For research programmes concerning gas formation processes that can affect long-term safety, see Section 20.2.18 "Microbial processes" and 20.2.19 "Metal corrosion processes".

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 when anaerobic conditions prevail.

The need for research on these different processes will be identified on the basis of SR-PSU. Currently identified research needs are described under each individual process.

20.2.1 Overview of processes

The processes that affect the conditions in the waste and the waste containers can be divided into four main categories: radiation-related processes, thermal processes, hydraulic processes, mechanical processes, chemical processes and radionuclide transport. A number of different processes may occur under each main category and interact with each other or with other processes.

Radiation-related processes

Radiation intensity is dependent on the inventory of radionuclides and the geometry of the waste.

The intensity of the radiation occurring 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. The following radiation-related processes are dealt with in this chapter:

- Radioactive decay, Section 20.2.2.
- Radiation attenuation and heat generation, Section 20.2.3.
- Radiation-induced degradation of organic matter, Section 20.2.4.
- Water radiolysis, Section 20.2.5.

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. 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, since freezing alters the integrity of the concrete. The following thermal processes are dealt with in this chapter:

- Heat transport, Section 20.2.6.
- Phase change/freezing, Section 20.2.7.

Hydraulic processes

The water flow through the waste is determined by the permeability of the different structural parts and components in the repository, as well as by the pressure gradient. The flows through the different repository parts are so low that erosion of the different 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. High 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 only of importance for the degradation of bitumen. The magnitude of the water flow in the repository and in the waste is determined to a high degree by the surrounding groundwater flow. The following hydraulic processes are dealt with in this chapter:

- Water uptake and transport under unsaturated conditions, Section 20.2.8.
- Water transport under saturated conditions, Section 20.2.9.

Mechanical processes

The waste and the waste containers in the different repository parts will be subjected to external mechanical impact and internal mechanical stresses caused by the weight itself and volume changes. This affects the distribution of stresses in the waste and the waste containers, which can in turn lead to fracturing. If gas is generated and cannot escape, this can lead to a build-up of pressure and stress, causing fracturing. Finally, the waste will be affected by any deformations in the rock (falling blocks, rock movements, earthquakes etc). Fall of ground is dealt with in Section 22.2.10.

The following mechanical process is dealt with in this chapter:

- Fracturing, see Section 20.2.10.

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 the 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 (Pedersen 2001).

The following chemical processes are dealt with in this chapter:

- Advective transport of solutes, Section 20.2.11.
- Diffusive transport of solutes, Section 20.2.12.
- Sorption, Section 20.2.13.
- Colloid formation and colloid transport, Section 20.2.14.
- Dissolution, precipitation and recrystallization, Section 20.2.15.
- Chemical degradation of organic compounds, Section 20.2.16.
- Water uptake/swelling, Section 20.2.17.
- Microbial processes, Section 20.2.18.
- Metal corrosion, Section 20.2.19.
- Gas formation and gas transport, Section 20.2.20.

Radionuclide transport

Several of the processes presented in this chapter affect the transport of radionuclides within and out of 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 packages or inside the shafts in the silo or the cells 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 mechanism, considering only the radionuclide concentration. On the other hand, co-precipitation with non-radioactive substances could be an important retardation process for certain nuclides. Chemical, mechanical and microbial degradation of bitumen affects radionuclide release from the bitumen-stabilized waste. Besides these, there are several other processes that affect radionuclide transport and that are handled by codes used for safety assessment calculations.

The following radionuclide transport processes are dealt with in this chapter:

- Speciation of radionuclides, Section 20.2.21.
- Radionuclide transport in the aqueous phase, Section 20.2.22.
- Radionuclide transport in the gas phase, Section 20.2.23.

20.2.2 Radioactive decay

The radionuclides in SFR will eventually be transformed into stable substances by radioactive decay. This process gives rise to alpha, beta, gamma and neutron radiation as well as new nuclides. These new nuclides may also be radioactive and continue to decay until a stable nuclide has been formed. Almost all the energy formed in the process is converted to heat, see Section 20.2.3 "Radiation attenuation and heat generation".

The process is fundamentally important, especially since it describes how radiotoxicity changes with time.

Decay products may have chemical properties that distinguish them from the parent nuclide, which can affect their release and transport properties.

Newfound knowledge since RD&D 2010

Within SR-PSU, the half-lives of the radionuclides that have been considered relevant to the safety assessment have been gone through and updated, where considered necessary. Half-lives are generally well-known, with the exception of selenium-79, whose half-life has been discussed in the literature. The most recently published value is 327,000 ± 8,000 years (Jörg et al. 2010). Previous estimates vary widely, for example (He et al. 2002). The most recently published value of the half-life of silver-108m is 435.7 years (Schrader 2010). This differs only slightly from a safety assessment perspective from the previously used value of 418 years (SKB 2008a).

SKB has no programme for studying half-lives, but keeps track of relevant research and updates the database when necessary.

20.2.3 Radiation attenuation and heat generation

The radiation from radioactive decay interacts with the waste form and/or other materials in the repository. The released energy is transferred to the surrounding materials. Most of the energy will be converted into thermal energy. The heat generated as a result of radiation attenuation is nevertheless considered to be negligible in SFR.

The scientific literature on the subject has been reviewed by experts since RD&D Programme 2010 and is presented in the process report that describes the waste form and the waste packages. This report constitutes supporting material for SR-PSU.

No further research, development or demonstration is considered to be needed in this field today. New developments are being monitored and will be acted on when appropriate.

20.2.4 Radiation-induced degradation of organic matter

Organic matter present in the waste can be affected by the ionizing radiation that is emitted by the waste. This can cause molecules to be excited or ionized either directly or indirectly via radicals produced by water radiolysis, see Section 20.2.5 "Water radiolysis". This leads to the production of a large quantity of organic radicals, and ultimately to stable molecular products.

The scope of this process is a function of the radiation intensity, which declines as the radioactivity decays. In the short time perspective, the process will be dominated by radiation from cobalt-60, followed by cesium-137. In the long perspective (more than 1,000 years), different radionuclides will take on importance for the process in different parts of SFR, depending on their radionuclide content.

Sulphate and oxalate might form in connection with the radiolytic degradation of ion exchange resins (Van Loon and Hummel 1999). Sulphate is expected to form ettringite (a hydrated calcium-aluminium sulphate) on reacting with cement and concrete. Oxalate affects the sorption of metals under neutral to acidic 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 pH levels has been shown to primarily generate mono- and dicarboxylates and carbonates (Van Loon and Kopajtic 1991). Of these, oxalate could be a conceivable complexing agent, but according to the above line of reasoning, oxalate is expected to have a negligible impact on the sorption of radionuclides.

The scientific literature on the subject has been reviewed by experts since RD&D Programme 2010 and is presented in the process report that describes the waste form and the waste packages. This report is included as supporting material for SR-PSU.

No further research, development or demonstration is considered to be needed in this field today. New developments are being monitored and will be acted on when appropriate.

20.2.5 Water radiolysis

The water that enters SFR and the waste package will be affected by the ionizing radiation. This contributes to the excitation and ionization of water molecules when the chemical bonds of the water are broken. The radiation intensity in SFR is so low that radiolysis of water is not judged to have any significant impact on long-term safety. The work being done by SKB to gain better knowledge and process understanding concerning water radiolysis is mainly being done within the framework of the research on the spent nuclear fuel, see Section 23.2.4 "Water radiolysis".

20.2.6 Heat transport

Heat transport between the different components 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 essentially be determined by heat exchange with the surrounding rock and groundwater, while the influence of the waste on the temperature can be regarded as negligible.

SKB sees no need today for a research programme on heat transport in the short-lived low- and intermediate-level waste.

20.2.7 Phase change/freezing

Waste and waste containers, as well as internal (stabilizing) and external (embedment) grout, can freeze if the permafrost reaches repository depth. The extent of this freezing is directly dependent on how long the permafrost period lasts and on how deep the repository is located.

The effect of freezing is dependent on the material type in question. For example, metals in the waste are not directly affected by freezing, while grout and concrete waste containers could burst due to freezing in the same manner as the concrete barriers. Metal containers could, however, burst due to freezing if the contents expand to a sufficient degree.

Conclusions in RD&D 2010 and its review

This process was described only briefly in RD&D Programme 2010 with a reference to the research on freezing of the engineered concrete barriers. SSM did not express any viewpoints on this in its review.

Newfound knowledge since RD&D 2010

SKB did not conduct any research on phase change/freezing of the short-lived low- and intermediate-level waste during the preceding period. However, a couple of studies regarding freezing of concrete barriers have been conducted and are described in Section 22.2.3 "Phase change/freezing". Since grout can be expected to be affected in the same way as concrete in the event of a freeze, these studies are considered to be relevant in this area as well.

Programme

SKB sees no need today for a separate research programme concerning phase change/freezing of the short-lived low- and intermediate-level waste and the internal and external grout, but intends to coordinate this work with the research on freezing of the concrete barriers, see Section 22.2.3 "Phase change/freezing".

20.2.8 Water uptake and transport under unsaturated conditions

Prior to closure, the repository is kept dry by drainage pumping. After closure, the groundwater will saturate fractures and pores in the rock near the repository. The groundwater then continues to fill voids in the rock vaults and gradually saturate other engineering materials and waste packages. Water uptake and water transport under unsaturated conditions only occurs during a relatively brief initial period. According to calculations, it takes 25 years to saturate the least permeable repository part in SFR (Holmén and Stigsson 2001). The saturation of the repository constitutes the starting point for transport processes that can release radionuclides.

Conclusions in RD&D 2010 and its review

RD&D Programme 2010 describes water uptake in bitumen-stabilized waste. Water uptake is a prerequisite for radionuclides to be released from the waste matrix. In its review of the RD&D programme, SSM commented that the longest times for release of radionuclides, 1,000 years or more, can only apply to e.g. waste stabilized with an ideal process that creates a completely homogeneous product.

SKB's models of radionuclide transport do not assume any retardation effect related to the release of radionuclides from bitumen matrices.

No further research, development or demonstration is considered to be needed in this field today. New developments are being monitored and will be acted on when appropriate.

20.2.9 Water transport under saturated conditions

When the repository has become saturated with water, the hydraulic gradient will drive the water flow through the waste. The size of the flow is determined by the gradient and the conductivity of the waste packages and the backfill material. The presence of cavities or fractures in the waste also have an influence. Water transport through waste packages, conditioned and unconditioned waste, as well as waste matrices, is regarded as relatively rapid. What limits the flow through the waste is primarily surrounding barriers of concrete and bentonite. Viewed over a long time, the flow through the waste will change. Changes in the groundwater flow due to land uplift are one reason. Furthermore, a gradual degradation of barriers and waste is expected, which affects the permeability of the materials to water.

Programme

Detailed studies of the water flow through individual waste packages are not currently planned, since the flow through barriers is considered to be a limiting factor. A programme for modelling of water flows linked to concrete degradation is described in Section 22.2.5 "Water transport under saturated conditions".

20.2.10 Fracturing

A number of processes in SFR can lead to the formation of fractures in the waste, the waste packages, or the grout. The extent to which a material can fracture is determined by its properties. A brittle material such as concrete has a greater tendency to fracture under an imposed load than a tougher material such as steel.

Of the material types occurring in SFR, internal and external grout (in and around containers) and concrete moulds and tanks are judged to be the ones most subject to fracturing, while steel moulds, ISO containers and steel drums are expected to be considerably less fracture-prone.

The effects of fracturing in the waste, its containers or grout are difficult to foresee. The process as such is, however, handled pessimistically, and no safety function is assumed for the waste or its containers. For the concrete tanks containing dewatered ion exchange resins, however, where the integrity of the tank is the only barrier, the effect of fracturing can be more substantial.

The most important processes that can lead to fracturing in the waste, waste containers and grout are:

- Dissolution, precipitation and recrystallization, Section 20.2.15.
- Gas formation caused by chemical degradation of organic compounds, Section 20.2.16.
- Water uptake/swelling, Section 20.2.17.
- Metal corrosion, Section 20.2.19.
- Gas formation caused by metal corrosion, Section 20.2.19.
- Phase change/freezing, Section 22.2.3.

Of these processes, water uptake/swelling, metal corrosion and phase change/freezing are in particular judged to be able to cause extensive fracturing in waste and waste packages. It is also possible that dissolution, precipitation and recrystallization could lead to fracturing in concrete containers containing sulphate ion exchange resin due to reactions between the components of the concrete and the degradation products of the sulphate ion exchange resin, see also Section 22.2.15 “Dissolution, precipitation and recrystallization”. However, it is judged that the gas formed by the other processes can be released from the waste packages at a sufficiently low pressure that the integrity of the container is not jeopardized.

All of the above processes are described in greater detail in the relevant sections.

Conclusions in RD&D 2010 and its review

RD&D Programme 2010 presented the research on fracturing in the short-lived low- and intermediate-level waste, its containers and internal and external grout in general terms with reference to the processes considered to have a potential to cause fracturing. In its review, SSM did not express any viewpoints on the programme, but wondered why fracturing due to sulphate attack on concrete was not taken up under this heading.

Programme

The programme on processes that can cause fracturing in short-lived low- and intermediate-level waste, waste containers and internal and external grout is presented in the relevant section, see above. In addition to these processes, dissolution, precipitation and recrystallization of concrete barriers is dealt with in Section 22.2.15 “Dissolution, precipitation and recrystallization”.

20.2.11 Advective transport of solutes

Advective transport of solutes in cement-conditioned waste is dealt with in the same way as in concrete barriers, see Section 22.2.11 “Advection and mixing”. No transport limitation is assumed for other waste forms.

20.2.12 Diffusive transport of solutes

Diffusive transport of solutes in cement-conditioned waste is dealt with in the same way as in concrete barriers, see Section 22.2.12 “Diffusion”. No transport limitation is assumed for other waste forms.

20.2.13 Sorption

Sorption of radionuclides is one of the most important retarding mechanisms 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.

Sorption of radionuclides is expected in all repository parts and in the surrounding rock. But the greatest potential for sorption exists in the repository parts that contain large amounts of concrete, such as in walls, embedment grout or waste packages. The quantity of concrete present in SFR has a large specific surface area of different CSH (calcium silicate hydrate) phases.

Sorption of radionuclides can also occur on other materials than cement/concrete, for example bitumen, steel, paper, plastics, etc. Sorption on these materials is not taken into account in SKB’s safety analysis reports for SFR. However, sorption on bentonite and sand/gravel is taken into account, see Chapter 25 “Buffer and backfill”.

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.

The importance of the safety function “sorption in concrete barriers” is strongly linked to the chemical properties of individual radionuclides. Some radionuclides will not be sorbed under any conditions, in which case the safety function does not exist at all. But most radionuclides can be assumed to sorb under the conditions that exist in SFR. For many radionuclides, sorption is linked to the chemical environment in the repository. The different radionuclides will, however, be affected in different ways by the chemical environment.

The chemical environment in the cement pores, with a 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 alpha-isosaccharinic acid, sorb to cement, possibly by complexation with calcium-rich solid phases (Bradbury and Van Loon 1997, Van Loon and Glaus 1997, 1998). Some of the radionuclides are redox-sensitive, and for some, sorption is much weaker under oxidizing conditions, see Section 20.2.21 "Speciation of radionuclides".

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 in SFR, according to the process report being produced in support of SR-PSU, is isosaccharinic acid, see Section 20.2.16 "Chemical degradation of organic compounds".

Conclusions in RD&D 2010 and its review

RD&D Programme 2010 described how the uncertainties in different radionuclides' sorption coefficients (K_d) in concrete, bentonite and sand/gravel have been estimated and determined. In the data report being produced within SR-PSU, the uncertainties in the K_d values will be handled in a formal manner. The resultant values, which will be used in the safety assessment in SR-PSU, are based on a review of relevant literature data. In its review of RD&D Programme 2010, SSM pointed out that SKB should develop a systematic method and models to handle the influence of complexation on sorption of radionuclides in the final repository.

Newfound knowledge since RD&D 2010

A model has been constructed for assessing the influence of organic complexing agents on sorption of radionuclides on cementitious materials. The model takes into account speciation and solubility in the presence of organic ligands. The sorption processes that are assumed to take place on the cementitious materials have been taken into account and implemented via their equilibrium constants (K_{ads}). This is relevant in the assessment of whether the waste is suitable for deposition in SFR with reference to its content of organic complexing agents. The results of studies of how degradation products from the filter aid UP2 (polyacrylonitrile-based polymer) affect the sorption of cesium(I), cobalt(II) and europium(III) have been reported (Holgersson et al. 2011). This work, in combination with the data on how degradation products of UP2 influence the sorption of europium(III) presented by Duro et al. (2012b), indicate that degradation products from the filter aid UP2 do not have any significant negative effect on sorption.

New K_d values have been determined within SR-PSU. K_d values for the different degradation stages of concrete are presented in this material, which makes it possible to choose K_d values for different time periods.

Programme

SKB has started to study the sorption of radionuclides on cement. The purpose of the studies is to gain a better understanding of the sorption mechanisms that occur in SFR in the prevailing environment with a high pH, low Eh, and the presence of complexing agents. An important part of the work is to acquire knowledge on how the interaction between radionuclides and cement is affected by the presence of organic complexing agents and calcium ions.

Other studies of importance are to evaluate the sorption properties of aged cement. The chemical composition of aged cement can be modelled by means of thermodynamic calculations, for example with the program PHREEQC (Cronstrand 2007). Based on this knowledge it should be possible to arrive at new sorption coefficients for the different ageing stages of the cement. This can in turn lead to greater knowledge of the sorbing properties of chemically partially degraded barriers, and how these properties affect future radionuclide releases from the repository.

When certain radioactive cations are transported out through the concrete, either as hydroxide complexes or as organometallic complexes, they will be affected by a constantly changing chemical environment. This is because groundwater enters and alters the chemical composition of the concrete at different depths in the concrete structure. The altered environment affects the speciation and solubility of the different radionuclides. How it affects the retardation of radionuclides may be the subject of further modelling studies.

20.2.14 Colloid formation and colloid transport

The stability and transport of colloids is dependent on the chemical composition of the water, i.e. the ionic strength, redox potential and pH of the solution, but also on what kind of colloid is present in the solution as well as the properties of the colloids. Other parameters of interest may be quantities and types of microbes, other biomass, quantities and composition of gas, density and viscosity (if the water has a very high salinity).

Colloids are characterized according to their pH_{pzc} (the pH at which the colloid's surface charge is zero, called point of zero charge). High ionic strength in solution leads to the precipitation of colloids, since high ionic strength reduces the repulsive force of the electric double layer. The colloid aggregates or precipitates, i.e. sediments.

In the cement pore water with alkaline pH values, colloid formation can on the other hand be favourable, since colloids there may be negatively charged to a higher extent, i.e. result in negatively charged surfaces. At a high pH, the concentration of hydroxide ions $[OH^-]$ is high, which in this case can increase the stability of the colloids.

If there are organic substances in solution, the stability of the colloids can increase in the waste (bitumen). Bitumen colloids are relatively stable at high pH-values, and Liu et al. (2002) found that their stability was affected in the presence of calcium ions.

Organic compounds are present in all natural waters and can comprise a large portion of the colloidal pool in aquatic media. In a recent review by the Nuclear Decommissioning Authority (NDA) concerning colloids in the near-field of a cementitious repository, the authors concluded that there was no site-specific information on colloid generation and stability of organic materials in the near-field whose confidence can be increased by further studies (Swanton et al. 2010).

The conditions for colloid generation from the bentonite in the silo can change during a glacial cycle. Beneath an ice sheet or during a prolonged period of global warming, low-salinity water could penetrate down to repository depth. If it comes into contact with the bentonite buffer, there is a potential for montmorillonite colloids to be released for further transport, see Section 25.5.19 "Colloid release/erosion".

Conclusions in RD&D 2010 and its review

In its review of RD&D Programme 2010, SSM contended that SKB should study the long-term stability of the closure with respect to degradation of concrete and chemical erosion of bentonite by glacial meltwater.

Moreover, SSM said that a closed SFR constitutes a rather unique environment with a complex chemistry and that the presence of colloids cannot be ruled out for the reasons given by SKB. The concrete environment cited by SKB is probably very unevenly distributed in time and space within the final repository.

Newfound knowledge since RD&D 2010

SR-PSU deals with colloid formation and colloid stability and its importance for safety with regard to SFR. The driving force for reaction is very high at the interface between cement and bentonite, or cement and mineral surfaces, since a sharp pH gradient will initially range over pH values between 8 and 13. Here there is a potential for colloids to form. Stability criteria are specific for each type of colloid and cannot be generalized. The release and stability of cement/bentonite colloids in this specific hydrochemistry should be investigated.

Programme

As an initial test, the release of colloids from cement to water with a high pH will be investigated, followed by the stability of these colloids in an aquatic environment. Then the interaction between bentonite and cement and the release of colloids from this interface will be investigated, followed by an investigation of the stability of any particles formed by bentonite and cement. The tests with cement or cement/bentonite in contact with water will be done with varying pH and ionic strength.

Colloid concentration, particle size distribution and surface charge will be determined by means of different measurements in solution. The metal content of water will also be measured with ICP-AES (Inductively Coupled Plasma – Atomic Emission Spectroscopy). In parallel with leaching tests, classic stability tests will be performed where cement, cement and bentonite and complexing agents are slurried in different chemical environments, and the concentration in solution is measured as a function of time. Cells where plugs of material are installed and then equilibrated with water from two directions are used for leaching tests. After equilibration, water is circulated through the cells and leaching of colloids is measured.

At the Äspö HRL, tests are being conducted with a number of steel cylinders containing e.g. bitumen. The environment in the steel cylinders simulates the environment expected to exist in the different repository parts after closure with regard to concrete and oxygen-free conditions. During the coming RD&D period, a container with bitumen and cement is planned to be opened and a method for analysis will be developed.

20.2.15 Dissolution, precipitation and recrystallization

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. In the assessment of long-term safety for SFR, it is assumed 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. This suggests that the impact on surrounding concrete should be small. However, it cannot be ruled out that the dissolved salts (especially during the first 1,000 years) could form such high local concentrations that nearby concrete could be affected, leading to porosity changes and possibly also fracturing. SKB has previously investigated the consequences of deposited salts in SFR, mainly in the form of evaporator concentrates (Gaucher et al. 2005). These studies indicate that if concrete is exposed to high concentrations of evaporator concentrates, the concrete nearest the evaporator concentrates could possibly degrade, entailing leaching of portlandite and decalcification, at a more rapid rate than otherwise. The degradation rate is dependent to some extent on the supply of calcium ions. The rate of degradation of concrete increases if the supply of calcium ions is limited. This means that the rate of degradation of the concrete barriers close to cement-conditioned evaporator concentrates is lower than close to bitumen-conditioned concentrates, since the supply of calcium ions in the compartments containing bitumen-conditioned waste is lower than in the compartments containing cement-conditioned waste.

Conclusions in RD&D 2010 and its review

In RD&D Programme 2010, this process was described under the headings “Expansion/contraction of the waste” and “Fracturing”. SSM pointed out that current knowledge of the degradation of sulphonated polystyrene is inadequate. The degradation of such ion exchange resin could generate large quantities of sulphate, which could affect the waste form and its surrounding concrete structures.

SSM wanted to remind SKB that one way to improve the credibility of a relatively fracture-free silo over a long time would be to better estimate the rate of the relevant degradation mechanisms.

SSM was doubtful to the claim that the additional expansion volume in the silo would be sufficient or available to take up all volume expansion in the waste.

Newfound knowledge since RD&D 2010

Cleaning agents containing organic complexing agents that previously have been used at the nuclear power plants have been replaced with a carbonate-based agent that only contains inorganic complexing agents. This will lead to higher levels of carbonates in certain waste types in SFR. The waste that will contain these elevated levels will be solidified in either cement or bitumen. SKB has studied the consequences of elevated carbonate concentrations in the waste for the barriers and the waste containers. This has been done by thermodynamic modelling where altered mineral composition in the concrete due to elevated concentrations of carbonates has been studied.

The study shows that local effects on the surrounding barrier cannot be ruled out due to locally elevated carbonate concentrations. Waste packages that are not deposited next to the engineered barrier will have a smaller impact on the degradation rate due to the increased buffering effect of the surrounding concrete.

Programme

The programme initiated to gain a better understanding of dissolution, precipitation and recrystallization is described in Section 22.2.15 "Dissolution, precipitation and recrystallization" and Section 22.2.16 "Pore water speciation and concrete interactions".

20.2.16 Chemical 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 resulting organometallic 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 pH-values 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 (Ewart et al. 1991, Duro et al. 2012b). Most of these substances are not relevant to the long-term safety of SFR, while others may be of importance, as will be discussed in a report that will be submitted in support of SR-PSU. The greatest influence is that of isosaccharinic acid (ISA), which occurs in two diastereomeric forms, alpha and beta (Greenfield et al. 1993), Figure 20-2.

Conclusions in RD&D 2010 and its review

In RD&D Programme 2010, SKB described in general terms the work that had been done and was planned to gain a better understanding of the importance of the effects of organic complexing agents on the sorption of radionuclides.

SSM thought it was important to plan initiatives in this area based on a systematic analysis of the occurrence of different substances and of the complexing properties of the possible degradation products. Such an analysis should of course include not just degradation products, but also substances that are present in the repository from the beginning.

Newfound knowledge since RD&D 2010

The work that has been done at Chalmers University of Technology to study the degradation of cellulose to the two diastereomeric forms alpha- and beta-isosaccharinic acid has led to the discovery of a synthetic pathway for the synthesis of alpha- and beta-isosaccharinic acid in their purest forms (SKBdoc 1378692), see Figure 20-2.

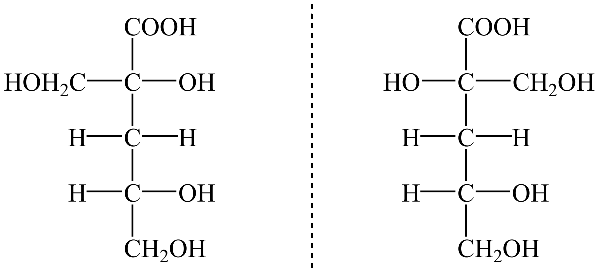


***Figure 20-2.** Alpha-isosaccharinic acid at left and beta-isosaccharinic acid at right.*

A project has been carried out to quantify the amount of cellulose that will be present in the waste at repository closure. In the case of BMA, the amount of cellulose has been quantified for each cell and the amount of isosaccharinic acid formed has been calculated with the aid of rate constants presented in Glaus and Van Loon (2008).

The degradation products of the filter aid UP2 are judged not to contribute to reduced sorption of radionuclides (Duro et al. 2012b).

Some of the studies conducted by SKB in this area are strongly linked to radionuclide sorption. Some of the results obtained in this discipline are therefore presented in Section 20.2.13 "Sorption".

Programme

Concrete cylinders containing ion exchange resins have been placed in the Äspö HRL to study the degradation of the waste and how the degradation products are transported in the concrete material. The experimental set-up is not such that the results can be used to make a quantitative assessment of the degradation rate of the different waste components. Instead, the focus is mainly on studying degradation products and how they spread in the grout. At the same time, laboratory experiments with a similar set-up are being conducted to serve as a reference as to when it may be appropriate to retrieve the experiments in the Äspö HRL.

20.2.17 Water uptake/swelling

When the waste, especially ion exchange resins, absorbs water it increases in volume and the entire waste matrix swells. In a bitumen matrix, water uptake occurs by diffusion into ion exchange resins and salt in evaporator concentrates. Since water moves through bitumen while the waste material is solidified, the bitumen matrix can be regarded as a semipermeable membrane and the process can be regarded as osmotic. Due to the stresses caused by the swelling, fractures can arise in the matrix and an interconnected pore system can be created. Swelling can also affect surrounding waste packages and barriers if not enough expansion volume is available. The degree of swelling is affected by a number of factors, for example quantity, type and quality of ion exchange resins and how they were pretreated when the waste was stabilized. The composition and quantity of evaporator concentrates present in the waste is also important.

Conclusions in RD&D 2010 and its review

Expansion of the waste was described in RD&D Programme 2010. A prerequisite for avoiding damage to surrounding barriers is that sufficient expansion volume is available. If the void in a waste package is insufficient, additional void may be required outside the packages. In its review, SSM commented that they wished to see an update of the state of knowledge concerning this process and particularly mention the silo repository. A broader study of the mechanical impact of swelling waste is described in Section 22.2.7 "Pressure from swelling waste".

Table 20-2. Swelling pressure for bitumen-stabilized waste as a function of expansion volume.

Relative expansion volume (%)	0	10	20	30
Average swelling pressure (MPa)	6	0.7	0.3	0.1
Maximum swelling pressure (MPa)	12	2.5	0.75	0.22

Newfound knowledge since RD&D 2010

A literature review within the framework of the safety assessment for SR-PSU has resulted in a compilation of data describing the relationship between swelling pressure and available expansion volume for bitumen-stabilized waste in SFR. These data are summarized in Table 20-2.

New developments are being monitored and will be acted on when appropriate.

20.2.18 Microbial processes

The organic material in the waste in the rock vault for low-level waste (1BLA), primarily cellulose, comprises a possible source of energy and nutrients for microorganisms. The microbial activity is dependent on the flow of water through the repository, since this is necessary for the transport of nutrients and energy to microorganisms, but also for the removal of toxic degradation products formed by fermentation. The pH will be around 12.5 initially, but is expected to decline to around 9 or less after about 2,000 years. During this time, most of the cellulose will be converted to isosaccharinic acid (ISA), which could be used as a source of carbon for microbes. The pH in other parts of SFR is expected to be high initially (higher than 13). As a result, the microbial activity is expected to be low, but will increase with time as the pH declines while the cement undergoes progressive degradation, ending up at a pH of 12.5.

Organic carbon from waste and reduced inorganic molecules such as hydrogen from anaerobic corrosion processes and methane from organic substances are potential electron donors and energy sources for microbial processes in SFR. As microbial oxidation of these energy sources proceeds, the microorganisms reduce electron acceptors in a particular order: oxygen, nitrate, manganese, iron, sulphate, sulphur and carbon dioxide. As long as the substances from oxygen down to sulphur in this chain are available, microbial formation of methane will be suppressed.

Newfound knowledge since RD&D 2010

The process report that has been produced within SR-PSU goes through the updated state of knowledge in the area. There are also references to the most recent research findings concerning alkaliphilic microorganisms.

Programme

At present, based on the current state of knowledge regarding microbes and the literature available on the subject, SKB can only assume which microorganisms are present in the groundwater in SFR. SKB therefore deems it necessary to survey which microorganisms that actually are present and what conditions these have to be active in the environment that will prevail in SFR after closure. SFR contains a great deal of organic carbon in both ion exchange resins and cellulose. They contain carbon-14, which could potentially reach the ground surface. However, only the carbon in cellulose is deemed to be available for microbes. Alkaline hydrolysis of cellulose under anoxic conditions mainly forms isosaccharinic acid (ISA), and only a small quantity (less than 10 percent) of low-molecular carbon compounds are judged to be consumed quickly by the microbes (Glaus et al. 1999). If it constitutes the predominant carbon source, ISA can be used by some microbes. Even if ISA-degrading microbes are not active when the repository is closed, it is possible that a mixed population could adapt to using ISA as a carbon source. This would happen when the ISA concentration increases as a result of increased alkaline degradation of cellulose. Methanogens (methane-producing archaea), however, have their optimal pH at 6–8 and their upper pH limit is in general around 10. When ISA is broken down, carbon dioxide and methane are formed in equal quantities. The carbon dioxide is absorbed by the concrete, while the methane gas can lead to a build-up of pressure.

To find out how much methane that can be formed under the conditions prevailing in the repository and whether there are microbes with the potential to degrade ISA in SFR, experiments will be conducted in cooperation with TVO Teollisuuden Voima Oyj) in Finland. Microbes will be taken from an SFR-like repository and tests will be conducted in the laboratory to investigate whether any species will be able to start using ISA as a carbon source.

20.2.19 Metal corrosion

Corrosion, i.e. degradation of metal by a chemical process, is one of the processes that has the greatest impact on a final repository, its engineered barriers and its ability to limit releases of radionuclides. The corrosion rate has a great impact on the life of the engineered barriers and determines the rate at which the metallic waste is degraded and induced activity is released.

Large quantities of metal are present in SFR in the form of waste, waste containers and iron reinforcement in concrete waste containers. This metal will corrode during the life of the repository, but the dominant corrosion process and the corrosion rate are dependent on the environment in the repository. Prior to and shortly after closure of the repository, when oxygen is still available, corrosion will be dominated by aerobic processes, while the long-term processes will be dominated by anaerobic corrosion mechanisms.

The clearest effect of corrosion is the fracturing caused by reinforcement corrosion. This occurs because the corrosion products occupy a larger volume than the original iron material, so the structures burst from the inside (Betongföreningen 2007).

Fracturing can also be caused by high gas pressures inside the repository. The underlying process is anaerobic corrosion, which produces hydrogen gas.

Corrosion of one micrometre of iron and steel corresponds to the production of about 4.5 litres of hydrogen gas per square metre if the process follows equation 20-1 below.

$$3Fe(s) + 4\,H_2O \rightarrow Fe_3O_4(s) + 4H_2(g) \qquad (20\text{-}1)$$

The corrosion rate of a metal depends on the surrounding environment. Important factors that affect corrosion are the presence of water and the water chemistry, mainly pH, Eh and the concentration of dissolved salts. Of these factors, pH is probably the most important, since a high pH puts iron and steel in a passive state, resulting in a low corrosion rate. Steel in contact with cementitious materials such as concrete and embedment grout is in a passive state, since the pH in this environment is about 12.5.

The passive state in the concrete can, however, be broken by carbonatization, where the portlandite ($Ca(OH)_2$) is transformed to calcium carbonate ($CaCO_3$) due to carbon dioxide penetration or chemical degradation of organic compounds, see Section 20.2.16 "Chemical degradation of organic compounds". The passive state can also be broken by penetration of chloride ions from the groundwater. When the passive state has been broken, the corrosion rate will increase considerably. This means that both degradation of the repository structures and the rate of release of radionuclides will increase.

Conclusions in RD&D 2010 and its review

Plans and a programme for joint research concerning corrosion of short-lived and long-lived low- and intermediate-level waste were presented in RD&D Programme 2010. Among other things, the then-just-initiated experimental programme in the Äspö HRL was presented, along with plans for laboratory experiments.

In its review of RD&D Programme 2010, SSM did not express any objections to the planned programme, but pointed out that a programme of research on aluminium corrosion had not been presented. They considered that this could be a deficiency if the uncertainties regarding this mechanism are great in an early phase after closure. SSM also noted that it was important that any future literature study of corrosion rates must focus on environments similar to that in SFR.

Newfound knowledge since RD&D 2010

SKB has carried out a literature study for the purpose of creating an updated picture of the state of research regarding corrosion of steel, aluminium and zinc in different environments. This study has shown that the corrosion rates previously assumed by SKB and used in previous analyses have been overly pessimistic. Tables 20-3, 20-4 and 20-5 present the corrosion rates that will be used in the coming safety assessment for SFR.

Programme

Concrete cylinders containing iron and steel samples have been placed in the Äspö HRL. The purpose is to study degradation of waste and how the degradation products are transported in the concrete material and interact with it. The experimental set-up is not such that the results can be used to make a quantitative assessment of the degradation rate of the different waste components. Instead, the focus is mainly on studying the degradation products and how they are spread in and interact with the grout. At the same time, laboratory experiments with a similar set-up are being conducted to serve as a reference as to when it may be appropriate to retrieve the experiments in the Äspö HRL.

SKB has also initiated a survey of the quantities of aluminium which the waste producers intend to deposit in SFR. The purpose of this work is to obtain a more correct picture of the gas formation caused by corrosion of aluminium during the early post-closure phase of the repository.

20.2.20 Gas formation and gas transport

Gas is formed in SFR as a result of a number of different chemical processes that occur in the waste, the most important of which are water radiolysis, microbial degradation of organic material and metal corrosion. The gases formed by microbial degradation of organic material mainly comprise methane and carbon dioxide. Anaerobic metal corrosion gives rise to hydrogen gas, as described in Section 20.2.19 "Metal corrosion". Radiolysis of water gives rise to oxygen and hydrogen gas, but this process is expected to make a smaller contribution to gas formation in SFR since water radiolysis is of secondary importance for SFR, see Section 20.2.5 "Water radiolysis".

The evolved gases will dissolve in water until saturation. Certain gases, such as carbon dioxide, can also react chemically with water to form carbonates. Dissolved gases can be transported through the waste by advection and diffusion. If the gas concentration exceeds the water's level of saturation,

Table 20-3. Corrosion rates for carbon steel in different environments.

Repository environment	Corrosion rate (μm/yr)	Reference
Alkaline aerobic	0.1	Blackwood et al. 2002
Alkaline anaerobic	0.05	Smart et al. 2004
Neutral pH aerobic	60	Kuron et al. 1985
Neutral pH anaerobic	2.8	Simpson and Weber 1988, Schenk 1988, Simpson et al. 1985

Table 20-4. Corrosion rates for stainless steel in different environments.

Repository environment	Corrosion rate (μm/yr)	Reference
Alkaline aerobic	0.02	Treadaway et al. 1989
Alkaline anaerobic	0.01	Mihara et al. 2002
Neutral pH aerobic	0.3	Kritsky et al. 1987
Neutral pH anaerobic	0.2	Naish et al. 2001

Table 20-5. Corrosion rates for aluminium and zinc.

Repository environment	Corrosion rate (μm/yr)	Reference
Alkaline anoxic conditions	1,000	Moreno et al. 2001

free gas is formed as bubbles in the water, which permits transport in a two-phase flow. The resulting increase in pressure in the system can hasten the transport of radionuclides in solution through the waste matrix. Upon formation of a gas phase, transport of gaseous radionuclides may also be facilitated, mainly radioactive methane gas that has been formed. When the gas pressure has increased to the point where it exceeds the resistance of the waste matrix, fractures will form so that gas can be released, see Section 20.2.10 "Fracturing".

Conclusions in RD&D 2010 and its review

This process is not described explicitly in RD&D Programme 2010, and SSM did not express any viewpoints on this in its review.

Newfound knowledge since RD&D 2010

SKB did not conduct any research on gas formation and gas transport in short-lived waste during the past RD&D period beyond what is described in Section 20.2.18 "Microbial processes" and Section 20.2.19 "Metal corrosion".

Programme

SKB does not currently have a separate research programme for gas formation and gas transport in short-lived waste. The work being done in this area is described in Sections 20.2.18 and 20.2.19. However, a specific programme may be formulated as a result of the coming assessment of long-term safety in SFR.

20.2.21 Speciation of radionuclides

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 way in which the pH changes over time will be presented in SR-PSU. Redox-sensitive radionuclides may be affected by changed redox conditions inside the repository.

The oxygen that is present in the repository will rapidly be consumed, mainly due to corrosion of steel and oxidation of dissolved iron(II). A low redox potential will be retained in the different repository parts as long as iron(II) ions are present. Since SFR is a relatively near-surface repository, the possibility that oxidizing conditions may arise in the repository cannot be ruled out, mainly due to penetration of oxygenated meltwater from future continental ice sheets.

Organic complexing agents may influence speciation and sorption for certain radionuclides, which affects radionuclide transport out of 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, complexing ability with organic complexing agents and sorbing capacity. This in turn affects radionuclide transport and water composition.

Conclusions in RD&D 2010 and its review

The programmes described in RD&D Programme 2010 that could possibly be realized have not been realized. SSM considered the most important question in this context rather to be how speciation affects radionuclide sorption. This issue should be investigated based on a more systematic approach than has been taken previously. In its review of SAR-08, SSM pointed out that there is a risk that the repository's redox-buffering capacity could eventually be depleted and oxidizing conditions could arise. This could result in a considerable dose contribution for redox-sensitive radionuclides, particularly technetium-99. SSM pointed out that this issue should be studied by SKB.

Newfound knowledge since RD&D 2010

SKB has investigated the reducing capacity of SFR by means of theoretical studies and modelling. The constructed models are based on twelve representative waste types that are deposited in SFR. The model has taken into account various geochemical processes that are relevant to the evolution of the repository's redox capacity.

The results indicate that corrosion of steel in containers, waste and reinforcement can sustain reducing conditions for a long time. A sensitivity analysis has been carried out with respect to the corrosion rate of steel. The results of this analysis suggest that the reducing environment in the repository will be maintained even with higher corrosion rates. The possible penetration of oxygenated groundwater (glacial meltwater) down into the repository has also been taken into account in the report. The model does not consider when such a case might occur; instead, the consequences are studied for the waste type S.13 (drums with ashes from Studsvik Nuclear AB). This is because S.13 has the lowest reducing capacity of the waste types studied. Penetration of oxygenated groundwater is also studied at the time when all iron(0) is assumed to have been corroded (Duro et al. 2012a).

New developments in the field are being monitored and will be acted on when appropriate.

20.2.22 Radionuclide transport in the aqueous phase

Extensive calculations of radionuclide transport for SFR were carried out within the SAR-08 safety assessment project (SKB 2008a). The basis for these calculations (division into elements and the hydrogeological model used) was similar transport calculations that were carried out in the previous safety assessment project "SAFE – Safety Assessment of Final Repository for Radioactive Operational Waste" (Lindgren et al. 2001). These were implemented in another calculation program in conjunction with SAR-08. The time period covered by the assessment was also extended to include the impact of climate change.

Newfound knowledge since RD&D 2010

After SAR-08 development has continued, and Ecolego will be used to model radionuclide transport in SR-PSU. SKB now has a model of SFR and its future extension in Ecolego, where both more complex and more simplified transport models will be analyzed.

Input data for this analysis comes in the form of the process descriptions presented in Chapters 20, 21 and 22 in RD&D Programme 2013 as well as the geometry descriptions and waste descriptions in SR-PSU. In addition, input data comes from other types of modelling activities, such as the hydrogeological calculations presented in Chapter 26 "The geosphere", and detailed calculations of flow in the near-field presented in Section 20.1.5 "Hydrovariables".

Programme

For a description of the work being done by SKB in the area, see Sections 22.1.4 "Hydrovariables", 22.1.5 "Mechanical stresses" and 22.2.5 "Water transport under saturated conditions".

20.2.23 Radionuclide transport in the gas phase

A number of gas-generating processes will proceed in SFR after closure. During an initial phase, gas will primarily be generated by corrosion of the aluminium and zinc present in parts of the deposited waste, see Section 20.2.19 "Metal corrosion". Other gas-generating processes are microbial degradation of organic material and corrosion of iron and steel. Some of the generated gas will be dissolved in water and be transported in the aqueous phase. No need for a separate handling of radionuclide transport in the gas phase has been identified in previous safety assessments.

Newfound knowledge since RD&D 2010

The importance of this transport mechanism and the need to study it explicitly is included in the safety assessment accompanying the application under the Nuclear Activities Act for the expansion of SFR.

Programme

The importance of radionuclide transport in the gas phase is being evaluated in the SR-PSU safety assessment, which will be completed in the spring of 2014. A possible research programme will be formulated based on the needs identified in the safety assessment.

21 Long-lived low- and intermediate-level waste

This chapter describes SKB's scientific research linked to the long-lived low- and intermediate-level waste and the repository components with which this waste will come into contact.

Since a final repository concept for the long-lived waste has not yet been finalized, the processes will be described in somewhat more general terms than is the case for the short-lived waste. This means that the link between waste and rock vaults will be via the materials used rather than a given waste being linked to a given rock vault. A point of departure in the research programme described in this chapter is that cementitious materials will be widely used in the repository, in the form of either internal (stabilizing) or external (embedment) grout, or in the form of repository structures and backfill. If it emerges from a future safety assessment that other materials or material combinations are to be preferred for long-term safety, this research programme will need to be revised. At the present time, however, SKB believes that assuming that the repository will contain cementitious materials, ensuring a high pH, is the most appropriate point of departure.

The research SKB plans to conduct to gain a better understanding of the degradation of the engineered concrete barriers in the final repository for short-lived radioactive waste, SFR, and the future final repository for long-lived waste, SFL, is dealt with in Chapter 22, while the bentonite research is dealt with in Chapter 25. Chapters 26 and 27 describe the research SKB is conducting to gain a better understanding of the geosphere and surface ecosystems.

Conclusions in RD&D 2010 and its review

RD&D Programme 2010 described the scientific research for the long-lived low- and intermediate-level waste in general terms, and only a few areas were touched upon. Otherwise, reference was made to the programme for scientific research linked to the short-lived low- and intermediate-level waste. SSM noted that the research programme for SFL largely conformed to the one for SFR and would have wished that the account for SFL to followed the same systematics as the one for SFR.

Comments by SSM concerning general issues linked to SFL are addressed in Chapter 6.

Newfound knowledge since RD&D 2010

Since the publication of RD&D Programme 2010, work on the SFL concept study has begun and the study is being compiled for presentation in December 2013. This work has pointed to the need to structure and report the research that will be required for future safety assessments in a similar manner as for the safety assessment for the extension of SFR.

This chapter therefore describes the research on the presumed initial state of the waste and the processes that are expected to affect the waste after repository closure. The point of departure in the description has been that SFL will be a geological repository and that concrete, gravel and bentonite may be components in the engineered barriers. Since this is the first RD&D programme where the research on long-lived low- and intermediate-level waste is described in this manner, the focus is on describing variables and processes as well as programmes, rather than conclusions and newfound knowledge since RD&D Programme 2010.

For a description of the origin and management of the waste and the work of updating the reference inventory for SFL, see Chapter 6.

21.1 Initial state in the waste

The initial state for the waste in SFL is defined as the state existing at closure. At closure, drainage pumping in the repository will cease, after which the repository will fill with water and the waste will become water-saturated. In this section, an attempt is made to present an initial state for the waste in SFL, despite the fact that a definitive repository concept is lacking. The attempt is based on the assumption that the same repository components that are used in SFR will also be used in the future concept for SFL. This means that concrete and/or bentonite will interact with and be affected by the groundwater that will enter the repository after closure. It is therefore assumed that the processes that can be expected to occur in SFL will be similar to those expected to occur in SFR.

To determine the initial state in the waste, it is important to know how the waste is conditioned. Today, core components are planned to be placed in waste containers of steel that are embedded in a cementitious material, which makes it relatively simple to determine the initial state. The legacy waste is also planned to be placed in steel containers. Considerable uncertainty is associated with the determination of its initial state, due to incomplete knowledge of the composition of the waste. SKB's knowledge concerning the legacy waste will increase as a result of the work of updating of the reference inventory, which will make it easier to correctly determine the initial state of the repository.

21.1.1 Variables

The initial state is the starting point for an assessment of long-term safety. The state is described by the initial values of a number of variables, which characterize the waste in a suitable manner for the safety assessment, see Table 21-1. The description applies not only to the waste itself, but also to the cavities (voids) that exist in and between the waste packages and into which water will penetrate. Processes such as advection and mixing will take place in the cavities.

21.1.2 Geometry

SFL will contain waste types with different geometries that will be packaged in waste containers with different properties, depending on the type of waste. One geometry parameter for the waste containers besides their dimensions is the fraction of cavities present in the waste packages. The term "cavity" here refers to the pore system in the cement-based grout that is used to fill the space between the individual packages in the repository and to fill up the voids inside the waste package. It also includes the volumes in and between waste packages that contain neither waste nor cement-based grout. The proportion of cavities in the repository affects advection and diffusion and thereby the transport of radionuclides.

Programme

SKB's work in this area is currently focused on development of waste containers, conditioning methods and repository structures, see Section 8.2 "Final repository for long-lived waste". In the development of waste containers for SFL, it is necessary to take into account the fact that there is today a great deal of waste that is already conditioned in various types of waste packages, for example drums and moulds.

Table 21-1. Variables for the waste in SFL.

Variable	Definition	Section
Geometry	Geometric dimensions of all waste containers.	21.1.2
Radiation intensity	Intensity of alpha, beta and gamma radiation as a function of time and space in the waste.	21.1.3
Temperature	Temperature as a function of time and space in the waste.	21.1.4
Hydrovariables	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.	21.1.5
Mechanical stresses	Mechanical stresses as a function of time and space in the waste containers.	21.1.6
Total radionuclide inventory	Total occurrence of radionuclides as a function of time and space in the different parts of the waste.	21.1.7
Material composition	The materials of which the different components of the waste are composed, excluding radionuclides.	21.1.8
Water composition	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.	21.1.9
Gas variables	Composition of gas (including any radionuclides) in the waste and in the cavities in the waste as a function of time and space.	21.1.10

In order to create a standardized transport and handling system for SFL, the intention is to place existing waste packages in new steel containers with standardized base dimensions. The space between the individual waste packages is planned to be filled with a cement-based grout to reduce the void volume and stabilize the individual waste packages during transport. The development of this cement-based grout, as well as external (embedment) grout and application methods, will begin when development of the waste containers is finished.

21.1.3 Radiation intensity

Radiation intensity is dependent on the inventory of radionuclides and the geometry of the waste. The radiation intensity in the waste planned to be deposited in SFL is not judged to be sufficiently high to have other than a marginal impact on the evolution of the repository.

Programme

SKB's programme in this area is mainly focused on the updating of the reference inventory and the associated determination of the radionuclide content in the waste, see Section 6.4 "Updating of the reference inventory". Further plans in the area will be made when this work is finished.

21.1.4 Temperature

The initial temperature in the repository is determined by the temperature of the ambient rock, and the annual mean temperature is expected to be around +15°C. No heat-generating processes are judged to occur in the waste initially, mainly due to the fact that there is not enough water, microbial activity or radiation energy.

Programme

SKB will await the results from the updating of the reference inventory before making any plans in this area. Of primary interest here is the legacy waste, for which certain uncertainties still exist.

21.1.5 Hydrovariables

The boundary conditions that control the water flow through the waste are given by the surrounding pressure field. This field is in turn dependent on the repository's layout and location in the rock, the hydrogeological properties of the rock and the properties of the engineered barriers. The local flow pattern is also affected by the size, shape and material properties of the waste and its containers. There is no flow of water through the waste during operation, since the repository is drained by pumping. When the pumping ceases the repository will fill with water. This constitutes the starting point for transport processes that can release radionuclides.

Programme

The initial versions of a model for evaluation of the flow through the waste will be developed within the framework of the safety evaluation for SFL.

21.1.6 Mechanical stresses

During operation and at closure, the waste and waste containers will be subjected to various kinds of loads, which may cause mechanical stresses, mainly compressive stresses, in the waste packages. During the operating period, the loads will mainly be caused by stacking of the waste packages as well as by embedment grouting in conjunction with closure (cf. handling in the silo in SFR). When the repository is backfilled and water-saturated, the forces that then act on the repository structures may also be propagated into the waste containers and the waste.

SKB's plans for development of waste containars and the conditioning method for the SFL waste are presented in Section 8.2 "Final repository for long-lived waste". Container and conditioning method have been chosen under the provision that the loads that act on the repository prior to and at closure may not adversely affect the waste containers or waste contained therein.

Other processes that could cause mechanical stresses in waste or waste container – such as evolution of gas caused by corrosion of the waste, pressure from swelling waste, pressure from bentonite and fall of ground – are not expected to occur to any appreciable extent prior to closure and are therefore not deemed to have any appreciable effect on the initial state.

SKB does not at present have any programme for research on mechanical stresses in waste and waste containers.

21.1.7 Total radionuclide inventory

The waste that is intended to be deposited in a future SFL consists primarily of neutron-irradiated core components from the NPPs and legacy waste managed by SVAFO. The radionuclide inventory for the core components is based primarily on calculations, while the legacy waste is characterized by means of measurements.

Programme

In order to continuously update and reduce uncertainty in the total radionuclide inventory for SFL, SKB will systematically evaluate the calculation methodology and models for determination of the activity content in reactor internals. By comparing the results of different methods and models with measured activity, SKB intends to clarify how approximations in methods and models affect the uncertainty in the calculation results and in this way establish the best available technology for determining the activity content in reactor internals. Furthermore, SKB will follow SVAFO's work with characterization of the legacy waste.

21.1.8 Material composition

The waste intended to be deposited in the future SFL can be divided into two main categories: waste from the Swedish nuclear power plants (NPPs) and legacy waste from the development of the Swedish programme, see also Section 6.4. "Updating of the reference inventory".

Most of the waste from the NPPs consists of metals, above all carbon steel and stainless steel that arise in conjunction with repairs and upgrades as well as dismantling and demolition of the NPPs. Most of the metallic waste consists of core components and PWR reactor pressure vessels (RPVs).

The composition of the legacy waste is not equally uniform, but can instead be described as complex. The many thousand waste packages that are stored at the Studsvik site contain various metals, ashes, organic compounds and, to some extent, even liquids.

Conclusions in RD&D 2010 and its review

The material composition of the long-lived waste was described in general terms in RD&D Programme 2010. In its review, SSM stated its expectation that the new reference inventory will be of at least as good quality as the one that was prepared for the safety assessment performed in 1999 (SKBdoc 1416968).

Newfound knowledge since RD&D 2010

SKB has in recent years worked in cooperation with the NPPs and other waste suppliers to compile a new reference inventory. In addition to a compilation of the radionuclides that occur in the waste, this work also includes a compilation of materials and quantities of other waste that are intended to be deposited in SFL, see also Section 6.4. "Updating of the reference inventory".

21.1.9 Water composition

See Section 20.1.9 "Short-lived low- and intermediate-level waste, Water composition" and 22.1.7 "Concrete barriers, Water composition".

SKB does not have any plans to study water composition for SFL during the coming RD&D period.

21.1.10 Gas variables

Gas variables are described in Section 20.1.10.

21.2 Processes

A number of processes will over time alter the state in the waste and in any cavities that may occur inside and between waste packages. Certain processes occur no matter what, while others are only possible under certain specific conditions.

21.2.1 Overview of processes

The processes that can be expected to affect the waste, the waste containers and materials that may be used to stabilize the waste are described in this section. In addition, processes that can affect any remaining voids and pores are described.

The processes in question can be divided into six main classes: radiation-related processes, thermal processes, hydraulic processes, mechanical processes, chemical processes and radionuclide transport. A number of different processes may occur under each main class and interact with each other or with other processes.

Radiation-related processes

As they decay, radionuclides in the waste emit ionizing radiation of different types and intensities. The radiation may interact with other components in the waste and create new chemical species. If the radiation is intense, this can lead to the formation of reactive radicals, while this only occurs to a limited extent if the radiation is less intense.

The following radiation-related processes are discussed in this chapter:

- Radioactive decay, Section 21.2.2.
- Radiation attenuation and heat generation, Section 21.2.3.
- Radiation-induced degradation of organic matter, Section 21.2.4.
- Water radiolysis, Section 21.2.5.

Thermal processes

In SFL, a thermal interaction will take place between the waste and the repository structures and surrounding bedrock during the entire operating lifetime of the repository. As a result of this interaction, heat-generating processes in the waste may affect the waste packages, the engineered barriers and the surrounding bedrock, and conversely, thermal processes in the bedrock may affect the engineered barriers and the waste. Thermal processes in the concrete barriers are dealt with in Section 22.2.2 “Heat transport” and 22.2.3 “Phase change/freezing”.

Heat-generating processes are only expected to occur to a limited extent, since the properties of the SFL waste are not of that character. However, information may emerge from the updating of the reference inventory that requires a revaluation of this expectation.

The repository temperature will be controlled by the temperature of the surrounding bedrock. Under normal conditions, this process is of little importance for the waste, its containers and any surrounding materials. However, when freezing occurs during a permafrost period, frost-sensitive solid materials such as concrete and any liquids in the waste could be affected. Since the repository will be located at sufficiently great depth to ensure that freezing will not take place, thermal processes in the waste and its containers are judged to be of subordinate importance for the long-term safety of the repository.

The following thermal processes are discussed in this chapter:

- Heat transport, Section 21.2.6.
- Phase change/freezing, Section 21.2.7.

Hydraulic processes

The water flow through the waste is determined by the geometry and permeability of the various structural parts and components in the repository, as well as by the pressure gradient from the surrounding rock. The objective of minimizing the flow through the waste is of central importance for the repository's design and location in the rock. As a result, erosion of the different waste containers is assumed to be negligible in comparison with chemical degradation. Corrosion is an example of a chemical degradation process that can generate gas and thereby affect the flow through the waste. High 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. Other simultaneous water and gas flows may arise when the waste is water-saturated after closure.

The following hydraulic processes are discussed in this chapter:

- Water uptake and transport under unsaturated conditions, Section 21.2.8.
- Water transport under saturated conditions, Section 21.2.9.

Mechanical processes

Waste and waste containers in SFL will be subjected to both internal and external mechanical forces, which will give rise to mechanical stresses in the materials. Depending on the mechanical properties of the material, these stresses may result in fracturing or plastic deformation of the material. In SFL, the concrete that was to be used for grouting can serve as an example of the former, while the planned steel containers and the core components in the waste more closely represent the latter material type. Fall of ground is dealt with in Section 22.2.10.

The following mechanical process is dealt with in this chapter:

- Fracturing, see Section 21.2.10.

Chemical processes

After SFL has been closed and sealed, a chemical interaction will take place between the repository and the groundwater in the surrounding bedrock. The interaction will occur with waste, waste containers, grout and the repository structure. The groundwater or solutes in it may affect the waste, the waste containers and the engineered barriers, but the reverse is also possible.

These chemical processes will lead to changes in the properties of the waste and the waste containers over time.

The following chemical processes are discussed in this chapter:

- Advective transport of solutes, Section 21.2.11.
- Diffusive transport of solutes, Section 21.2.12.
- Sorption, Section 21.2.13.
- Colloid formation and colloid transport, Section 21.2.14.
- Dissolution, precipitation and recrystallization, Section 21.2.15.
- Chemical degradation of organic compounds, Section 21.2.16.
- Water uptake/swelling, Section 21.2.17.
- Microbial processes, Section 21.2.18.
- Metal corrosion, Section 21.2.19.
- Gas formation and gas transport, Section 21.2.20.

Radionuclide transport

The following radionuclide transport processes are dealt with in this chapter:

- Speciation of radionuclides, Section 21.2.21.
- Radionuclide transport in the aqueous phase, Section 21.2.22.
- Radionuclide transport in the gas phase, Section 21.2.23.

21.2.2 Radioactive decay

No account was given of the process of radioactive decay in RD&D Programme 2010, and SSM did not express any viewpoints on this in its review. SKB has not conducted any studies of radioactive decay of long-lived low- and intermediate-level waste beyond what is reported in Section 20.2.2 “Radioactive decay”. If new information is forthcoming from the update of the reference inventory for SFL, however, initiatives may be needed in the area.

21.2.3 Radiation attenuation and heat generation

The radiation emitted during radioactive decay can interact with the waste and/or other materials in the repository. The released energy is transferred to the surrounding materials in the form of thermal energy and will thereby be extinguished. The waste in the future SFL is not expected to emit appreciable quantities of heat-generating radiation, so the process is deemed to be irrelevant to SFL.

No account was given of this process in RD&D Programme 2010, and SSM did not express any viewpoints on it in its review. SKB did not conduct any studies of radiation attenuation and heat generation in long-lived low- and intermediate-level waste during the past period and sees no need today for such a research programme. If new information is forthcoming from the update of the reference inventory for SFL, however, this standpoint may be reconsidered.

21.2.4 Radiation-induced degradation of organic matter

Organic matter in the waste can be broken down by ionizing radiation and form new chemical species. The scope of the degradation is dependent on the type and intensity of the radiation. Some organic waste will be present in the legacy waste in SFL. But virtually no organic matter will be present in the waste from the NPPs.

Due to the low radiation intensity in the legacy waste, this process is deemed to be of little importance for long-term safety in SFL.

No account was given of this process in RD&D Programme 2010, and SSM did not express any viewpoints on it in its review. SKB did not conduct any studies of radiation-induced degradation of organic matter in long-lived low- and intermediate-level waste during the past period and sees no need today for such a research programme. If new information is forthcoming from the update of the reference inventory for SFL, however, this standpoint may be reconsidered.

21.2.5 Water radiolysis

The water that enters SFL and the waste packages will be affected by the ionizing radiation. This leads to excitation and ionization of water molecules, breaking the chemical bonds in the molecules. The radiation intensity in SFL is expected to be so low that radiolysis of water is not judged to have any significant impact on long-term safety.

No account was given of this process in RD&D Programme 2010, and SSM did not express any viewpoints on it in its review. SKB did not conduct any studies of water radiolysis in repositories for long-lived low- and intermediate-level waste during the past period and sees no need today for such a research programme. If new information is forthcoming from the update of the reference inventory for SFL, however, this standpoint may be reconsidered.

21.2.6 Heat transport

Heat transport between the different components can 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 essentially be determined by heat exchange with the surrounding rock and groundwater, while the influence of the waste on the temperature can be regarded as negligible.

No account was given of this process in RD&D Programme 2010, and SSM did not express any viewpoints on it in its review. SKB did not conduct any research on heat transport in the long-lived low- and intermediate-level waste during the past period and sees no need today for such a research programme. If new information is forthcoming from the update of the reference inventory for SFL, however, this standpoint may be reconsidered.

21.2.7 Phase change/freezing

Freezing of waste and waste containers, as well as grout, can occur if permafrost reaches repository depth. The scope of freezing is directly dependent on how long the permafrost period lasts and the depth at which the repository is located.

The effect of freezing is dependent on the material type in question. For example, metals in the waste are not expected to be directly affected by freezing, while grout and concrete waste containers could burst due to freezing in the same way as the concrete barriers. Metal containers could burst due to freezing if the contents expand to a sufficient degree.

No account was given of this process in RD&D Programme 2010, and SSM did not express any viewpoints on it in its review. SKB did not conduct any research on phase change/freezing of the long-lived low- and intermediate-level waste during the past period. However, a couple of studies regarding freezing of concrete barriers have been conducted and are described in Section 22.2.3 "Phase change/freezing". Since grout can be expected to be affected in the same way as concrete in the event of a freeze, these studies are considered to be relevant in this area as well.

SKB sees no need today for a separate research programme concerning phase change/freezing of the long-lived low- and intermediate-level waste and its grout, but intends to coordinate this work with the research on phase change/freezing of the concrete barriers, see Section 22.2.3.

21.2.8 Water uptake and transport under unsaturated conditions

Prior to closure, the repository is kept dry by drainage pumping. After closure, the groundwater will saturate fractures and pores in the rock near the repository. The groundwater then continues to fill voids in the rock vaults and gradually saturate other engineering materials and waste packages. When the repository becomes water-filled, transport processes are initiated that can lead to release of radionuclides.

SKB does not at present have any research programme on water uptake and transport under unsaturated conditions in SFL.

21.2.9 Water transport under saturated conditions

When the repository has become saturated with water, the hydraulic gradient will drive the water 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 presence of cavities or fractures in the waste also has an influence. What limits the flow through the waste is primarily surrounding barriers of concrete and bentonite. Viewed over a long time, the flow through the waste can change, partly as a result of changed groundwater flow, and partly due to gradual degradation of barriers and waste.

Detailed studies of the water flow through the waste are not currently planned.

21.2.10 Fracturing

A number of processes in SFL can lead to fractures in the waste, the waste containers, or the grout. The extent to which a material can fracture is determined by its properties. A brittle material such as concrete has a greater tendency to fracture under an imposed load than a tougher material such as steel.

Of the material types in SFL, internal and external grout and the concrete moulds with legacy waste are considered to be most susceptible to fracturing, while the planned steel containers and core components are expected to be less fracture-prone. The tendency of the legacy waste to fracture cannot be judged today, since not enough is known about its exact composition.

The effects of fracturing in the waste, its containers and grout are difficult to foresee. However, the process as such is judged to be less critical for long-term safety, since neither the waste nor its containers have been credited with any safety function in the long-term safety assessment. Furthermore, the water flow through the repository is affected more by fracturing in the concrete barriers than in the waste and its containers.

The following processes that can lead to fracturing in the waste, waste containers and grout are discussed:

- Phase change/freezing, Section 21.2.7.
- Gas formation caused by chemical degradation of organic compounds, Section 21.2.16.
- Water uptake/swelling, Section 21.2.17.
- Metal corrosion, Section 21.2.19.
- Gas formation from metal corrosion, Section 21.2.19.

Fracturing in the engineered concrete barriers is dealt with in Section 22.2.9.

No account was given of the process of fracturing in RD&D Programme 2010, and SSM did not express any viewpoints on it in its review. SKB did not conduct any research on fracturing in the long-lived low- and intermediate-level waste during the past period. Newfound knowledge concerning the processes that can cause fracturing is presented in the sections cited above.

SKB sees no need today for a separate research programme concerning fracturing in the long-lived low- and intermediate-level waste or its grout. Research concerning fracturing in cementitious materials is coordinated with research in the concrete barriers and is presented in Section 22.2.9 "Fracturing".

21.2.11 Advective transport of solutes

Advective transport of solutes in cement-conditioned waste is dealt with in the same way as in concrete barriers, see Section 22.2.11 "Advection and mixing". No transport limitation is assumed for other waste forms.

21.2.12 Diffusive transport of solutes

Diffusive transport of solutes in cement-conditioned waste is dealt with in the same way as in concrete barriers, see Section 22.2.12 "Diffusion". No transport limitation is assumed for other waste forms.

21.2.13 Sorption

Sorption of radionuclides is one of the most important retarding safety functions in SFL. Sorption is expected to take place on cement, see Section 20.2.13 "Sorption", but also on minerals in the bedrock. Cement will be a commonly occurring material in SFL, so that knowledge regarding sorption can be regarded as the same as that described in Section 20.2.13.

21.2.14 Colloid formation and colloid transport

A number of physical and geochemical factors affect the stability of colloids and, as a consequence, their transport. The stability of colloids in solution is defined by hydrochemical parameters. Mobility and retention in transport are linked to the physical properties of the fractures, where colloids can be filtered and retarded in fracture-filling materials as well as by sorption to mineral surfaces. The gradient of water velocities, the size distribution of pores, and the rugosity and tortuosity of the sorbing mineral surfaces and pores influence the mechanical filtration of colloids by the medium (Ryan and Elimelech 1996). Colloid filtration can therefore take place via either mechanical or electrostatic processes. See also Section 20.2.14 “Colloid formation and colloid transport”.

21.2.15 Dissolution, precipitation and recrystallization

The chemical and mechanical properties of cement and concrete used for grouting or to cast covers on the waste packages will be altered with time by reactions with the groundwater or substances dissolved in it. These substances may come from the surrounding rock, the waste or other waste packages.

Degradation and dissolution of cement and concrete in the waste releases ions and mobilizes substances that become available for transport. Processes such as water transport can alter the water composition, and speciation will be altered when chemical equilibria are established. This can lead to precipitation and immobilization of substances dissolved in the water, or an increase in the solubility of a substance.

How quickly dissolution takes place depends on the material composition of the waste, see Section 21.1.8 “Water uptake and transport under unsaturated conditions”, and how it is conditioned and packaged. Chemical degradation of organic compounds is described in Section 21.2.16 “Chemical degradation of organic compounds”, while metal corrosion is described in Section 21.2.19 “Metal corrosion”.

Reactions between solutes and cementitious materials in waste containers and grout can also lead to clogging of pores and small fractures. This can in turn prevent gases from being conducted out of the repository in a controlled fashion. This process is dealt with in Section 22.2.15 “Dissolution, precipitation and recrystallization”.

This process is described only cursorily in RD&D Programme 2010 in the chapter “Management of long-lived low- and intermediate-level waste”, and SSM did not express any viewpoints on it in its review. SKB did not conduct any research in the area during the past period. This work has instead been coordinated with corresponding studies of the concrete barriers and is presented in Section 22.2.15 “Dissolution, precipitation and recrystallization”.

SKB sees no need today for a separate research programme in the area, but will continue to coordinate this research programme with corresponding studies of the concrete barriers.

21.2.16 Chemical 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 great extent on the oxidation number of the radionuclide and whether the resultant organometallic 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 pH-values that are generated in the cement pore water.

Chemical degradation of organic compounds can also result in gas production, which may lead to pressure build-up inside the structures. This may also lead to the release of radionuclides in the form of gas molecules. Another possibility is that generated carbon dioxide hastens the carbonatization of cementitious materials, such as grout or waste containers.

According to the reference inventory from 1998 (SKBdoc 1416968), the quantity of organic matter in SFL is limited and derives primarily from the legacy waste. The assessment is thereby made that this process will not be of crucial importance to the long-term safety of SFL.

No account was given of this process in RD&D Programme 2010, and SSM did not express any viewpoints on it in its review. SKB did not conduct any research on chemical degradation of organic compounds in the long-lived low- and intermediate-level waste during the past period. SKB sees no need today for a separate research programme in the area, but has chosen to coordinate this with research concerning chemical degradation of organic compounds in the short-lived waste, see Section 20.2.16.

If new information is forthcoming from the update of the reference inventory for SFL concerning organic material in the legacy waste, however, this standpoint may be reconsidered.

21.2.17 Water uptake/swelling

After closure and resaturation of the repository, the waste will be in constant contact with water, which means that different processes may affect the waste.

Changes in the chemical and/or mechanical properties of the waste could cause the waste to expand or contract. Contraction is not expected to have any impact on the long-term properties of the repository, whereas expansion could give rise to high mechanical stresses and possibly fracturing in the waste containers.

Most of the waste in SFL will consist of metals, while the quantity of organic waste is expected to be limited. Expansion of the waste in SFL is therefore judged to be caused mainly by metal corrosion, see Section 21.2.19.

In waste packages containing large quantities of waste water or water bound in grout, freezing could lead to expansion and thereby high stresses in the container, see Section 21.2.7 "Phase change/freezing". In the light of the objective of locating the repository at a depth that remains frost-free even during a permafrost period, it is judged that this process will not have any adverse impact on the long-term safety of the repository.

No account was given of this process in RD&D Programme 2010, and SSM did not express any viewpoints on it in its review. SKB did not conduct any research on water uptake/swelling in the long-lived low- and intermediate-level waste during the past period and sees no need today for such a research programme in the area. Since it is judged that whatever swelling may occur is mainly linked to metal corrosion, reference is made to the research programme for metal corrosion, see Section 21.2.19. If new information is forthcoming from the update of the reference inventory for SFL concerning the content of organic swellable material in the legacy waste, however, this standpoint may be reconsidered.

21.2.18 Microbial processes

Microbial processes are not judged to be of crucial importance with regard to the long-lived low- and intermediate-level waste consisting of induced steel.

No account was given of this process in RD&D Programme 2010, and SSM did not express any viewpoints on it in its review. SKB did not conduct any research on microbial processes in the long-lived low- and intermediate-level waste during the past period and sees no need today for such a research programme in the area. If new information is forthcoming from the update of the reference inventory for SFL concerning organic material in the legacy waste, however, this standpoint may be reconsidered and a separate research programme may be formulated.

21.2.19 Metal corrosion

At closure, SFL will contain a large amount of metals of various kinds, mainly in the form of steel core components, but also waste containers. If the repository will contain concrete structures, the assumption in matters concerning metal corrosion should be that steel reinforcement will be used.

The rate at which steel corrodes has a great impact on the long-term safety of the repository, since the release rate for induced activity in the metallic waste is directly linked to the corrosion rate. The anaerobic corrosion process also gives rise to the formation of hydrogen gas. A high corrosion rate leads to a high gas formation rate, which can cause pressure build-up inside the repository structures. If the gas formation rate is low, however, it is possible for the evolved gas to slowly diffuse out through the repository structures, see also Section 22.2.19 "Gas production".

Since the environment in SFL will probably resemble that in SFR with respect to pH, Eh and water composition, the processes that occur in SFR can also be expected to occur in SFL.

Conclusions in RD&D 2010 and its review

The research on metal corrosion of the long-lived low- and intermediate-level waste was presented only briefly in RD&D programme 2010, with reference to the research on metal corrosion in the short-lived waste. In its review, SSM could not see a clear picture of what the programme for corrosion and gas formation due to corrosion in the long-lived waste will contain and expects to see an improvement in this respect in RD&D programme 2013. SSM also said that central issues for SFL are the need for knowledge of corrosion of metals and the consequences of gas formation due to corrosion.

Newfound knowledge since RD&D 2010

Since RD&D Programme 2010, SKB has updated its knowledge concerning corrosion of metals, mainly iron and steel, in the repository environment. It has been found in this work that the corrosion rates previously assumed to apply in an alkaline oxygen-free repository environment have probably been considerably overestimated, see Section 20.2.19 "Metal corrosion".

These new data have consequences for SFL as well, since the environment in this repository, and in particular the repository part where the largest fraction of metals will be placed, will probably be characterized by high pH and a minimal supply of oxygen. With a low corrosion rate, the gas formation rate will also be low and there will be a higher likelihood that the evolved gas can be transported away in a controlled manner.

Programme

SKB intends to coordinate the research on metal corrosion of the long-lived low- and intermediate-level waste and the associated consequences with the research on metal corrosion of the short-lived low- and intermediate-level waste during the coming period as well. SKB's programme for research on corrosion is presented in Section 20.2.19 "Metal corrosion". The programme for research concerning effects of gas production on the concrete barriers is presented in Section 22.2.19 "Gas production".

21.2.20 Gas formation and gas transport

Of the processes that can contribute to gas formation in the long-lived waste, metal corrosion is believed to stand for the greatest contribution, see Section 21.2.19. This is due to the fact that a large portion of the waste consists of core components and reactor internals of carbon steel and stainless steel. Organic material is also present in long-lived waste from the Swedish nuclear research programmes as well as in long-lived non-nuclear waste, which means that microbial degradation of these waste types can also give rise to gas formation.

Gas transport through the long-lived waste in a future SFL is determined by how the waste is conditioned, the choice of waste containers and whether grout is used.

Conclusions in RD&D 2010 and its review

This process was not described in RD&D programme 2010. In its review, SSM offered the opinion that metal corrosion and the consequences of gas formation resulting from it is a crucial issue for SFL.

Programme

SKB did not conduct any research on gas formation and gas transport in long-lived waste during the past period beyond what is described in Section 21.2.19. There is no need today for a separate research programme concerning gas formation and gas transport in long-lived waste. As a result of the coming safety evaluation for SFL, however, a specific programme may be formulated.

21.2.21 Speciation of radionuclides

Radionuclides released from the waste will occur in SFL in the form of various chemical species in solution. The exact form in which each individual radionuclide occurs is determined by the surrounding environment, where pH, Eh and oxidation number have a particularly great influence. A finished repository concept does not exist today for SFL, but cement and concrete will probably be important components. In addition, steel will be present in the form of both waste containers and waste, as well as in the form of reinforcement in the concrete structures. This means that the environment in SFL will probably resemble that in SFR after closure, and speciation of radionuclides will therefore resemble that in SFR, see Section 20.2.21 "Speciation of radionuclides". It should, however, be realized that the radionuclide inventory in SFL will differ from that in SFR and that there is at this time no compilation of knowledge on the speciation of radionuclides in SFL.

This process was not described in RD&D Programme 2010, and SSM did not express any viewpoints in its review. SKB did not conduct any studies concerning speciation of radionuclides in the long-lived low- and intermediate-level waste during the past period. Certain studies concerning speciation of radionuclides in the short-lived waste are, however, relevant to the long-lived waste as well, see Section 20.2.21 "Speciation of radionuclides".

At this time SKB does not have any specific programme for research on the speciation of radionuclides in the long-lived waste. However, as the reference inventory is concretized and more details are determined on the choice of barrier material in the repository, a separate research programme for speciation of the long-lived waste may become necessary.

21.2.22 Radionuclide transport in the aqueous phase

Handling of radionuclide transport in the aqueous phase is presented in Section 20.2.22.

21.2.23 Radionuclide transport in the gas phase

Handling of radionuclide transport in the gas phase is presented in Section 20.2.23.

22 Concrete barriers

This chapter describes the scientific research SKB plans to conduct to gain a better understanding of how the function of the engineered barriers of concrete in repositories for low- and intermediate-level waste changes during the time periods dealt with in the safety assessments for the different repositories. The research SKB is conducting to gain a better understanding of other barrier and backfill materials used for low- and intermediate-level waste is presented in Chapter 25 “Buffer and backfill”.

Concrete barriers occur in repositories for low- and intermediate-level waste as floors, walls and covers in the barrier systems that enclose the waste. Some of the concrete barriers contain metal parts such as reinforcement and form ties. An example of what the concrete barrier can look like is shown in Figure 22-1.

The purpose of the concrete engineered barriers is to limit and retard the release of radionuclides. The concrete barriers are therefore ascribed the safety functions “Limited advective transport” and “Good sorption”, see SAR-08. In order to evaluate long-term safety, different performance indicators are linked to the safety functions. The state of the indicators is used in the assessment of the long-term safety of SFR to evaluate whether the safety function is maintained.

The safety performance indicators for the safety function “Limited advective transport” are hydraulic conductivity and temperature. The safety performance indicators for the safety function “Good sorption” are high pH, reducing conditions, low concentrations of complexing agents and available sorption surface area.

The diffusion properties of the concrete are also of importance for limiting and retarding the release of radionuclides. The concrete also has a great influence on the chemical environment in the barriers and the waste.

Plugs are a type of engineered structure installed in tunnels leading to rock vaults for the purpose of limiting the water flow. Plugs are also made in part of concrete. Concrete also occurs as grout around the waste.

Conclusions in RD&D 2010 and its review

In RD&D Programme 2010, SKB described briefly how research on the concrete barriers is organized and that it is based on a process-based methodology.

Newfound knowledge since RD&D 2010

A report is being prepared within SR-PSU that deals with processes related to the engineered barriers in SFR. The report includes process descriptions for the concrete barriers and the processes that have been identified as being relevant to long-term safety for SFR.

Figure 22-1. *Drill core taken from the concrete barrier in the rock vault for intermediate-level waste in SFR. Grains of aggregate and the cement matrix can be distinguished in the picture.*

Programme

The current assessment of the need for research on concrete barriers is presented in this chapter. The conclusions from the safety assessment for the extended SFR, SR-PSU, may have a great influence on the direction and scope of the programme.

22.1 Initial state of concrete barriers

The initial state of concrete barriers is defined as the state that exists in the barriers at closure. In conjunction with closure, the repository's drainage pumping will cease and the repository will fill with water.

The properties of the concrete barriers at closure are dependent on technology decisions regarding the design of new barriers and the maintenance of existing barriers, which is described in Chapter 8 "Technology development for final disposal of low- and intermediate-level waste". The properties of the barriers are also affected by the environment prevailing in the repository during the operating period and possible grouting of the waste. This is determined by operating and deposition procedures.

Conclusions in RD&D 2010 and its review

The research programmes for the initial state of the different variables are described under each individual variable in RD&D Programme 2010.

Newfound knowledge since RD&D 2010

Certain assumptions are made in SR-PSU regarding the properties of the barriers at closure. These assumptions are based on the expected result of the technology decisions described in Chapter 8 "Technology development for final disposal of low- and intermediate-level waste".

Programme

The research programmes for the initial state of the different variables are described under each individual variable.

22.1.1 Variables

The initial state of the concrete barriers is described by a number of variables. An overview of these variables and references to the sections where each individual variable is described are given in Table 22-1.

Table 22-1. Variables for description of the initial state.

Variable	Definition	Section
Geometry	Volume and dimensions.	22.1.2
	Porosity and pore structure of the barriers.	
Temperature	Temperature.	22.1.3
Hydrovariables	Size, direction and distribution of water flow.	22.1.4
	Water saturation, water flows and water pressures.	
	State of aggregation of the water (water or ice).	
Mechanical stresses	Stresses and strains in the barriers.	22.1.5
Material composition	Quantities, composition and surface properties of materials in barriers.	22.1.6
	Type and quantity of chemicals.	
	Type and quantity of organic matter and components that can be used by microbes as nutrients and energy sources.	
Water composition	Composition of water (including radionuclides).	22.1.7
	Redox, pH, ionic strength, concentration of dissolved species, type and quantity of colloids and/or particles, quantity and composition of dissolved gas.	
	Density and viscosity.	
Gas variables	Quantities, composition, volume, pressure and degree of saturation.	22.1.8
	Size, direction and distribution of gas flow.	

22.1.2 Geometry

The geometry of the engineered barriers is determined by 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 how the repository was constructed and any maintenance measures before closure.

In the case of concrete, fracture distribution and fracture geometry are important geometry parameters. Fractures may have been created prior to closure due to volume changes during hardening, drying and re-wetting, as well as reinforcement corrosion.

The pore structure of cementitious materials such as concrete and grout includes gel pores, capillary pores and air pores in increasing order of size. The desired pore structure of a given material is dependent on the desired properties of the material. For example, the embedment grout used in the silo is designed to have an open pore structure in order to permit effective gas transport out of the repository. The structural concrete used in the engineered barriers, on the other hand, is designed to be as dense as possible to ensure that transport of radionuclides through this concrete can only take place by diffusion in a stagnant medium.

As the concrete ages, its pore structure changes. This may entail that the pores grow due to leaching or that they are clogged by mineral precipitation. Processes that can influence pore geometry are dealt with in Section 22.2.15 "Dissolution, precipitation and recrystallization". Most processes that could lead to an altered pore geometry are, however, so slow that they do not affect the initial state of the concrete. A phenomenon of particular interest is therefore the carbonatization and possible densification of the surface layer of the concrete that can occur during the operating period and that transforms portlandite to calcium carbonate via a reaction with carbon dioxide or carbonate ions.

Conclusions in RD&D 2010 and its review

SKB judged that more research was needed regarding the effect of the void and the backfill on hydrovariables.

Newfound knowledge since RD&D 2010

The research programme for judging the status of the concrete structure in SFR is described in Chapter 8 "Technology development for final disposal of low- and intermediate-level waste". Parts of the research programme described in Section 20.1.2 "Geometry" (of the waste) also apply to the geometry of the barriers.

Programme

SKB is currently conducting studies linked to the embedment grout used in the silo in SFR. The work today is mainly focused on technical aspects such as manufacturing methods and methods for evaluation of the hydraulic properties of the grout. The purpose of these studies is to obtain a more homogeneous grout whose properties in the initial state should be easy to determine.

SKB is also currently planning to have a simpler study conducted of the pore structure in the concrete in the rock vault for intermediate-level waste, BMA.

22.1.3 Temperature

The initial temperature in the repository is determined above all by the temperature of the surrounding rock, since low- and intermediate-level waste generates very small amounts of thermal energy due to radiation attenuation.

22.1.4 Hydrovariables

The hydrovariables are water saturation, water pressure and water flows in the concrete barriers. The state of aggregation of the water is also taken into account.

The boundary conditions that control the water flow through the concrete barriers are given by the surrounding pressure field. This field is dependent on the repository's layout and location in the rock, the hydrogeological properties of the rock, the properties of the backfill material in the rock vaults and the external boundary conditions that apply for the flow model. The local flow pattern is also affected by the size, shape and material properties of the concrete barriers. There are no water flows during deposition, since the repository is drained by pumping during the operating phase. When the drainage pumping ceases, the repository will fill with water. However, the degree of water saturation in the barriers at closure is initially low. A certain amount of water can occur initially via penetration from water-bearing fractures in the rock, and the water content of the different repository 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 reinforcement to corrode. The concrete barriers will eventually become water-saturated after closure, once drainage pumping has ceased. A hydrogeological model has been used to calculate how long it takes to fill and saturate the repository with groundwater (Holmén and Stigsson 2001). The calculations show that the void (porosity) inside the silo is saturated last and that this can take 25 years. The time it takes to completely saturate BMA, BLA and BTF (see Chapter 4, Figure 4-1 regarding repository parts in SFR) is only a few years.

22.1.5 Mechanical stresses

Mechanical stresses occur to a varying extent in all types of structures. Both compressive and tensile stresses arise in concrete structures as a consequence of different processes and loads. The floor in a rock vault is exposed to compressive loads from waste, grout and backfill. Walls are loaded by lateral forces, which result in compressive stresses on one side and tensile stresses on the other. These lateral forces can, for example, derive from an internal gas pressure caused by gas production, see Section 22.2.19 "Gas production", and from swelling waste due to water uptake/swelling, see Section 20.2.17 "Water uptake/swelling", and finally from external loads from backfill or groundwater pressure.

Corrosion of steel reinforcement in concrete structures can also lead to considerable local tensile stresses in the material due to the fact that the corrosion products occupy a larger volume than the pure metal, see Section 22.2.18 "Metal corrosion". Tensile stresses can also arise in connection when the structures are built due to hydration (i.e. hardening) of the concrete.

Finally, fall of ground can lead to very great stresses, mainly on the cover of the repository structures, see Section 22.2.10 "Fall of ground".

Of the processes mentioned above, the initial state is mainly affected by pressure from waste on the bottom pad of the concrete structure and pressure on other structural parts from backfill material and the hydrostatic water pressure associated with resaturation after closure. Corrosion of reinforcement in particular during the operating period can lead to local spalling of the concrete cover, but since this can be repaired prior to closure its impact on the initial state is judged to be limited.

Conclusions in RD&D 2010 and its review

In RD&D Programme 2010, this area is only described briefly in the chapter "Engineered barriers in SFR". Reference was made there to the then-planned investigation programme for the state of the concrete structures in SFR, whose purpose was to survey the status of the engineered barriers and other installations after more than 20 years of operation. The viewpoints SSM expressed in its review are presented in the different sections for the above-mentioned processes that can cause mechanical stresses.

Newfound knowledge since RD&D 2010

A status assessment was made of the concrete structures in SFR in 2010 and 2011. The investigations revealed local reinforcement corrosion, which had caused the concrete cover (the layer of concrete outside the outermost reinforcement layer) to spall off in some places. The concrete also exhibited locally elevated chloride concentrations, caused by groundwater seepage. This seepage has probably accelerated the reinforcement corrosion. Penetrating fractures in the concrete structure were also noted during this work, which had probably occurred when the structures were built due to shrinkage during hydration of the concrete. See further Section 8.1 "Final repository for short-lived low- and intermediate-level waste".

Programme

In order to secure the initial state of SFR, SKB is currently devising a programme for repairing damage discovered on the concrete structures, see Section 8.1.

The knowledge gained from the status assessment of the concrete structure is also being put to use in the design and construction of the new concrete structures in the extension of SFR, as well as in the design of concrete structures in the future SFL. Focus areas identified for future facilities include choice of concrete mix design, method of construction, thickness of the concrete cover and monitoring of the climate in the repository during the operating period.

22.1.6 Material composition

Concrete consists of a mixture of cement and aggregate in the form of sand, gravel and crushed rock, where the cement is the binding matrix. Other ingredients are water and in some cases additives such as superplasticizers, water reducers or foam reducing agents. The properties of the concrete are affected by the proportions of cement and aggregate, but also by the water/cement ratio (w/c ratio) used in mixing the concrete.

Different kinds of cementitious materials can be used in the repositories for low- and intermediate-level waste, such as concrete in repository structures or grout for stabilization/solidification or embedment. The chosen composition of these materials is dependent on the desired properties, but there is basically no difference with regard to the main components, cement and aggregate. The main difference between different cementitious materials lies rather in the choice of water/cement ratio and type and amount of different additives.

The chemical composition of the concrete is affected very little during the operating period. The main processes are hydration, which takes a long time, and carbonatization of the portlandite in the surface layer of the concrete, which occurs by reaction with carbon dioxide. In addition, the composition of the material may undergo some changes due to interactions with incoming groundwater.

Conclusions in RD&D 2010 and its review

Concrete composition was dealt with only briefly in RD&D Programme 2010. SSM did not express any viewpoints on this in its review.

Newfound knowledge since RD&D 2010

As a part of the investigation programme that has been carried out regarding the status of the concrete structures in BMA in SFR, a number of analyses have been performed to determine the chloride and carbonate content of the concrete. In addition, measurements have been made of the pH in the concrete in an attempt to determine the present depth of carbonatization.

These studies have shown that the total carbonate content in the concrete is 6.5 ± 0.2 weight percent calcium carbonate per kilogram of concrete, but only very limited carbonatization of the portlandite in the surface layer could be demonstrated. Measurements of the chloride content showed elevated concentrations in the outermost 100 millimetres, which can be attributed to the accumulation of penetrating groundwater on the concrete structures.

Programme

SKB is planning further investigations of the mineral composition and structure of the concrete in BMA. The purpose of these studies is to obtain a more complete body of input data for determining the initial state of the concrete structures. These investigations can then serve as a basis for developing concrete mix designs and design solutions for future repositories.

22.1.7 Water composition

The composition and pH of the water entrapped in the concrete pores (pore water) is affected by the chemical environment, i.e. minerals in the concrete, and the groundwater in the surrounding bedrock. Initially, these properties are determined entirely by the composition of the concrete, and at closure it can be expected that the pore water in the concrete will have the high pH and composition that is characteristic of a hardened concrete.

Conclusions in RD&D 2010 and its review

RD&D Programme 2010 gave only a cursory account of the water composition of the concrete, and a programme was presented for sampling of the concrete in BMA. One of the purposes of this sampling programme was to measure the chloride content of the concrete at certain selected places in the structure. SSM did not express any viewpoints on this in its review.

Newfound knowledge since RD&D 2010

Since RD&D Programme 2010, SKB has taken samples of the concrete in BMA in particular and measured e.g. chloride content, pH and relative humidity in the concrete. The investigations showed elevated chloride contents down to about 100 millimetres and in some cases reduced pH in the surface layer, an indication of incipient carbonatization.

Programme

During the coming period, SKB will continue to evaluate the results of the investigations and explore possible corrective measures.

22.1.8 Gas variables

The gas variables are quantity, composition, volume, partial pressure, degree of saturation, and size and direction of gas flows. There are no gases in the concrete structure initially. They start to form when the deposited waste begins to degrade and the concrete reinforcement begins to corrode.

At present, SKB has no programme for determining gas composition and gas flows in the initial state in the concrete barriers. For a research programme concerning gas formation processes that could affect long-term safety, see Section 22.2.19 “Gas production”.

22.2 Processes

A number of processes will eventually alter the state of the concrete barriers and their cavities. Certain processes occur under all conditions, while others are only possible under special conditions.

This section describes the processes that can be expected to affect the concrete barriers during the lifetime of the repository. The description covers both SFR and SFL, since most of the processes expected to occur can be regarded as independent of repository type. In cases where a specific process differs substantially between the two repositories, it is dealt with specifically for each repository.

22.2.1 Overview of processes

The processes that affect conditions in the concrete barriers can be divided into five main classes: thermal processes, hydraulic processes, mechanical processes, chemical processes and radionuclide transport. There may be a number of different processes under each main class that interact with one another or with other processes.

Thermal processes

The effect of the low- and intermediate-level waste on the temperature in the repository is negligible under normal circumstances. This assessment is based on the knowledge that the heat-generating processes in the low- and intermediate-level waste, such as radioactive decay (Sections 20.2.2 and 21.2.2) and radiation attenuation and heat generation (Sections 20.2.3 and 21.2.3), are very limited. The temperature of the repository will therefore be determined by the temperature of the surrounding rock.

As long as the temperature is above the freezing point for the concrete pore water, the influence of the temperature on the properties of the concrete barriers is negligible. However, freezing of the concrete during a period when permafrost reaches repository depth would cause the concrete to fracture and diminish its capacity to retard radionuclide transport out of the repository (Emborg et al. 2007).

The following thermal processes are dealt with:

- Heat transport, Section 22.2.2.
- Phase change/freezing, Section 22.2.3.

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 mainly determined by the surrounding groundwater flow. The groundwater flow changes slowly over time as a function of e.g. land rise and climate evolution.

The following hydraulic processes are dealt with:

- Water uptake and transport under unsaturated conditions, Section 22.2.4.
- Water transport under saturated conditions, Section 22.2.5.
- Gas transport and water solubility, Section 22.2.6.

Mechanical processes

During the lifetime of the repository, the concrete barriers will be subjected to both internal and external mechanical forces that will give rise to mechanical stresses. As long as the strength limits of the concrete are not exceeded, these processes will not have a negative effect on its retention capacity. If these limits are exceeded, however, fractures may occur in the concrete and its capacity to limit releases of radionuclides may be affected.

The following mechanical processes are dealt with:

- Pressure from swelling waste, Section 22.2.7.
- Pressure from bentonite, Section 22.2.8.
- Fracturing, Section 22.2.9.
- Fall of ground, Section 22.2.10.

Chemical processes

During the lifetime of the repository, the properties of the concrete barriers will be affected by chemical interactions with surrounding groundwater or substances dissolved in the groundwater. The concrete properties may also be affected by changes in the waste due to similar chemical reactions, see Sections 20.2 and 21.2.

Certain processes affect the properties of the concrete barriers directly and concretely by causing fracturing and opening up flow paths. Other processes affect the ability of the concrete barriers to retain the radionuclides by creating complexing agents or changing the pore water composition in the concrete.

The following chemical processes are dealt with:

- Advection and mixing, Section 22.2.11.
- Diffusion, Section 22.2.12.
- Sorption, Section 22.2.13.
- Colloid transport and filtration, Section 22.2.14.
- Dissolution, precipitation and recrystallization, Section 22.2.15.
- Pore water speciation and concrete interactions, Section 22.2.16.
- Microbial processes, Section 22.2.17.
- Metal corrosion, Section 22.2.18.
- Gas production, Section 22.2.19.

Radionuclide transport

The following radionuclide transport processes are dealt with in this chapter:

- Speciation of radionuclides, Section 22.2.20.
- Radionuclide transport in the aqueous phase, Section 22.2.21.
- Radionuclide transport in the gas phase, Section 22.2.22.

22.2.2 Heat transport

Under normal conditions, the repository temperature is determined by the temperature of the surrounding rock, while the influence of the low- and intermediate-level waste on the temperature in the repository is deemed to be negligible. This means that when the temperature on the ground surface above the repository changes, this will, with some delay, affect the temperature of the concrete. Under normal conditions, the properties of the concrete are not affected by temperature changes, but if the repository is reached by permafrost it could burst due to freezing, affecting its ability to retard radionuclide transport (Emborg et al. 2007).

Conclusions in RD&D 2010 and its review

Heat transport in the concrete barriers was described only briefly in RD&D programme 2010. SSM did not express any viewpoints on this in its review, but called for a better body of data to assess the risk and scope of the damage to cement barriers that could be caused by freezing, see further Section 22.2.3 "Phase change/freezing".

Newfound knowledge since RD&D 2010

SKB has not conducted any research concerning heat transport in the concrete barriers during the period in question.

Programme

SKB is not planning to conduct any research concerning heat transport in concrete during the coming period, but is following international research in the area.

22.2.3 Phase change/freezing

The frost resistance of the concrete is dependent on its pore structure and degree of water saturation. The gel pores and capillary pores fill quickly with water in a normal outdoor climate. After a long period of water absorption, the larger air pores 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 (Emborg et al. 2007).

The freezing point of water in the concrete declines with diminishing pore size. For example, water does not freeze in a pore with a diameter of 150 angstroms until –20°C. At the normal freezing temperature, the water in the gel pores and in the finest capillary pores is therefore still not frozen. The larger the fraction of the total pore water that remains unfrozen at a given temperature and the larger the fraction of air pores that are present in the concrete, the smaller is the risk that the concrete will be damaged at temperatures below 0°C.

However, the moisture loading time in a geological repository is so long that all pores in the concrete will have become water-filled before permafrost reaches repository depth. The exact temperature at which the concrete freezes is, however, dependent on its pore structure, which is in turn dependent on how the concrete has been prepared as well as its age and the chemical processes it has previously been exposed to.

Conclusions in RD&D 2010 and its review

The then-fresh results of the theoretical study of freezing of concrete in SFR were reported in RD&D Programme 2010 (Emborg et al. 2007). The main conclusion of this study was that the concrete in SFR will burst at a temperature of between 0 and –5°C and that the concrete could, after a freeze-thaw cycle, be expected to have a structure similar to gravel.

In its review, SSM called for an improved body of data for assessing the risk and scope of damage to cement barriers caused by freezing.

Newfound knowledge since RD&D 2010

Experiments involving freezing of concrete cores from BMA were conducted in 2011 (Thorsell 2011). The investigation showed that the concrete bursts at between –3 and –7°C. The overwhelming majority of the test bodies burst at –5°C, and after freezing the concrete took on the form of gravel, Figure 22-2.

SKB has also had another modelling exercise done on the consequences of freezing of concrete in a geological repository. Here the main focus was on the concrete types used in the construction of BMA and the silo. However, the results can be regarded as generally applicable, since the concrete used to build BMA and the silo can be considered to be of a standard type.

In this study (Luping and Bager 2013), the general conclusions are that the concrete used in the structures in SFR will not burst in the manner described by Emborg et al. (2007). The reason for this is that the concrete is expected to be under external pressure from water that has frozen at a lower temperature in surrounding, more porous materials.

Figure 22-2. *Concrete sample from BMA after freezing at –5°C.*

Programme

The experimental investigation of concrete samples from BMA (Thorsell 2011) can be considered to confirm the conclusions drawn in Emborg et al. (2007). Both of these studies have, however, dealt with very young concrete. In reality, the concrete will have been subjected to chemical transformation by leaching and reactions with dissolved waste and the surrounding groundwater for several thousand years before it freezes for the first time. These processes, which are described in Section 22.2.15 "Dissolution, precipitation and recrystallization", will affect the pore structure in the concrete and thereby also its freezing properties.

A research project has been under way since 2010 in cooperation with Chalmers University of Technology for the purpose of investigating the chemical and mechanical properties of aged concrete, see also Section 22.2.15. The objective of this project is to first develop a method for accelerated ageing of solid cement specimens, and then follow this up with a study of their properties after different periods of ageing. The possibility of conducting measurements of the freezing properties of the aged material is being discussed in the project.

The conclusions in Luping and Bager (2013), which point to a possibility that concrete under pressure might not be affected by freezing in the way described in Emborg et al. (2007), warrant some attention. In order to try to verify the conclusions drawn by Luping and Bager (2013), an experimental investigation of freezing of concrete under pressure is therefore being conducted. The work is an introductory study and has been conducted in the form of a degree project at Lund University that was presented in 2013.

22.2.4 Water uptake and transport under unsaturated conditions

At repository closure, the degree of water saturation in the concrete is low and the water flow in the barriers is negligible. Some water may be present initially due to penetration from water-bearing fractures in the rock or condensation. The presence of water affects the chemical process in the concrete, such as the tendency of the concrete reinforcement to corrode. The concrete in the barriers will become water-saturated after closure. According to calculations, it takes 25 years to saturate the least permeable repository part in SFR (Holmén and Stigsson 2001). The time it takes to completely saturate other repository parts is considerably shorter.

No further research, development or demonstration is considered to be needed in this field today. New developments in the field are being monitored and will be acted on when appropriate.

22.2.5 Water transport under saturated conditions

The water flow through concrete barriers is determined by the pressure gradient, the geometry of the barrier and the permeability of the concrete, which is described by its hydraulic conductivity. Concrete of high quality has low conductivity and thus serves as an effective flow barrier. Processes that degrade the concrete also affect its hydraulic conductivity. The formation of penetrating fractures creates local flow paths, which can have a great influence on the effective conductivity of the barrier. Bursting of the concrete due to freezing is another example of high conductivity impact.

Newfound knowledge since RD&D 2010

Method and model development for calculations of water flows in the repository's near-field have been carried out within the framework of the safety assessment for SR-PSU. The models describe the existing SFR and the extension with a refined geometric description of repository parts, concrete and bentonite barriers, waste and backfill, see Figure 22-3. Calculations have been performed to analyze the water flow as a function of gradual degradation of the concrete barriers in the repository and further how this influences the transport of radionuclides out of the repository.

Programme

Development of flow models, which include degraded barriers, is ongoing. Local degradation effects, such as the occurrence of fractures or areas of substantially altered permeability, will be further studied. Since degradation processes that affect concrete are often controlled by chemical

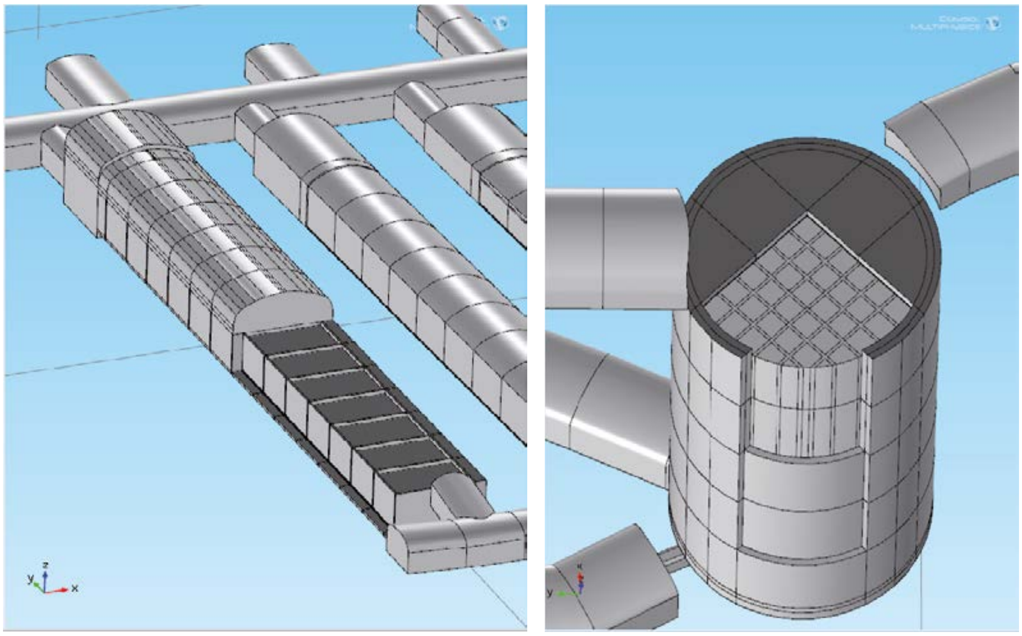


Figure 22-3. *Geometric description of BMA and silo in the flow model for the repository's near-field. Certain sub-geometries on top of the repository parts have been removed to clearly show the discretization of the barriers in the repository parts.*

reactions, it is deemed to be of interest to handle flow and reactive mass transport interlinked in the same model, with a common geometric description of repository barriers. Moreover, a study is being initiated to interlink established calculation tools for flow modelling and geochemical modelling.

22.2.6 Gas transport and water solubility

Gas formation processes that begin when SFR has been closed may lead to build-up of gas pressure in the concrete barriers. Gas will dissolve in the concrete pore water until it is saturated. Solubility is dependent the partial pressure of the different gases (linearly correlated at equilibrium). Dissolved gas can be transported by advection and diffusion, see Section 20.2.20 "Gas formation and gas transport".

Since the outflow of gas that is formed by corrosion of concrete reinforcement is limited when the pores in the concrete barrier are water-saturated, gas will accumulate and an internal gas pressure will be built up.

When the pressure exceeds the capillary force in the pores, water will be displaced until a gas phase has been formed that can then be transported. Passages for gas transport can form in the barrier and gas can flow out as long as the pressure difference exceeds the capillary pressure. However, gas can also be transported in the form of a "bubble flow", i.e. without the formation of a cohesive gas phase (Neretnieks and Ernstson 1997).

In BMA, evolved gas can be transported out of the repository through small fractures in the concrete or through joints between the concrete components. In the silo, gas can be transported out through sand-filled gas evacuation pipes in the lid.

SKB does not currently have a separate research programme for gas transport and water solubility in concrete barriers. However, a specific programme may be formulated as a result of the coming assessment of long-term safety in SFR.

22.2.7 Pressure from swelling waste

Water uptake in hygroscopic waste can cause swelling, which is described in Section 20.2.17 "Water uptake/swelling". If insufficient expansion volume is provided, such waste can generate pressure that can exert a mechanical force on waste containers as well as surrounding concrete barriers.

Newfound knowledge since RD&D 2010

The mechanical force exerted by swelling of bitumen-stabilized waste on surrounding concrete barriers in BMA and the silo has been studied within the framework of SR-PSU. The effects were analyzed by means of structural mechanical calculations in the finite element software Comsol Multiphysics. It was concluded that for BMA, an expansion volume (in addition to the one provided by the waste package) was necessary to ensure that concrete barriers are not damaged. In the case of the silo, only the expansion volume inside the waste package was assumed to be available. The corresponding steady-state swelling pressure was set as a boundary condition. The calculation model showed that the pressure did not create any harmful stresses in the outer concrete walls of the silo.

Programme

No further research, development or demonstration is considered to be needed in this field today. New developments in the field are being monitored and will be acted on when appropriate, for example if new waste types are developed.

22.2.8 Pressure from bentonite

Bentonite swells in the presence of water and will, when confined, exert pressure on its surroundings. The silo is currently the only repository structure where bentonite is in direct contact with a concrete barrier and can exert mechanical force. When the repository has been filled with water and the bentonite has become saturated with water, a swelling pressure on the order of 100 kilopascals is expected. The pressure acts radially against the concrete in the outer walls of the silo, whose structure is designed to absorb loads of this kind. In relation to the compressive strength of undegraded concrete, the pressure from the bentonite is very moderate and is not expected to cause mechanical damage. Viewed over a long time, however, the concrete may fracture and be degraded by chemical processes. Its strength properties may then finally be reduced so that the pressure from the bentonite further degrades the concrete.

Programme

The pressure that is built up when the silo buffer swells will be modelled as a part of SR-PSU. The buffer in the silo will be sampled to make updated determinations of its hydromechanical properties.

22.2.9 Fracturing

The principal functions of the concrete barriers are to minimize advective and diffusive flow of water through the waste and effectively retard releases of radionuclides via sorption. The radionuclides are sorbed on the amorphous and crystalline minerals in the concrete. The flow in a fracture-free structure will be diffusive and sorption of radionuclides will be effective due to a large contact surface area with the concrete components. The flow in a fractured structure will, however, be dominated by advective flow and sorption will be reduced due to the smaller contact surface area between water and the concrete components.

A number of processes can lead to the occurrence of fractures in the concrete barriers, the most important of which are:

- Water uptake/swelling of the waste, Sections 20.2.17 and 21.2.17.
- Phase change/freezing, Section 22.2.3.
- Dissolution, precipitation and recrystallization, Section 22.2.15.
- Metal corrosion, Section 22.2.18.
- Gas production, Section 22.2.19.

Fractures in the concrete can also be created during construction when the concrete hydrates (hardens).

The prerequisites for fracturing and the mechanisms behind the different processes that can lead to fracturing are discussed in the above-mentioned sections.

Conclusions in RD&D 2010 and its review

Fracturing in the concrete barriers was described briefly in RD&D Programme 2010 with reference to those sections where the processes that could lead to fracturing were described.

In its review, SSM thought that SKB needed to develop the safety assessment so that it took into account slow and gradual deterioration of the concrete barriers due to fracturing and degradation.

Newfound knowledge since RD&D 2010

Newfound knowledge on the processes that can lead to fracturing in the concrete barriers is presented in the relevant sections, see above.

Programme

SKB is not currently planning a separate research programme concerning fracturing in concrete barriers. The programme for research concerning fracturing in concrete barriers is instead presented in the relevant sections, see above. For programmes of research on the effects of fracturing in the concrete barriers on their transport properties, reference is made to Sections 22.2.4 "Water uptake and transport under unsaturated conditions" and 22.2.5 "Water transport under saturated conditions".

22.2.10 Fall of ground

The rock vaults in SFR were constructed using conventional blasting and rock support measures during the period 1983–1986. The rock support consists of fully embedded rock bolts and shotcrete. The rock support is inspected regularly and reports are made every five years to SSM. No significant ageing tendencies had been observed up to and including 2011. Minor measures in the form of scaling and supplementary shotcreting have, however, been adopted during the operating period in those parts of the construction tunnels that were left unsupported. Monitoring of rock movements with extensometers in the silo and at the points where the access tunnels are intersected by the Singö Zone (a major zone of weakness in the rock) have not revealed any rock deformations during the operating period. SKB judges that the rock surrounding the vaults has good bearing capacity and that the rock support is still in good condition.

In the supporting material accompanying the applications for the extension of SFR, the stability of the rock vaults is judged to be marginally affected by the fact that support elements such as rock bolts and shotcrete no longer retain their loadbearing capacity. When the loadbearing capacity, in particular of the rock bolts, has declined it can be expected that occasional rock blocks can come loose and fall into the rock vaults. Backfilling with crushed rock protects the concrete barriers from any blocks that fall from the roof and walls of the rock vault.

Newfound knowledge since RD&D 2010

A rock mechanical three-dimensional simulation of the stability of the rock has been done for existing rock vaults (BLA and BMA). The study will be included as supporting material for the applications for extension of SFR. The geometries of the rock vaults and the geological documentation on minor deformation zones and the distribution of fractures compiled during the construction period have been used, see Figure 22-4. The simulation was performed in several steps with progressively reduced strength parameters for the fractures. No rock support was taken into consideration in the modelling. The analysis shows that the rock vaults are stable even with the assumption that there is no cohesion and the friction angle is reduced to less than 15°. The good stability can be attributed to the rock stress situation, along with the shape and location of the rock vaults in relation to the dominant fracture systems. Falling of individual blocks cannot be ruled out, however.

SKB does not plan to carry out any more research on long-term stability during the coming RD&D period.

22.2.11 Advection and mixing

Solutes can be transported in and out through the concrete barriers by advection, dispersion and diffusion. In this context, advection is a transport process where a solute accompanies the bulk water flow. Advection is given by the flow (Darcy flow) divided by the porosity. Porosity is therefore important when advection is described. The water flow through concrete barriers is determined by the pressure gradient and by the hydraulic conductivity of the concrete, see Section 22.2.5 "Water transport under saturated conditions".

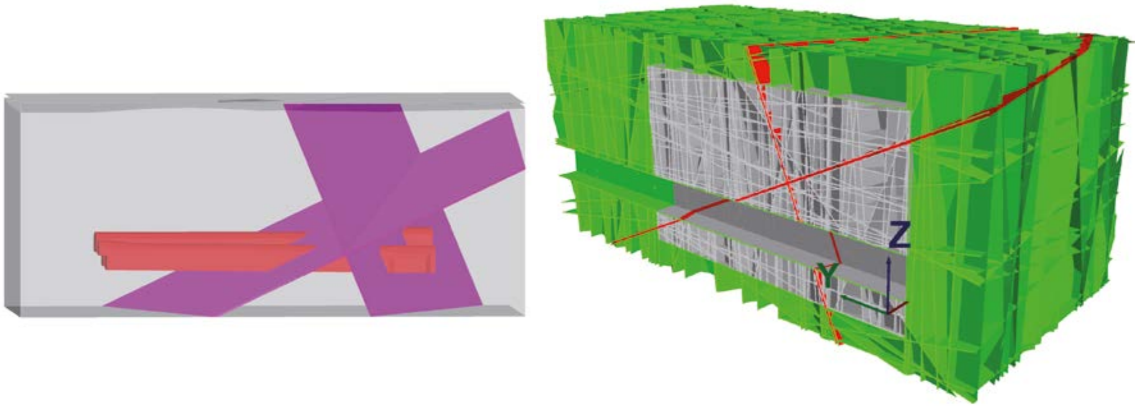


Figure 22-4. Numerical model for simulating degradation of the rock mass around BLA and BMA. The minor deformation zones observed during the construction period are shown at the left. A section through one of the eight realizations with stochastically distributed fractures is shown at the right. Red = minor deformation zones, grey box = more detailed fracture network nearest the rock vaults.

In an intact concrete barrier, the hydraulic conductivity is so low that all transport of solutes takes place by diffusion. The low hydraulic conductivity is defined as one of the barrier's safety functions, see Section 18.3 "Assessment of long-term safety of SFR". It is therefore very important to understand the processes that influence the properties of the concrete. This is described in e.g.:

- Phase change/freezing, Section 22.2.3.
- Fracturing, Section 22.2.9.
- Dissolution, precipitation and recrystallization, Section 22.2.15.
- Pore water speciation and concrete interactions, Section 22.2.16.
- Metal corrosion, 22.2.18.

How the advective transport of radionuclides is handled in safety assessments is described in Section 20.2.22.

Programme

See Section 22.2.5 "Water transport under saturated conditions".

22.2.12 Diffusion

In a well-functioning concrete barrier, diffusion will be the dominant transport mechanism. In a concrete repository where diffusion is the dominant transport mechanism, the outward transport of radionuclides will be low.

Effective diffusivity in concrete is an important parameter both for radionuclide transport (Section 20.2.22) and for the description of concrete degradation, Section 22.2.15.

Programme

In general, uncertainties in diffusivities are handled by pessimistic data or sensitivity analyses. No specific programme is planned for determining diffusivity in concrete.

22.2.13 Sorption

See Section 20.2.13 "Sorption".

22.2.14 Colloid transport and filtration

See Section 20.2.14 "Colloid formation and colloid transport".

22.2.15 Dissolution, precipitation and recrystallization

Chemical compounds such as minerals and organic additives in the concrete 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. What reactions take place and what chemical equilibria are established in the concrete are dependent on the following factors: the concentrations of substances in the concrete and its pore water, substances dissolved in the groundwater, and solutes from bentonite barriers, waste containers, the waste and waste matrices.

As the concrete ages, its pore structure also changes. The pores may grow due to leaching or they may be clogged by mineral precipitates. What effects this transformation has on long-term safety depends on the desired properties of the material. For example, clogging of the external (embedment) grout can lead to reduced gas transport and a build-up of pressure inside the structure. On the other hand, clogging of the pores in the concrete barriers can reduce the inflow of water through them, reducing releases of radionuclides.

Changes in the pore structure can also lead to changes in the freezing properties of the concrete, since the freezing temperature of the water is affected by the size of the pore in which it is enclosed, see Section 22.2.3 “Phase change/freezing”. Finally, changes in the pore structure can lead to fracturing when swelling minerals are deposited in pores or small fractures. The effects of changes in the pore structure of cementitious materials on their transport properties is discussed in Section 22.2.4 “Water uptake and transport under unsaturated conditions” and in Section 22.2.5 “Water transport under saturated conditions”.

Many phase transformations can occur without having any significant impact on the integrity of the repository. Others, on the other hand, such as formation of ettringite or thaumasite, are in some cases believed to have a negative effect on the strength of the concrete structures.

Ettringite can be formed by a number of different processes, see e.g. Skalny et al. (2002). However, ettringite and thaumasite formation always include a reaction between sulphate from either an external or an internal source and various minerals in the cement. In sulphate-resistant concrete, which is the type used in the construction of SFR, the amount of tricalcium aluminate (C3A) is limited, since a high concentration of C3A is considered to constitute a risk factor for ettringite formation.

There are two possible sources of sulphate for a concrete barrier in a geological repository: sulphate-containing groundwater and internal sources in the concrete. The sulphate concentration in the concrete is not judged to comprise a risk factor in SFR, since it is deemed to be low. Degradation of sulphonate ion exchange resins has been proposed as a possible process that could contribute some sulphate, but this process has not yet been fully clarified, so no conclusions can be drawn regarding its contribution to ettringite formation.

Conclusions in RD&D 2010 and its review

These processes are dealt with under the heading “Chemical cement and concrete degradation” in RD&D programme 2010. SKB reported there that two projects had been initiated for studies of ageing of concrete and interactions of concrete with its surroundings and embedded waste.

- The project “Long-term durability cement” includes three parts: two experimental parts and one modelling part. The sub-project “Concrete and Clay” is being carried out in the Äspö HRL and aims at studying interactions between cement and different waste forms as well as between cement and surrounding materials.
- The project “Chemical and mechanical properties of aged cement” is being conducted in cooperation with Chalmers University of Technology. The purpose of this project is to develop a method for accelerated leaching of solid, fairly large cement and concrete specimens and study the properties of the specimens as a function of degree of leaching.

In its review, SSM asserted that SKB needed to develop the safety assessment so that it could take into account slow and gradual deterioration of the concrete barrier due to fracturing and degradation. SSM was also of the opinion that SKB should better substantiate its argumentation concerning the reasons for ruling out certain concrete degradation processes such as interaction between cement and various components in the waste. SSM also wanted SKB to clarify certain questions linked to the risk of earthquakes, such as that chemical transformation can affect the strength properties of the concrete and thereby its resistance to quake movements.

Newfound knowledge since RD&D 2010

A licenciate thesis was presented in 2013 within the project "Chemical and mechanical properties of aged cement" (Babaahmadi 2013). The method that has been developed entails that cement is leached electrochemically with lithium hydroxide and ammonium nitrate. In this way an acceleration factor of about 600 has been obtained compared with non-accelerated leaching.

SKB has also studied interactions between cement and bentonite, mainly linked to sealing of boreholes but also applicable to concrete barriers in contact with bentonite such as in the silo in SFR (Pusch and Ramqvist 2011). The conclusions in this report are that the swelling properties of bentonite changed somewhat after three years' contact with concrete, at the same time as the strength of the concrete was affected.

Furthermore, since 2010 SKB has initiated international collaboration in the project "Long-term cement studies", LCS, headed by Nagra of Switzerland. The main purpose of this project is to improve the general understanding of the mechanisms involved in interactions between cement, water and solutes dissolved in the water as well as between cement and the bedrock. This is being done by a combination of new experiments in Nagra's underground laboratory in Grimsel, studies of natural analogues and modelling studies. The background of the project and the experiments being conducted in Grimsel are described in Rüedi (2010).

A test unit for studies of interactions between injection grout and granite was recovered within LCS in 2012. The experiment had two main purposes: to test and evaluate the methods for retrieval and analysis of remaining specimens deposited in the rock, and to investigate the interaction between cement and granite in the bedrock. The study showed that the methods used worked well, albeit not completely optimally, and that some, albeit very little, mineral alteration had occurred in the cement grout in the interface with the bedrock (Rüedi et al. 2012).

Finally, SKB has had a study conducted in order to be able to assess, by means of thermodynamic modelling, the long-term function of the concrete barriers at elevated carbonate concentrations, see Section 20.2.15 "Dissolution, precipitation and recrystallization".

Programme

SKB's research programme concerning dissolution, precipitation and recrystallization of concrete barriers includes a number of areas.

Additional specimens will be deposited in the Äspö HRL during the coming RD&D period within the "Concrete and clay" project. In this phase the focus is on experiments for studying interactions between concrete and other possible repository components such as crushed rock, bentonite, bedrock and solutes in the groundwater. These experiments will largely be designed for long-term deposition in the rock and are not planned to be analyzed until retrieval. SKB also envisions the possibility of conducting monitored experiments where changes in the properties of the materials are continuously studied. Studies of how a pH plume propagates through the bedrock may also be conducted, but this has lower priority at the present time.

The project "Chemical and mechanical properties of aged cement" is planned to continue until the autumn of 2015, when it will be concluded with a Ph. D. dissertation. With the method that has now been developed, sufficiently large specimens can be fabricated to be used in studies of mechanical properties. The planned work includes studies of how the strength properties of the material are affected when portlandite is leached out, as well as how the pore structure and freezing properties of the material are altered by leaching. At present, specimens are being produced and further plans are being made for the remaining project period.

The focus in the LCS project in the next few years is on retrieval, analysis and modelling of the experiments in Grimsel. Aside from this, a sub-project is proceeding within LCS in cooperation with Swiss Federal Laboratories for Materials Science and Technology, EMPA, to update and gather new data on cement-related mineral phases and thereby improve the body of input data for thermodynamic modelling of degradation of cement.

22.2.16 Pore water speciation and concrete interactions

Initially the pore water in the concrete will have a pH of 13 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 reinforcement to corrode.

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. How this process is affected by the waste is described more fully in Section 20.2.15 "Dissolution, precipitation and recrystallization".

Conclusions in RD&D 2010 and its review

RD&D Programme 2010 did not describe the process explicitly, but it was taken up under the section "Pore water composition in the engineered barriers for SFR". SSM had no viewpoints on that section in the RD&D programme.

Newfound knowledge since RD&D 2010

The programmes described in RD&D Programme 2010 concerning quantification of the chloride concentration in the structural concrete in BMA have been concluded. The study indicates that the chloride concentration can locally be higher than the chloride concentration in the penetrating groundwater.

Programme

Calculations for pore water speciation will be carried out within the framework of SR-PSU. The thermodynamic modelling will take into account the cement composition at different leaching stages as well as the composition of the penetrating groundwater at different points in the future evolution of SFR.

22.2.17 Microbial processes

See Section 20.2.18 "Microbial processes".

22.2.18 Metal corrosion

Metal that is embedded in concrete structures such as reinforcement, form ties and other items such as ladder stabilizers will corrode during the life of the repository. The rate of this corrosion process and its effects are dependent on the surrounding environment, but also on technical details such as the concrete cover and the choice of concrete quality. A thicker concrete cover provides better protection against reinforcement corrosion than a thin one, since it affects the time of initiation of a number of processes. For example, the time before chloride-containing groundwater or the carbonatization front reaches the reinforcement is prolonged by a thicker concrete cover. This prolongs the time before the passivity of the iron is overcome and the corrosion rate increases.

The effects of reinforcement corrosion are dependent on the corrosion rate. At very low corrosion rates, the corrosion products have time to be carried away and the process then leads to a reduction in the cross-sectional area of the reinforcement. At higher corrosion rates, corrosion products that occupy a larger volume than the pure metal can instead form on the steel items. They may then create a higher internal mechanical pressure, which can in turn lead to fracturing of the structures, mainly in the form of local spalling of the concrete cover (Betongföreningen 2007).

In addition to the direct effect of reinforcement corrosion on the load-bearing capacity of the concrete structure, the anaerobic corrosion process can also lead to the formation of hydrogen gas. If the corrosion rate is high, this can in turn lead to the build-up of a gas pressure inside the repository structures.

Conclusions in RD&D 2010 and its review

RD&D Programme 2010 gave only a brief presentation of the research on corrosion of metal embedded in the concrete barriers, with reference to the research on metal corrosion in the short-lived low- and intermediate-level waste. SSM did not offer any specific comments on this in its review.

Newfound knowledge since RD&D 2010

Since RD&D Programme 2010, SKB has updated its knowledge concerning corrosion of metals, primarily iron and steel, in the repository environment, see Section 20.2.19 "Metal corrosion". In this work, which has been carried out within SR-PSU, it has been found that the corrosion rates previously assumed for an alkaline oxygen-free repository environment have probably been overestimated.

These new data will also be considered to apply to the long-lived low- and intermediate-level waste, as well as to metal embedded in the concrete barriers, since they will be exposed to the same type of environment as the waste in SFR.

Programme

SKB intends during the coming period to coordinate the research on metal corrosion and its consequences in a joint programme including short- and long-lived waste as well as corrosion of metal embedded in the concrete barriers. SKB's programme for research on corrosion is presented in Section 20.2.19 "Metal corrosion".

22.2.19 Gas production

The main source of gas formation in the concrete barriers is anaerobic metal corrosion of reinforcement and other metallic components that are embedded in the concrete barriers, such as form ties and stabilizers for ladders or other items. Another possible source of gas production is degradation of organic additives to the concrete, such as superplasticizers or stabilizers, where the cellulose in the embedding grout for the silo can serve as an example. The quantity of gas that can be formed by chemical degradation of these additives is, however, small in relation to the amount that can be formed by metal corrosion and is therefore not deemed to be of crucial importance to long-term safety.

Conclusions in RD&D 2010 and its review

This process was not described explicitly in RD&D programme 2010, but was instead handled within the research on metal corrosion of short-lived low- and intermediate-level waste. SSM did not express any viewpoints on this in its review. Other comments from SSM concerning the research on gas formation are presented in Section 20.2.19 "Metal corrosion".

Newfound knowledge since RD&D 2010

SKB has not conducted any research on gas formation in concrete barriers. However, the literature review conducted within SR-PSU concerning corrosion rates for metals in different environments has a bearing on this area as well, see Section 20.2.19 "Metal corrosion".

Programme

SKB does not intend to conduct any research aimed solely at the area of gas production in the concrete barriers during the coming period. For research programmes concerning underlying processes, see Section 20.2.16 "Chemical degradation of organic compounds" and Section 20.2.19 "Metal corrosion".

22.2.20 Speciation of radionuclides

Speciation of radionuclides in the concrete barrier is similar to that in the waste, which is described in Section 20.2.21.

22.2.21 Radionuclide transport in the aqueous phase

Handling of radionuclide transport is presented in Section 20.2 "Processes".

22.2.22 Radionuclide transport in the gas phase

Handling of radionuclide transport is presented in Section 20.2 "Processes".

23 Fuel

Most of the nuclear fuel that will be deposited in the Spent Fuel Repository consists of spent nuclear fuel from the operation of the twelve Swedish reactors in Forsmark, Ringhals, Oskarshamn and Barsebäck. These are either boiling water reactors (BWRs) or pressurized water reactors (PWRs). SKB will also manage the smaller quantities of fuels of other types that are interim-stored in Clab, for example MOX fuel. In SKB's safety assessment, SR-Site, the quantity of spent fuel is estimated according to a reference scenario (SKB 2010o). It includes the spent fuel stored in Clab today, plus that arising during the planned operation of the ten reactors still in operation. In the reference scenario, the operating times of the four reactors in Ringhals and the three in Forsmark are set at 50 years, while the operating times of the three reactors in Oskarshamn are set at 60 years. The two reactors in Barsebäck were closed after about 24 and 28 years of operation. The planned operating times for the reactors in Forsmark and Ringhals 3 and 4 have since been increased to 60 years.

Most of the fuel used in the reactors consists of uranium oxide fuel. There are also smaller quantities of mixed oxide fuel, MOX, expected to be used in Oskarshamn. Quantities and more details concerning spent nuclear fuel can be found in SKB (2010d, p 15).

Differences in radionuclide content between PWR and BWR fuel are marginal viewed from a safety assessment perspective. After interim storage in Clab, MOX fuel has a higher decay heat than uranium fuel, which means that less fuel can be deposited in each canister or that a longer decay time is needed.

23.1 Initial state

The initial state of the fuel describes the properties the spent fuel is expected to have when it has been placed in a sealed canister and deposited in a deposition hole. The technological development that is planned for management of the spent nuclear fuel in accordance with final disposal requirements is described in Chapter 11.

23.1.1 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 23-1 (next page).

23.1.2 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. The gap inventory is important to estimate, since it can give rise to pulse releases, also known as the instant release fraction (IRF). Fission gases (for example krypton or xenon) 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 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 (FGR). The FGR is important to know and calculate, mainly for safe operation of the reactors. Information on FGR is used in SKB's safety assessments to estimate the gap inventory of certain segregated and mobile fission products.

Table 23-1. 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 alpha, beta, gamma 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.	No programme. Knowledge sufficient.
Gap inventory	Occurrence of radionuclides as a function of time and space in gaps and grain boundaries.	Section 23.1.2
Material composition	The materials of which the different components in the fuel assembly are composed, excluding radionuclides.	No programme. Knowledge sufficient.
Water composition	Composition of water (including any radionuclides and dissolved gases) in the cavities in the fuel and canister.	See "Gas composition", Section 23.1.3
Gas composition	Composition of gas (including any radionuclides) in the cavities in the fuel and canister.	Section 23.1.3

Conclusions in RD&D 2010 and its review

SKB's safety assessments use information on fission gas release for fuel from light water reactors (BWRs and PWRs) with a burnup of up to 60 MWd/kgU (megawatt-days per kilogram of uranium) and MOX fuel with a burnup of up to 50 MWd/kg HM (megawatt-days per kilogram of heavy metal) in order to estimate the gap inventory of certain segregated and mobile fission products. In view of the limited quantity of data on fuel with a burnup of up to 60 MWd/kgU, the calculated FGR for typical Swedish BWR fuel (Oldberg 2009) and PWR fuel (Nordström 2009) was used to estimate the gap inventory, for e.g. iodine and cesium. These calculations take into account the reactor's linear power, since the linear heat ratingaffects the FGR more than fuel burnup, although the two are often related. The plans for uprating the Swedish reactors entail an expected increase of the FGR in the future.

In its review of RD&D Programme 2010, SSM thought that SKB should examine the evolution of the HBU (High BurnUp) structure and determine whether releases of fission products in particular will increase substantially after a late canister failure and whether this could lead to a substantially higher pulse release. The question of ASIED (Alpha Self-Irradiation Enhanced Diffusion) should also be further studied, according to SSM.

Newfound knowledge since RD&D 2010

The study of release of segregated radionuclides from PWR fuel segments and fuel fragments with high burnup and measured FGR is continuing and is being supplemented with analyses of other radionuclides, such as iodine-129 and selenium-79. Results of more than one year of leaching of four PWR segments have been published during the period (Zwicky et al. 2011). Data on releases of IRF nuclides such as cesium-137 and iodine-129 have been published in a separate study (Ekeroth et al. 2012). The results show that the method used to prepare the fuel specimens is important. This is particularly true of high-burnup fuels, where the fuel-clad gap is usually closed. When specimens are prepared, the cladding can either be removed or left in place. Removing the cladding exposes the gap to the aqueous solution and the gap inventory is released more quickly than if the cladding remains in place. In the study, fuel with a burnup of between 43 and 75 MWd/kgU and an FGR of between 0.9 and 5 percent was investigated. The results showed that the ratio between iodine-129 and FGR was nearly one for the whole burnup interval, which agrees with what can be expected based on the diffusion coefficients for iodine and xenon in uranium dioxide (UO_2). The ratio between cesium-137 and FGR approaches a value of 0.6, which corresponds to the ratio between the diffusion coefficients for cesium and xenon in uranium dioxide (Ekeroth et al. 2012).

In a collaboration between the Paul Scherrer Institute (PSI), Nagra (Switzerland) and Studsvik Nuclear AB, new data have been obtained regarding the Instant Release Fraction (IRF) of cesium-137, iodine-129 and selenium-79 from BWR, PWR and MOX fuel with a burnup of 50–75 MWd/kgU (Johnson et al. 2012). Data show that the fission gas release (FGR) can be used as an upper limit for the IRF of cesium-137 and iodine-129. Johnson et al. (2012) show that the IRF of cesium-137 is less than the FGR for all fuels in the study, and that the IRF of iodine-129 is comparable to the FGR, particularly in cases where the fuel was separated from the cladding. This study also gives, for the first time, experimental data that can be used to determine the maximum value for the IRF of selenium-79 from high-burnup fuel. This was possible after a method development of analysis of selenium-79 described in Johnson et al. (2012). The experiments also showed that the pores in the outer rim of the pellet do not contribute to the IRF.

The gap inventory is an important parameter of the fuel's initial state and must be estimated for use in SKB's safety assessments. Based on the observations that have been done for the ratio of FGR and gap inventory for certain fission products, Johnson et al. (2012) propose that FGR statistics for a large number of fuel assemblies irradiated in a whole core can be used to estimate the IRF.

ASIED was considered to contribute considerably to redistribution of radionuclides in the fuel in the French PRECCI programme up to around 2006, but more recent modelling and experimental data (Ferry et al. 2008, 2010) show that the contribution made by ASIED is insignificant, even in a long time perspective. This is the basis for the handling of the phenomenon in SR-Site (SKB 2010o, pp 41-43).

Programme

As a consequence of the plans for increased power level and increased burnup, FGR and IRF, i.e. the quickly released fraction of radionuclides from the fuel, will be substantially higher than for low-burnup fuel. At the same time, improvements are constantly being made in the form of new fuels that give limited FGR, even at higher burnup and linear heat rating. An example is ADOPT fuel (Advanced Doped Pellet Technology) from Westinghouse (Backman et al. 2010) and chromia-doped fuel from AREVA (Delafoy and Zemek 2010). New experiments are planned for leaching of doped pellets and comparison with ordinary pellets burned in the same reactor and with known FGR. The results are expected to provide a better estimate of the IRF and contribute to more realistic ratios between FGR and IRF. These planned experiments will be carried out within the framework of the EU project "FIRST Nuclides", initiated in 2012, where Studsvik Nuclear AB will carry out measurements of IRF from different fuels and burnups with support from SKB.

23.1.3 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.

Drying of fuel and the initial state with respect to remaining quantity of water after drying of the fuel are described in Section 11.8 "Water and water content". Stress corrosion cracking (SCC) on nodular iron caused by radiolytically produced nitric acid is discussed in Section 24.2.7 "Stress corrosion cracking of insert".

The effect of the hydrogen on the mechanical properties of the nodular iron is discussed in Section 24.2.2 "Deformation of insert".

23.2 Processes in fuel/cavities

A number of processes will with time alter the state in the fuel and in 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.

23.2.1 Overview of processes

The radionuclides in the fuel will eventually be transformed into stable substances by radioactive decay. 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 generation by 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 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 protective cladding tubesare perforated 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 conversion of the fuel matrix with 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's void volume. This is determined by the chemical conditions in the canister's cavities. 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.

If water is present between a sufficiently large number of fuel rods, the energy of the neutrons may be moderated. Low-energy neutrons can subsequently cause fission of certain nuclides in the fuel, releasing more neutrons. Criticality may be achieved under certain conditions, 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 23-2.

The research programme for the other processes in the fuel is discussed in the following sections.

23.2.2 Radioactive decay

Conclusions in RD&D 2010 and its review

The half-lives of the relevant radionuclides are generally well known. Selenium-79 is an exception, and its half-life has been discussed in the literature. The most recently published value is 327,000 years (Jörg et al. 2010). Previous estimates are 280,000 years (He et al. 2002) and 377,000 years (Bienvenue et al. 2007).

The most recently published value of the half-life of silver-108m is 435.7 years (Schrader 2010). This differs insignificantly from previous values, 418 years and 438 years (Schrader 2004).

Table 23-2. Fuel processes that are not judged to require any research programme, or that are handled within technology development. The processes are not commented on in the review of RD&D Programme 2010.

Radiation attenuation/heat generation (see Chapter 11 "Technology development, fuel handling").
Heat transport.
Water- and gas transport in canister cavities, boiling/condensation.
Thermal expansion/cladding failure.
Advection and diffusion (see Section 25.5.10 "Advection" and Section 25.5.11 "Diffusion").
Residual gas radiolysis/acid formation (see Section 24.2.7).
Solution of gap inventory.

Newfound knowledge since RD&D 2010

SKB has no programme for studying half-lives, but keeps track of relevant research and updates the database when necessary.

Programme

The field is judged today not to require any further research, development or demonstration. New developments in the field are being monitored and will be acted on when appropriate.

23.2.3 Induced fission – criticality

Conclusions in RD&D 2010 and its review

The criticality analyses that were carried out for SR-Site show that the effective multiplication factor (k_{eff}) is under 0.95 inside a water-filled canister for all types of spent fuel if burnup credit is used. 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, which has most recently been shown by Nicot (2008).

In its review of RD&D Programme 2010, SSM judged that SKB needs to give an account of the validation of ORIGEN-S and estimate the uncertainty in the determinations. SSM also said the method for calculating burnup credit needs to be presented.

Newfound knowledge since RD&D 2010

SKB, Posiva and Nagra have met twice during the period at a series of workshops on analysis of criticality in a Spent Fuel Repository and the role this plays in a long-term perspective. The first workshop was held in Stockholm in December 2010 and the second in Helsinki in December 2011. The purpose of these meetings was to share information and coordinate the work on criticality in a long-term perspective, summarize the current state of knowledge and identify uncertainties and future research needs.

Programme

SKB's programme for criticality analyses, including those with a bearing on long-term safety, will be conducted via a project called "Criticality analysis" SKB. This is described in Section 11.9 "Criticality".

23.2.4 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 2010 and its review

Studies were presented in RD&D Programme 2010 showing that small quantities of hydrogen in the water suppress 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.

Newfound knowledge since RD&D 2010

When it comes to homogeneous alpha radiolysis, i.e. radiolysis in bulk solution, a previous study (Pastina and LaVerne 2001) has shown that hydrogen in solution had no appreciable effect on the hydrogen peroxide concentration, contrary to the results of numerical simulation . In the study by Pastina and LaVerne (2001), in which the solution was exposed to an external radiation source, the dose rate was assumed to be homogeneously distributed in the entire water volume. New calculations (Trummer and Jonsson 2010) show that this assumption was incorrect. In a water solution that is irradiated with alpha particles, the dose is deposited in a very limited volume of the solution (Trummer and Jonsson 2010). The new calculations that have now been performed use this limited volume and result in much better agreement with the experimental results. The same study reports at what limit values of dissolved hydrogen oxidative dissolution of spent fuel no longer occurs, i.e. the threshold value at which fuel oxidative dissolution ceases due to the hydrogen effect. The study suggests that low concentrations of hydrogen (on the order of 10^{-9} molar) are sufficient to counteract the effect of alpha radiolysis of fuel that is several tens of thousands of years old (Trummer and Jonsson 2010).

Programme

Mechanistic studies of actinide oxide surfaces in contact with water and exposed to different types of radiation are being conducted in cooperation with the Institute for Transuranics Elements (ITU) in Germany by using thin layers of oxides produced from the gas phase.

An ongoing study of homogeneous alpha radiolysis in the presence of bromide in plutonium solutions will continue. Certain difficulties have arisen in preparing the solutions with the desired plutonium-238 content with a nearly neutral pH.

23.2.5 Metal corrosion

Metals and alloys of different types and with different functions are present in a spent fuel assembly. The fuel rods are encased in Zircaloy cladding. Other parts consist of stainless steel, Inconel, Incoloy or Zircaloy. Control rods, which will be encapsulated and emplaced together with PWR fuel, consist of an alloy of 80 percent silver, 15 percent indium and 5 percent cadmium. These metal parts will contain some radionuclides that have arisen due to neutron activation, which will be released to the water when the metal corrodes. It was pessimistically assumed in the SR-Can safety assessment that this corrosion is rapid, and the activation products were included in the Instant Release Fraction (IRF). It is more realistically assumed in the SR-Site safety assessment that it takes some time for the metal parts to corrode. The exception is the control rods, which are assumed (very pessimistically) in SR-Site to be included in the IRF. Corrosion of Zircaloy and other metals is controlled by the material composition, the chemical environment in the canister and the temperature. The process affects the release of a certain fraction of the radionuclide inventory.

Conclusions in RD&D 2010 and its review

Prior to RD&D Programme 2010, it was judged that the area did not require an active research programme. Metal corrosion was thereby listed as one of the processes that only required monitoring by SKB. New needs have emerged in conjunction with the SR-Site safety assessment, however.

Newfound knowledge since RD&D 2010

The release of activation products from the cladding will be controlled by the corrosion rate of Zircaloy, which is transformed to zirconium oxide. Dissolution of the oxide layer is controlled by the solubility of zirconium oxide in the surrounding water and the removal of dissolved zirconium. The solubility of zirconium oxide in water is very low, around 10^{-9} molar (Brown et al. 2005). It is assumed in SR-Site that the Zircaloy cladding does not have any protective function, but corrosion proceeds slowly; available data indicate that it takes at least 100,000 years for the cladding tubes to corrode completely (SKB 2010o).

It is assumed that the release of radionuclides present in Crud (Chalk River Unidentified Deposit, an undesirable deposit on the fuel parts that forms in the reactor) takes place immediately when water enters the canister. The same is assumed for the silver alloy in the encapsulated control rods. This is a pessimistic assumption, particularly as regards the radioactive silver isotopes, which can be expected to be released at the rate of corrosion of metallic silver (McNeil and Little 1992). Relevant experimental data on corrosion of the control rod alloy are currently lacking, however. The radionuclide silver-108m has proved to be of great importance for scenarios that involve early failures of engineered barriers. In lieu of data, the process has been handled very pessimistically, which has affected further handling in the safety assessment as regards solubility of silver-108m and stable silver.

Programme

Based on experience from SR-Site, leaching of control rods for PWR fuel will be investigated under different conditions with respect to silver release.

23.2.6 Fuel dissolution

Conclusions in RD&D 2010 and its review

In its review, SSM thought that SKB should make further efforts to gain a better understanding of the mechanisms of the effects of hydrogen on fuel dissolution. In this connection, SKB should systematically report additional experimental evidence and modelling results. SKB should further strive to quantify the threshold value of hydrogen concentration above which water radiolysis is prevented, and determine whether such a value can be maintained when the bentonite buffer has been partially or entirely eroded away. SSM considered that SKB should investigate the possibility of catalyst poisoning in case metallic particles activate hydrogen.

Newfound knowledge since RD&D 2010

In view of the foreseen increase in 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. The extended study included leaching of four fuel segments with high burnup (55–75 MWd/kgU) by use of the same leaching procedure as before (Series 11, Forsyth 1997). All in all, the results for the entire burnup interval show that releases from fuel do not increase proportionately with burnup (Zwicky et al. 2011). The cumulative released fraction of uranium (Figure 23-1) is not higher than equivalent values for fuel from Series 11 (burnup 21–49 MWd/kgU).

Since the rim of the pellet has a higher burnup than the central parts, the impact of the high-burnup rim on the quantity of oxidized uranium in solution was investigated. By comparing the ratio of uranium-236 to uranium-235 in the leachant for each time interval with the average value (the inventory value), conclusions can be drawn regarding which part of the pellet the uranium was leached from. Uranium from the high-burnup rim has a higher uranium-236/uranium-235 ratio. The results show that in all leachants, the ratio is lower than the inventory value at the start, followed by a gradual increase up to the average value, see Figure 23-2 (next page). This indicates a slower dissolution rate for the high-burnup rim. The same applies to the release rate for other redox-sensitive nuclides such as molybdenum and technetium.

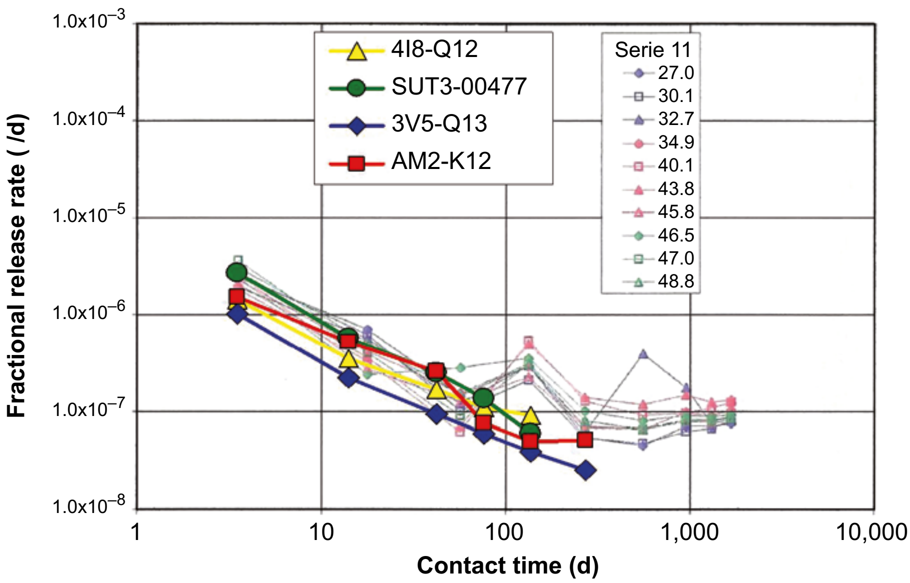


Figure 23-1. *Released fraction of uranium for high-burnup fuel (Zwicky et al. 2011) in comparison with Series 11 for fuel with a burnup of 30-49 MWd/kgU (Forsyth 1997).*

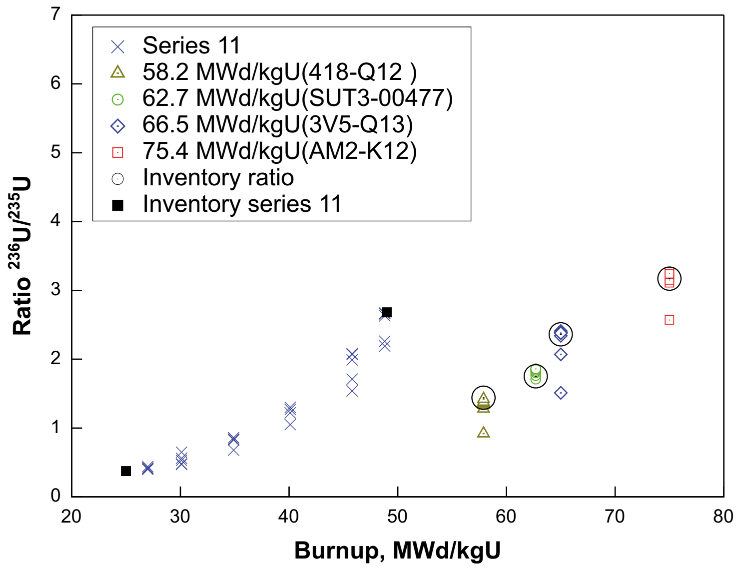


Figure 23-2. *Ratio between uranium-236 and uranium-235 in leachant from high-burnup fuel (unfilled symbols) for different contact periods compared with the inventory value (circle) (Zwicky et al. 2011) and similar data from lower-burnup fuel (Forsyth 1997) (X, inventory squared). In all leachants, the ratio between the two uranium isotopes is lower than the inventory value to start with, followed by a gradual increase up to the maximum value.*

During the first tens of thousands of years in the repository, the high concentrations of reductants (iron(II) and hydrogen) formed by anoxic iron corrosion are expected to effectively consume the oxidants produced by radiolysis. In experiments with fuel under argon (Cui et al. 2011), it is noted that water radiolysis contributes to an oxidative dissolution of the fuel. This causes release of actinides and fission products, whose concentrations are measured in the leachant. To simulate conditions in the final repository, two small metal foils of iron and copper were added to an experiment with fuel under argon. Analyses of the leachant showed that the release of the fission products cesium, molybdenum and strontium was not appreciably affected, while the concentrations of uranium, neptunium and technetium fell considerably. This shows that, despite the strong radiation from the fuel, the contact with iron is able to create reducing conditions (Cui et al. 2011). Precipitates of iron(III) oxide also contribute to reduced release of radionuclides in water.

The mechanism behind the effect of the hydrogen on fuel dissolution has been linked in a number of studies to the catalytic effect of metallic epsilon particles (containing molybdenum, palladium, technetium, rhodium and ruthenium). The hypothesis is that these metal particles, which occur in the fuel matrix, activate hydrogen on the surface of the fuel. Metallic particles, extracted from spent fuel by dissolution of the uranium dioxide matrix in hot phosphoric acid, have been investigated (Cui et al. 2012) with the objective of better characterizing the particles. Analysis with a transmission electron microscope (TEM) showed that these particles are less than one micrometre in diameter, and often composed of aggregates of smaller particles (10–20 nanometres). Furthermore, the metallic particles that are formed in the high-burnup rim of the fuel pellet are smaller and contain more palladium than those formed in the inner part of the pellet. This is due to the variation in local temperature and the neutron irradiation that takes place during the time in the reactor. The cell parameters for the alloy have been determined by X-ray diffraction and EXAFS (Extended X-ray Analysis Fine Structure). The measurements show a very homogeneous alloy.

In almost all fuel leaching experiments in autoclaves under hydrogen atmosphere, a decrease is noticed in the concentrations of redox-sensitive nuclides during the course of the experiment. This decrease proceeds for weeks or months, depending on the fuel's surface area, the temperature and how well the fuel was washed with carbonate solution prior to the experiment (to remove the pre-oxidized layer). The decrease indicates that a reduction and precipitation of the redox-sensitive nuclides, including uranium, is taking place somewhere in the system. In order to determine whether the precipitation took place on the autoclave walls, in one experiment the fuel was removed and the autoclave was analyzed for uranium after rinsing with nitric acid (Albinsson et al. 2003). Only one percent of the quantity of precipitated uranium was found on the autoclave walls; the rest had probably been precipitated on the fuel surface. It is not possible to confirm this by observation of the fuel surface in, for example, a scanning electron microscope (SEM), since uranium dioxide cannot be observed on the fuel surface.

In order to investigate whether redox-sensitive nuclides precipitate on the fuel surface, an experiment was performed where a redox pair not present in the fuel was added to the solution: CrO_4^{2-}/Cr^{3+}. Chromium(VI) in the form of chromate is very soluble at the pH assumed in the experiment, while chromium(III) forms chromium hydroxide ($Cr(OH)_3(s)$) if the concentration of chromium(III) is higher than in the order of 10^{-6} molar. The purpose of the experiment was to distinguish whether certain areas of the fuel surface cause reduction of chromium(VI) to chromium(III), which could then be detected via precipitation of dark green chromium hydroxide. The results are published and show a homogeneous layer of chromium hydroxide covering the entire fuel surface (Puranen et al. 2012). If oxidants were to be produced somewhere at the surface, they would oxidize chromium(III) to yellow chromium(VI) and dissolve the chromium precipitate. Another observation from the test is that reduction and precipitation are a surface effect: a few millimetres up in the fuel basket (a basket of gold mesh in which the fuel is placed), where the gamma field is virtually unchanged, there is no green precipitate. It is only found in those parts of the gold mesh that are in contact with fuel powder.

Hydrogen peroxide is catalytically dissociated on oxide surfaces and in bulk solution, with oxygen and water as end products. The first step in the process is cleavage of the hydrogen peroxide molecule into two hydroxyl radicals (OH radicals) (Hiroki and LaVerne 2005). In an experiment with SIMFUEL in an argon atmosphere (Nilsson and Jonsson 2011), the concentrations of hydrogen peroxide and uranium in solution were analyzed as a function of time. The results show that 99.8 percent of the hydrogen peroxide is dissociated at the surface of SIMFUEL and only 0.2 percent causes oxidation of uranium measured as uranium(VI) in solution. In order to investigate how fission products influence oxidative dissolution of fuel, experiments were performed with uranium dioxide doped with yttrium as well as pure uranium dioxide and pure yttrium oxide, in the presence of hydrogen peroxide (Trummer et al. 2010). The results show that the decrease in the oxidative dissolution of yttrium-doped uranium dioxide compared with pure uranium dioxide cannot be explained by catalytic decomposition of hydrogen peroxide on yttrium oxide. The probable explanation is instead that yttrium in the lattice increases resistance to oxidation, so that a larger fraction of the hydrogen peroxide decomposes. A possible mechanism behind the hydrogen effect is also discussed in the study: When hydrogen peroxide decomposes on the metal oxide surface, the first step is the formation of two hydroxyl radicals, which starts a chain of reactions finally producing water and oxygen. In the presence of hydrogen, this chain is broken when the hydrogen reacts with the hydroxyl radicals, producing hydrogen atoms and water. This leads to a reduced production of hydrogen peroxide and oxidized uranium in solution.

The mechanistic difference between uranium dioxide and doped uranium dioxide material, such as SIMFUEL, has been investigated (Lousada et al. 2013). A special reagent (Tris buffer) was used to analyze hydroxyl radicals formed by splitting of hydrogen peroxide molecules. The results show that the difference in oxidative dissolution of the two materials is primarily dependent on differences in redox reactivity. In conjunction with this, new values are calculated for the relative yield of different oxidants for fuel dissolution.

Investigation of the mechanism behind the hydrogen effect requires an understanding of what happens with the hydrogen peroxide that is formed by radiolysis. There are above all two possible reaction pathways when hydrogen peroxide reacts with the uranium dioxide surface: electron transfer and oxidation, or catalytic decomposition of the hydrogen peroxide. In a study comparing the reactivity of different uranium dioxide materials containing different additives (SIMFUEL, uranium dioxide with yttrium oxide, uranium dioxide with palladium, or uranium dioxide with both additives) with hydrogen peroxide, no appreciable differences are revealed between the materials (Pehrman et al. 2010). However, the ratio between dissolved uranyl species and consumed hydrogen peroxide is much lower for doped uranium dioxide. This may be due to different rates for the two different reaction pathways, i.e. between electron transfer (oxidation of uranium) or catalytic decomposition of hydrogen peroxide. How much of the hydrogen peroxide is consumed by catalytic decomposition could be estimated by analyzing the primary product of the decomposition (the hydroxyl radical) with a suitable reagent as a function of time. This was compared with the measured redox activity of doped pellets. The results show that a larger quantity of hydrogen peroxide decomposes catalytically on pure uranium dioxide than on doped pellets, but the differences are not great, at most 30 percent. The differences are, however, much greater when it comes to redox reactivity, and the results show clearly that pure uranium dioxide is more reactive than doped uranium dioxide, and that this difference declines with the activity of the oxidant. These results explain the observed differences in oxidation yield for doped pellets and uranium dioxide.

The effect of sulphide ions on radiation-induced dissolution of spent fuel has been investigated by the use of simplified model systems (Yang et al. 2013). The reaction between sulphide and hydrogen peroxide is rapid, and three to four hydrogen peroxide molecules are consumed per oxidized sulphide. The fact that sulphide reacts quickly with hydrogen peroxide also means that the small quantities of sulphide ions that may be present in the groundwater after contact with copper and iron cannot reach the fuel surface as long as it is producing hydrogen peroxide. Experiments with gamma radiolysis in the presence of sulphide show that release of radiolytically oxidized uranium decreases with the concentration of sulphide in solution. Sulphide also reduces uranium(VI) in anoxic solutions. The effect of sulphide on uranium(VI) reduction by noble metal particles in the presence of hydrogen shows that sulphide does not affect the rate of reduction of uranium(VI) by the palladium-hydrogen system. Thus no poisoning of the palladium catalyst can be detected in the presence of relatively high concentrations of sulphide (1 millimolar). According to Yang et al. (2013), this may be due to the high concentration of dissolved hydrogen, which causes desorption of sulphide from the palladium surface.

A critical review of the models used to describe radiation-induced dissolution of uranium dioxide fuel has been published (Eriksen et al. 2012). The models are based on a large amount of experimental data obtained from chemical and electrochemical experiments with various types of materials that simulate spent fuel. Use of rate constants that relate to the surface of the solid phase yields results that are in much better agreement with experimental data than previous radiolytic models (Christensen 1998), which assume dissolution of a surface layer and rate constants for homogeneous systems. Use of rate constants that are determined experimentally, together with verification of reaction mechanisms, is important for both chemical and electrochemical modelling.

In the EU project MICADO (Model Uncertainty for the Mechanism of Dissolution of Spent Fuel in Nuclear Waste Repository) and in cooperation with KTH, six different models describing fuel dissolution in a final repository during geological time periods were judged with respect to uncertainties. The project, which was concluded in 2010, mainly investigated model and parameter uncertainties and their impact on the whole system during the various stages of a repository. The results are now available in Grambow et al. (2010, 2011).

In the EU project REDUPP and in cooperation with Stockholm University, leaching tests were conducted with a focus on changes in the sample surface during the course of the dissolution. In order to gain a better understanding of the dissolution processes, materials with a fluorite structure were used, i.e. the same mineral structure as uranium dioxide. Results from these studies have been reported (Godinho et al. 2011, 2012, Evins and Vähänen 2012, Stennett et al. 2013, Corkhill et al. 2013). So far the results that have emerged have shown that, under conditions far from equilibrium, the dissolution rate varies depending on the crystallographic orientation and structure of the exposed surface (Godinho et al. 2011, 2012, Corkhill et al. 2013). The less stable crystal planes dissolve faster and leave the more stable crystal planes (111) and (100) in contact with the solution. This leads to a change in the topography of the surface and increases its area, without for that reason increasing its reactivity; instead, this seems to increase its stability, since the surface consists to a greater extent of stable crystal planes (Godinho et al. 2012).

One sub-project in REDUPP relates to the effect of natural groundwater on the dissolution of alpha-doped uranium dioxide. Here the possibility that trace elements in the water may affect the dissolution process is investigated, and the results are compared with those previously obtained with synthetic groundwater. The preliminary results published to date (Evins and Vähänen 2012) show that no effect of the alpha doping was detected in experiments where the surface area to volume ratio (SA/V) was five per metre ($5\ m^{-1}$). The results agree with previous experiments performed with synthetic groundwater. There are, however, signs that the results can be affected by an increased SA/V.

Programme

Research activities are planned during the coming period both to obtain data on fuel dissolution under repository-like conditions and efforts to shed light on the mechanisms of the different processes that contribute to fuel dissolution.

Leaching of high-burnup fuel under oxidizing conditions, as well as under hydrogen, will continue during the coming years. These tests are being done to confirm and obtain better data on the leaching properties of high-burnup fuel. New types of fuel, so-called doped fuel, have been developed recently. This fuel has additives such as chromium or aluminium that increase the grain size in the fuel pellets and thereby reduce FGR. These new types of fuel have better properties in the reactor, and will become more common in the future. Knowledge of IRF and the dissolution rate of doped fuel will be needed to describe its behaviour in the final repository.

Both experimental mechanistic studies and modelling studies will be conducted to get a better understanding of fuel dissolution under repository conditions, as demanded by the regulatory authorities (SSM 2011).

The EU project REDUPP will continue up to the end of March 2014, and the sub-projects will continue to investigate the effects of dissolution on the surface structure of the studied oxides cerium dioxide, thorium dioxide and uranium dioxide. A modelling study is included, with a focus on the link between the experimental results and the theoretical calculations of the relative stability of the crystal planes. Dissolution of alpha-doped uranium dioxide is continuing, with an increased surface area to volume ratio (SA/V). Studies are also being done of gadolinium-doped uranium dioxide, also with relatively high SA/V.

23.2.7 Speciation of radionuclides, colloid formation

In the scenario with a damaged canister, 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.

Conclusions in RD&D 2010 and its review

RD&D Programme 2010 presented data from thermodynamics and kinetics showing that a simultaneous release of radium and barium in sulphate-rich water causes co-precipitation of radium with barium as radium-barium sulphate. It was, however, noted that further kinetic studies are required to find out how rapidly radiogenic radium is uptaken by already precipitated barite ($BaSO_4$(s)).

In its review of RD&D Programme 2010, SSM wrote that SKB should study, report and compile current research linked to formation of colloids and amorphous phases based on uranium, plutonium or other actinides and their possible relevance to radionuclide transport when colloid particles are not filtered by an eroded buffer. SKB should further consider whether scenarios of repository evolution with periods of reduced sulphate concentrations may be of importance with respect to altered solubility conditions, decay, biosphere conditions, etc. SSM also wanted SKB to investigate the importance of any new species (mixed metal complexes) for the solubility of radionuclides.

Newfound knowledge since RD&D 2010

In order to better understand the redox processes that are expected to occur in a damaged canister, a study has been conducted of the interaction between selenium (redox-sensitive) and iron corrosion products under oxygen-free conditions. The study showed that selenium(IV) can be immobilized by reduction to selenium(0) on the magnetite that covered the iron surface (Puranen et al. 2010a). The study also showed that the reaction was faster in the presence of uranyl ions, possibly due to a stepwise mechanism involving reduction of uranyl and oxidation of the magnetite layer, which is then transformed to a mixed oxyhydroxide of iron(II) and iron(III). Iron oxides and speciation of selenium on the iron surface were investigated by Raman spectroscopy and micro-XANES (X-Ray Absorption Near Edge Structure) in order to determine the oxidation numbers of uranium and selenium on the surface.

Reduction of selenium(IV) and selenium(VI) was tested with hydrogen and metallic palladium, a known hydrogen catalyst. Selenite (selenium(IV)) was found to adsorb on palladium and was reduced to selenium(0) in the presence of hydrogen. No sorption or reduction was observed for selenate (selenium(VI)), which is due to the high barrier to a stepwise electron reduction of selenate (Puranen et al. 2010b). By the use of an electron accelerator, plus different reagents in the solution, it could be proved that there is a noticeable barrier to a one-electron reduction of selenate and only very strong one-electron reductants can reduce selenate to selenite.

Uranyl, carbonate and hydrogen peroxide are expected to be present in a final repository for spent nuclear fuel. Present-day knowledge concerning the complex species that can be formed in solution is currently rather limited. In a new study, speciation and stability constants for mixed peroxide-carbonate complexes have been determined by the use of both classical methods (potentiometric titration) and NMR (Nuclear Magnetic Resonance) (Zanonato et al. 2012).

The effects of iron(II) in solution on the radiolytic dissolution of uranium dioxide and plutonium dioxide with plutonium-238 have been studied (Amme et al. 2012). Hydrogen peroxide is produced on the surface of the plutonium dioxide and is consumed by reaction with iron(II) ions and on the surface of the uranium dioxide by oxidation of the surface. The reaction of iron(II) ions with hydrogen peroxide produced by radiolysis, known as the Fenton reaction, produces a hydroxyl radical, which is a strong oxidant, and iron(III). How these processes affect the dissolution of actinide oxides is discussed. Uranium(VI) in solution is reduced by iron(II), and since iron(II) also consumes radiolytically produced hydrogen peroxide, a high concentration of iron(II) in solution will delay large-scale oxidative dissolution of uranium dioxide.

The oxidizing dissolution of actinide oxides such as uranium dioxide, neptunium dioxide and plutonium dioxide caused by hydrogen peroxide solutions without complexing agents has been studied in cooperation with the Institute for Transuranium Elements (ITU) in Germany (Pehrman et al. 2010). The results show that hydrogen peroxide is consumed most in the presence of uranium dioxide, followed by neptunium dioxide, and least with plutonium dioxide. The same applies to the concentration of oxidized actinide ions in solution, but owing to the relatively high hydrogen peroxide concentrations used in the experiment, uranium and neptunium peroxide are precipitated. Since resistance to oxidation increases from uranium dioxide to neptunium dioxide to plutonium dioxide, the fraction of hydrogen peroxide that undergoes catalytic decomposition on the oxide surface is expected to increase in a similar fashion.

Uptake of radium in synthetic barite ($BaSO_4$) was investigated in a study conducted at the Paul Scherrer Institute (PSI) in Switzerland (Curti et al. 2010). In a first step, the rate at which recrystallization of barite proceeds was studied by uptake of barium-133 from solution. The study indicates that recrystallization of barite proceeds at a constant rate and that the distribution of barium-133 in newly formed barite is homogeneous. In a second step, uptake in barite of radium-226 from solution was studied, and it was found that radium barite is formed as a solid solution, and that this takes place at a relatively fast rate and in thermodynamic equilibrium with the solution. It is also noted that the solid solution that is formed between radium sulphate and barite is non-ideal, which is somewhat unexpected.

Scenarios with reduced sulphate concentrations, which alter solubility conditions for e.g. radium sulphate, and release of colloids are handled pessimistically in SR-Site. The calculations in the main scenario for the corrosion case are done without solubility limits; instead, the entire dissolved inventory is assumed to be in solution. The exception is uranium-238, owing to the large quantity present in the fuel, which is assumed to be reduced on the iron surface or its corrosion products and form amorphous uranium dioxide. This is the main source of radium formation in the canister, via decay to thorium-230, which is also considered to be sorbed in the interior of the canister.

Programme

Studies of redox kinetics for different oxidized forms of radionuclides (including technetium(VII), neptunium(V) and plutonium(V), plutonium(VI)) on fresh and corroded iron surfaces will continue in cooperation with Studsvik, Nagra, ITU and PSI.

Co-precipitation of radium with barium will be further studied in experiments involving solution analyses and characterization of the solid phaseby spectroscopic and microscopic methods (TEM images). The study will be carried out in cooperation with Forschungszentrum Jülich in Germany.

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.

An experiment with fuel and iron powder in an argon atmosphere is under way in Studsvik. The products that are deposited on the iron surface will be analyzed at the end of the experiment, and this is expected to provide a better understanding of possible new phases or species that can form in the repository environment.

23.2.8 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 2010 and its review

RD&D Programme 2010 presented the results of the French programme (Ferry et al. 2007, 2008), which indicated that helium build-up did not cause increased fracturing in uranium dioxide fuel with a burnup of up to 47.5 MWd/kgU. It was thereby deemed sufficient that SKB's programme for this process only entailed monitoring of current international research in the area.

Newfound knowledge since RD&D 2010

The French programme has continued its calculations of helium production in fuel, now in uranium dioxide fuel with a burnup of up to 60 MWd/kgU. A model has been used to judge the development of the fuel's microstructure before expected contact with water (Ferry et al. 2010). The conclusion is that after 10,000 years, the fuel's microstructure will not have been adversely affected by the helium build-up. This is in line with previous results (Ferry et al. 2007, 2008).

Studies that relate to helium diffusion and migration in the fuel have been published during the period. They refer to previous ab initio studies (Yun et al. 2009) and indicate the importance of having a detailed understanding of the processes in the solid phase that lead to the formation of helium bubbles in the uranium dioxide lattice (Yakub et al. 2010). Molecular dynamics simulations have been carried out for this purpose (Yakub 2011), indicating a higher solubility of helium in UO_{2+x} compared with stoichiometric uranium dioxide.

Programme

Since RD&D Programme 2010, needs have emerged for further studies to shed light on how helium released from the fuel can in the long term contribute to pressure build-up in the canister. Modelling studies have therefore been started for the purpose of calculating the quantity of helium that is formed over time in both uranium oxide and MOX.

23.2.9 Chemical transformation of the fuel matrix

Over the long term, and depending on the chemical environment, the spent fuel may undergo chemical transformation. In a reducing environment with siliceous water, there is a possibility that the uranium dioxide matrix will be transformed to the mineral coffinite (Janeczek and Ewing 1992). If there is a chemical driving force for this reaction, it could entail that radionuclides in the fuel matrix are released due to congruent dissolution (Grambow et al. 2011).

Conclusions in RD&D 2010 and its review

The process is not described in RD&D programme 2010.

Newfound knowledge since RD&D 2010

In recent years, a number of studies have been carried out to learn more about coffinite and uranothorite (U, Th silicate). Some uncertainties still exist concerning how coffinitization affects fuel dissolution in a long time perspective (Grambow et al. 2011). A review of published literature shows that it is difficult at the present time to ascertain under what conditions, above all at what temperatures, coffinite is stable in relation to uraninite and quartz (Evins and Jensen 2012). A study of a uranium mineralization in the Athabasca Basin shows that coffinite was formed at at temperatures below 50°C (Mercadier et al. 2011). This agrees with other observations from natural systems that suggest that coffinite precipitates at relatively low temperatures, below 130°C (Deditius et al. 2012). There are, however, other studies that also link coffinite formation in nature to higher temperatures, 125–170°C (Min et al. 2005, Kříbek et al. 2009). In nature, thorite is usually formed at temperatures between 200–400°C (Deditius et al. 2012), and previous calorimetric studies also show increased stability for thorite at temperatures higher than room temperature (Mazeina et al. 2005). The laboratory experiments that have been conducted aimed at synthesizing pure coffinite have run into difficulties, however (Pointeau et al. 2009), and new studies being conducted are focusing on synthesis of uranothorite with a composition of $Th_{1-x}U_xSiO_4$ (Costin et al. 2012). The published syntheses of coffinite and uranothorite have been performed at hydrothermal temperatures, 250°C (Pointeau et al. 2009, Costin et al. 2011, 2012), but it should be noted that uranium(IV) silica colloids have been produced in the laboratory at room temperature (Dreissig et al. 2011). In an experiment where the reaction time was extended to 672 hours, it was noted at 250°C that a nanocrystalline mixture of oxides was transformed to consisting mostly of uranothorite (Costin et al. 2011). This suggests that the synthesis is kinetically controlled by an alteration process that includes dissolution and co-precipitation. Since uraninite – in a reducing environment – can be preserved for very long time spans (for example in Oklo), this can be interpreted as meaning that even if there were a thermodynamic driving force for the alteration process, it is a very slow process (see for example Grambow et al. 2011).

Programme

The field is judged today not to require any further research, development or demonstration. New developments in the field are being monitored and will be acted on when appropriate.

24 Canister

24.1 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 Spent Fuel Repository. The requirements and design premises related to the canister's barrier function in the final repository are described in Section 12.1 "Requirements and premises", with reference to (SKB 2010c) in the supporting material for the application.

24.1.1 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 copper shell and insert. The initial values of these variables can be taken from either information on properties of manufactured and inspected canisters or other documented information, see Table 24-1.

The values of the variables that define the initial state for the canister can be ascertained from the reference design and relate to e.g. the thickness of the copper shell (variable: geometry) and the concentrations of impurities and other substances in the copper (variable: material composition). The initial radiation at 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, see Section 12.1 "Requirements and premises". They are defined in such a way that the canister shall 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 cast 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 12 "Technology development". The research planned in this area for analysis of the initial state is continued development of models for analysis of mechanical loads (see further Section 12.3 "Canister design – analyses of the canister"). The research efforts concerning the processes that affect long-term changes in the canister, for example deformation and corrosion, are described in Section 24.2.

Table 24-1. Variables for the copper shell and the iron insert and related properties (design parameters), as well as sources where information on the variable is provided.

Variable	Definition	Comment
Geometry	Geometric dimensions of the canister components, including manufacturing tolerances. This also includes a description of possible defects from manufacturing, welding and the like.	Reference canister and initial state, (SKB 2010c).
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 is described in Section 11.6 "Radiation protection and dose estimates".
Temperature	Temperature as a function of time and space in the canister components.	Verification of the design premises regarding temperature is described in Section 12.1 "Requirements and premises".
Mechanical stresses	Mechanical stresses as a function of time and space in the canister components.	Residual stresses from manufacturing 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 (SKB 2010c).Corrosion products (SKB 2010o).

24.2 Canister processes

24.2.1 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 radiation could also have some effects on the materials in the insert and copper canister. Otherwise the radiation has a negligible impact on other processes in the canister, with the exception of corrosion of the insert and the copper canister. This is documented for each process in the so-called influence tables presented in the process description in the SR-Site safety assessment (SKB 2010o).

The insert and the copper shell can be deformed mechanically by external loads. The creep properties of the copper are important, especially initially when the copper creeps and the gap between the copper shell and the insert decreases. 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 insert. Corrosion of nodular cast iron is primarily of importance if water enters the canister through a hole in the copper shell.

Transport of radionuclides in the near-field can take place via advection and diffusion in the aqueous phase or in the gas 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 (SKB 2010o). The processes radiation attenuation/heat generation, heat transport, corrosion of insert and galvanic corrosion (galvanic corrosion between copper shell and insert) are not judged to require a research programme and are not commented on in the review of RD&D Programme 2010.

24.2.2 Deformation of insert

Chapter 12 "Technology development, canister" describes how the work with the canister's reference design is being pursued and how it is being evaluated for different mechanical loads, based on the requirements made on the canister.

This section only describes creep in nodular cast iron and the effect of hydrogen on material properties in nodular cast iron.

Conclusions in RD&D 2010 and its review

The conclusion in RD&D Programme 2010 regarding creep in nodular iron was that it proceeds at a declining rate (logarithmic creep). Supplementary creep testing to get a more uniform body of data was stipulated for the continued programme. The programme also included further studies of hydrogen-charged material and its material properties.

In its review of RD&D Programme 2010, SSM approved of SKB's plan for further studies of the effects of hydrogen on material properties, both under monotonous loading and under creep loading. SSM also expressed the opinion that SKB should supplement the analysis of material properties with an analysis of suitable load cases where these properties are found to be particularly critical.

The Swedish National Council for Nuclear Waste regarded with satisfaction the fact that SKB has initiated research on creep and the influence of hydrogen on material properties in the cast iron insert.

Newfound knowledge since RD&D 2010

The conclusions from the preliminary results of creep testing of the nodular cast iron that were presented in RD&D Programme 2010 remain valid and have now also been reported (Martinsson et al. 2010). The observed logarithmic creep entails that the deformation is declining and that the total strain can be calculated, even after a very long time. The creep strain is judged to be less than 0.1 percent in the insert in the repository, which means that creep in nodular cast iron has a negligible effect on the properties of the insert.

Work has continued on studying hydrogen-charged nodular cast iron, above all to determine how far into the material the hydrogen penetrates with electrolytic charging. The same multi-step method as was developed to study hydrogen penetration in copper (Martinsson and Sandström 2012) has been used for the nodular cast iron. The method entails that round specimens are hydrogen-charged and turned on a lathe to different depths and the hydrogen concentration in the remaining piece is analyzed by melt extraction. Preliminary results show that a profile of the hydrogen content is formed in nodular cast iron as well with up to 50 times more hydrogen in the surface than in the original material. At longer charging times, the hydrogen goes deeper in (up to a millimetre or so after charging for 100 hours), while the concentration near the surface reaches saturation (the profile becomes flatter). After completed charging, a rapid degassing of the material takes place, especially at elevated temperatures, e.g. 100°C.

Programme

No further creep testing of nodular cast iron (specimens from several inserts, a wider temperature interval, etc) is planned at present, since SKB considers it to be of lower priority.

The results of the work with the effects of hydrogen on nodular cast iron and its material properties will be compiled and reported. Drying of fuel and the initial state with respect to remaining amount of water after drying of the fuel are described in Section 11.8 "Water and water vapour". In order to assess possible hydrogen penetration, modelling work similar to that done for hydrogen penetration in copper may be considered, see Section 24.2.3. To evaluate the effects on the strength of the insert, load cases must also be identified where hydrogen penetration might be of importance. The load cases are being identified within the framework of the updated design analysis, see Section 12.3 "Canister design – analyses of the canister".

As an additional tool for understanding the strength of copper and nodular cast iron, SKB is supporting a Ph. D. project at KTH where the role of defects is being studied by calculations of dislocation dynamics. The method entails that in a network of dislocations, the dislocations are allowed to interact with each other and with other defects, and the movement of each dislocation is calculated and studied. This is planned to be used to study the effects of phosphorus on the plastic and creep properties of copper and for segregation of copper in nodular cast iron under irradiation.

24.2.3 Deformation of copper canister from external pressure

Chapter 12 "Technology development, canister" describes how the work with the canister's reference design is being pursued and how it is being evaluated for different mechanical loads, based on the requirements made on the canister. Deformation of the copper shell due to external pressure exerted by different loads is dealt with integrated with the insert.

Research on the creep properties of copper and other material-related properties is reported in this section.

Conclusions in RD&D 2010 and its review

SKB's programme to describe the properties of the copper material was planned to cover creep properties in particular, with several parallel lines of inquiry: development of fundamental equations for FSW material, implementation in models using the Finite Element Method (FEM) to describe different load cases, and effects of cold working as well as of multiaxial stress states. The programme also included the effects of hydrogen on the copper via controlled hydrogen charging of the material, including modelling of solubility and diffusion of hydrogen in copper.

In its review of RD&D Programme 2010, SSM expressed the opinion that SKB should continue its efforts to determine the effects of phosphorus and sulphur on the creep properties of copper. SSM also found that SKB had not demonstrated sufficiently that phosphorus-doped copper exhibits a ductile behaviour even at very long creep times. In this respect, SSM also found that the link between creep brittleness and the copper's content of phosphorus and oxygen needs to be further studied with respect to reduction of copper(I) oxide (Cu_2O) to form copper and water vapour. SSM further deemed that it is important in future FEM analyses to describe the canister's entire load history in view of the fact that cold-worked copper has lower creep ductility than copper that has never undergone plastic deformation.

SSM agreed with SKB that the possible influence of hydrogen on the creep properties of copper should be further investigated. SSM urged SKB pursue deeper knowledge in this area by investigating whether hydrogen can affect the creep properties of copper or cause embrittlement, and why the hydrogen concentration in copper is many orders of magnitude higher than the solubility of hydrogen in copper at equilibrium.

The Swedish National Council for Nuclear Waste observed that SKB has thoroughly investigated the creep properties of copper, but that a validated creep model is still lacking. Furthermore, the Council wrote that the creep ductility of copper in particular decreases with increasing cold deformation, which occurs for different reasons, for example during manufacturing or due to rock shear. If the integrity of the copper shell can be maintained under different loads in the repository, this must be shown by creep modelling. Studies of creep in welds are as yet too few, and in future SKB will have to study in greater depth the creep properties of friction stir welds and geometric discontinuities that exhibit the highest strains in design analysis.

Newfound knowledge since RD&D 2010

The fundamental creep model without fitting parameters that was presented in 2009 and was formulated using basic dislocation mechanisms (Sandström and Hallgren 2009) has been found to be applicable for further situations. The model has been published in the scientific literature and compared there with data from creep specimens with round notches, where multiaxial stress states are present (Sandström 2012). The model has also been compared with stress-strain curves for cold-worked material and been found to describe the creep curves for these cases as well (Sandström and Hallgren 2012). The fundamental creep model can at least roughly cover stresses down to 40 megapascals (which has been verified experimentally), whereas the best available methods for extrapolation of creep data today can be used to predict failure down to stresses of 150 megapascals at 75°C with reasonable certainty. Since creep also occurs in the canister at these lower stresses, the developed model is important for predicting the behaviour of the canister after long periods of time. In order to be able to simulate full canisters with the finite element method, however, the creep model used must be fitted to the large deformations that can occur, so that e.g. cold working can be included in the loading history.

Preliminary results are also available from studies using transmission electron microscopy (TEM) of creep-tested cold-worked copper, which can probably explain the longer creep life of the material with finer and more stable “subgrains”.

The influence of local cold working has been studied by creep testing of flat specimens with conical indentations. Preliminary conclusions show that neither creep life nor creep ductility of the specimens were affected by the indentations, since the material creeps more around the harder local cold-deformed areas.

A model for crack growth in copper has been set up (Wu et al. 2011, 2013). Two types of creep damage are taken into consideration, which are denoted ductile and brittle respectively. Ductile creep rupture occurs when creep ductility exceeds a given value. The creep strain is calculated with the fundamental creep model. Brittle creep rupture is based on the extent of intergranular creep cavities, which is calculated with a model for formation and growth of creep cavities (see below concerning the influence of phosphorus). Ductile rupture dominates at lower temperatures, 22 and 75°C. At higher temperatures, 175 and 215°C, and especially if the stress state is decidedly multiaxial, brittle rupture is obtained at the crack tip. Crack growth was studied by means of creep testing of CT (compact tension) specimens. The model was found to be able to represent the influence of the temperature on crack growth, as well as the influence of the stress state, which was varied in the experiments by testing CT specimens with reduced cross-section (with side grooves).

The influence of loading time in creep testing has been studied (Andersson-Östling and Sandström 2011), and the total strain has been found to be 30–50 percent (elongation, or engineering strain), which agrees with previous testing with shorter loading times. The loading strain increases with increasing loading time, at the expense of the creep strain. A model where initial plastic deformation and creep (primary and secondary) are combined was well able to describe the creep testing data. The fundamental models for plastic deformation and creep described above were used. Plastic deformation dominates the strain during loading until the stress is close to the constant load level. After that, all of the strain comes from creep.

The work with fundamental equations for friction-stir-welded material as well has been pursued further and is described in Sandström et al. (2013). Stress-strain curves have been used in the same way as for the parent metal and described with the same equations as for the parent metal. The largest differences are that the yield strength is higher (and exhibits greater variation). Based on the stress-strain curves, a creep model has then been derived. Dynamic recovery plays a large role in both the model for plastic deformation and the one for creep. When dislocations of the opposite sign meet, they eliminate each other. The interaction distance between the dislocations is set to twice the radius for the cores of the dislocations. This interaction distance is slightly smaller in the models for welded metal than in the models for the parent metal. This can be assumed to be due to a more complex structure in the weld, which makes it more difficult for the dislocations to interact. These fundamental constitutive equations for welded joints have been verified by comparisons with tensile tests at different temperatures and strain rates, as well as by creep tests where the primary and secondary phases have been well reproduced.

Work of implementing the fundamental creep equations in finite element models has been carried out. Simplifications are needed in the numerical integration of the creep equations, and three different methods have been used. Continued work is, however, required before the equations can be used in finite element models for canister calculations.

Model studies of a full-sized canister are being conducted to investigate loads, both even and uneven, on the outer surface of the canister and how they affect stresses and strains in the material, especially in the weld. An elasto-plastic analysis is being carried out where the very low strain rates that occur during creep in the canister are taken into account. The fundamental models for parent metal and welded joint described above are used. Other work with analyses of loads on a full-sized canister are presented in Section 12.3 “Canister design – analyses of the canister”.

Residual stresses in the weld area have also been analyzed, by linking heat transport and elasto-plastic deformation in a finite element model (Jin and Sandström 2012). The modelling results indicate that the distribution of residual stresses is sensitive to the angle and asymmetrical in relation to the weld line. However, the maximum tensile stresses – both modelling results and values measured by hole drilling and X-ray diffraction – do not exceed 50 megapascals.

The influence of phosphorus on the creep properties of copper has been further studied. In a previous study (Sandström and Andersson 2008), the effect of phosphorus on creep strength has been described as being due to the fact that the phosphorus atoms (which are dissolved in the copper) form “atmospheres” around dislocations, thereby locking the dislocations so that they cannot move freely. An extra stress is needed to get the dislocations to break loose, resulting in elevated creep strength. In another previous study, it was assumed that phosphorus atoms could in the same way reduce grain boundary sliding and thereby reduce cavity formation (Sandström and Wu 2007). This would explain why phosphorous doped copper has a much higher creep ductility than phosphorus-free material at higher temperatures (> 160°C). This is still a viable mechanism, but not the only conceivable one. In a study involving tensile testing of copper with and without addition of phosphorus, no appreciable difference in grain boundary sliding could be observed, and the author draws the conclusion that the explanation that phosphorus locks the dislocations is not correct (Pettersson 2010). At least some of the difference in creep strength was instead considered to be due to intergranular cracking. Results from the continued work indicate that one explanation why the measured grain boundary sliding in Pettersson (2010) does not differ between the materials could be that the tensile testing proceeds so rapidly that the phosphorus atoms do not have time to diffuse to moving grain boundaries (Sandström and Wu 2013).

SKB’s work with analyses and modelling of how phosphorus affects the formation and growth of creep cavities is continuing. Pettersson (2010) obtained a grain boundary sliding that was considerably greater than that Sandström and Wu (2007) had used. In order to study the importance of the magnitude of the grain boundary sliding, Pettersson’s higher value has been used in the new study. It turns out that the difference in creep ductility between copper with and without phosphorus can be explained. In the new study, the interaction energy between phosphorus and the grain boundaries obtained from ab initio calculations is used (Korzhavyi et al. 2001). Several different mechanisms are conceivable for this interaction, but since they have almost the same interaction energy, the calculated creep ductility is not appreciably affected and the exact choice of mechanism is not crucial. The new model for formation and growth of cavities is, in the same way as the fundamental creep model, formulated without the use of fitted parameters. Some of the small differences in ductility and strength between copper with and without phosphorus which Pettersson observed in his tensile tests cannot be explained by the cavity model. They can, however, be described when the difference in deformation behaviour between the materials is taken into account.

The model under development can reproduce the temperature dependency of the ductility of copper both with and without phosphorus (Cu-OFP, oxygen-free phosphorus-doped copper, and Cu-OF, oxygen-free copper). Model calculations also show that the high creep ductility of copper to which phosphorus has been added also applies at longer times to rupture, see Figure 24-1. As discussed above, the cavity model can also explain the influence of temperature and stress state on the crack propagation rate during creep crack growth. The influence of the stress state in particular has previously been difficult to describe without the use of empirical approaches.

In order to gain a better understanding of the effects of phosphorus and other elements, a thermodynamic evaluation has been made of the copper-hydrogen-oxygen-sulphur-phosphorus system (Magnusson and Frisk 2013). Solubilities have been calculated for hydrogen, oxygen, sulphur and phosphorus, and the most stable phases have been identified. Sulphur is mainly present in sulphides. Phosphorus is mainly present dissolved in the copper matrix, but also in phosphates, where most of the oxygen is bound. Hydrogen is probably present in gaseous form, since the hydrogen quantities in Cu-OFP exceed the solubility of hydrogen in the copper matrix and the quantities that may be bound in phosphate.

A number of studies have been conducted to investigate the influence of hydrogen on the copper material. The conclusion of the efforts to develop a method for controlled hydrogen charging of copper was that cathodic charging of bars gave reproducible results. Charging of foils in a similar manner resulted in a large spread in the results, probably due to the influence of the surface structure. Thermal charging primarily led to an outgassing of hydrogen, and at higher temperatures undesirable grain growth (Martinsson et al. 2013). Hydrogen penetration in the bars was analyzed by means of both a three-step profile with turning-down and subsequent melt extraction and by means of GDOES (Glow Discharged Optical Emission Spectroscopy). The results obtained with the two methods agreed and showed a penetration depth of 50 micrometres (Martinsson and Sandström 2012). A diffusion-based model for hydrogen penetration and formation of hydrogen bubbles (pores) was developed and has been shown to be capable of describing the amount of hydrogen gas and the number and surface area of the bubbles (Martinsson and Sandström 2012). The model has also be used to investigate the effect on hydrogen penetration of charging intensity, charging time, grain size and critical nucleus size for bubble formation (Martinsson et al. 2013). The conclusion for the copper canister in the repository is that hydrogen from copper corrosion will not give higher concentrations than 0.6 weight-ppm (which is the limit SKB has set for the hydrogen content of copper) further in than a couple of hundred micrometres at the relevant grain sizes.

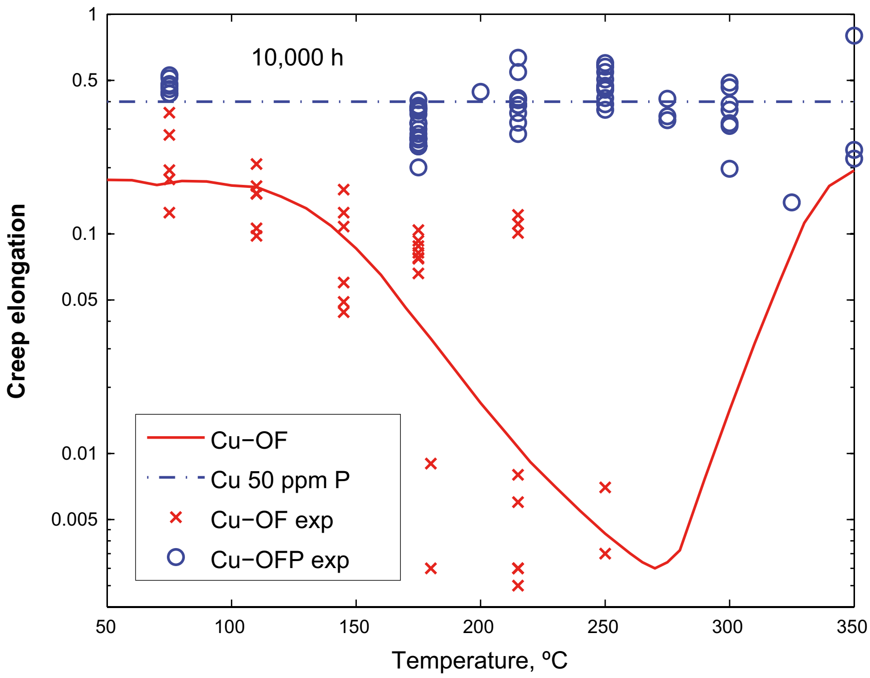


Figure 24-1. *Creep elongation (creep ductility) as a function of temperature for Cu-OF and Cu-OFP for a rupture time of around 10,000 hours (somewhat more than a year). Measured values are compared with the results from the model under development, which is based on initiation and growth of cavities.*

To gain a better understanding of solubility and diffusion of hydrogen in copper, ab initio studies are being conducted of structure and binding energy for vacancy-impurity complexes in copper. Preliminary results show that up to six hydrogen atoms can be bound in a vacancy. If an oxygen atom is bound in a similar manner to a vacancy, not more than one hydrogen atom can be bound. On the other hand, a hydrogen molecule or a water molecule cannot fit into such a single vacancy; larger defects (such as grain boundaries) are required in order for hydrogen embrittlement to occur.

In summary, fundamental models have been developed for i) plastic deformation, ii) creep in parent metal and welds, iii) initiation and growth of creep cavities, iv) crack growth and v) penetration of hydrogen into copper. Constitutive equations have now been developed for welded joints as well and used for calculation of plastic deformation in canisters. These models, where fitted parameters are not used, are utilized to describe the behaviour of the canister during very long time periods.

Programme

In order to provide a deeper understanding for the material properties of copper and thereby a better knowledge base for the assessments of post-closure safety, the research programme in material properties will continue. The process of creep in copper in the most important issue, but the research is being conducted (as before) along several parallel lines. The most important activities comprise:

- Further study of the effects of hydrogen, oxygen, sulphur and phosphorus on the properties of copper. The stability of copper is be investigated by thermodynamic modelling, and kinetic aspects of this stability will also be modelled. Two specially developed copper materials – both with a high concentration of oxygen and sulphur, and one with a high and one a low concentration of phosphorus – will be creep-tested. Ordinary canister copper will be creep-tested during simultaneous hydrogen charging (cathodic electrolytic charging). The latter can then be compared with similar experiments done in Finland (Yagodzinskyy et al. 2012). Creep testing is also being done to try to quantify grain boundary sliding in the material, as a comparison with the measured grain boundary sliding during tensile testing (Pettersson 2010).
- Further testing will be done to investigate crack growth and the effects of slow loading with the purpose of verifying the developed creep models. Welded metal, of material with the weld quality that is expected for the refined welding method now specified (see Section 12.6 “Sealing and testing of the weld”), thereby needs to be further tested. Similarly, copper with inhomogeneous sound attenuation will be creep-tested.
- The possibilities of using other creep equations will be investigated, in particular to see how cold working of the copper should be accounted for in the models, for example by including kinematic and isotropic hardening. Furthermore, work remains to be done to be able to implement the fundamental creep equations in finite element models.

The development work with computations (mainly using the finite element method) for a full-sized canister (isostatic load, inhomogeneous load, shear, including creep of copper) is being done within the framework of the PSAR, see Section 12.3 “Canister design – analyses of the canister”.

24.2.4 Thermal expansion

Conclusions in RD&D 2010 and its review

In RD&D Programme 2010, the process was not deemed to require a research programme, and the process was not commented on in the review.

Newfound knowledge since RD&D 2010

In its review of SKB’s application for the final repository for spent nuclear fuel in Forsmark, SSM requested clarifying information regarding difference in thermal expansion (SKBdoc 1323062, 2. Skillnad i termisk expansion, C05). Clarification was desired regarding to what extent SKB had, in the calculations in Section 3.4.4 in SKB (2010o), taken into account the fact that local strains based on the geometry of the canister and defects in the welded area (e.g. JLH, joint line hooking) can cause higher strain levels than the calculated homogeneous strain level.

SKB provided clarification on 2 Feb. 2012 (SKBdoc 1333256), which showed that thermal load alone does not give rise to any stresses on the canister with the initial gap of the canister in the reference design. A supplementary account with an integrated analysis of external loads, different temperature evolution, internal gas pressure and thermal expansion was submitted to SSM in 2013.

Programme

Continued work in the area is included in the work described in Section 12.3 "Canister design – analyses of the canister".

24.2.5 Deformation from internal corrosion products

Conclusions in RD&D 2010 and its review

Miniature canisters with pre-drilled holes (one millimetre in diameter) were installed in the Äspö HRL during 2006. The purpose of the experiment, called MiniCan, was to investigate how water gets into the canister if there is a hole in the copper shell and how corrosion of the insert proceeds.

Five miniature canisters are installed individually in cages of stainless steel that holds the surrounding bentonite clay in place. Each canister has a set of measurement instruments installed inside the steel cage, but outside the copper canister. They include electrodes for various electrochemical measurements to monitor the corrosion process, as well as carefully weighed iron and copper coupons for post-analysis of mass loss (gravimetry) as a direct measure of actual corrosion. Technical details concerning the design and execution of the experiment are given in Smart and Rance (2009).

Up to and including RD&D Programme 2010, no canister had yet been retrieved for analysis. However, measurements of groundwater chemistry (in both borehole and canister) as well as electrochemical measurements had been going on since the start of the experiment. The conclusions that could be presented in RD&D Programme 2010 were a relative general increase in the concentration of iron(II) ions, and generally declining pH. Furthermore, it was pointed out that both of these hydrochemical changes could be the result of microbial activity in the experimental environment. Another conclusion drawn in RD&D Programme 2010 was that the potentials decrease as the oxygen present initially in the bentonite's pore water is consumed. This was also confirmed by gas analyses. The electrochemical measurements (AC impedance, polarization resistance, electrochemical noise as well as resistance) of the corrosion of both iron and copper have, however, shown unreasonable and in some cases contradictory results (such high corrosion rates that the electrodes would be completely dissolved).

In its review of RD&D Programme 2010, SSM took a positive view of SKB's plans to retrieve a miniature canister to compare gravimetric and electrochemical evaluation of the copper corrosion rate.

Newfound knowledge since RD&D 2010

The chemical and electrochemical measurements on the canisters in the Äspö HRL have continued and additional measurement results have been published (Smart et al. 2011, 2012a). In short, it can be said that the increase in the concentration of iron(II) ions that was pointed out in RD&D Programme 2010 has in some cases continued and in other cases been virtually unchanged. Similarly, the decline in pH has in some cases continued and in other cases seems to have stopped. Otherwise, no considerable changes have occurred in the chemistry of the water or the electrochemical measurements.

In order to investigate the question of microbial influence on the course of the experiment, analyses of microbes and dissolved gases (which may derive from microbial metabolism) have been performed on water samples from both canisters and boreholes (Lydmark and Hallbeck 2011). One of the more important results is that sulphate-reducing bacteria (SRB) are present in the experiment and that their concentration (quantified as most probable number and specified in number of cells per millilitre of water) in the space inside the bentonite has increased since 2007. For one experiment, a water sample was also taken in the borehole outside the bentonite buffer. It was found that the concentration of sulphate-reducing bacteria there was 100–1,000 times lower than inside the bentonite. This may be a consequence of the fact that the activity of the sulphate-reducing bacteria is dependent on nutrients present in the bentonite (organic matter) or of hydrogen formed when iron components (installed inside the bentonite buffer) in the experiment corrode.

Since RD&D Programme 2010, experiment 3 has been interrupted and the canister has been retrieved and analyzed with respect to e.g. corrosion of iron and copper, changes in the dimensions of the canister and occurrence of microorganisms (Smart et al. 2012b). There has been little corrosion of the iron insert during the relatively short time the experiment has been in progress. Canister 3 had two pre-drilled holes in the copper shell. Inside each of them were pitted areas in the iron insert about ten cubic millimetres in size; otherwise there has been a relatively even corrosion of the surface of the iron insert. Measurements of the outside dimensions of the copper canister before and after exposure show no measurable change (accuracy of measurement ± 0.05 millimetre). Similarly, measurements of tensile stresses in the copper canister have been made throughout the experiment without any changes being detected (Smart et al. 2012b).

The gravimetric analysis of the iron coupon is important for the quantitative evaluation of the corrosion. The iron coupon (one cubic centimetre) has been completely transformed to corrosion products (above all iron sulphide). No metallic iron could be detected in the post-analysis (Smart et al. 2012b). Since corrosion of iron has been extensive due to microbially formed sulphide, a precipitate of iron sulphide (FeS) has been formed as a coating on both iron and copper surfaces in the experiment. This coating of iron sulphide has also been formed on the copper electrodes used to monitor the corrosion process electrochemically, which is a possible explanation for the contradictory results. Both impedance and polarization resistance in the copper electrode were affected by the electrical properties of the surface coating of iron sulphide (Smart et al. 2012b).

When canister 3 was retrieved, all parts of the canister, the bentonite buffer and the water in the borehole were sampled for microbial analysis. The analysis confirmed previous sampling and analysis of the water. Sulphate-reducing bacteria occur to different extents on all sub-areas (Hallbeck et al. 2012).

Programme

Operation of the MiniCan project will continue as the canisters remain in the boreholes. SKB plans to interrupt the experiment and analyze canister 5 (without bentonite) during 2013 and canister 4 (compacted bentonite) during 2014/2015. There are currently no concrete plans for the two remaining experiments, canisters 1 and 2. Since they have bentonite of the same degree of compaction as canister 3, they are at present of less interest to analyze.

24.2.6 Radiation effects

Gamma and neutron radiation can affect the material properties of copper and nodular cast iron. The effect of radiation on corrosion is discussed in Section 24.2.8 "Corrosion of copper canister".

Conclusions in RD&D 2010 and its review

It was stated in RD&D Programme 2010 that the theoretical studies would continue, above all with calculations of solubility and diffusivity of copper in nodular cast iron, and with studies of the mechanisms for precipitation of copper at low temperatures, especially in interaction between copper clusters and vacancies. Experiments with irradiation of nodular cast iron were planned for validation of the models.

In its review of RD&D Programme 2010, SSM took a positive view of SKB's plans and the supplementary experimental studies. Furthermore, SSM considered that SKB should study whether Late Blooming Phases (LBP) is a possible mechanism for embrittlement of the nodular cast iron insert. The phenomenon has been detected in reactor pressure vessel steel, where other alloying elements (e.g. nickel, phosphorus, manganese and silicon) cause embrittlement effects that appear after a relatively long time and in such a way that the embrittlement rate increases markedly after a certain time in operation.

Newfound knowledge since RD&D 2010

Calculations have been carried out for the experimental part to investigate how the production of defects in the matrix by electron irradiation could be used to study the mobility and precipitation of copper in the iron matrix. The kinetics must be accelerated with an increased dose rate. The preliminary conclusion is that an electron beam with an energy of 2.5 megaelectronvolts (MeV) at 175°C would be possible to use. The samples that can be irradiated must, however, be fairly small, and it can be difficult to obtain representative samples from cast inserts. It has also been difficult to get the planned experiments in France going (via the French network of irradiation facilities, EMIR).

The interaction between vacancies and solutes (alloying elements or impurities) in the iron matrix has been studied by ab initio electronic structure calculations (Gorbatov et al. 2011a, b). The effects of magnetic state have been investigated by means of the extreme cases completely ordered (ferromagnetic) and completely disordered (paramagnetic). In the case of copper (and other impurity atoms) it can be concluded that the vacancy-impurity interaction is attractive and stronger at a higher degree of magnetic order. This agrees with experimental results where the diffusion of certain impurity atoms in the iron matrix is higher than the self-diffusion of the iron. The magnetic state also has a great influence on the elastic anisotropy of iron and iron-chromium alloys (Razumovskiy et al. 2011a, b). With this as a basis, the work of studying the solubility and diffusivity of copper in iron will continue.

Programme

Since it has been difficult to get experiments with electron irradiation of nodular iron specimens in France going, a feasibility study has been ordered from Studsvik Nuclear AB to make an unprejudiced attempt to devise an experimental plan.

The work of modelling the properties of the iron will continue. Thermodynamic modelling of phase diagrams (Calphad, CALculations of PHAse Diagrams) and ab initio calculations for continued analysis of vacancy-solute interactions are being used to investigate whether Late Blooming Phases can occur in the nodular cast iron. The work with ab initio calculations will take into account dislocation dynamics, among other things (Ph. D. project at KTH, see Section 24.2.2 “Deformation of insert”).

24.2.7 Stress corrosion cracking of insert

Conclusions in RD&D 2010 and its review

In RD&D Programme 2010, the process was not deemed to require a research programme, and the process was not commented on in the review.

Newfound knowledge since RD&D 2010

In its review of SKB’s application for the final repository for spent fuel in Forsmark, SSM requested (on 19 Dec. 2011) clarifying information (SKBdoc 1323062) regarding stress corrosion cracking in the cast insert. The immediate question concerned why water in liquid form cannot occur in the canister, as was claimed in the supporting material for SR-Site (SKB 2010o, Section 3.5.3).

SKB provided clarification (on 2 Feb. 2012, SKBdoc 1333256) which corrected the non-updated claim that liquid water cannot occur (the claim pertained to canisters sealed with an electron beam weld) and provided an updated analysis. The essential conclusions in this analysis are that no studies showing stress corrosion cracking on nodular cast iron in nitrate solution have been found in the literature (Reynaud 2009), but that any nitric acid formed primarily causes general corrosion. The mechanism (slip dissolution) that is considered to be the cause of stress corrosion cracking in concentrated nitrate solution (King 2010a) does not work in the presence of high general corrosion. In order for stress corrosion cracking to occur, tensile stresses must also be present in the material. In the repository, tensile stresses in the insert will only occur locally and in small areas (Dillström et al. 2010).

Programme

The process is not judged to require further research.

24.2.8 Corrosion of copper canister

Conclusions in RD&D 2010 and its review

The results of several research areas were presented in RD&D Programme 2010. The question that was given the most attention was the noticable discussion of whether copper corrodes in pure water, where no clear interpretation could be made of the various experiments that had been conducted. Other topics that discussed up were the morphology of growing copper sulphide films, possible differences in the corrosion tendency of welded or cold-worked copper, microbial sulphide production in compacted bentonite, and evaluations of if and how electrochemical measurement methods can be used to estimate corrosion in in situ tests at the Äspö HRL.

In the review of RD&D Programme 2010, a number of consultation responses were submitted on the process of corrosion of the copper canister, which are described in SSM's review report (SSM 2011). SSM's review comments and the Swedish National Council for Nuclear Waste's viewpoints are summarized here below.

SSM assessed that more research was needed in the area of copper corrosion in pure oxygen-free water in order to achieve a satisfactory understanding of the corrosion mechanism for copper in pure oxygen-free water. Furthermore, the possible impact of the groundwater chemistry on the proposed corrosion mechanism needs to be explored. SSM said that SKB should try to experimentally verify the corrosion rate for copper in order to obtain further evidence in support of the hypothesis that the copper corrosion rate is limited by mass transport of sulphide. SSM also said that SKB should study the relationships between the corrosion rate of copper exposed in a laboratory environment, archaeological findings and possible natural analogues in order to link theories of copper corrosion with models of copper corrosion in the repository environment. Corrosion mechanisms that produce hydrogen gas may also lead to hydrogen uptake by the copper metal. The analysis should also include an investigation that clarifies the possible impact of an elevated hydrogen concentration in the copper shell on the mechanical integrity of the canister.

SSM found that more research is needed on the corrosion properties of copper from the time of deposition until the surrounding buffer has been resaturated. SSM considered that the influence of radiolysis on the corrosion of copper should be elucidated before and after the surrounding buffer is saturated with water.

SSM contended that SKB should further study the causes of migration of corrosion products in bentonite and how this can affect the conceptual model that SKB uses for copper corrosion in the safety assessment. Furthermore, the forms of occurrence of the corrosion products in bentonite and the swelling and mechanical properties of the bentonite should be further investigated with respect to the concentration of copper corrosion products in the bentonite. Possible changes in the properties of the bentonite should then be related to the evolution of the repository. SSM also said that SKB should characterize copper corrosion products on the copper canisters in the Prototype experiment in the Äspö HRL that was planned to be retrieved during 2011.

SSM took a positive view of the work being done to study the risk for the establishment of biofilms of sulphate-reducing microbes on the surface of the copper shell and their influence on copper corrosion. SSM contended that SKB should shed light on the interaction between different types of microbes in relation to the availability of different energy sources.

SSM also expressed the opinion that the electrochemical assessment of differences in the corrosion rate between weld metal and parent metal needs to be complemented by long-term exposure. This to gravimetrically quantify the difference in corrosion rate between weld metal and base material.

The Swedish National Council for Nuclear Waste was gratified to see that SKB is taking copper corrosion research seriously and is focusing more on studies of copper corrosion and noted that experiments are presented to a reference group that includes critical scientists, environmental organizations and municipal representatives. The Council further noted that the studies of copper corrosion under repository-like conditions are aimed at avoiding the problems associated with the interpretation of results of more short-duration and limited laboratory experiments. It is important to obtain in-depth knowledge concerning the evolution of the environment in compacted bentonite and the entire final repository system and how this affects the corrosion rate.

Newfound knowledge since RD&D 2010

The individual topic within the area of copper corrosion on which SKB has focused most is the question of whether copper corrodes in pure oxygen-free water to a much greater extent than is predicted by the thermodynamics of known copper compounds. A number of new studies have been presented, but the results have not yet led to an unequivocal interpretation of what happens to copper in pure water.

Experiments at KTH involving measurement of hydrogen gas pressure have continued and show a hydrogen pressure in the millibar range (Hultquist et al. 2011, 2013). The maximum hydrogen gas pressure reached in a closed system increases with increasing temperature, and the authors point out that the maximum pressure has a similar temperature dependency as the concentration of hydroxide ions in the equilibrium for the water's autoprotolysis (Hultquist et al. 2011). Studies of copper surfaces exposed to oxygen-free (anoxic) water are presented in Hultquist et al. (2013). Corrosion products in the form of oxide grains of a size up to 0.5 micrometres are observed in studies with a scanning electron microscope (SEM), X-ray photoelectron spectroscopy (XPS) indicates a superficial product that contains both hydroxide and oxide, while secondary ion mass spectroscopy (SIMS) detects oxygen in the outermost 0.3 micrometre and hydrogen further in. No analyses or discussions of the oxidation number of copper in the products are presented. Further, several of the copper specimens have undergone long-term exposure to the atmosphere prior to analysis, making the results difficult to interpret.

SSM has financed a study in cooperation with Studsvik Nuclear AB, Aalto University in Helsinki and VTT, Technical Research Center in Finland. The experiment with a set-up for measurement of hydrogen gas pressure in an ultrahigh vacuum system at Studsvik Nuclear is presented in Becker and Hermansson (2011). The results show that hydrogen gas is evolved in the first two experiments with a copper foil in pure water, while an experiment with a platinum foil produced only about 1/25th of this hydrogen gas evolution. The measured quantity of hydrogen gas is less than that which corresponds to the amount of copper that could be measured in the aqueous solution in the experiments with a copper foil. The aqueous solution contained substances that could derive from the glass vessel with copper and water, while other metals encountered were less expected (copper in the platinum experiment, and copper and palladium that could be washed off of the glass after the experiments). The pressure gauge experiments were interrupted after about 500 hours and no further studies have been conducted with the test set-up at Studsvik Nuclear.

Experiments conducted on behalf of SKB have also detected hydrogen. They involve experiments with copper rods in glass vials (Bengtsson et al. 2013), and with copper in water in a set-up with ultrahigh vacuum. The results from the development work with the vial experiments show hydrogen gas evolution of up to 6 millibars at 70°C. It was also concluded that the presence of oxygen prevents hydrogen gas evolution, and that hydrogen gas evolution continues after emptying of the hydrogen gas. One example of hydrogen gas evolution from vials with copper and water is shown in Figure 24-2.

The experiments with ultrahigh vacuum are being conducted at the Ångström Laboratory at Uppsala University. They include both set-ups with pressure measurements (see Figure 24-3) and analyses of copper foils in water in glass vessels sealed with palladium membranes, performed in a glove box with an oxygen-free environment. Copper foils that have been exposed to ultrapure oxygen-free water for six months have been analyzed by XPS and the water by mass spectrometry (ICP-MS, Inductively Coupled Plasma – Mass Spectroscopy). The preliminary results of the surface analyses reveal neither monovalent nor divalent copper. The oxygen present on the surface is not bound to copper, but rather to e.g. carbon or in water. The analyses of the water show a copper concentration of about 5–6 ppb, which is equivalent to about 10^{-7} molar.

More than 20 years ago, the National Board for Spent Nuclear Fuel initiated an experiment with copper wires in water in vials with palladium membranes. The vials have been stored since then with SP, the Technical Research Institute of Sweden, and now the copper wires have been analyzed (Möller 2012). The analyses could not detect more corrosion than had been measured in similar experiments that were interrupted after two years. It could not, however, be clarified how permeable the membrane had been to hydrogen gas, since the vials had been stored in a horizontal position and the membrane may have been partially wetted by the water.

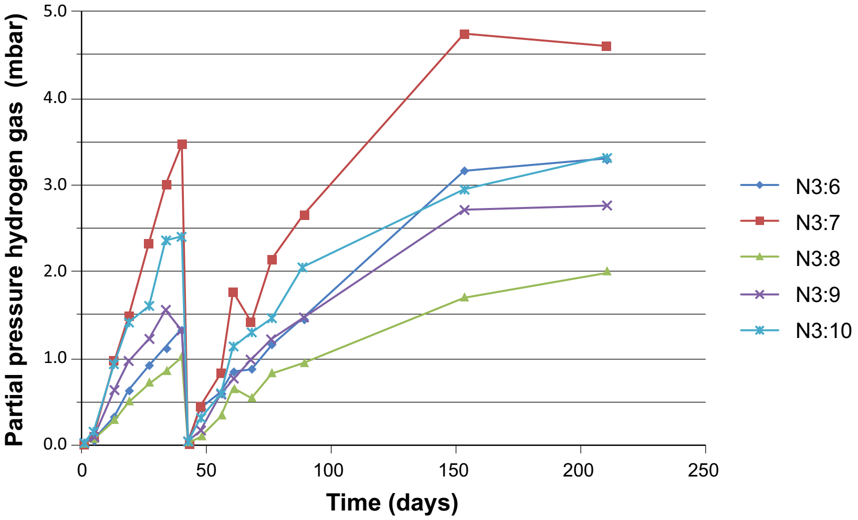


Figure 24-2. *The partial pressure of hydrogen gas in experiments with copper in water in vials. After 40 days, the vials were emptied of hydrogen gas, which was replaced with nitrogen, after which the measurements of hydrogen gas continued. Each curve represents one vial (experiment series N3, see Bengtsson et al. 2013).*

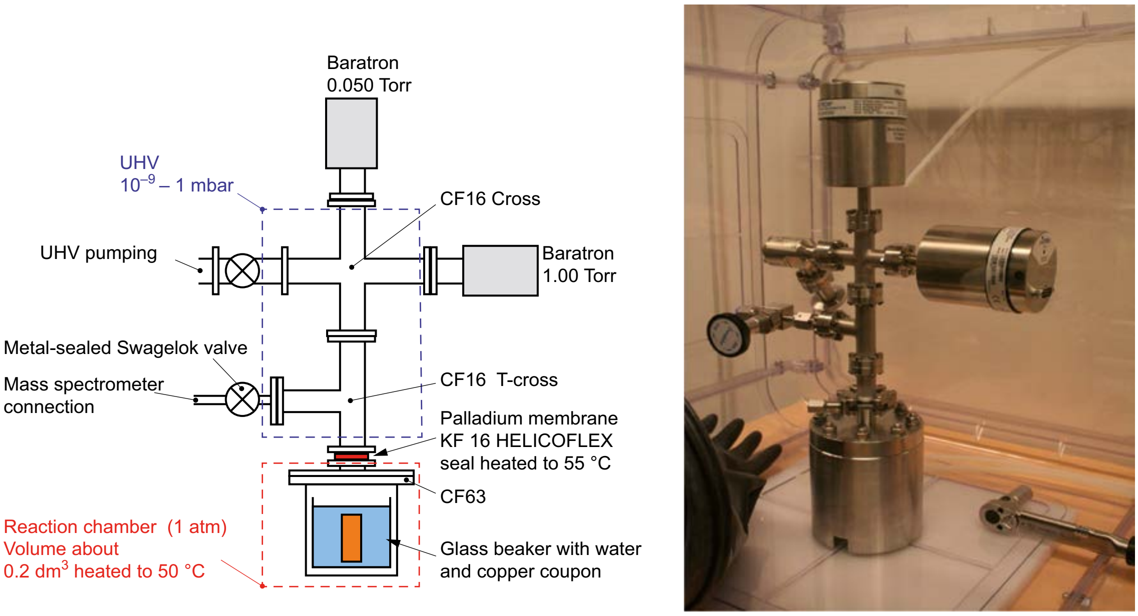


Figure 24-3. *Experimental set-up for investigation of copper corrosion in pure water with ultrahigh vacuum.*

The electrochemical studies of copper in water have continued, and a kinetic model containing two adsorbed species has been developedconstructed (Bojinov et al. 2010, Betova et al. 2013a). No reducible layer could be found on the copper surface during the experiment. Experiments with a duration of up to four months indicate more electrochemically active properties of the absorbed layer, which is however stabilized after 2,000–2,500 hours. The kinetic model has also been extended with the solubility of copper(I) oxide and the disproportionation equilibrium for copper. Experiments with chloride in the solution have been initiated (Betova et al. 2013b).

SKB is also continuing the search for possible unknown copper compounds with oxygen and hydrogen, via both experiments and calculations. Copper hydride, CuH, has been produced with the aid of hypophosphoric acid from a copper(II) solution and its structural and optical properties have been studied by X-ray diffraction and infrared spectroscopy (FTIR, Fourier Transform Infrared Spectroscopy) (Korzhavyi et al. 2012). The copper hydride can be preserved in cold water, but is not stable in air or vacuum at room temperature. Quantum mechanical calculations (DFT, Density Functional Theory) have been carried out to investigate the thermodynamic stability of different configurations of copper-oxygen-hydrogen compounds. Calculations for copper(I) oxide, copper(II) oxide and copper hydride have been performed and compared with experimental thermodynamic data (Korzhavyi and Johansson 2010, Korzhavyi et al. 2011). Different possible configurations for copper(I) hydroxide have been investigated, and a structure that strongly resembles that of both copper(I) oxide and ice has been proposed (Korzhavyi et al. 2012). Calculations show that this compound is metastable and decomposes into copper(I) oxide and water (Korzhavyi and Johansson 2010). Different ways of producing copper(I) hydroxide have been investigated, and one obtained product could possibly be characterized as a hydrated copper(I) hydroxide, $Cu_2O \times H_2O$ (Soroka et al. 2013). In summary, these studies have not proved the existence of any previously unknown copper-oxygen-hydrogen compound that could explain the hydrogen gas evolution observed from copper in oxygen-free water.

The reactivity of copper surfaces in water has been studied. A literature review (Johansson and Brinck 2012) has gone through both experimental and theoretical studies. Quantum mechanical calculations (density functional theory) have been carried out to study dissociation of water and desorption of hydrogen gas on the ideal [100] surface (Johansson et al. 2011). The conclusion drawn from these studies is that surface reactions can lead to the formation of hydrogen gas in limited amounts. On an ideal [110] or [100] copper surface, 2.4 nanograms per square centimetre (ng/cm^2) of hydrogen gas could form if it is assumed that water cleavage and the hydrogen-forming reaction proceed until a monolayer of hydroxyl groups has been formed. If the oxidation progresses to a monolayer of oxide, or if a microscopically rougher surface is assumed, this amount could increase, but hardly even by a factor of 10. This is not sufficient to explain the hydrogen gas evolution that has been observed from copper in oxygen-free water.

Additional survey reports have been published. Results from studies of the properties of copper(I) oxide and oxidation of copper, above all the initial oxidation studied at an atomic level, have been compiled (Korzhavyi and Johansson 2011). A review of published results for corrosion in pure water were presented at the conference "The Fourth International Workshop on Long Term Prediction of Corrosion Damage in Nuclear Waste Systems" in July 2010 (King and Lilja 2011, Åkermark 2013, King et al. 2013a) and compiled in December 2010 in an SKB report (King 2010b). The state-of-the-art report on copper corrosion has been updated as a basis for SR-Site (King et al. 2010). The material is also summarized in King et al. (2013b). The calculations of corrosion for SR-Site are collected in SKB (2010p).

The effects of gamma radiation on corrosion of copper are being studied in a Ph. D. project at KTH. Experiments have been conducted with copper cubes in pure water in a nitrogen atmosphere at radiation dose rates of 0.37 and 0.77 kGy/h. The surfaces and the aqueous solution have been investigated by means of various spectroscopic methods. The initial studies (Björkbacka et al. 2012, 2013) show higher corrosion for irradiated specimens than for unirradiated ones. Corrosion appears as both formation of copper(I) oxide and local cavities. However, it must be pointed out that these dose rates are approximately 1,000 times greater than the maximum dose rate at the outside of the canister in the repository, why more studies of the mechanism must be done to understand how the results can be extrapolated to a lower dose rate.

Morphology and film growth have been studied by means of electrochemical impedance measurements and scanning electron microscopy for copper in sulphide solution at different concentrations (Chen et al. 2010, 2011a). At a higher concentration (5×10^{-4} molar) the film becomes more compact and adherent (see Figure 24-4a). The film exhibits a parabolic growth rate, which indicates that diffusion of copper(I) through the film is rate-determining. At a low sulphide concentration (5×10^{-5} molar), a porous film is formed (see Figure 24-4b) whose growth is nearly linear. This indicates that growth is limited by diffusion of the sulphide in the pores on the film (Chen et al. 2011b). If the sulphide in the solution is depleted, diffusion in the solution becomes rate-determining.

The study of bacterial coverage on copper and titanium surfaces in an environment of compacted bentonite saturated with groundwater has been completed (Persson J et al. 2011). The question concerned the possibility of formation of a biofilm of sulphide-producing bacteria on copper in compacted bentonite. The presence of bacteria could be detected, with an estimated maximum coverage of 3.7 percent. This can be considered to be at the lower end of what is called a biofilm in other known systems. The number of bacteria was higher on the titanium surface than on the copper surfaces. Neither bentonite density nor addition of hydrogen gas or lactate had any effect on the number of bacteria on the surfaces. Bacterial activity in the circulating water increased significantly with such additions, however.

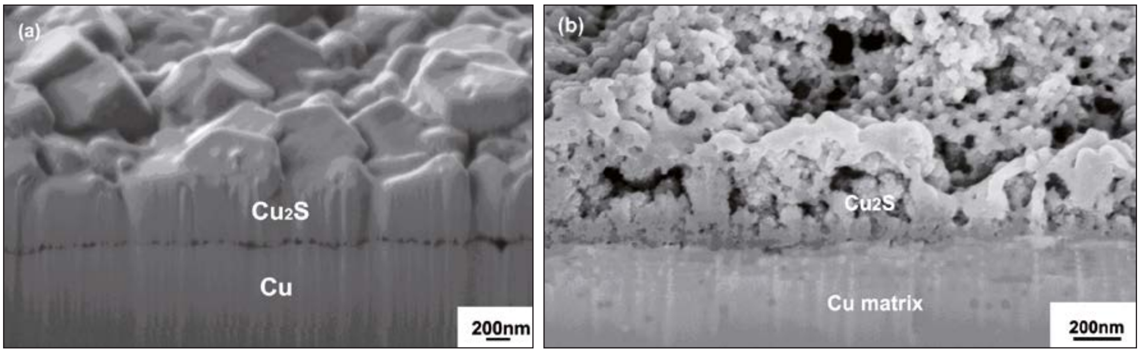


Figure 24-4. *Cross-section of copper electrodes with copper sulphide, formed in oxygen-free water after 4,000 hours. The composition of the solution was a) 0.1 molar NaCl + 5×10^{-4} molar Na_2S, b) 0.1 molar NaCl + 5×10^{-5} molar Na_2S (Chen et al. 2011a).*

The studies with the electrodes from the LOT experiment in the Äspö HRL (A2 test parcel) have continued with both ER (electrical resistance) electrodes (Rosborg et al. 2011a) and EIS (electrochemical impedance spectroscopy) (Rosborg et al. 2011b). The ER electrodes were retrieved and examined after more than four years of operation (Rosborg et al. 2012). The corrosion products consisted of cuprite (Cu_2O) and paratacamite ($Cu_2(OH)_3Cl$). The average corrosion rate was 3 micrometres per year. The corrosion was unevenly distributed over the electrode surface, but without any signs of active pitting. A comparison was done between the different types of electrodes (ER and EIS) and both types of measurements can be used to follow the corrosion rate over time, at least in an oxidizing environment as in these experiments. The measurements with the electrodes show a declining corrosion rate which eventually decreases to less than 1 micrometre per year.

Electrochemical real-time measurements of corrosion in the Prototype Repository have been made via copper electrodes installed in the bentonite buffer on a number of occasions between 2004 and 2010 (Rosborg 2013a). They show that the average corrosion rate has been lower than 1.3 micrometres year. Previous experience of these measurement methods and correction for the oxidation state of the actual corrosion products (mainly Cu_2O) suggests that the actual corrosion rate rather has been closer to 0.4 micrometre per year. It should be noted that measurements of the potential in the Prototype Repository just prior to retrieval show that the immediate chemical environment has been more or less oxidizing throughout the exposure period, but mildly oxidizing during 2010 (–30 to –60 millivolts against the standard hydrogen electrode). Post-test analysis of a copper electrode that had been installed in the bentonite buffer in deposition hole 5 showed that cuprite (Cu_2O) is the principal corrosion product (Rosborg 2013b). Malachite ($Cu_2(OH)\,{}_2CO_3$) also occurs on the surface of the electrode. The copper electrode shows no unmistakable signs of pitting.

The ring on top of the lid from the canister from deposition hole 5 has been analyzed with respect to localised corrosion (Taxén et al. 2012). Scanning electron microscopy shows the occurrence of up to 6-micrometre-deep pits that may represent localised corrosion. Evaluation of the importance of these results is hindered by the fact that the canister material was not characterized before deposition. The characterization that has now been done will serve as a reference for future analyses of the canister surface on canisters that have been deposited in the Prototype Repository for a longer time.

As described in Section 24.2.5 "Deformation from internal corrosion products", a canister from the MiniCan project has been retrieved and analyzed after about five years' exposure in the Äspö HRL. Gravimetric analysis of the mass loss from a copper coupon during the test period gave a corrosion loss of 0.15 micrometre per year, which is an order of magnitude lower than previous measurements in the LOT experiment (Karnland et al. 2009, 2011). It is important to point out that this corrosion rate cannot be used in any simple way for extrapolation in time, since an oxygen-containing period is included in the measurement and since a large amount of sulphide has been produced locally by microorganisms due to the supply of hydrogen gas from corrosion of iron. No signs of localised corrosion attacks have been detected by optical microscopy, but this is being subjected to further post-test analysis.

In summary, important new results have been obtained concerning above all copper corrosion in pure water since RD&D 2010. However, with regard to this and other corrosion-related topics where progress has been made, further work is needed to reach conclusions that can be used directly in the assessment of the long-term safety of the KBS-3 repository.

Programme

SKB's studies of corrosion on copper will continue in order to provide further knowledge concerning the details in the corrosion mechanisms, which is necessary in view of the central role played by the integrity of the canister in the safety assessment. The largest efforts will continue to be devoted to studies of copper in pure anoxic water along essentially the same lines as in the current work. Measurements of hydrogen gas evolution are continuing both under high vacuum at Uppsala University and in vial experiments. Linked to this, analyses are being performed of aqueous solutions, copper surfaces and other exposed parts of the experimental set-ups. The electrochemical experiments are continuing, where measurement of hydrogen gas is under development.

Conceivable copper-oxygen-hydrogen compounds will continue to be studied, experimentally by studies of metastable compounds (including copper hydride) and theoretically by quantum chemical calculations. Further quantum chemical studies will be carried out to gain a better understanding of the reactivity at the interface between copper and water, and the effects of solvents on this reactivity will be investigated, among other things. The reference group for copper corrosion that has been working for three years will continue to operate for the purpose of giving viewpoints to SKB on further experiments, above all the experiments at Uppsala University, and giving the participants insight into SKB's work on the subject.

The Ph. D. studies of the effects of gamma radiation on corrosion of copper are continuing, with a focus on understanding the mechanisms involved in oxidation. Of vital importance is also to investigate whether any dependency on dose or dose rate can be identified.

The focus in the electrochemical studies of copper in sulphide solution is on determining the site of the cathode reaction and studying in different ways what determines the formation of the film and its properties, in solutions with sulphide and higher concentrations of chloride.

The role of sulphides for corrosion in the repository is also being studied in other studies. A literature review is being conducted regarding the solubility of pyrite (FeS_2) in bentonite. Sulphidation of copper(II) compounds is being investigated by X-ray spectroscopy. Another attempt is being made to determine whether sulphate-reducing bacteria can be active and produce sulphide in bentonite with different degrees of compaction. The experimental set-up has been specially designed to avoid sulphide production in the circulating liquid phase, which has complicated the interpretation of previous experiments.

The development of electrochemical methods to investigate the difference in the corrosion tendency of different copper materials (welded, cold-worked etc) continues.

In the analysis of canister 3 in the MiniCan project, no attacks of localised corrosion could be identified by optical microscopy, but the possibility that this has happened on a smaller scale cannot be ruled out. SKB is therefore planning further post-test analyses of the copper material. Scanning electron microscopy (SEM) will be used in these analyses. The analyses will also include an evaluation of possible effects of stress corrosion cracking.

In response to SSM's request for supplementary information for the licence application, a probabilistic analysis will also be initiated of the probability of localised corrosion, including the effects of possible salt enrichment on the canister surface, see further Section 24.2.11 "Precipitation of salt on canister surface", along with a supplementary study of the effect on corrosion of earth currents, see further Section 24.2.10. "Earth currents – stray current corrosion".

24.2.9 Stress corrosion cracking of copper canister

Conclusions in RD&D 2010 and its review

RD&D Programme 2010 stated that the research programme would continue in the area of stress corrosion cracking (SCC), above all with experimental studies in a sulphide-containing environment and with simulated groundwater with ammonium.

In its review of RD&D Programme 2010, SSM discussed experiments conducted at VTT in Finland (Arilahti et al. 2010) showing that sulphur can diffuse into the grain boundaries of copper in artificial groundwater containing sulphide. SSM judged that more research is needed in this area to clarify whether sulphide-containing water can also cause brittleness phenomena in Cu-OFP. SSM also said that SKB should conduct further studies of the geochemical evolution of different substances and microbial activity that can give rise to SCC in copper as a function of time.

The Swedish National Council for Nuclear Waste pointed out that SCC is usually caused by residual stresses and that residual stresses must for this reason (among others) be carefully measured and analyzed, especially in the welds in the canister.

Newfound knowledge since RD&D 2010

The critical review of proposed mechanisms for SCC on copper has been completed (King and Newman 2010). The conclusions in the study, as stated already in RD&D Programme 2010, 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-founded mechanism for SCC under reducing conditions.

The behaviour of copper in simulated groundwater with ammonium has been studied in a joint project with Posiva (Kinnunen and Varis 2011). The studies were conducted as slow strain rate tests under potential control (to simulate an oxidizing environment in the repository) with 0.1, 0.01 and 0.001 mole of ammonium per litre. Scanning electron microscope (SEM) images were analyzed, along with electrochemical impedance spectroscopy (EIS) measurements. The conclusion was that it was only at the highest concentration of ammonium (0.1 mole per litre) that SCC could be observed, and then especially at a potential just above the Cu^{+}/Cu^{2+} equilibrium. At lower potentials (where monovalent copper is stable), and at the lower ammonium concentrations, no SCC was noted; the fractures were completely ductile. At the highest potentials, where divalent copper is stable and with 0.1 and 0.01 mole of ammonium per litre in the solution, there was more extensive general corrosion.

How the risk of SCC can be judged on the basis of the surrounding chemical environment is discussed in the background material to SR-Site, in the state-of-the-art report on copper corrosion (King et al. 2010, Section 6.2.2.1). The possibility of using threshold values for the concentrations of nitrite, ammonium and acetate in the analysis is limited, since they must then be specified for many different environments. In previous studies the limit of 0.001 mole per litre has been mentioned for nitrite (King et al. 2010), and in the above study (Kinnunen and Varis 2011), no SCC was obtained at 0.001 mole of ammonium per litre. In SR-Site, representative groundwater data were determined for nitrite, ammonium and acetate – see Section 6.1 in the Data Report (SKB 2010q) – and these are at most 0.0002 mole per litre (for ammonium under "submerged" conditions, i.e. beneath the sea).

When it comes to SCC and cracking in copper in sulphide-containing solutions, the work at VTT has continued within the Finnish KYT programme. In the experiment, CT specimens were loaded in simulated groundwater with 10, 100 and 200 milligrams of sulphide per litre. The general conclusion was that SCC in in sulphide-containing groundwater at room temperature could not unequivocally be shown (Arilahti et al. 2011a). No certain conclusions could be drawn regarding crack growth during the testing (Arilahti et al. 2011b). The fracture surface was also analyzed in the study with optical microscopy and SEM/EDS (scanning electron microscopy/energy dispersive X-ray spectroscopy), which detected sulphide inclusions in the grain boundaries and to greater amounts nearer the fracture plane (but not only there). With a penetration depth of 10 millimetres in six weeks, this would entail a relatively high diffusion constant, on the order of 10^{-11} square millimetres per second (mm^2/s) (Arilahti et al. 2011a, b).

The analysis of the CT specimens has continued with STEM analyses (analyses with a Scanning Transmission Electron Microscope) of reference material and material that has been in simulated groundwater with 200 milligrams of sulphide per litre (Pakarinen 2011). All specimens, both reference specimens and material that has been in sulphide-containing water, contained small sulphur-rich precipitates, which appeared as holes in the matrix after electrolytic polishing. Small quantities of phosphorus were noted in the grain boundaries. Chlorine and other elements present in the groundwater were found in one grain boundary.

In another study in the Finnish KYT programme (Arilahti et al. 2012), round specimens with notches were loaded in simulated groundwater with 10 milligrams of sulphide per litre and compared with unloaded flat specimens. The specimens were ruptured by fatigue in air and the fracture surfaces were analyzed by means of SEM/EDS. Small quantities of chloride, sulphur, calcium and potassium were found at many places on the fracture surfaces, with a penetration depth of 1 millimetre per week. The fracture surfaces were (up to 30 percent) covered with copper oxides, both on loaded and unloaded specimens. No clear effect could be noted on yield strength, ultimate strength or elongation at rupture after exposure to sulphide-containing groundwater. The explanation offered by the authors is the hypothesis that sulphide reacts with the copper and probably creates a pathway for other substances from the groundwater in the grain boundaries, including ions with oxygen from water molecules that react to give copper oxide. How this interaction between sulphide ions (in solution or in the form of copper sulphide) and oxygen-containing ions takes place is, however, not described in detail.

In order to further investigate SCC in sulphide-containing water and to permit comparison with the above Finnish studies, as well as the previous Japanese experiments (Taniguchi and Kawasaki 2008), SKB has carried out an experimental study (Bhaskaran et al. 2013). There, copper has been tested by slow strain rate testing, constant strain rate testing, electrochemical analyses and surface analyses, as well as examination by EBSD (Electron Backscatter Diffraction) of orientation effects of the grains. The effects of anodic or cathodic polarization were also tested. The analyses were done in 5–50 millimolar sulphide solution (equivalent to 0.16–1.6 milligrams of sulphide per litre) with sodium chloride or simulated groundwater, at 25 and 80°C. SCC could not be observed in any case. The corrosion products were in general porous and could easily be detached, but fine-grained, compact, adherent corrosion products also occurred. Some correlation could be observed between thin films and (100) orientation on the surface of the grains, but no definite conclusions could be drawn. The results are difficult to interpret, especially with the hypothesis that the corrosion is determined by the diffusion of sulphide to the copper surface. New findings have been forthcoming in the latter area, however, indicating different rate-determining steps dependent on the sulphide concentration, see Section 24.2.8 "Corrosion of copper canister".

In summary, it can be concluded that several groups have in different ways tried to repeat the Japanese experiments, but have not been able to detect any SCC. There are, however, questions concerning how the experimental observations are to be interpreted and what actually happened in the experiments.

Residual stresses in the copper shell are dealt with in Section 24.2.3 "Deformation of copper canister under external pressure".

Programme

The research programme for SCC (stress corrosion cracking) on copper is continuing, focused mainly on sulphide-containing groundwater. Further experiments are needed, since the results and interpretations obtained thus far are contradictory. The focus will be on investigating whether SCC can occur in an environment that resembles the repository environment, even though some efforts may be focused on repeating and if possible explaining other difficult-to-interpret experiments.

An ongoing post-test analysis of the retrieved canister from the MiniCan experiment may give additional knowledge on whether SCC occurs in copper in sulphide-containing groundwater.

For studies of the geochemical evolution and effects of microbial activity, see Section 26.17 "Microbial processes", where literature studies of acetogenic microbes are discussed.

24.2.10 Earth currents – stray current corrosion

Conclusions in RD&D 2010 and its review

RD&D Programme 2010 did not specify a research programme for the process and the process was not commented on in the review.

Newfound knowledge since RD&D 2010

The process is dealt with in SR-Site in the "Fuel and canister process report" (SKB 2010o, Section 3.5.6). The analysis included an estimate of potential gradients in the Forsmark area, as a basis for the description of corrosion. Measurements in packed-off boreholes were used as a approxiy for unsaturated bentonite (with high electrical resistance) and gave a potential difference of about 0.5 volt over five metres. Data from open boreholes were used to modelsaturated bentonite, which typically gave 0.05 volt, in some cases up to 0.25 volt.

Different conceivable corrosion reactions in the vicinity of an external electrical field were discussed, and the conclusion was that with these potential differences over the canister, corrosion will still be limited by the amounts of oxygen (limits the cathodic reaction) and sulphide (limits the anodic reaction) that reach the canister. Reduction of water with the formation of copper(I) oxide is limited by dissolved hydrogen that is present in the groundwater at these potential differences. The margins to localised corrosion decrease with an external electrical field, but increase with decreasing oxygen concentration.

A limited study involving measurements of polarization resistance in the transition between saturated bentonite and copper in an oxygen-free environment was done in a previous attempt to assess earth current corrosion by comparisons of different types of electrical resistance in the repository (Taxén 2011). The measurements made by AC technique gave no useful results, while measurement with direct current gave a polarization resistance of 3,100 ohm square metres (Ωm^2).

Programme

SKB is planning an update of the assessment, especially with respect to the case with partially saturated bentonite and with results from the new studies with probabilistic analyses of localised corrosion that are planned, see Section 24.2.8 "Corrosion of copper canister".

24.2.11 Precipitation of salt on canister surface

Conclusions in RD&D 2010 and its review

RD&D Programme 2010 did not specify a research programme for the process.

In its review of RD&D Programme 2010, SSM said that it is fully possible that canisters may be exposed to atmosphere-like conditions with a gas gap between the copper shell and the buffer material, especially at low groundwater flow and thereby a long resaturation time in the buffer. For this reason, SSM considered that SKB should examine under what conditions salt precipitation can occur on copper surfaces and the possible effects of this on corrosion of the copper shell.

Newfound knowledge since RD&D 2010

How corrosion under unsaturated conditions is dealt with in SR-Site is described in the background reports "Fuel and canister process report" (SKB 2010o, Section 3.5.7) and the updated state-of-the-art report on copper corrosion (King et al. 2010, Section 4.3.2). The analyses there are based on the same background data as that reported in RD&D Programme 2007. The conclusion was that the uncertainties lie more in the precipitation of salts than in the effects of salts on the corrosion process.

Programme

Studies of vapour transport in the canister–bentonite gap are described in Section 25.5.5 "Water transport under unsaturated conditions".

SKB is planning broadened studies of localised corrosion with probabilistic analyses, see Section 24.2.8 "Corrosion of copper canister".

25 Buffer and backfill

This chapter deals with research concerning the long-term function of 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. Some research described in the chapter also pertains to the clay buffer in the silo in SFR. When this is the case, this is stipulated in the text.

25.1 Buffer

The buffer's main function is to limit the flow of water around the canister. This is achieved by means of low hydraulic conductivity, so that diffusion is the dominant transport mechanism, and by means of a swelling pressure that makes the buffer self-sealing. The buffer is also supposed to hold the canister in place in the deposition hole, dampen the shear movements of the rock and retain its properties during the period being analyzed. In addition, the buffer is supposed to limit microbial activity on the canister surface and filter colloidal particles. The buffer must not significantly degrade the function of the other barriers. Quantitative design premises are defined in SKB (2009a) and are documented in Section 13.1.1 "Buffer".

Water saturation and swelling processes comprise part of the buffer's long-term evolution and cannot be determined in the initial state. Based on analyses of these processes, requirements can instead be stipulated on geometry and density in the initial state.

25.2 Backfill – requirements and functions

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 Section 13.1.2 "Backfill and plug".

25.3 The bentonite barrier in the silo in SFR

The bentonite in SFR is an integrated part of the barrier system in SFR and is analyzed in SR-PSU. The research programme for the bentonite in SFR is presented together with the buffer in this section.

The requirements on the buffer material in the silo in SFR are (Pusch 1985):

- Water flow through the material shall be insignificant.
- Gas shall be able to be evacuated so that the creation of local gas bubbles with very high pressure is prevented.
- Homogeneity shall be high.
- The properties of the material shall be preserved for a long time.
- The pressure from the buffer mass may not give rise to unacceptable deformations in the concrete structure during any phase: construction period, waste deposition or post-closure.

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.

25.4 Initial state

25.4.1 Overview

The initial state of the buffer is the state that exists when the equipment used during installation has been removed and all of the buffer's constituents have been installed in the deposition hole.

The groundwater inflow into the deposition hole and how it affects the buffer is not taken into account for the initial state. The buffer properties that must comply with the design premises and are related to long-term safety are:

- Material composition.
- Installed density.
- Installed geometry.

These properties are to some extent dependent on each other. The desired ensity, for example, is dependent on a given material composition.

The properties have traditionally been determined as an average value for an entire deposition hole. However, this presumes that the installed components (blocks and pellets) and remaining voids are homogenized to a sufficient degree. It is therefore necessary to analyze how the installed configuration will evolve in the repository in order to give feedback to the requirements on the installed buffer properties. This is discussed in Sections 25.4.4 "Bentonite composition" and 25.5.9 "Swelling".

In the process report for buffer, backfill and closure (SKB 2010m), the buffer is described with the aid of a number of variables. Most of these variables' values in the initial state are determined by the properties shown in Table 25-1.

Table 25-1. Variables for buffer and backfill.

Variable	Property	Initial state	Definition
Water content	Material composition		Water content as a function of time and space in buffer and backfill.
Gas content	Material composition		Gas contents (including any radionuclides) as a function of time and space in buffer and backfill.
Bentonite composition	Material 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.
Montmorillonite composition	Material composition		Chemical composition of the montmorillonite (including any radionuclides) in time and space in buffer and backfill. This variable also includes material sorbed to the montmorillonite surface.
Pore water composition	Material composition		Composition of the pore water (including any radionuclides and dissolved gases) in time and space in buffer and backfill.
Hydrovariables (pressures and flows)	Material composition		Flows and pressures for water and gas as a function of time and space in buffer and backfill.
Stress state	Installed density		Pressure as a function of time and space in buffer and backfill.
Pore geometry	Installed density		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.
Buffer geometry	Installed geometry		Geometric dimensions of buffer and backfill. A description of e.g. boundary surfaces inward towards the canister and outward towards the geosphere.
Radiation intensity	–	Calculated	Intensity of alpha, beta, gamma and neutron radiation as a function of time and space in buffer and backfill.
Temperature	–	Calculated	Temperature as a function of time and space in buffer and backfill.
Structural and stray materials	–		Composition of any engineering materials in the deposition holes.

25.4.2 Water content

In SR-Site it is assumed that the compacted buffer 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 ten 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 could be filled with water, but in SR-Site it is assumed that they are dry.

The initial content has no direct bearing on the long-term function of the repository. It does, however, influence how buffer blocks and pellets can be fabricated, handled and stored (this is discussed in Chapter 13 "Technology development of buffer, backfill and closure").

25.4.3 Gas content

The initial gas content follows from the water content and the porosity. If 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. At present, the intention is not to fill the pellet-filled gap between blocks and rock with water, which means that the gas content in the buffer is about 85 percent. The air in a deposition hole occupies approximately 16 percent of the total volume of buffer. The uncertainties in initial gas contents are not important for long-term safety.

25.4.4 Bentonite composition

Bentonite is the name of a naturally occurring clay material that is rich in montmorillonite and varies in composition depending on how it was 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 cubic metres of GEKO/QI, which is a sodium-converted calcium bentonite. The requirement specification says that the montmorillonite content shall be at least 65 percent.

Conclusions in RD&D 2010 and its review

SSM took a positive view of the fact that SKB has developed criteria for the concentrations of sulphur and carbon in the buffer. SSM thought that SKB should investigate whether there is also a need to develop criteria for concentrations of other substances that could affect the buffer's pH and redox buffering.

Further, SSM said that it is not enough to rely on simple acceptance criteria as a basis for material selection. Whether or not other material properties not covered by the criteria are of no importance for repository function depends entirely on how great the variations can be between possible material candidates.

In the NEA review of SR-Site (NEA 2012), the IRT (International Review Team) points out the importance of evaluating differences between the properties of different commercial bentonites with respect to sealing capacity, piping and erosion stability.

Newfound knowledge since RD&D 2010

The commercial bentonites that are of interest as buffer materials have a relatively constant composition. The smectite content (primarily montmorillonite) normally lies around 80 percent. In order for the requirement on density of the buffer to be relevant, a montmorillonite content in the interval 75–90 percent is needed. The requirement defined in SKB (2010f) is a content of 80–85 percent. Bentonite material for block fabrication will undergo comprehensive quality inspection prior to pressing. This inspection includes determination of the montmorillonite content (see Chapter 13 for plans).

Different bentonites can however have different swelling pressures at a given density. If data on FEBEX (Figure 25-1) are compared with data on MX-80 and IBECO-RWC (Figure 25-2), it will be seen that FEBEX does not give a higher swelling pressure at a given density, even though its montmorillonite content is higher. This means that the simple definition regarding montmorillonite content and density previously used by SKB is not universally applicable.

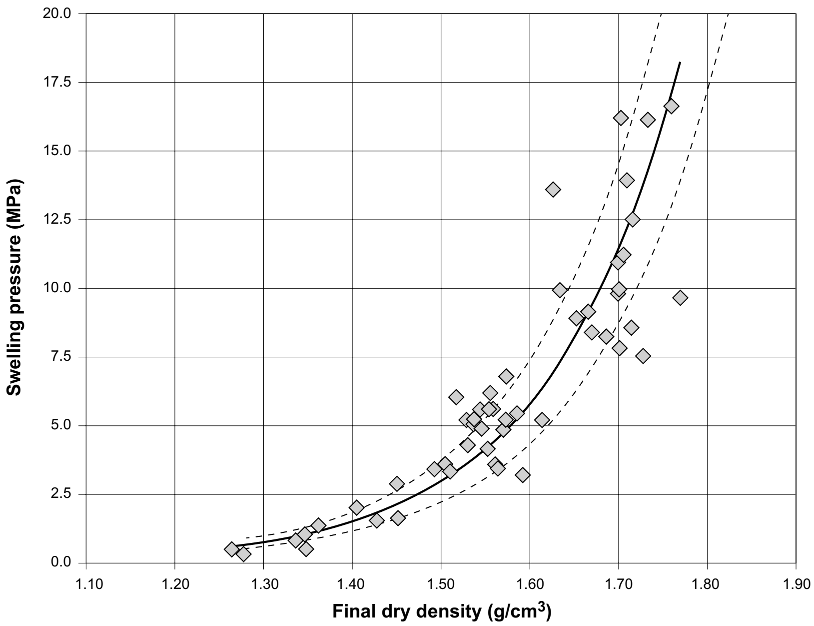


***Figure 25-1.** Swelling pressure for FEBEX bentonite as a function of dry density measured with deionized water (Villar 2002).*

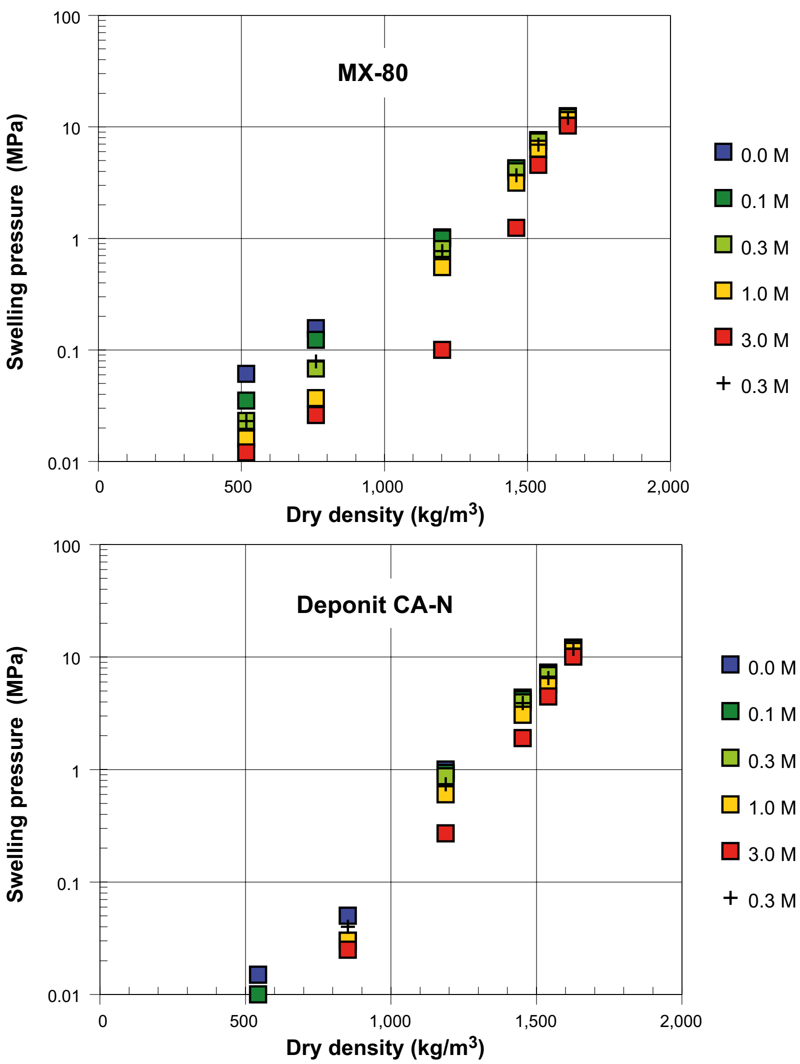


***Figure 25-2.** Swelling pressure for MX-80 in sodium chloride solution and for Ibeco-RWC (Deponit CA-N) in calcium chloride solution (SKB 2006c).*

In selecting buffer materials, material-specific relationships should therefore be plotted between density and swelling pressure and between density and conductivity. Material-specific requirements on density and montmorillonite content can then be derived from these relationships.

In the underground test "Alternative Buffer Materials" (ABM), more than ten different bentonites are being tested and compared with each other to permit an optimal choice of buffer material. The key parameters here are availability, safety provided and cost. The schematic design of ABM is shown in Figure 25-3. The hydromechanical tests of the material have revealed great variation in the delivered clays. The flow limit varied from 68 percent (Friedland) to 545 percent (MX-80) and the grain density was around 2.7 grams per square centimetre (g/cm^2), with the exception of the more iron-rich clays, which were between 2.8 and 2.9 g/cm^2. Most of the clays had a high cation exchange capacity (CEC), which is typical of bentonite clays. Extractable salts in the form of chlorides and sulphates varied between the clays, and values of up to 0.3 weight-percent chloride and 0.5 weight-percent sulphate were measured. Chemical extraction using the CBD (citrate-bicarbonate-dithionite) method gave in some cases not only high iron contents, but also elevated concentrations of manganese and silicon. This elevation of silicon (e.g. Calcigel, Rokle) was not linked to the quantity of quartz detected by X-ray diffraction, which indicates the presence of amorphous silica in some clays. The mineral phases were determined semiquantitatively and agreed very well with elemental analysis (ICP/AES).

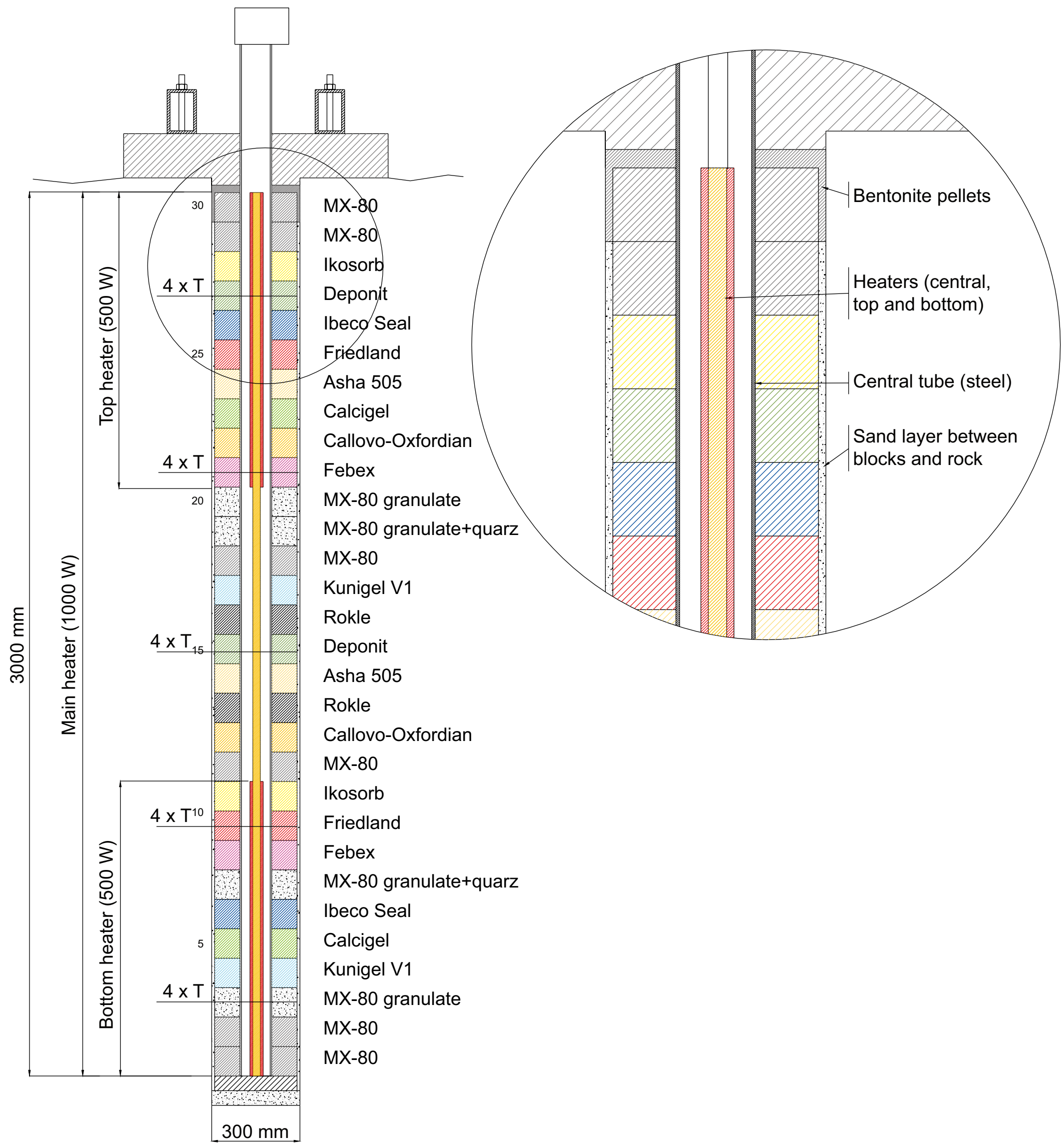


Figure 25-3. Schematic design of the test packages used in the Alternative Buffer Materials (ABM) test. Each test package contains a dozen or so clays, and a total of six test packages have been installed at a depth of 400 metres in the Äspö HRL.

Microbial abundance varied widely between the different clays. Kunigel was virtually sterile, while Friedland and Ibeco Seal contained high numbers of all microbes examined. Iron reducers were detected in varying amounts in all clays, and small amounts of sulphate reducers were detected in roughly half of the materials.

No difference in hydraulic conductivity was observed between the reference samples and samples from ABM package 1. However, both Asha 505 and Deponit CA-N had a lower swelling pressure after the test, which could perhaps be attributed to ion exchange. The concentration of chloride in the clays was evenly distributed both radially and vertically in the test package. Some clays showed an enrichment of sulphate towards the warmer part. The cation composition was relatively constant radially in the clays, but varied greatly vertically through the package, see Figure 25-4. The top part of the package had much more calcium than the bottom part, which was attributed to a greater interaction with the calcium-rich Äspö water in the top part due to more water-bearing fractures. The examined clays exhibited magnesium enrichment towards the warmer interior, which could not be linked to any identified phase. Powder X-ray diffraction on clay samples did not detect any montmorillonite transformation. The exception was several samples that had been in direct contact with the iron heater, where there were indications of the presence of trioctahedral clay minerals. Artefacts were observed in the contact with the heater due to the lubricant used for block pressing, which created uncertainties. Low levels of mesophilic bacteria were detected in some samples after the test, while all other microbe species were below the detection limit, which shows that microbes that are potentially corrosive or buffer-degrading failed to survive this tough test environment (130°C) (Svensson et al. 2011).

A study has been initiated to enhance in-house competence at SKB for bentonite characterization and sampling and to build up a clay laboratory. This has been done in the form of a knowledge transfer programme, whose first part is focused on fundamental chemical and mineralogical characterization. A consignment of an Indian clay for backfilling was analyzed at a consultancy firm and simultaneously by SKB's personnel at the Äspö HRL. This permitted running comparison of the analysis results. The analyses and techniques performed were cation exchange capacity, ion exchange and purification of the clay fraction, and powder X-ray diffraction. The results will be reported during 2013.

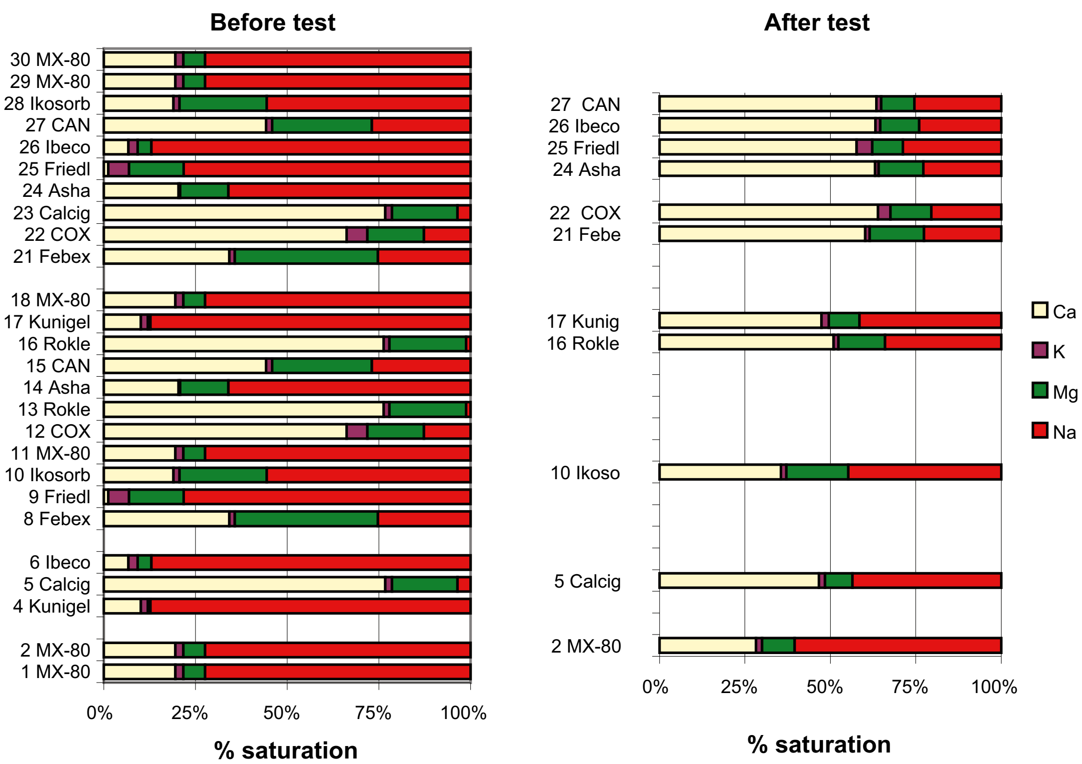


Figure 25-4. *Relative cation distribution in ABM package 1 before and after the test. Note that calcium was enriched in the top part of the package (Svensson et al. 2011).*

In the autumn of 2012, three new ABM packages were installed as a part of SKB's long-term experimental programme. The packages were very similar to the original ABM packages installed in 2006. Some clays that were not true bentonites (Friedland and Callovo-Oxfordian) were removed and replaced by new bentonites or a bentonite-like clay (saponite). Saponite is a different swelling clay mineral than montmorillonite, which is normally present in bentonite. Saponite was included for scientific reasons and not because it was intended as a potential buffer material.

Analyses of previous field tests with bentonite have revealed that the lubricant used in the pressing of the blocks creates uncertainties concerning what exactly happens in the contact between canister and clay. For this reason, 1–2 millimetres of the inside of the clay blocks, which is in contact with the iron heater, was removed and a number of 24-millimetre-high copper cylinders were installed in holes drilled into selected clay blocks, see Figure 25-5. In the KBS-3H concept (see Chapter 16) and internationally, there are plans to replace iron with titanium, since titanium is expected to have less impact on the bentonite. Small titanium tubes were therefore also installed in selected clay blocks.

Programme

At the same time as the choice of buffer and backfill materials needs to be optimized, the fundamental characterization methods that are intended to be used for arrival inspection of the bentonite (Karnland 2010) need to be optimized. The methods will be optimized with respect to speed, accuracy and precision. This is done in order to be able to analyze large quantities of clay in an industrialized process with sufficient accuracy to guarantee that the requirements are met.

This is of importance for further development of the design premises, since the statistical variability in material quality is directly linked to the desired buffer properties. The tolerance in the requirement on buffer density must be adapted to the expected variability in the montmorillonite content.

This work is continuing as a material project aimed at enhancing in-house competence for independent execution of the critical analyses prior to operation of the final repository for spent nuclear fuel. The work is also described in Section 13.3 "Material studies, bentonite" and forms a bridge between technology development and research. SKB is building up the activity at the Äspö HRL with the goal of independently executing as much as possible of the inspection programme. In 2013, an X-ray laboratory was established at the Äspö HRL for the purpose of characterizing the mineralogical composition of the clay by means of powder X-ray diffraction (XRD) and the chemical composition of the clay by means of X-ray fluorescence spectroscopy, see Figure 25-6. This equipment will be used to develop the methodology for determining the montmorillonite content and the quantity of accessory minerals. Other measurement methods need to be implemented and developed for determination of amorphous phases and the total amount of iron(II) and iron(III) in the bentonite. Methods need to be developed for handling and sampling a large quantity of clay in a representative way. Equipment for crushing and grinding large quantities of bentonite clay in a reproducible way was installed at the Äspö HRL in the spring of 2013.

Figure 25-5. *Copper pellets were installed in different bentonite blocks in the newly installed ABM packages in 2012 in order to gain better knowledge of possible interactions between bentonite and copper.*

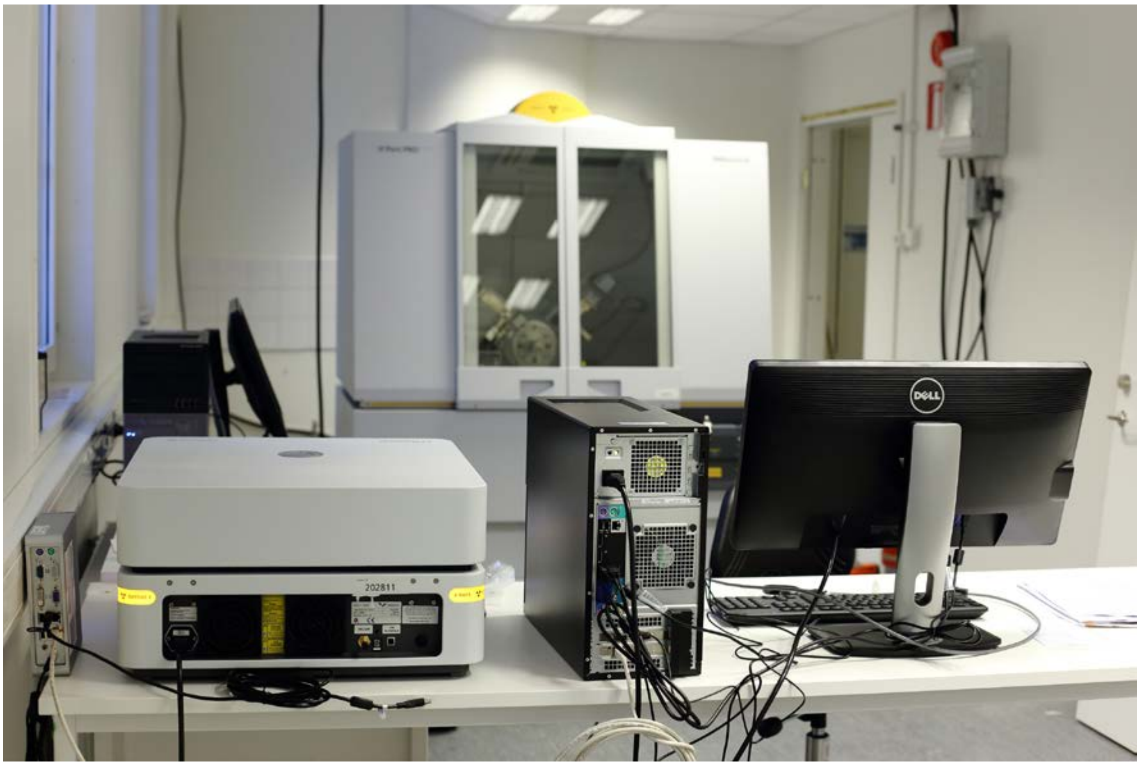


Figure 25-6. *An X-ray laboratory has been installed at Äspö to analyze bentonite clay. An X-ray diffractometer (XRD) for phase identification can be seen in the background. An X-ray fluorescence spectrometer (XRF) for determination of elemental composition can be seen in the foreground.*

The results from ABM as well as other experiments indicate that differences exist in long-term stability between different bentonites, see further 25.5.14 "Montmorillonite transformation".

At present, SKB is not planning any further experiments with different bentonite materials in the Äspö HRL.

25.4.5 Montmorillonite composition

The mineral montmorillonite is characterized by nanometre-thin mineral layers with the ideal structural formula:

$$M^{z}_{(x+y)/z}\, n(H_2O)\, (Al_{(4-x)} Mg_{x})\, (Si_{(8-y)} Al_{y})\, O_{20}(OH)_4$$

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. 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.

Depending on what the positive counterions are, they hydrate to a varying degree. A sodium-dominated montmorillonite is considered to have a water layer between the layers at a relative humidity of about 50 percent, while a calcium-dominated montmorillonite at the same relative humidity has two water layers around the hydrated ions (Brindley and Brown 1980).

Conclusions in RD&D 2010 and its review

There are no direct viewpoints from the review of RD&D programme 2010.

Newfound knowledge since RD&D 2010

Studies using X-ray absorption spectroscopy (Fe-, K-XANES) have been conducted on the Max-lab synchrotron on purified montmorillonites and their original clays selected from the ABM experiment. The goal was to study the ratio between iron(II) and iron(III) in the clays in order to correctly determine the structural formulas of the montmorillonites. The investigated montmorillonites were all dominated by iron(III), as were most of the raw bentonites, which however often contained a small amount of iron(II). The origin of iron(II) in the raw clay is probably pyrite or siderite. The results will be published in a Ph. D. thesis during 2013.

Programme

The montmorillonite's layer charge is the essence of its swelling and sealing properties. In order to better understand the differences between montmorillonites of different origins, further studies are needed to learn more about determining the layer charge. The layer charge can either be determined experimentally by means of different methods (for example, swelling with acryl ammonium ions in XRD) or be calculated from a correct structural formula. The determination of a structural formula is sensitive, because it is sometimes very difficult to purify montmorillonite and because certain elements, such as aluminium, can be located at different places in the crystal structure and iron can have different oxidation numbers, which affects the resulting charge. Greater knowledge of the redox chemistry of iron in montmorillonite and deeper studies of silicon and aluminium, with e.g. solid-phase NMR or X-ray absorption spectroscopy, would furnish more information and reduce the uncertainties in the determination of the mineral's structural formula. Greater knowledge of the different silicon phases that accompany the clay fraction would probably also be valuable.

A planned activity is evaluation of different preparatory methods to determine whether they can be used in a better way to purify montmorillonite in different types of bentonite. Supplementary analytical measurements may otherwise be used to adjust for the influence of the secondary minerals. The activity will above all take place in-house at the X-ray laboratory on Äspö, see Section 25.4.4 "Bentonite composition". Any differences in the long-term stability of clays with different montmorillonite compositions will be studied in the test with alternative buffer materials, ABM, at the Äspö HRL, see Section 25.4.4 "Bentonite composition".

25.4.6 Pore water composition

When the bentonite is delivered, 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 dispersed material in the supernatant (the aqueous solution above a slurried and centrifuged sample). This quantification provides an opportunity to calculate the initial composition of the pore water.

The initial pore water composition does not have a direct bearing on the long-term function of the buffer or backfill material.

25.4.7 Hydrovariables

The hydrovariables are water flow, water pressure, gas flow and gas pressure. Initially it is relevant to describe water and gas 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 creates a gas pressure (air) of one atmosphere (approx. 0.1 megapascal) and a water pressure of 0–0.1 megapascal 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 megapascals.

25.4.8 Stress state

The swelling pressure starts to build up when buffer and backfill come into contact with external water, see Sections 25.5.5 "Water transport under unsaturated conditions" and 25.5.9 "Swelling". 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.

25.4.9 Pore geometry

In order to realize the defined buffer functions, there must be a specific counterion concentration in 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 density/porosity.

According to the design premises in SKB (2009a), the buffer in the saturated state has a density 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/^{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 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. The grain densities of a large number of alternative buffer materials have been determined (Karnland et al. 2006).

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 (Pusch 2003).

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 (2,370 kg/m^3 after saturation). It is assumed today that the top bed will have the same properties as the bottom bed.

25.4.10 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 performance. The previously specified dimensions of 35 centimetres on the sides of the canister, 50 centimetres underneath the canister and 150 centimetres above the canister still apply for KBS-3V (Karnland et al. 2009). The dimensions may be somewhat different for KBS-3H.

The buffer in the silo in SFR consists of a bottom and a top bed plus a wall fill, see Figure 25-7.

25.4.11 Radiation intensity

The initial dose rate on the canister surface was calculated in SR-Site to be a maximum of 500 milligrays per hour (mGy/h). The radiation is dominated by the radionuclide cesium-137. The dose rate is used to assess radiolysis of pore water and radiation-induced changes of the montmorillonite.

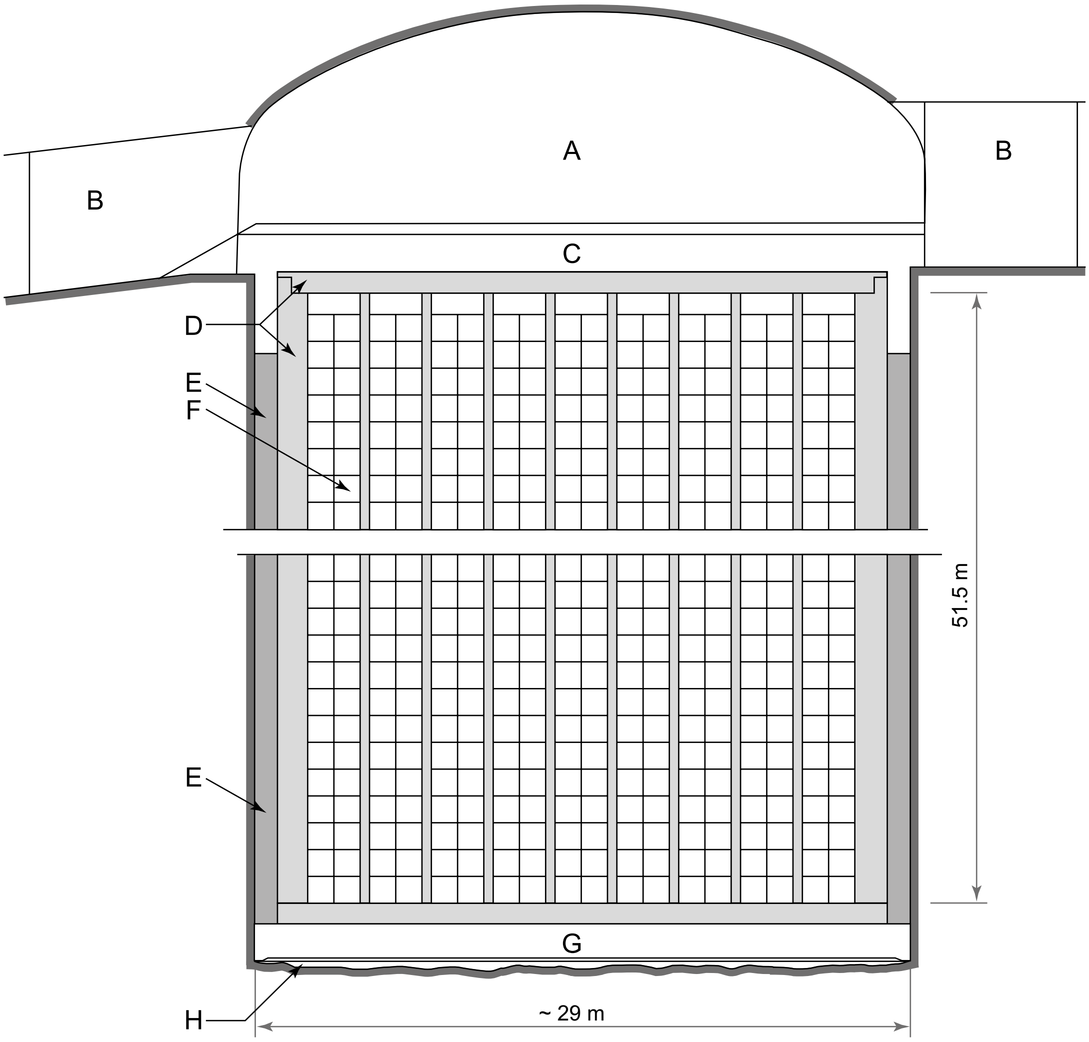
25.4.12 Temperature

At deposition, the buffer and the backfill have the same temperature as the ambient environment. 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, the heat from the deposition machine, the 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 25.5.3 "Heat Transport".

25.4.13 Structural and stray materials

At the present time it is not expected that any engineering materials will be left in the deposition holes. However, some type of levelling bed is still planned at the bottom of the holes, see Section 13.6 "Installation of buffer and backfill". The bottom plate was treated as a separate subsystem in SR-Site.



A Packed sand fill
B Concrete plug
C Clay-based top bed
D Concrete silo
E Clay-based wall fill
F Grouted waste packages
G Clay-based bottom bed
H Concrete pad

Figure 25-7. *Schematic section through the silo in SFR (Pusch 1985).*

25.5 Processes

An in-depth understanding and handling of the processes that take place over time in the repository system is fundamental for the safety assessment. The primary sources of information for this are the results of decades of research and development work carried out by SKB and other organizations. In a broader perspective, these results are in turn based on knowledge that has emerged over centuries of scientific and technical development. The research and development work has led to the identification of a number processes that occur in the engineered barriers and the natural systems and that are of importance for long-term safety.

This section describes the identified processes in the clay barriers.

25.5.1 Overview of processes

Buffer

On emplacement, the buffer comes into contact with the hot canister surface. The thermal energy is spread through the buffer by heat transport and its temperature increases. The gamma and neutron radiation emitted by the canister decreases in intensity due to radiation attenuation in the buffer.

A negative capillary pressure exists initially in the pores in the buffer, causing water to be transported in from the surrounding rock. After the buffer has been saturated with water, this water inflow is very slow. Gas transport can occur during the saturation process, when water vapour can flow from the hotter parts of the buffer to condense in the outer, colder parts. Originally there is also air in the buffer, which can leave the buffer by dissolving in the pore water. This process is called gas dissolution. After water saturation, gas transport can occur if a canister should be damaged, leading to corrosion of the insert in the canister accompanied by evolution of hydrogen gas.

If there is an inflow of water into the deposition hole that is greater than the capacity of the bentonite to absorb water, channels may form in the pellet gap between the buffer blocks and the rock. This phenomenon is called piping and can lead to loss of the buffer from the deposition hole.

On absorbing water, the buffer and backfill swell and a swelling pressure is built up. The swelling pressure is different in the buffer and backfill, which therefore interact mechanically. The swelling pressure is decisive for the mechanical interaction between canister and buffer, which can cause the canister to move in the buffer. The swelling pressure also lends the buffer its self-healing capacity. On heating, the pore water in particular can expand due to thermal expansion.

The chemical evolution in buffer and backfill is determined by a number of transport and reaction processes. Solutes in the water can be transported by advection and diffusion. In the buffer, advection occurs almost exclusively during the water saturation process, after which diffusion dominates. By means of osmosis, the salinity of the groundwater in particular can affect the physical properties of the buffer. Ion exchange and sorption replace the buffer's original content of charge-compensating counterions with other ionic species. Chemical transformation of the buffer's swelling minerals can occur, leading to altered buffer properties. Other minerals undergo various dissolution and precipitation reactions in the buffer. On swelling, the buffer penetrates out into the fractures in the surrounding rock, where it can form colloids which can be carried away by the groundwater. This can lead to gradual erosion of the buffer. The clay can be transformed by radiation effects and the pore water can be decomposed by radiolysis. Finally, microbial processes might possibly occur in the buffer.

After water saturation, radionuclide transport in the buffer is expected to take place exclusively by diffusion in the pores of the buffer, and possibly also on the surfaces of the clay particles. As long as there is enough bentonite left in the deposition hole, neither advection nor colloid transport is expected in a saturated buffer. Radionuclides can be sorbed to the surfaces of the clay particles. A crucial factor for this is the chemical form of the radionuclide, which is determined by the chemical environment in the buffer via the process of speciation. Together with the transport conditions, the rate of radioactive decay determines to what extent radionuclides from a breached canister will decay before reaching the outer boundary of the buffer.

Backfilling

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 research programme for the different processes in buffer and backfill is dealt with in the following sections.

Silo buffer

During the filling phase in SFR, the rock cavern around the silo 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.

25.5.2 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 good enough for the needs of the safety assessment.

25.5.3 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.

The thermal evolution of the near-field is of importance as general input data for the mechanical, chemical and hydrological process. The temperature criterion that is directly relevant for safety is the maximum buffer temperature. According to the criterion, this temperature may not exceed 100°C. The temperature is pessimistically chosen to avoid mineral alterations in the buffer with good margin.

Conclusions in RD&D 2010 and its review

SSM was not convinced that a quasi-steady-state heat transport can be assumed in the calculations of heat transfer, considering that the heat source in the canister is time-dependent. SKB should investigate whether the assumption of steady-state heat transport in the buffer is reasonable and whether heat storage in the buffer is negligible.

SSM further deemed that SKB should investigate how a local drying-out of the buffer nearest the canister would affect the temperature evolution.

Newfound knowledge since RD&D 2010

The analysis in SR-Site of the thermal evolution is based on the guidelines for thermal dimensioning and the calculation methods established by Hökmark et al. (2009) as well as on the results in the design report (SKB 2009b) with respect to Layout D2 for Forsmark when these guidelines were applied. This is described in the production report for the rock line (underground openings, SKB 2010i) and in SR-Site.

The numerical calculations in the design report are adequate and sufficient to show that the safety assessment's requirement of 100°C is met for all canisters. However, these calculations only pertain to the first 20 years after deposition and only apply to canisters that have been deposited in rock volumes that consist for the most part of rock with low thermal conductivity. They cannot be used to estimate how many canisters will actually reach the maximum temperatures, which approach the limit specified in the design process. It should be noted that the majority of the canisters will be deposited in rock that has properties that are roughly the same as the average values for the domain and consequently reach lower maximum temperatures.

In SR-Site, an average of less than one deposition position of a total of 6,000 would reach a peak buffer temperature in excess of 95°C, which means that the design requirement would be satisfied with a margin of 5°C in this analysis. A very large majority of the canisters, around 98 percent, will have a margin of 10°C or more. Furthermore, the maximum temperatures are overestimated due to the following considerations:

- All canisters are assumed to be deposited with the nominal canister spacing over the entire repository. In reality, some deposition holes will be excluded. Canisters next to excluded positions will have lower temperatures.
- All canisters are assumed to be deposited in the the central parts of the deposition areas. In reality, around 1,000 canisters will be deposited close enough to the ends of the tunnels for the peak temperatures there to be lower.
- All deposition holes are assumed to be completely dry, with a 10-millimetre air-filled gap between the canister and the bentonite. In reality, the degree of saturation will vary. Some of the holes will be sufficiently close to saturation that the wet hole model, rather than the dry hole model, applies. This will reduce the peak temperatures.

Programme

Heat transport is handled generally in Chapter 26.3 "Heat transport".

25.5.4 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. For buffer density, the critical temperature (Tc) lies in the range between –4°C and –10°C. All water does not freeze at the same temperature, and even at temperatures as low as –50°C , the montmorillonite still has two water layers left, i.e. only the third layer and any additional water will freeze to ice (Svensson and Hansen 2010).

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.

Conclusions in RD&D 2010 and its review

SSM wanted SKB to justify its assumption of one-type pore water in the theory. Furthermore, SKB should investigate whether other important mechanisms such as electrostatic interaction should be included in the theory.

SSM also thought that SKB should investigate whether ice lenses, which studies of SFR have shown can occur (Emborg et al. 2007), can also be formed in a frozen eroded buffer. SKB should more thoroughly examine and report the consequences of freezing of the backfill.

Newfound knowledge since RD&D 2010

The probability that the buffer clay will freeze at repository depth was evaluated in SR-Site. With the most pessimistic combination of all uncertainties, the uncertainty range for the −4°C isotherm reaches a maximum depth of 316 metres. This shows that even in the most pessimistic case, temperatures that could cause freezing of the buffer do not reach repository depth in the reference evolution.

The reference density of the borehole seals are the same as in the buffer. But the seals will most likely be exposed to temperatures below the critical temperature due to their vertical extent up to about 100 metres beneath the surface, see Figure 25-8. This vertical extent also creates a thermal gradient in the borehole seal. In Forsmark, the measured geothermal gradient in the uppermost 1,000 metres is lower than 0.020°C per metre (Sundberg et al. 2009). During periods of permafrost, the borehole seals could connect parts of the crystalline rock in which the temperature is over 0°C to parts where ice formation occurs. Hence, this design could give rise to frost heave as water is transported from locations where it is liquid to locations where an ice lens is built up, as shown schematically in Figure 25-8.

The driving force for water transport in the bentonite is a suction gradient, and it concluded in this study that in the temperature range from 0°C to the critical temperature, this gradient is of the order of 1.2 megapascals per degree (MPa/°C) for a system of homogeneous density. When the geothermal temperature gradient is used, the suction gradient is directly seen to be 0.03 megapascal per metre, or three metres water column per metre. The latter quantity can directly be plugged into the expression for Darcy flow, and assuming a hydraulic conductivity of $K_h=10^{-13}$ metres per second (Karnland et al. 2006), a flow of $3 \cdot 10^{-13}$ metres per second is obtained.

This flow puts an upper limit on the rate of a possible ice lens growth. It should be noted that with the assumption of a constant geothermal gradient, this limit is independent of the distance the water must travel. Because the problem at hand in principle is one dimensional, this evaluated flux can directly be converted to an ice lens growth of approximately 10 micrometres per year. From this estimate

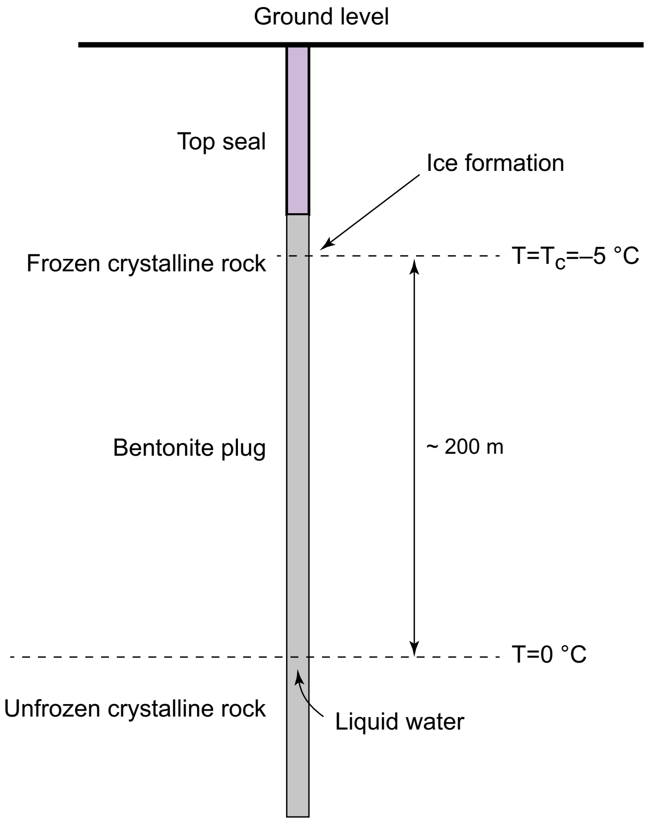


Figure 25-8. *Schematic illustration of possible ice lens formation in the borehole seal in the KBS-3 repository. A geothermal gradient of 0.025°C per metre and a critical temperature in the bentonite of –5°C is assumed here. (Birgersson et al. 2010).*

it is seen that ice lens formation will not cause a problem, as it at most will lead to a build-up of 10 centimetres over a period of 10,000 years. Since this water transport process is serial in character, it is the lowest value of hydraulic conductivity over the length in question that will determine q, i.e. the section with the lowest K_h will be rate-determining. Hence, this prediction is relatively robust. Furthermore, the process is only active when the temperature in the uppermost parts of the seal is below the critical temperature, which will only occur during parts of a permafrost period.

The situation is less favourable in the silo in SFR, however. The hydraulic conductivity in the silo buffer is considerably higher (on the order of 10^{-10} metre per second), which means that an ice lens could grow by 10 millimetres a year.

A prerequisite for ice lenses to form is that some of the clay is in contact with unfrozen groundwater, while the lens is formed in another part. At repository depth, the water in the surrounding rock will freeze before the water in the buffer/backfill freezes, so ice lenses cannot form.

Programme

Freezing of clay barriers at the depth of the Spent Fuel Repository in Forsmark is not deemed possible. No further research will be conducted in the area during the period. Freezing of the silo buffer will be evaluated in SR-PSU.

25.5.5 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, 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 diverge since the materials are different.

The safety functions for the buffer and the backfill assume that a completely water-saturated state exists. This should mean that the buffer and the backfill must be saturated in order to function as intended. However, a functioning buffer is not necessary as long as the deposition hole is unsaturated, since no mass transfer can occur between the canister and the groundwater in the rock in the unsaturated stage. The water saturation process therefore has no direct impact in itself on the safety functions of the buffer and the backfill. It is nevertheless important to understand the water saturation process, since it defines the state of the barriers during the early stage of repository evolution. Finally, ventilation of the deposition tunnels during long periods of time (before backfilling of the deposition holes and the tunnels) can lead to drying-out of the surrounding rock. The resulting air-filled pore volume in the rock can potentially serve as a sink for the water contained in the buffer during installation. If a significant amount of this water should be transported into the rock, it could potentially lead to a considerable increase in the maximum temperatures of the canister surface.

In SFR, the operating phase's drainage pumping will cease when the repository is closed. At this point, resaturation of the silo buffer will begin. Water transport in the unsaturated silo buffer occurs in principle in the same way as in the buffer in the Spent Fuel Repository, but with other material data and boundary conditions.

Conclusions in RD&D 2010 and its review

SSM found SKB's model for describing the water inflow to the buffer or the backfill to be inadequate for describing the clearly heterogeneous water inflow in bentonite exhibited in many laboratory experiments and field tests.

SSM also thought that SKB should better justify its assumption of an isothermal evolution in the backfill. Regarding vapour transport in an unsaturated buffer, SSM considered that SKB should also consider a possible convection mechanism in addition to molecular diffusion.

Newfound knowledge since RD&D 2010

An extensive analysis of the time for resaturation of backfill and buffer was done in SR-Site. Since the rock in Forsmark is expected to contain very few water-bearing fractures, which are normally spaced at a distance of over 100 metres, saturation of the backfill can be achieved in anything from < 100 years up to around 6,000 years. The longer times apply to deposition positions located far from the water-bearing fractures. As for saturation of the backfill, the time for saturation of the buffer is highly dependent on the local hydraulic conditions. A water-bearing fracture in or near the deposition hole will result in fairly rapid saturation. A deposition hole to which water is only supplied from the rock will, on the other hand, remain unsaturated for hundreds of years. In view of the expected low frequency of water-bearing fractures in Forsmark, saturation of the buffer can be achieved in anything from < 10 years up to around one thousand years. The longer times apply to deposition holes that are not in connection with water-bearing fractures. Furthermore, the thermal conductivity of the buffer is affected by the redistribution of the moisture contained in the buffer in deposition holes with a dry surrounding rock mass. This is taken into account in the analysis of the thermal evolution of the buffer.

Extensive THM modelling of the TBT (Temperature Buffer Test) has been carried out and reported in Åkesson et al. (2012a). Three groups modelled different aspects of the test with two different numerical models: CODE_BRIGHT and ABAQUS. The evaluation showed that the material models were able to satisfactorily describe some of the experimental results: the thermal evolution, the hydration process around the upper heater, the swelling pressure (i.e. the relationship between the void ratio and the effective stress at excavation) and the confining force. However, there were aspects which the models could not handle satisfactorily: the dehydration around the lower heater was generally exaggerated, the relative humidity and pore pressures around the lower heater were difficult to reproduce, and the calculated von Mises stresses were in some cases significantly lower than the experimental data.

Modelling of the Prototype Repository is being done as a part of the Äspö Task Force of Engineered Buffer Systems (TF EBS). The main purpose of the work in the TF EBS has been to describe the THM processes during the operation of the Prototype Repository and to predict their state at excavation. At this stage, the work has been focused on the buffer in the sixth deposition hole. An example of results is shown in Figure 25-9.

Questions have arisen as to whether water from rock fractures can be vaporized against the canister and be transported out into the backfill, thereby causing salt enrichment against the canister (sauna effect, see Figure 25-10). The general opinion is that this cannot occur to an extent that is harmful to the canister or the bentonite, but an experimental programme to verify this has been initiated.

A series of semi-quantitative tests have been performed to study one aspect of the sauna effect: uptake of water vapour and sealing capacity of bentonite pellets in temperature and humidity gradients. The results obtained thus far show that water condensation is an important mechanism for the water uptake process: water vapour condenses to liquid water somewhere in the system, after which water uptake occurs to the pellets and spreads from this point. Further, the importance of limiting the volume of the pellet fill for the sealing capacity has been demonstrated, see Figure 25-11. The graph shows the total quantity of fluxed vapour and the quantity of water absorbed in the pellet fill as a function of time in two tests, identical except that one is free to swell in the axial direction (unconfined) while the other is confined (which more closely resembles conditions in the Spent Fuel Repository), see Figure 25-12.

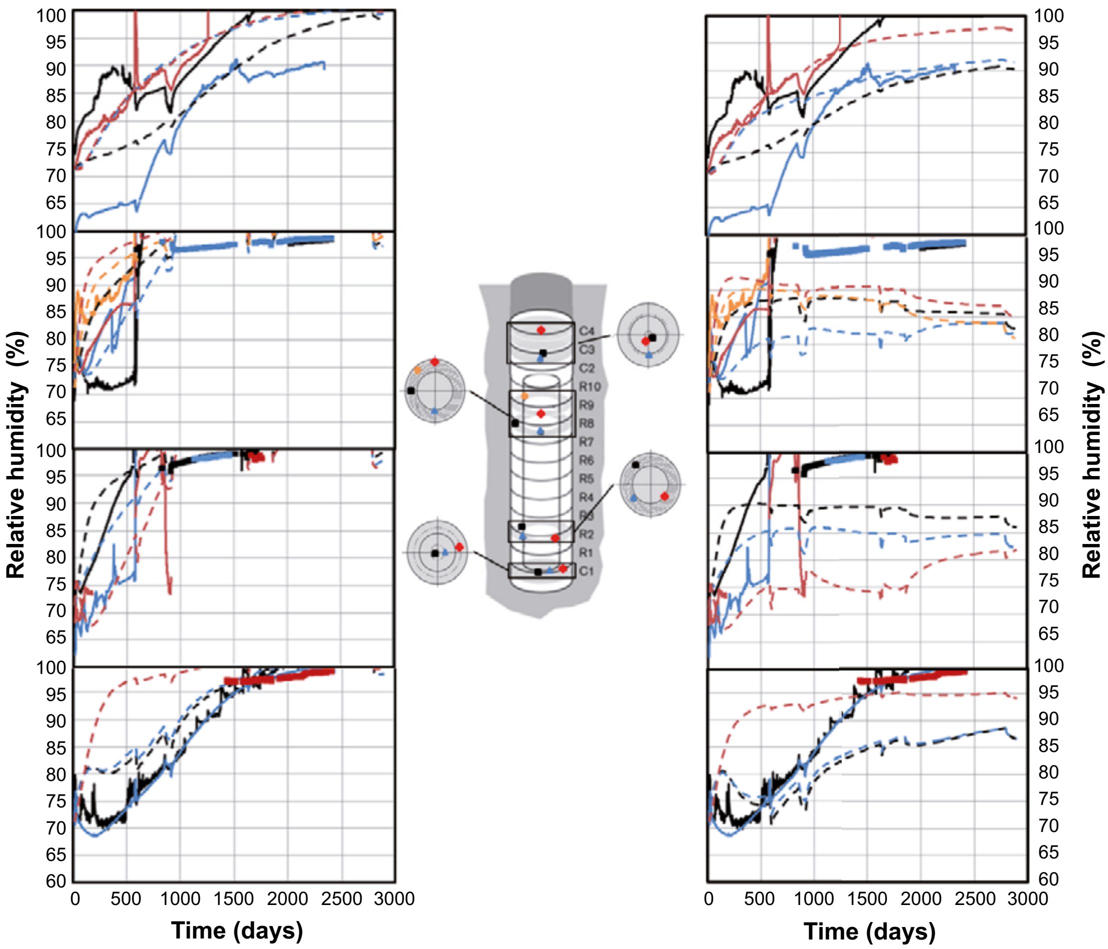


Figure 25-9. Evolution of the relative humidity at four different depths and several different radii in deposition hole 6 from the two installation models. The dashed lines represent model data, the solid lines represent experimental data from RH sensors, and the symbols represent data from suction sensors in the area (Malmberg and Kristensson 2013).

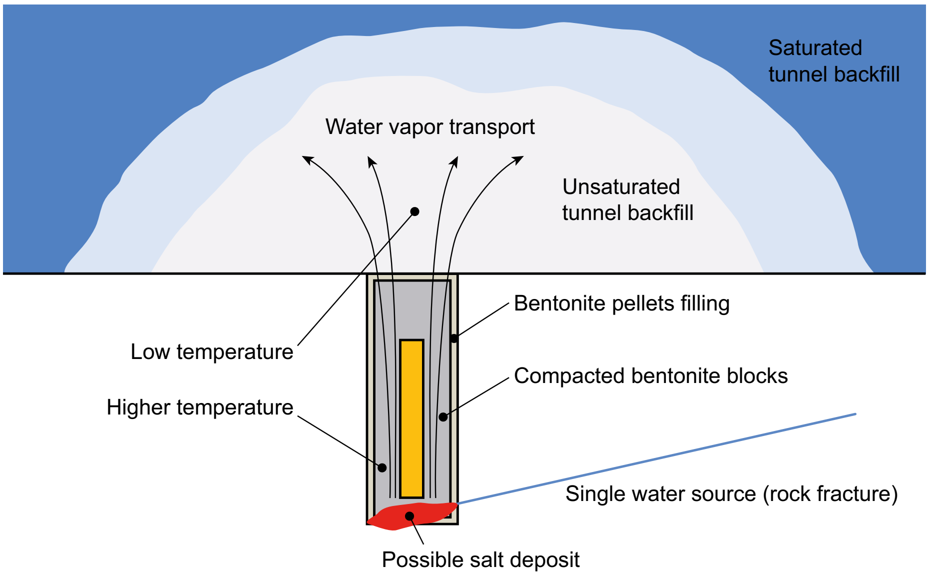


Figure 25-10. *Schematic illustration of the sauna effect where salt is enriched around the canister when vapour is transported out of the deposition hole (Birgersson and Goudarzi 2013).*

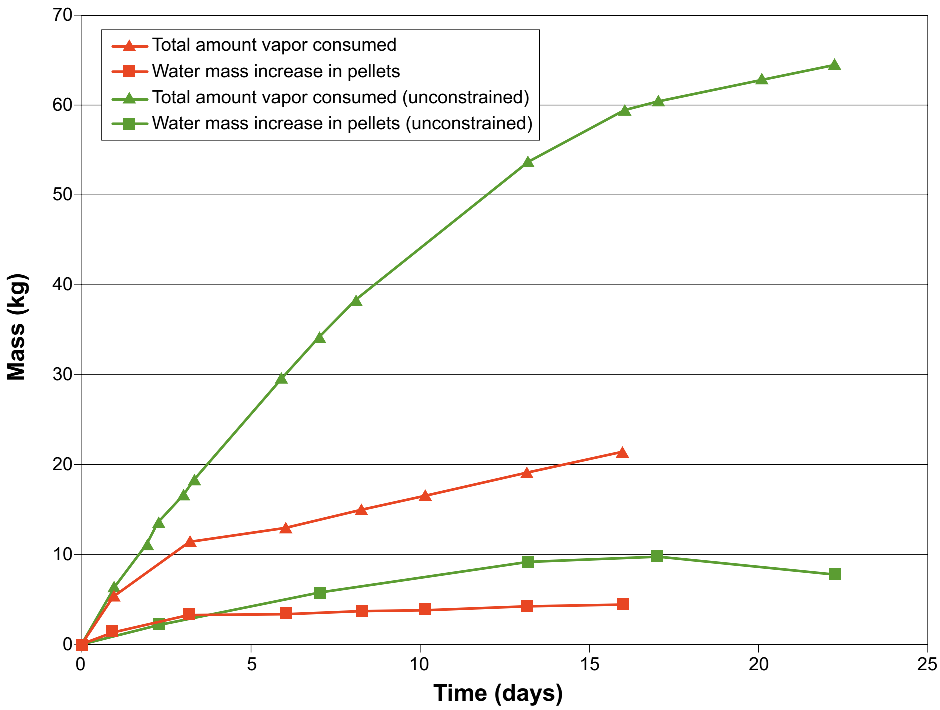


Figure 25-11. *Total quantity of fluxed vapour and quantity of water absorbed in the pellet fill as a function of time in two tests, identical except that one is free to swell in the axial direction (unconfined), while the other is confined (which more closely resembles KBS-3 conditions) Birgersson and Goudarzi 2013).*

As can be seen, the differences are great: while the confined system reaches a steady-state after just a few days, water accumulates (and is lost) in the unconfined case for 15–20 days. Further, it can be seen that water consumption is much less in the confined case (about 20 kg compared with 60 kg after 15 days) – a sign that the confined system has begun to seal and is much less permeable to vapour. It should be noted that the unconfined system also shows signs of sealing after an extended period – water consumption declines after about 15 days.

The interaction between the groundwater in the rock and the buffer is being studied by an Äspö Task Force. This is described in greater detail in the Section 26.4.5 "Task Force for Groundwater Flow and Transport of Solutes".

Figure 25-12. The picture shows a concluded test of the unconfined system (where the bentonite has been allowed to swell freely), before excavation (Birgersson and Goudarzi 2013).

Programme

The work of verifying and updating the models of water saturation of the buffer continues. Most of this work is being done within TF EBS.

The results of the tests concerning the sauna effect do not provide an unequivocal answer as to whether the issue can be dismissed or not. Scale tests with a real geometry in relation to the Spent Fuel Repository have been planned, but will not be carried out during the coming six-year period. But clear data on this issue should be available sometime in the middle of the 2020s.

Resaturation of the silo buffer in SFR is being modelled as a part of SR-PSU. This includes:

- Laboratory investigations of swelling pressure and hydraulic conductivity before and after ion exchange. The material is compacted to a pellet of suitable density, and deionized water is added after installation. In order to bring about an ion exchange whereby calciumbecomes the dominant cation, the deionized water is exchanged for a solution of calcium chloride. Measurement of swelling pressure is done continuously and determination of hydraulic conductivity is done at initial equilibrium and at equilibrium after the exchange from deionized water to calcium chloride solution.
- Laboratory experiments for input data to modelling of the wetting phase. The water retention curve (the relationship between the water potential and the water ratio) and the hydraulic conductivity are of importance for modelling the wetting phase. The water retention curve is determined by means of a climate chamber or psychrometers, depending on the measurement range. Water uptake tests are also performed to validate the resulting material model.
- Modelling of effects of ion exchange and mechanical impact of cement degradation. This includes:
  1) Analysis of effects of ion exchange in the bentonite.
  2) Analysis of timescale for ion exchange in the bentonite.
  3) Analysis of how cement degradation affects the silo.
- Modelling of the wetting phase and the water flow through the repository. These processes will be described with analytical and/or numerical models. The main purpose is to estimate the time for resaturation of the silo and to describe the character of the resaturation, as well as to study water transport in the saturated silo.

25.5.6 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. The reference bentonites in SR-Site project have a hydraulic conductivity of about 10^{-13} metre per second at the intended buffer density, 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 processes, above all loss of buffer due to colloid formation, see Section 25.5.19 "Colloid release/erosion".

The same model can be used for both 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 2010 and its review

No direct comments were offered in the review of RD&D programme 2010.

Newfound knowledge since RD&D 2010

There has recently been a debate about how the hydraulic conductivity of bentonite is affected by heat and how hydraulic conductivity should be measured. Pusch et al. (2010) reported that the hydraulic conductivity of MX-80 bentonite that is used in the Canister Retrieval Test in the Äspö HRL increased by three orders of magnitude after five years of heating to 95°C. However, the publication does not include a description of how hydraulic conductivity was measured. The same sample exhibited no change in swelling pressure. The authors nevertheless interpreted this as a change in clay mineralogy.

More cautious and systematic testing of samples from the same tests as performed by Dueck et al. (2011) showed no significant changes in either hydraulic conductivity or clay mineralogy compared with the reference samples. Harrington et al. (2013) also studied the hydraulic conductivity of samples from the Canister Retrieval Test. They decided to inject distilled water with a flow rate of only 1.6 microlitres per hour to minimize any disturbances in the system and avoid the risk of movement of dissociated mineral components (if any).

If the bentonite sample exhibited an actual conductivity of 2.0×10^{-11} metre per second as reported by Pusch et al. (2010), this would generate a gradient of only eight metres per metre (m/m). However, it turned out that the pressure gradient required to achieve this flow was over 3,000 m/m and not eight m/m as predicted. This gave an average hydraulic conductivity of 4.7×10^{-14} metre per second, a value close to what should be expected for unaffected material and also in line with the measurements in Dueck et al. (2011). This clearly shows that the hydraulic properties of bentonite do not change after heating for five years.

Programme

Water transport under saturated conditions is in principle equal to assessment of the hydraulic conductivity of the material in question. This is not a research area of its own, but is included as a natural part of several different projects.

25.5.7 Gas transport/dissolution

In the case when a copper canister is damaged and water can come into contact with the insert, hydrogen gas will be formed inside the damaged canister. Dissolved gas is transported slowly through the bentonite buffer. It is highly likely that a gas phase and a gas pressure will be built up inside the canister, and it is therefore important to show that this pressure will not lead to any negative consequences for the function 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 (Björkenstam 1997).

Conclusions in RD&D 2010 and its review

SSM thought that SKB should strive for a better understanding of the mechanism for gas transport. The reason for the difference between the results of LASGIT (Large Scale Gas Injection Test at the Äspö HRL) and equivalent laboratory tests should be investigated.

SKB should study gas transport through a partially eroded buffer and assess the maximum hydrogen gas content that can be maintained when the buffer is eroded.

More specifically, SSM asked for an update of the modelling of transport in the near-field of the Spent Fuel Repository of hydrogen gas evolved from the corrosion of the nodular iron insert following canister failure.

Newfound knowledge since RD&D 2010

An extensive experimental study of gas transport in bentonite has been carried out in the EU FORGE project, where the main conclusions are:

- There is a linear relationship between the gas flow rate and the pressure gradient in unsaturated or partially saturated bentonite – two-phase flow is the dominant transport mechanism (this applies even to saturated mixtures of sand and bentonite if the sand content is high enough).
- A high gas pressure can delay the saturation of the bentonite.
- Classical two-phase flow models cannot correctly representative gas transport in a saturated compacted bentonite. At a degree of saturation of 80–90 percent or higher:
  - No flow of gas will take place in the bentonite unless the applied pressure is equal to or higher than the swelling pressure.
  - at a pressure below the swelling pressure, the only transport mechanism is the ubiquitous diffusion of dissolved gas.
- If the gas pressure reaches a higher value than the pressure in the bentonite, a mechanical interaction occurs, which leads to either:
  - Consolidation of the bentonite, and/or.
  - formation of dilatant transport pathways.
- Consolidation: The gas volume in the clay causes a compression with an increase in clay density nearest the gas volume and a local swelling pressure to balance the gas pressure. Consolidation is clearly observed in bentonite samples with relatively low density, see Figure 25-13. There are still no clear experimental observations of consolidation at high density.
- At a critical pressure, dilatant pathways will form and the gas will become mobile. The transport pathways are unstable in time and space – the outflows are local after the gas breakthrough and no measurable dehydration has been identified in any samples.
- The transition from consolidation to when dilatant pathways are formed is unclear. The transition has been observed at different levels of gas pressure:
  - When the gas pressure reaches the test pressure (swelling pressure+water pressure) (for example seen in LASGIT).
  - At an excess pressure of about 20–30 percent.
  - At pressures 2–3 times higher than the test pressure (swelling pressure+water pressure).
- If there are interfaces between bentonite and other materials, the gas will preferably move in the interface. This does not appear to affect the transport mechanism, however.
- Self-healing of the bentonite always occurs after a gas transport event.
- In saturated systems, interfaces between bentonite blocks heal, and these are not preferential gas transport pathways.

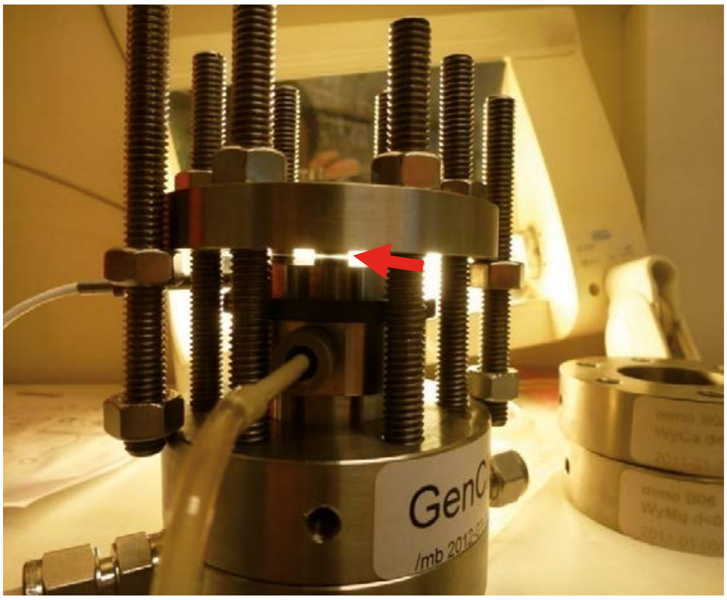


Figure 25-13. *Photograph of bentonite that has become consolidated by a gas pressure. The gas has compressed the bentonite and a gas-filled space has been formed (at the arrow).*

Programme

Since early canister failure was not expected in SR-Site, the problem with gas transport in bentonite is no longer a priority issue. In SR-Site, the only mechanisms that could damage the canister were erosion and shear, and in those cases the buffer is already lost (erosion) or strongly affected (shear) and the issue of gas transport is then of rather secondary importance.

A final gas transport test in LASGIT and a couple of laboratory tests will be performed in 2013 to strengthen the existing body of data. LASGIT will then be put in the dormant mode. It will, however, be possible to resume the test if it were found necessary.

25.5.8 Piping/erosion

A hydraulic problem during the operating phase concerns piping and related erosion effects in the buffer and the backfill. The inflow of water to the deposition holes will primarily take place through fractures and will contribute to the wetting of the buffer. If the inflow is concentrated to fractures that are filled with more water than the swelling bentonite can absorb, however, a water pressure will be created in the fracture that acts on the buffer. Since the swelling bentonite is initially a gel with a density that increases over time as water penetrates more deeply into the bentonite, the gel may be too soft to stop the water inflow. The result may be piping in the bentonite and a continuous water flow plus progressive erosion of bentonite particles. Competition is then created between the swelling rate of the bentonite and the flow rate through the buffer on the one hand and the buffer's erosion rate on the other.

The consequence of piping is always erosion of material that has been detached from the channels. The eroded material is transported in the channels until it reaches an area with more immobile water in the backfill, where it can sediment or continue out from the backfill into an open transport tunnel. After complete water saturation and homogenization of the buffer and the backfill and re-establishment of the hydrostatic water pressure, channels and openings caused by erosion will heal and a swelling pressure will be established. This presumes that the density and the resulting swelling pressure are high enough to overcome the internal friction. After the initial stage, there is very little risk that piping will occur again, since this would require a sharp and rapid increase in the local water pressure gradient in the rock at the contact with buffer or backfill.

The process has not even been discussed for SFR and is probably quite insignificant. 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 2010 and its review

SSM deemed that laboratory tests should be supplemented by theoretical studies to provide a quantitative understanding of the processes associated with piping and erosion.

SSM wanted SKB to justify its new criterion for the highest permissible water inflow volume for piping/erosion (150 cubic metres of water through a deposition hole during the buffer's entire wetting time). SKB should explain the influence of other factors besides total water volume, such as flow rate and groundwater pressure.

Newfound knowledge since RD&D 2010

A number of additional tests have been carried out to simulate erosion in a deposition hole, which mainly occurs in a vertical direction in the pellet fill (Sandén and Börgesson 2010). Based on these tests, an exponential erosion model has been proposed (Sandén and Börgesson 2010). The model relates the accumulated mass of eroded bentonite to the accumulated mass of eroding water. This model was used in the consequence analysis of piping in SR-Site.

A conceptual model for piping and erosion (see Figure 25-14) is currently being developed. The approach is to distinguish between three different processes – piping, sealing and loss of material – and to describe these processes as concisely as possible. Different variants of descriptions will be investigated. The ultimate goal is to derive mathematical equations for the different processes, which can in turn be combined with different types of mathematical models.

Piping is regarded as a hydraulic process with water transport through a pipe or channel that is maintained as long as the pore pressure is equal to or exceeds the swelling pressure in the surrounding bentonite. The flow rate is assumed to be related to the hydraulic gradient and the radius of the pipe in accordance with the Hagen–Poiseuille equation.

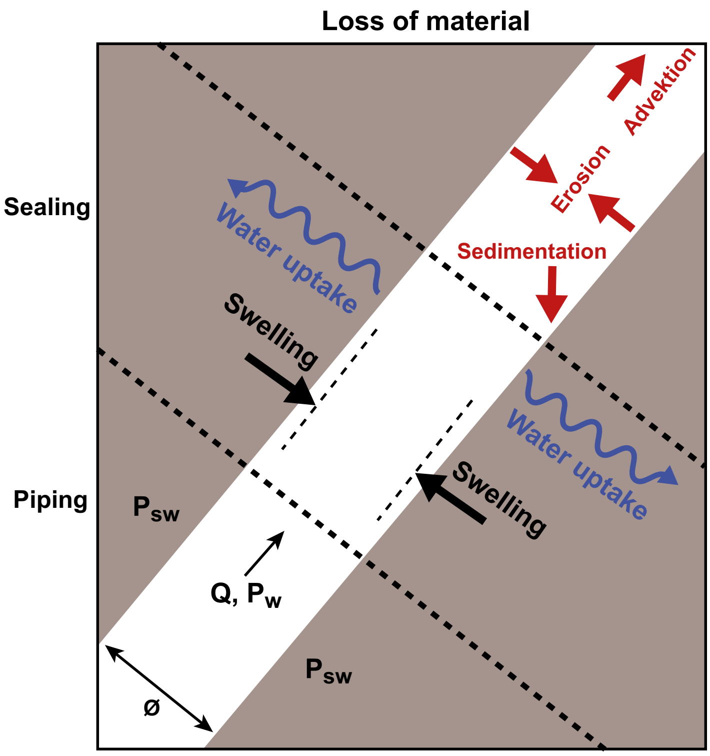


Figure 25-14. *Conceptual model for piping and erosion. Grey represents bentonite and white the eroded channel.*

The sealing process is regarded as a hydro-mechanical process that includes water uptake in the surrounding bentonite, which in turn leads to swelling. The first step in describing the process is to define a density profile for the bentonite around the pipe. The shape of the profile should yield a relationship between the radius of the pipe, the mass of lost material and the distance to a saturation front. The idea of such an approach is that it could make it possible to simulate a pipe that is sealed and the cessation of piping.

Loss of material is regarded as a complex process that includes erosion of bentonite out into an aqueous phase, sedimentation and advective transport of bentonite. Erosion is generally assumed to be controlled by the shear strength of the bentonite and the shear forces exerted by the flowing water. Sedimentation is also assumed to have a significant impact on the bentonite concentration in eroding water, since observed concentrations in vertical erosion tests are considerably lower than equivalent concentrations in horizontal erosion tests. The first step in describing this process is to formulate and analyze a mass balance with expressions for erosion and sedimentation.

Programme

The erosion tests within the Eva Project focus on what happens with pellets in the deposition holes when there is a water flow. The following subjects are being studied:

- Erosion as a function of time and accumulated flow.
- Effects of flow rate and salinity.
- Effects of geometry (length, cylinder/gap, open or closed system).
- Composition of the eroded bentonite.

25.5.9 Swelling

The primary function of the buffer is to ensure that the transport of various substances from the rock to the canister and from the canister to the rock is dominated by diffusion. The swelling pressure in the bentonite is expected to seal all gaps and ensure that the rock and the buffer are in good contact with each other. It is therefore important that the swelling pressure is maintained. A swelling pressure of one megapascal constitutes the criterion for the safety performance indicator for maintaining the self-healing capacity of the buffer. On the other hand, the swelling pressure may not exceed 15 megapascals in order to limit the pressure on canister and rock.

The swelling process has 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. The models have, however, been improved in recent years, and some new homogenization calculations have been done and will be done without this simplification, i.e. under unsaturated conditions.

Water uptake after deposition of 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. This residual inhomogeneity is of importance for the design premises and the configuration (pellets and blocks) with which the buffer is deposited. 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 magnitude 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 may 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 uncertainties mainly concern 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 caused by the weight of the canister.

The interaction between buffer and near-field rock is caused by e.g. swelling pressure from the buffer, convergence of deposition holes and shear movements in the rock. Convergence of deposition holes can largely be neglected, see also Section 26.10 “Time-dependent deformations”. In a KBS-3H-repository, the bentonite will first penetrate 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 25.5.8 “Piping/erosion”. Swelling occurs during the water saturation phase as well, see Section 25.5.5 “Water transport under unsaturated conditions”.

The swelling and compression properties of the backfill are important for the function of the Spent Fuel 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 channels, 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 Pusch (2003). 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 2010 and its review

SKB should study the evolution of the mechanical properties as a consequence of geochemical interaction over long periods of time, for example under altered conditions brought about by climate change.

SSM wanted SKB to study in greater depth the influence of hydrogeochemistry (mainly ionic strength) on the elastic, plastic and viscous properties of the clay material.

In the NEA review of SR-Site (NEA 2012), it was discussed whether the buffer’s swelling capacity (and not just the swelling pressure) should be a safety function in the KBS-3 concept.

Newfound knowledge since RD&D 2010

Extensive studies are under way to gain a better understanding of the self-healing capacity of the buffer and the backfill. Some of the results of the studies are reported in Dueck et al. (2011). One type of test has been outswelling of MX-80 bentonite in different geometries, see Figures 25-15 and 25-16.

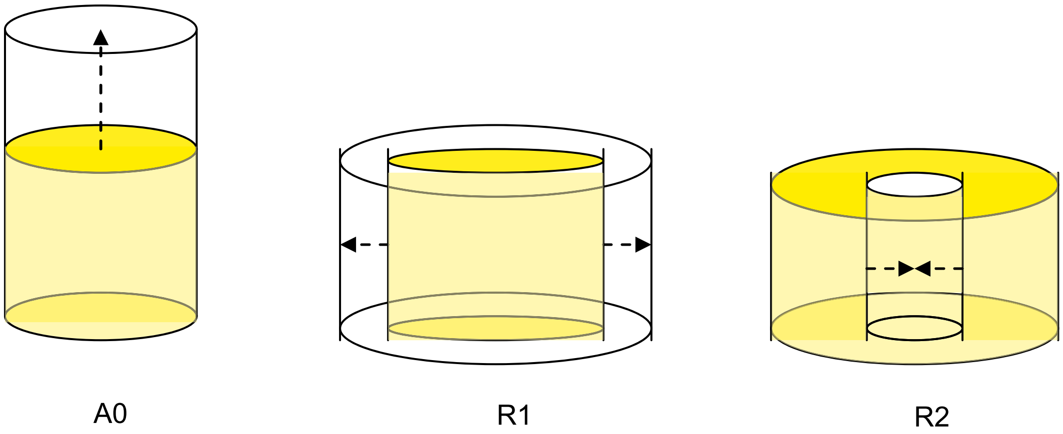


Figure 25-15. *Different geometries for swelling tests. A0 shows axial swelling, R1 radial outward swelling and R2 radial inward swelling (Dueck et al. 2011).*

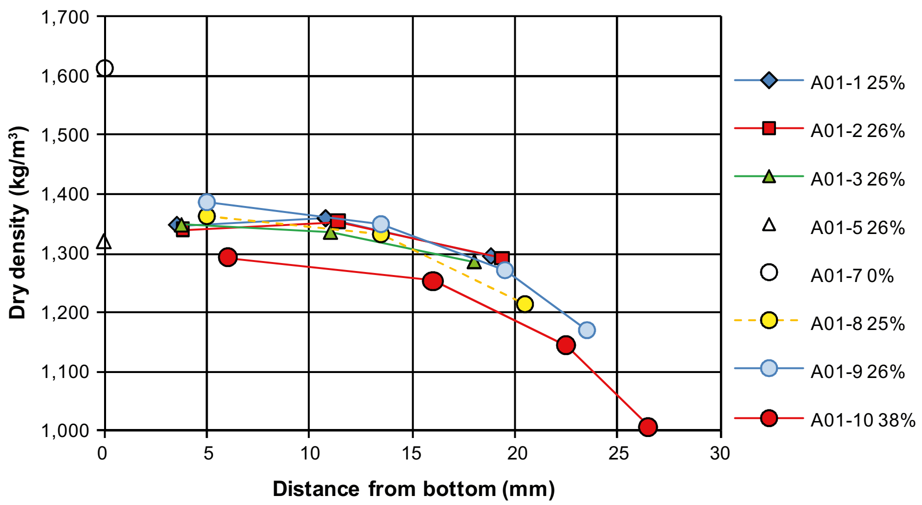


Figure 25-16. *Example of results from axial outward swelling tests with the A0 geometry from Figure 25-15. The percent figure indicates the swelling percentage.*

Programme

The buffer's capacity to homogenize and fill cavities is crucial for its function in the repository. This is particularly true offor:

1. Homogenization of buffer blocks and pellets in connection with water saturation after installation. The degree of homogenization is directly linked to the requirement on installed density and configuration.
2. Self-healing capacity after piping with erosion. This is indirectly linked to the requirements on inflows.
3. Self-healing capacity after colloid formation with erosion.

It is not practical to define a safety function for the buffer's swelling capacity. Swelling capacity is primarily related to initial state and is of less importance in the long term. However, swelling capacity may be related to homogenization capacity and be of fundamental importance for points 1 and 2 above.

A laboratory programme has been started where a number of fundamental pure test principles have been used with axial and radial swelling and simultaneous measurement of these pressures. The tests are also being used as a modelling task within TF EBS.

The laboratory part of this project consists of four parts:

A. Fundamental pure laboratory tests to obtain better knowledge of the constituent material parameters.

B. Friction tests to investigate friction between buffer and rock and between buffer and canister. This friction is a significant uncertainty factor but is at the same time very important for the buffer's healing capacity.

C. Scale tests of homogenization of cavities in bentonite to be used for predictions prior to excavation.

D. Long-term homogenization tests in tubes to evaluate the theories that predict that friction against walls creates residual density and swelling pressure gradients.

The fundamental laboratory tests will mainly be carried out in the large test set-up illustrated in Figure 25-17. The advantage of this equipment is that it provides better resolution of the density distribution and contains a number of sensors for measurement of radial swelling pressures.

The friction tests are continuing. Among other things, friction against other materials such as copper and rock will be studied.

A scale test of homogenization of bentonite on the LOT block scale (30-centimetre diameter) was installed in December 2012. Two large cavities against the simulated rock wall had been cut out in the bentonite block. The test, which involves swelling shut of the cut-out cavities and homogenization, will continue for most of 2013. Residual density gradients will be evaluated by careful sampling at excavation, which is planned for December 2013.

A long-term homogenization test in tubes started in March 2012. The remaining tests will be started during 2013. In these tests, some ten or so tubes (see Figure 25-17), 0.25–0.35 metre in length and 2.5–3.5 centimetres in diameter) will be filled with highly compacted bentonite in one half and bentonite pellets in the other. The pellet half is supplied with water. In some tests, swelling pressure will be measured, mainly axially at the ends but also radially in different places. The tests will be allowed to stand for different lengths of time to enable density distribution and swelling pressure evolution to be studied and residual creep effect to be analyzed. A total of ten such tests are planned in long tubes.

Most of the modelling is being done within TF EBS.

25.5.10 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. This can be sustained as long as the hydraulic conductivity is low. Hydraulic conductivity is discussed in Section 25.5.6 "Water transport under saturated conditions". Advective transport can therefore only occur if the buffer has been converted or been lost. This is discussed in Section 25.5.14 "Montmorillonite transformation" and 25.5.19 "Colloid release/erosion".

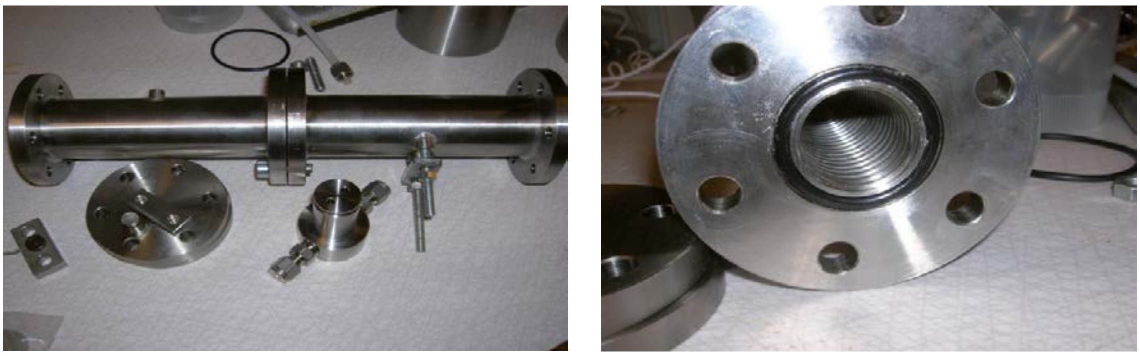


Figure 25-17. *Example of test set-up with long tubes. This tube is threaded to maximize friction between tube and bentonite.*

25.5.11 Diffusion

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 buffer coefficients in the buffer are reduced by a factor of 50 or more in relation to pure water.

Diffusion in the backfill and diffusion in the wall buffer in the silo in SFR can be handled in the same way as diffusion in the buffer.

Conclusions in RD&D 2010 and its review

SSM took a positive view of SKB's efforts to achieve a better understanding of transport properties in bentonite. SKB should, however, consider how its own approaches relate to other alternatives, for example the Maxwell-Stefan method (Krishna and Wesselingh 1997), and analyze whether an alternative conceptual understanding would entail consequences for the long-term safety of the Spent Fuel Repository.

Newfound knowledge since RD&D 2010

In the Alternative Buffer Material (ABM) test, a dozen or so different bentonites were placed side by side and allowed to equilibrate with Äspö water during the period 2006–2009. The different montmorillonites in the different bentonites had different counterion contents at the time of installation; some were dominated by sodium, others by calcium, etc. This made it possible to study ion exchange processes between the different bentonites and between the bentonites and the Äspö water at excavation (Svensson et al. 2011). An attempt has been made to model this with a diffusion and ion exchange model, which is being reported during 2013.

The model is based on a two-dimensional axis-symmetrical geometry of the deposition hole and includes coupled diffusion and cation exchange of sodium, potassium, calcium and magnesium (as a chloride solution) in a stack of 30 bentonite blocks of eleven distinct initial compositions, see Figure 25-18. In the model, ion diffusion is permitted between the individual bentonite blocks and between the bentonite blocks and the sand layer that fills the space between the blocks and the rock (the red zone around the bentonite package in Figure 25-18). The effective diffusion coefficients for individual bentonite blocks were estimated based on the dry density of the bentonite and an estimation of the temperature dependence of diffusion coefficients during the simulation.

Together with a simple description of the difference in hydraulic boundary conditions in the top and bottom part of the deposition hole, the relatively simple model succeeded very well in reproducing experimental data, see Figure 25-19. This is particularly true for the dominant counterions: calcium and sodium. The measured fractions of magnesium in the top part of the test were more difficult to calculate. This may possibly be due to the fact that other reactions than ion exchange have been involved.

Programme

SKB is not pursuing a separate programme to study diffusion or diffusivities. However, diffusion occurs as a natural process in many tests, and models and data can be evaluated/verified there. An example of this is the ABM test described above. The next package in the ABM test was excavated in 2013, and it may then be of interest to perform repeat the modelling of diffusion/ion exchange.

Figure 25-18. Three-dimensional representation of the two-dimensional axis-symmetrical model (generated by Comsol Multiphysics) showing the bentonite blocks and a sand layer surrounding the buffer (two centimetres thick, shown in brown). Colours indicate the type of bentonite block in the model. The total height of the stack is three metres (ten centimetres for each block), while the inside and outside diameters are 0.11 and 0.3 metres, respectively.

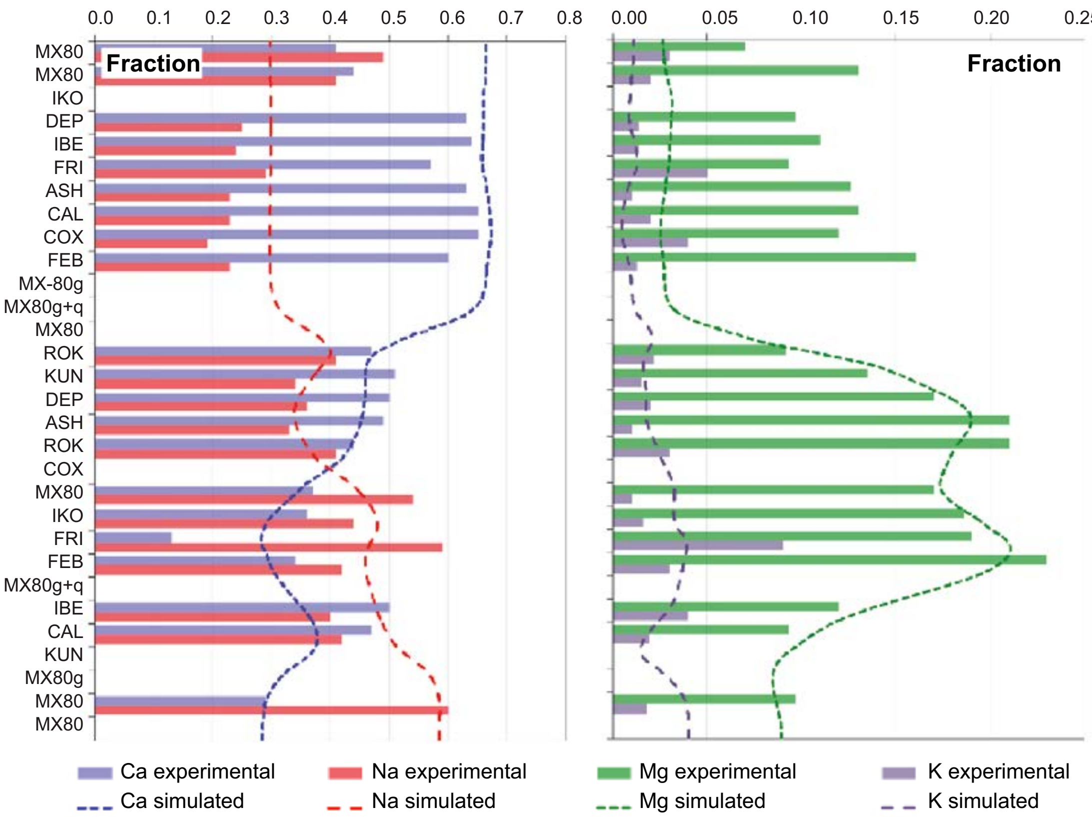


Figure 25-19. Experimental and calculated counterion fractions in ABM package 1, after 880 days in the rock.

25.5.12 Osmosis

The buffer's sealing properties, primarily high swelling pressure and low hydraulic conductivity, are intimately linked to the bentonite's affinity for water. The water affinity of bentonite material is mainly dependent on the proportion of montmorillonite and on variations in the mineral structure of the montmorillonite. 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. Other components in a repository system – such as host rock and salt in the groundwater – 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.

SKB uses the term "osmosis" to describe how the hydraulic conductivity and swelling pressure of the bentonite are affected by the salinity of the groundwater.

Conclusions in RD&D 2010 and its review

Osmosis is an important colloidal-chemical phenomenon in a system of bentonite clay and water and determines many properties of the bentonite clay. SSM agreed with SKB that a better understanding is needed of the properties of calcium-dominated systems. Moreover, SSM took a positive view of SKB's plans to undertake molecular dynamic simulations of ion distributions in the pore water and of how concentration differences with an external solution are maintained in equilibrium. The planned further studies of hydraulic conductivity and swelling pressure as a function of the salinity of the material in the silo in SFR were also deemed valuable.

Newfound knowledge since RD&D 2010

The phenomenon of osmosis is linked to erosion, and certain aspects are dealt with in Section 25.2.19 "Colloid release/erosion". Otherwise, an understanding of the influence of elevated salinity on the properties of the bentonite is sufficient.

Programme

Salinity may have great importance for the properties of the silo buffer in SFR, since the density of the buffer is so low. A programme of sampling and evaluation of the hydromechanical properties of the silo buffer is in progress.

25.5.13 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.

Different types of counterions are associated to the montmorillonite surface with differing strengths, depending on such factors as their valence, hydration capacity and electrical polarization capacity. The counterion distribution at equilibrium is therefore dependent on the concentrations of all positive ions in the external solution.

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. A bentonite in ion exchange equilibrium with a typical Swedish groundwater at repository depth is calcium-dominated.

Different sets of counterions lead to different physical properties in the bentonite. At low densities, for example, a calcium-dominated clay will have a lower swelling pressure than an equivalent sodium-dominated one. The buffer's capacity to form colloids is also dependent on the set of counterions.

The total ion exchange capacity is routinely determined in the laboratory by completely ion-exchanging the clay to a specific ion (e.g. copper ion, Cu^{2+}, or ammonium ion, NH_4^+).

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. Due to proton interaction with the layer edges, these edges may also be positively charged under certain pH conditions.

Sorption in the backfill and in the wall buffer in the silo in SFR can be handled in the same way as sorption in the buffer.

Conclusions in RD&D 2010 and its review

SSM thought that SKB's work with ion exchange and sorption should be extended to include the species that may occur in the Spent Fuel Repository environment and may be of great importance for the repository's long-term safety, for example divalent and trivalent iron, monovalent and divalent dissolved copper and other substances, such as ammonium.

Furthermore, SKB should study the interaction between ion exchange/sorption processes and other geochemical processes such as iron and copper corrosion, as well as microbiological processes involving different nitrogen compounds.

SSM suggested that SKB should study the link between electrostatic interaction and selectivity in a system with highly charged particles and a solution of low electrolyte concentration.

Newfound knowledge since RD&D 2010

Diffusion/ion exchange has been studied and evaluated in ABM, see Section 25.5.11 "Diffusion". In the field test, a dozen or so different bentonites were placed side by side and allowed to equilibrate with Äspö water during the period 2006–2009. The different montmorillonites in the different bentonites had different counterion contents at the time of installation; some were dominated by sodium, others by calcium, etc. This made it possible to study ion exchange processes between the different bentonites and between the bentonites and the Äspö water at excavation (Svensson et al. 2011). The international teams who are analyzing the ABM experiment have together made a unique study comparing different methods for cation exchange with each other. The teams have also selected a specific method and compared the scatter in the data from the different laboratories. This made it possible to identify factors that contributed to uncertainty and data scatter between the results of the different teams (Dohrmann et al. 2012a, b).

Ion exchange may of great importance for piping and erosion. Figure 25-20 shows results from the Eva Project, where water of different compositions has been used to generate piping in a pellet fill. Ion exchange takes place very rapidly in the bentonite closest to the channel, and in the case with calcium-dominated bentonite, swelling capacity is lost since the density low.

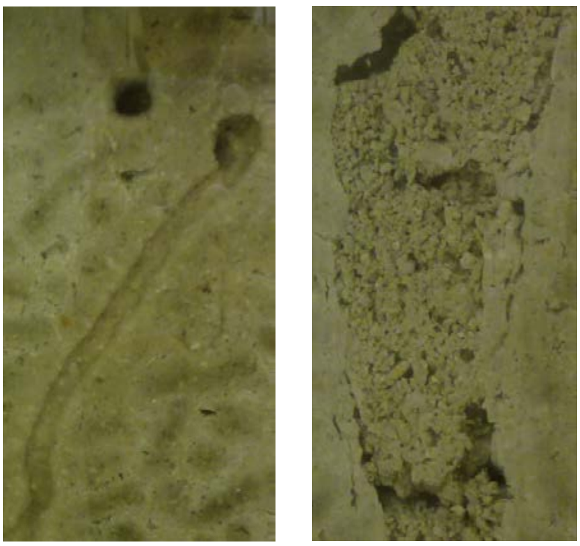


Figure 25-20. *Piping in a pellet fill. Left: sodium chloride solution. Right: calcium chloride solution.*

Ion exchange is also of great importance for the evaluation of colloid release/erosion, see Section 25.5.19. Karnland et al. (2011) report the ion exchange coefficients for sodium and calcium in bentonite with low water-to-solid mass ratios, and the results from the study thereby provide a reliable basis for calculating the distribution of cations in a bentonite buffer with a relatively high density. The results of the study agree well with measurements of ion exchange coefficients in systems with high water-to-solid mass ratios.

Programme

Ion exchange may have great importance for the properties of the silo buffer in SFR, since the density of the buffer is so low. A programme of sampling and evaluation of the hydromechanical properties of the silo buffer is in progress.

In the KBS-3 concept, the buffer is relatively thin and the effect of sorption is therefore only important for relatively short-lived radionuclides. The results from SR-Site show that the probability of early canister failure is low. The contribution of short-lived radionuclides to dose/risk is therefore relatively insignificant. In the silo in SFR, sorption on cement is the dominant retardation mechanism. Sorption studies of radionuclides in bentonite are therefore not prioritized.

Ion exchange is included as an integral part of many other processes, for example in Sections 25.5.8 "Piping/erosion", 25.5.9 "Swelling" and 25.5.19 "Colloid release/erosion".

25.5.14 Montmorillonite transformation

The desirable physical properties of the buffer, especially swelling pressure and hydraulic conductivity, are dependent on the montmorillonite's affinity for water. This affinity is affected by changes in the ion concentration in the groundwater and by changes in the mineral structure of the montmorillonite. The mineralogical stability of the montmorillonite is therefore of crucial importance for the function of the buffer.

Montmorillonite can be stable for hundreds of millions of years in its formation environment, but changes in the geochemical environment can lead to a relatively rapid alteration of the mineral structure.

Minerals occur in nature with a similar structure but with great differences in layer charge. If the charge is close to zero (for example in pyrophyllite), the mineral's affinity for water is insignificant, which results in radically different properties compared with montmorillonite. A minor increase in the montmorillonite's layer charge, and thereby a greater number of 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.

Fixing of charge-balancing ions is dependent to a high degree on the properties of the ion. Potassium ions, for example, are fixed at a lower layer charge than sodium ions, which are in turn fixed at a lower charge than calcium ions. Illite is a material with 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.

Thus, in order for a montmorillonite to be transformed in the direction 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.
- Change of valence in the structure (iron).

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 some uncertainty regarding how bentonite reacts at high pH.

Conclusions in RD&D 2010 and its review

SSM wanted SKB to undertake more in-depth studies of interactions between bentonite clay and low-pH cement as well as causes of the increased ion exchange capacity and the measured rheological changes in LOT. Moreover, the slow dissolution of montmorillonite due to hydrolysis and its effects on the long-term transformation of the material should be studied.

Newfound knowledge since RD&D 2010

The Cyprus Project has been concluded, Phase III has been reported (Alexander et al. 2011) and Phase IV will be reported during 2013. The purpose of the project was to study whether a mineralogical transformation had occurred of natural bentonite on Cyprus, where the bentonite has been in contact with groundwater with pHs of the same order of magnitude as low-pH cement. The alkaline groundwaters on Cyprus derive from serpentinization of ophiolites. In the vicinity of the abandoned village of Parsata, there are rich deposits of smectite that have been in contact with alkaline groundwater for very long periods of time (hundreds of thousands of years). Mineralogical studies of the contact zone between smectite and water show the reactions have been minimal. A possible transformation from smectite to palygorskite has been identified, but the scope is very small despite the long timescale.

The results of the LOT project showed that the magnesium concentration increased nearer the heater (Karnland et al. 2009). This has also been observed by Fernandez and Villar (2010) and may be related to a higher solubility of magnesium compared with other structural cations in smectite – in other words, magnesium dissolves first (incongruently). To delve deeper into this question, bentonite clay was studied in ABM (Section 25.4.4 “Bentonite composition”) with respect to increase of magnesium content and possible structural changes. The analyses showed that many of the ABM materials had an increased magnesium content in the contact surface with the heater (Kaufhold et al. 2013). In the infrared spectrum, many of the materials exhibited a slightly elevated intensity at 680 per centimetre (cm^{-1}), which can be explained by the presence of anhydrite or saponite (trioctahedral 2:1 minerals). Reflection from X-ray diffraction (XRD, d60) supports the theory of formation of trioctahedral minerals in certain of the samples, see Figure 25-21. The studies of the ABM materials indicate that they behave differently under simulated repository conditions after as short a time as about two years. However, no direct conclusions can be drawn from ABM about the stability of the different materials. The materials have been in both electrical and hydraulic contact with each other, which also means that they may have affected each other. The retrieval of ABM package 2 during 2013 will hopefully provide greater insight in the matter.

The interaction between cement and bentonite has been modelled as a part of the supporting material for SR-PSU. The purpose of the study was to shed further light on the significance of uncertainties in data and assumptions for the stability of the montmorillonite component in the silo buffer in SFR. An example of results is shown in Figure 25-22.

Programme

Different transformation models have been published, and a model constructed by Huang et al. (1993) has been used by SKB to justify the temperature criterion for a spent fuel repository. In this model, the kinetic expression and associated parameters have been systematically determined by laboratory tests and the results have been verified against natural systems. Since the conversion takes place slowly at low temperatures, the laboratory tests were carried out at much higher temperatures (more than 200°C) that will prevail in a spent fuel repository. Calculations of transformation in a spent fuel repository therefore include extensive extrapolation, which reduces the reliability of the calculations. SKB has therefore chosen not to use the kinetic model to assess montmorrillonite stability in SR-Site, but instead uses a comparison with natural systems. Studies of transformation at moderately elevated temperature are therefore being conducted. The process is accelerated by other possible variables, for example by an elevated calcium concentration and by maintaining a low silicon content in the montmorillonite/water system.

The laboratory tests are being performed with purified montmorillonite with different counterions (sodium, potassium, magnesium and calcium) that are exposed to equivalent salt solutions and repository-like temperatures. The solutions are analyzed as the test proceeds. After excavation of the tests, the montmorillonite is analyzed with respect to composition and swellability.

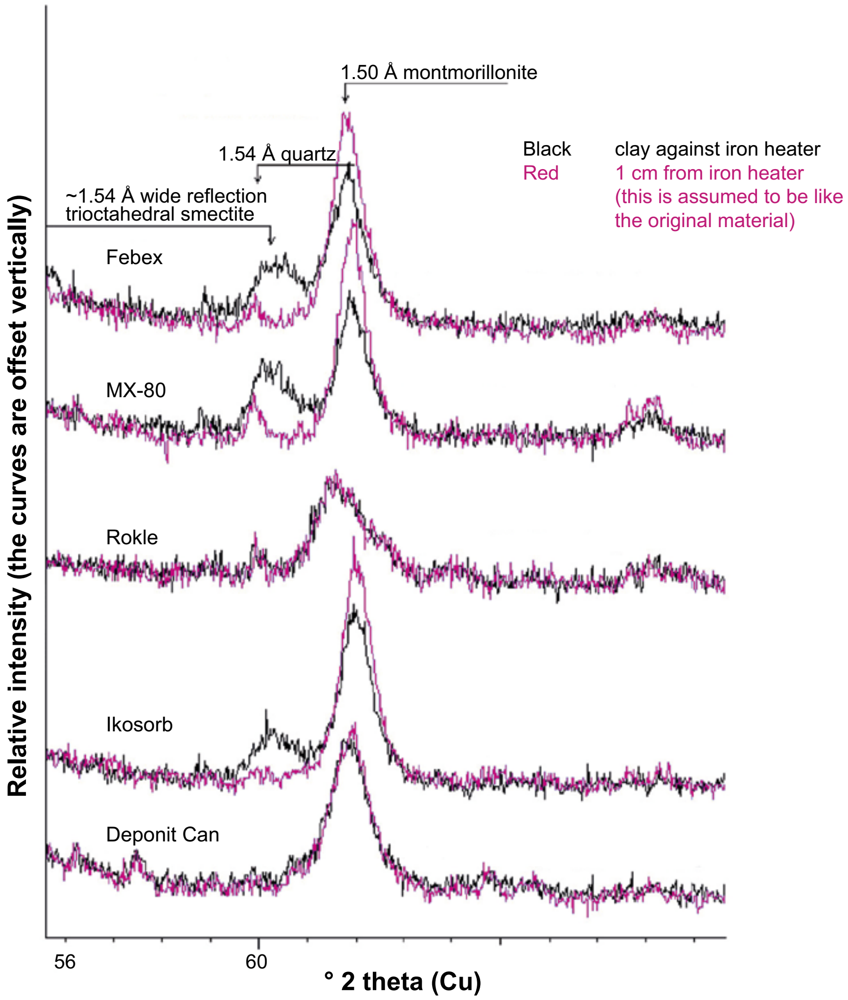


Figure 25-21. *XRD d(060) reflection of selected samples from ABM (Kaufhold et al. 2013). There is supplementary IR data and chemistry data that supports the observations.*

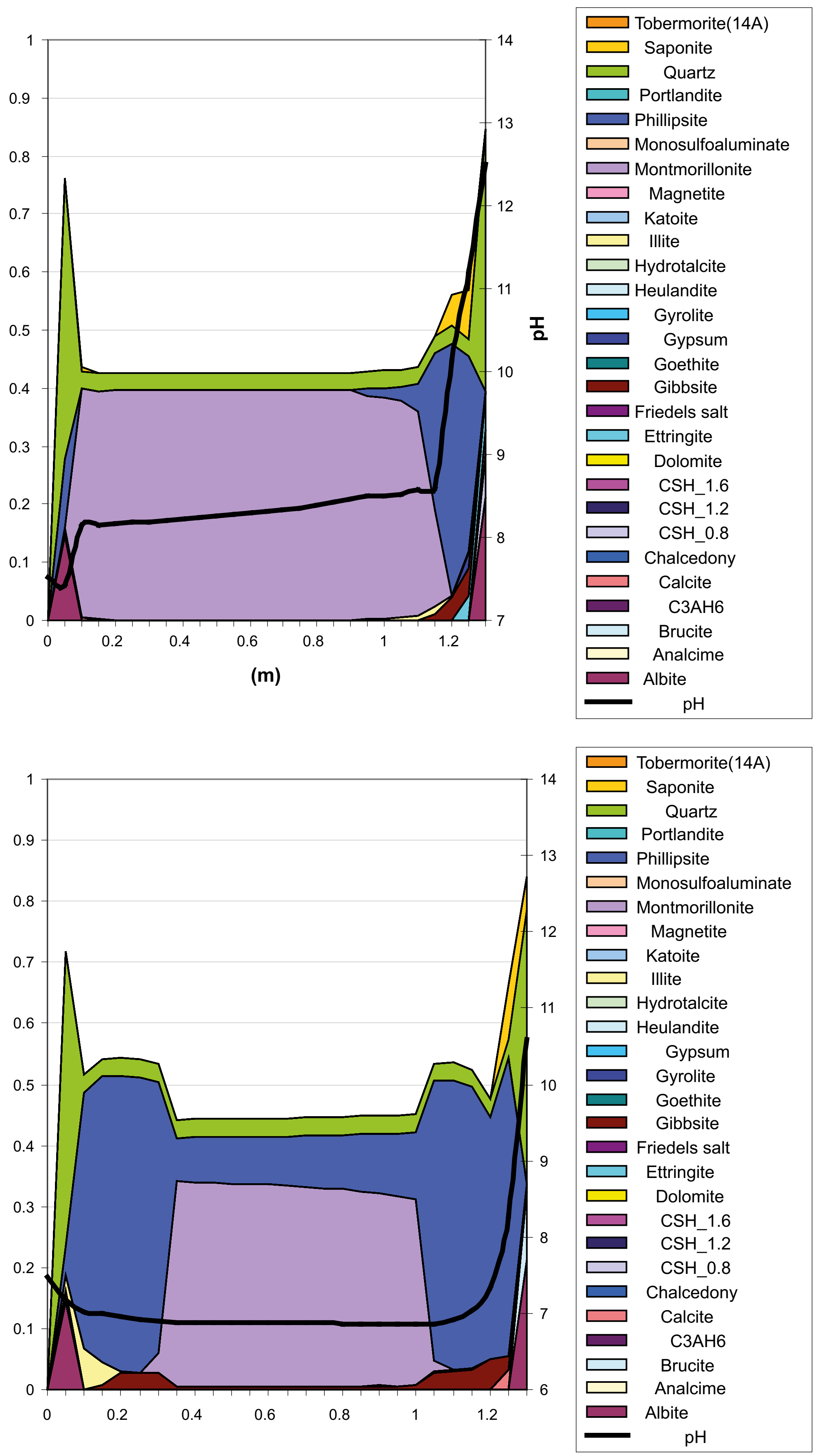


Figure 25-22. *Composition of the silo buffer after 30,000 years for the reference case (top) and for a case with elevated diffusivity (bottom).*

The dry conditions in Forsmark make it interesting to study montmorillonite alteration under unsaturated conditions. Couture (1985) reported that bentonite exposed to heated water vapour lost much of its swelling capacity. However, no mineralogical changes were identified. Temperature tests are therefore being conducted with montmorillonite with a varying degree of water saturation to determine if water vapour has any special effect on the swelling properties. If this is the case, the effect will be quantified for different exposure times and temperatures. The work also includes attempts to identify the mechanism behind such an effect.

Cement and bentonite are not completely compatible with each other. The interaction between cement and bentonite is however being studied both in the nuclear fuel programme (only low-pH cement) and in the programme for low- and intermediate-level waste, for the silo in SFR. In the nuclear fuel programme (KBS-3), the question is of secondary importance, since the requirement that cement components must have a pore water with a pH < 11 means that the effect of the interaction is minimized. In the silo in SFR, the interaction will take place, but there it can be accepted since the timescale is shorter. There are therefore in principle no unique issues for the concept in the final repository for long-lived waste, SFL. The requirements on the durability of the barriers in SFL are, however, tougher than those on the silo in SFR (longer timescales), which means that the requirement on the supporting material may be greater. The studies that may be needed are:

- further development and testing of coupled models,
- laboratory studies of individual processes,
- field tests (evaluation of the Concrete and Clay project),
- analogues,
- technology development (with compatible materials).

The detailed programme will be adapted to the prioritized repository concepts.

In the Concrete and Clay project, long-term tests with bentonite and cement will be installed. The project is one of SKB's long-term experimental programmes, and no results are expected during the period of this research programme.

25.5.15 Iron/bentonite

Metallic iron in an anaerobic environment normally corrodes very slowly, forming iron(II)hydroxide and hydrogen. In an environment with bentonite clay, it is possible that iron(III) in montmorillonite could interact with metallic iron by formation of iron(II). Reduction of iron(III) to iron(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.

Conclusions in RD&D 2010 and its review

In addition to the iron-bentonite interactions, SSM thought that SKB should also study the interaction between corrosion products from the copper canister and the bentonite buffer.

Newfound knowledge since RD&D 2010

In the ABM experiment, a number of different bentonites with montmorillonites with different properties are being investigated to determine if there are differences in properties or long-term stability. The tests are being conducted under accelerated but realistic conditions. Preliminary investigations indicate that at buffer density, no or very small observations can be made suggesting montmorillonite transformation (Svensson et al. 2011). On the other hand, an increase in the iron(II)/iron(III) ratio has been observed in several clays in ABM. This increase is linked to the formation of iron(II) phases by corrosion, but partial reduction of structural iron(III) in montmorillonite cannot be ruled out.

Similar conclusions are drawn in the TBT (Temperature Buffer Test), but without an increase in iron(II) in the clay (Åkesson et al. 2012b). These results contrast with tests done with a low clay/water ratio where various montmorillonite alteration reactions have been identified. The alteration product varies but is mainly dependent on pH and temperature (Guillaume et al. 2004, Jodin-Caumon et al. 2010, Charpentiera et al. 2006, Perronnet et al. 2008).

Corrosion of cast iron has also been studied in the presence of bentonite, for example in the EU project NF-PRO (Near-Field Processes). Further analyses provide continued support to the conclusions presented in RD&D programme 2010. In the near-field of the iron and steel specimens, the bentonite contains elevated concentrations of iron due to corrosion. Most of the iron occurs in the clay mineral and only a small fraction occurs in the form of amorphous or crystalline iron oxide.

The specimens that have now been analyzed have been exposed for a longer time (2–3.5 years) than the specimens analyzed in previous phases of the NF-PRO experiment (Milodowski et al. 2009). Analysis shows that the pore water near the corroding metal contains high salinities and that a precipitation of sodium chloride has formed. This has not been observed in the earlier phases of the experiment but is believed to be due to the fact that iron (steel) consumes pore water so that eventually the bentonite near the corroding metal is dried out. As the water is consumed, the local concentration of salts rises, and at saturation precipitation occurs.

Programme

Interaction can only occur between the iron insert and the bentonite buffer if the copper canister is defective. In SR-Site, this only occurs in the case with shear failure, and the interaction between the insert and the bentonite was assumed to have a negligible effect. This was, however, identified as a remaining uncertainty. Further studies are therefore needed in the area. The amount of montmorillonite needs to be better quantified before and after the experiments. Attempts to enrich alteration products should also be tested, along with better characterization of which iron phases are actually formed. Three new ABM packages were installed at the end of 2012. Another ABM package was excavated in the spring of 2013. On review of the ABM work, SKB has been recommended to use high-spatial-resolution analyses with Raman microspectroscopy, which would provide more information on the contact zone between iron and bentonite and permit correct identification of the corrosion products formed. It is also important to obtain greater knowledge on how the properties of the montmorillonite are affected if parts of the structural iron are reduced. A very large glovebox is being installed at the Äspö HRL during 2013 to enable certain types of specimens to be finely divided and analyzed in an oxygen-free environment.

25.5.16 Copper/bentonite

Copper and bentonite have been selected as barriers in the KBS-3 concept, since they should not affect each other to any appreciable extent. The solubility of metallic copper under repository conditions is so low that the quantity of copper ions that can interact with the bentonite is negligible. The solubility of iron(III) in the bentonite is also so low that it should not affect the copper.

Conclusions in RD&D 2010 and its review

In addition to the iron-bentonite interactions, SSM thought that SKB should also study the interaction between corrosion products from the copper canister and the bentonite buffer. SKB should study the chemical composition of copper that is initially corroded by residual oxygen and later by reductants such as sulphides in the buffer. Furthermore, SKB should elucidate whether the interactions between copper corrosion products and bentonite could lead to lower concentrations of copper ions in bentonite pore water compared with concentrations of copper ions at equilibrium with secondary copper minerals. If so, this could affect the corrosion process. The influence of the corrosion products on the long-term geochemical, colloidal-chemical and geomechanical properties of the buffer should be further studied.

Newfound knowledge since RD&D 2010

Since RD&D Programme 2010, the properties of MX-80 bentonite that has been ion-exchanged with copper(II) have been tested (to be reported during 2013). The conclusion is that the properties of the ion-exchanged bentonite are essentially the same as those of a bentonite that has been ion-exchanged with calcium. Carlsson (2008) has also studied the interaction between copper corrosion products and bentonite and cannot find any impact on the properties of the bentonite either.

Programme

In field tests conducted at the Äspö HRL (LOT), a marginal increase has been noted in the cation exchange capacity (CEC) of the bentonite buffer near heaters. One conceivable explanation for the increase is a redox reaction between the metal in the heaters and the bentonite. A study has been initiated with laboratory tests of bentonite in direct contact with metallic copper. The purpose is to identify whether such a process actually takes place, and if so to quantify it by measurement of the bentonite's CEC. The initial tests are being conducted with thin copper sheet/foil that is placed in a bentonite slurry which has in turn been placed in an anaerobic box. The tests are initially being performed with an iron-rich bentonite (from Ashapura Co.) and with a purified ion-exchanged montmorillonite from the same bentonite.

25.5.17 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.

Conclusions in RD&D 2010 and its review

In the review of RD&D Programme 2010, SSM did not have any viewpoints regarding dissolution/precipitation of impurities.

Newfound knowledge since RD&D 2010

The geochemical evolution in the buffer was modelled in SR-Site. The aspects that were taken into account were:

1. The influence of the period with elevated temperature.
2. The processes that take place when the bentonite is saturated.
3. The interaction of the water-saturated bentonite with the local groundwater.

No safety functions in the buffer are directly linked to this evolution, but an evaluation was done to determine whether the evolution can indirectly jeopardize the safety functions of the buffer.

The geochemical evolution of the silo buffer is described as a part of SR-PSU. Here, however, the focus is on the stability of the montmorillonite. This is described in Section 25.5.14 "Montmorillonite transformation".

Programme

SKB is not conducting any specific programme for dissolution/precipitation of impurities in the buffer or for the evolution of the buffer chemistry. However, these processes comprise a natural part of many other processes (see for example Sections 25.4.4, 25.5.8, 25.5.11, 25.5.14, 25.5.15 and 25.5.19.)

25.5.18 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 so important to take into account for the buffer that it is dealt with under its own heading. Typically, cementation is caused by mineral precipitations in bentonite, and there are therefore strong links to other processes.

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.
- 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.

Conclusions in RD&D 2010 and its review

SSM took a positive view of the fact that SKB has conducted new tests to study the effects of cementation on the buffer's geomechanical properties. SSM suggested that SKB should study the dependence of cementation on temperature and justify the chosen safety function indicator of a maximum temperature of 100°C. Moreover, SKB should study whether cementation can also affect the geochemical properties of the buffer.

Newfound knowledge since RD&D 2010

The studies of factors that affect the rheological properties of the buffer are continuing within the EU project PEBS (Long-term Performance of Engineered Barrier Systems). An example of results is shown in Figure 25-23.

Clearly, sample preparation is of importance for the test results.

Programme

The cementation studies will continue until the end of PEBS. The purpose is to improve the material database for the buffer. This is of particular importance for calculations of shear load on the canister.

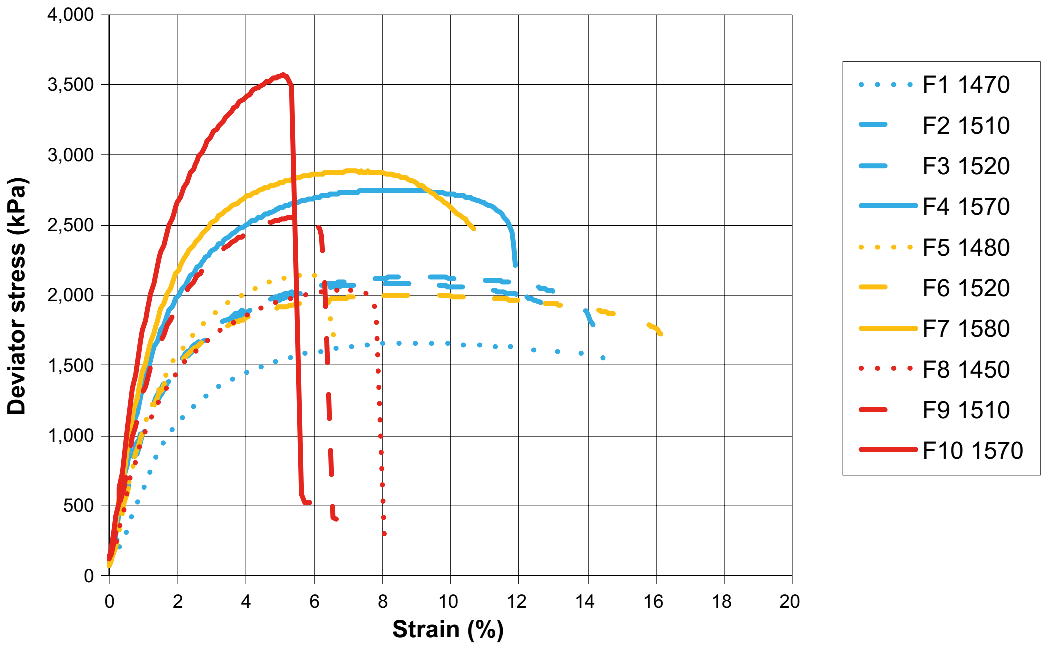


Figure 25-23. *Stress-strain diagram for MX-80. The blue curve represents basic material. The yellow curve represents washed material. The red curve represents the fine fraction (<2 millimetres) of the washed material.*

25.5.19 Colloid release/erosion

The uptake of water and resulting swelling of the bentonite buffer is hindered by the walls of the deposition hole, and a swelling pressure is developed in the bentonite. If fractures intersect the deposition hole, rigid swelling restrictions do not exist everywhere, and localized swelling continues into the fractures until equilibrium or steady-state conditions are reached. This free swelling may lead to separation of individual montmorillonite layers (dispersion) and some of the buffer could be transported away with the groundwater. The maximum free swelling of bentonite is strongly dependent on the charge and concentration of the ions in the interlayer spaces. At low concentrations of solutes in the groundwater, the interlayer distance between the individual montmorillonite layers may increase so much that the clay/water system takes on a sol character, i.e. single or small groups of montmorillonite layers behave like colloidal particles.

It was assumed in SR-Site that water with a cation content higher than four millimolar charge equivalents can prevent colloidal sol formation, provided that the calcium content in the montmorillonite is greater than 20 percent, irrespective of montmorillonite type.

If the pore water cation concentration is lower than four millimolar charge equivalents, the particles in the bentonite-water interface can swell/diffuse into the flowing water and be carried away. There is also a region where the gel/sol has so low a particle concentration that it is only a little more viscous than water and can flow away. A force-balance model for spherical colloids has been adapted to parallel clay layers an used in SR-Site to calculate the swelling of sodium montmorillonite into fractures filled with water of low ionic strength. Advective loss of montmorillonite is modelled by combining the force-balance model for swelling with a viscosity model for the repulsive montmorillonite gel and the Darcy equation for two-dimensional flow in a fracture intersecting the deposition hole.

Conclusions in RD&D 2010 and its review

In its review of RD&D Programme 2010, SSM took a positive view of the fact that SKB had taken the initiative and started the large research project "Bentonite Erosion". SSM judged that very significant results had been obtained within the project and that SKB's understanding of the process had been greatly deepened.

SSM concurred with SKB that chemical erosion of the buffer process is an important process in the long-term evolution of the Spent Fuel Repository that is of great importance for the long-term safety of the repository. It is one of the two identified processes in SKB's previous safety assessment SR-Can which can lead to canister failure within a million years (the other is earthquakes). In view of the fact that erosion could begin as early as after the first major glaciation (which could occur in about 60,000 to 70,000 years) and could jeopardize the integrity of an engineered barrier in a multiple-barrier system, SSM felt that it is important for SKB to have a sufficiently deep understanding of the process and that well-founded consequence analyses can be formulated for the case with a completely or partially eroded buffer (Apted et al. 2010).

SKB should strive to make models of the chemical erosion of the buffer internally consistent. The existence of unclarified fundamental differences between both assumptions and results of different models in the buffer erosion project may complicate the assessment of whether the understanding of the process is deep enough and whether the quantification of the process can be regarded as conservative or not. SSM took a positive view of the fact that SKB planned to study different aspects of the erosion process, such as the influence of calcium, and to conduct more carefully controlled experiments. SSM considered that SKB should also continue development of the mathematical modelling of chemical erosion relating to more realistic systems.

SKB should further study the effects of buffer erosion on other safety functions in the barrier system. An example of a particularly important question is whether the hydrogen gas concentration can be sustained at the level required to have an effect on fuel dissolution when the buffer erodes.

Other related questions which SSM deemed to be interesting are whether:

- Colloids that are formed directly from actinides (possibly from other radionuclides as well) and are not filtered can be transported more rapidly out of the Spent Fuel Repository and thereby affect the radionuclide transport calculations.
- Microbial processes that take place next to the canister or directly on the surface of the canister may be of great importance for general corrosion (Apted et al. 2010).
- Removal of uranium sorbed on colloid particles can affect fuel dissolution.

Newfound knowledge since RD&D 2010

A focus area for the erosion programme during the past period has been to study erosion in an artificial fracture instead of in a tube or through a filter, as was the case in the previous programme. This type of experiment is being conducted by several laboratories within the framework of the EU project BELBAR (Bentonite erosion: Effects on the Long term performance of the engineered Barrier and Radionuclide transport). An example of an experimental set-up is shown in Figure 25-24.

Both swelling and erosion can be studied in this type of experiment. The parameters included are the same as those used in the erosion model for a deposition hole in SR-Site (Moreno et al. 2010). Figure 25-25 shows the results of an outswelling test without water flow.

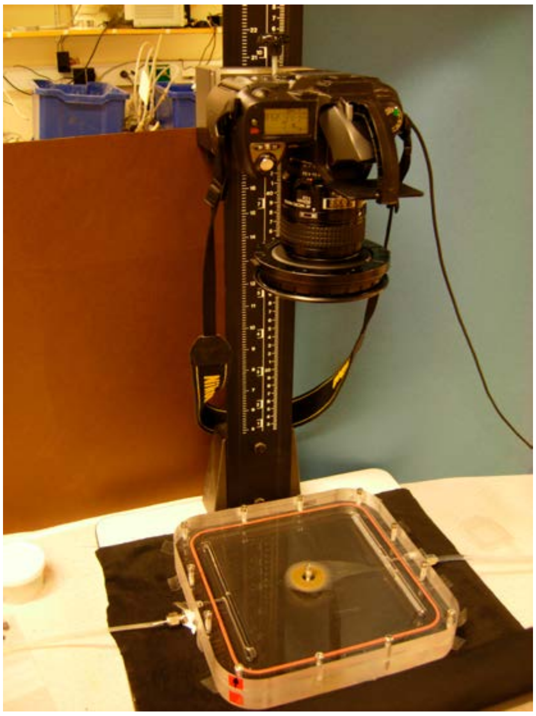


Figure 25-24. *Experimental set-up for a test with artificial fractures.*

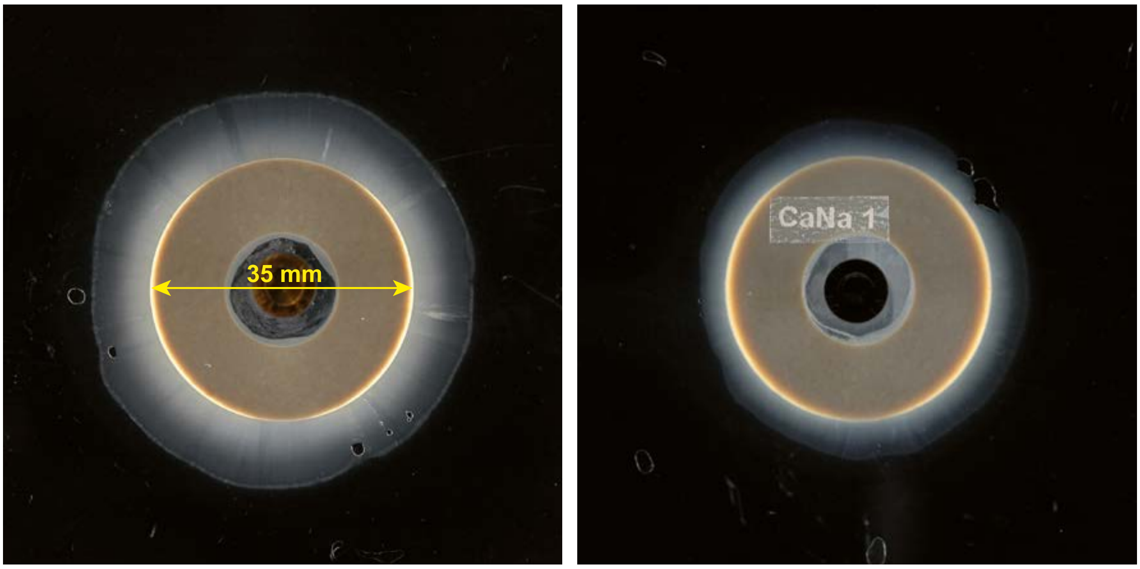


Figure 25-25. *Outswelling of bentonite in a 0.1 millimetre artificial fracture with distilled water. The left-hand picture shows a pure sodium montmorillonite, while the right-hand picture shows a 50/50 sodium/calcium montmorillonite.*

Three different swelling regions can be seen in both clays: a white region nearest the block with high density and viscosity, a transparent grey region and an outer region with white fringes. The outswelling is much greater for the pure sodium montmorillonite, however.

Pure calcium montmorillonite does not form colloidal sol, not even in distilled water. A study has therefore been carried out to determine how much sodium can be ion-exchanged into the clay without this property being changed (Hedström et al. 2011). An important conclusion is that as much as about 90 percent of the ion exchange capacity must be occupied by calcium in order for the clay to behave as a calcium montmorillonite in this respect. The study also showed that a pure calcium montmorillonite does not erode through a 100 micrometre filter, while an 80/20 calcium/sodium montmorillonite can easily pass through a 2 micrometre filter.

The model used to calculate erosion is relatively complicated. The solution can lead to numerical problems due to the mutual effect of three factors: the evolution of the chemistry in the gel/sol system, the forces that act on the smectite particles and the viscosity of the gel/sol, which determines how they can flow. A study has therefore been made to determine which mechanisms have a dominant effect on erosion and how simplifications could be made to obtain more transparent, faster and stabler calculations. It turns out that water chemistry is of crucial importance in the area where the gel/sol meets the flowing water, and it is therefore possible to uncouple the chemical evolution from the swelling in the model and thereby simplify the solution.

Programme

The topics are being studied in the EU BELBAR project and in separate SKB projects.

An extensive experimental programme will continue under SKB's auspices. The main elements in the programme are as follows:

- Production of purified sodium and calcium smectite based on four bentonite materials with varying layer charge and charge distribution.
- Erosion tests of sodium and calcium montmorillonites in an artificial fracture.
- Improve understanding of the homogenization process in conjunction with heavy erosion. To what degree will the clay be able to homogenize? These experiments will be done in pressure cells.
- Determination of phase diagrams for sodium and sodium/calcium montmorillonites.
- Rheological tests of mixed sodium/calcium montmorillonites with given sodium/calcium conditions.
- Develop the theoretical understanding of the process and try to relate theoretical concepts to the experimental observations.

The purpose is to achieve a better process understanding, which can in turn lead to a better ability to quantify the loss of bentonite, if any.

The model used for buffer loss in SR-Site will be further developed. It is important to gain a better understanding of how a completely or partially calcium-ion-exchanged bentonite erodes. The goal is to investigate the possible effects of how different mechanisms and factors affect the colloidal stability of dispersions of calcium-dominated bentonite in dilute solutions, such as correlations of ions, differences in sizes of small ions and heterogeneity of surface charges, etc. The results will be used to explain why calcium-dominated bentonite behaves quite differently compared with sodium bentonite.

Extensive research on bentonite erosion is also being pursued in BELBAR by other organizations than SKB. Hence, the project can serve as a scientific discussion forum where issues can be studied and elucidated in a much more effective way than if the topic were studied by a single organization.

25.5.20 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. Before canister failure, only gamma radiation can affect the buffer. In a case with a breached canister where radionuclides escape, it is also possible that the buffer may be exposed to alpha radiation. The topic is dealt with in SR-Site in a study by Pusch et al. (1993) for the case with gamma radiation. There, compacted MX-80 bentonite was exposed to a total gamma dose of 30 megagrays, which is several orders of magnitude more than will be the case in the repository. Mineralogical studies showed negligible differences compared with samples not exposed to radiation. Gu et al. (2001) showed that the amorphization dose for montmorillonite for alpha radiation is 30 gigagrays. Calculations have shown that the total dose from sorbed alpha emitters in the bentonite nearest the canister in the event of early canister failure is $8 \cdot 10^{15}$ alpha particles per gram of bentonite. This is equivalent to a dose of eight megagrays, which is several orders of magnitude lower than the amorphization dose.

Conclusions in RD&D 2010 and its review

SSM said in its review of RD&D Programme 2010 that SKB should give a better account of the issue of radiation-induced montmorillonite transformation. (No references were cited in RD&D Programme 2010, which made SSM's judgement more difficult.)

Further, SSM thought that SKB should heed viewpoints from the Royal Swedish Academy of Sciences, the Royal Institute of Technology (KTH) and Milkas (Hultén) concerning the effects of radiation on bentonite/water and uncertainties in conjunction with dose estimation.

Newfound knowledge since RD&D 2010

No new research has been conducted since RD&D Programme 2010. What was requested in the review of RD&D Programme 2010 was knowledge that existed, but where the references needed to be stipulated.

Programme

No further studies are planned in the area of radiolysis of pore water.

25.5.21 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. Since microbial formation of sulphide is primarily of importance for the properties of the canister, it is handled in Section 24.2.8 "Corrosion of copper canister".

Conclusions in RD&D 2010 and its review

The process mainly affects the canister and is therefore presented in Chapter 24.

Newfound knowledge since RD&D 2010

The process mainly affects the canister and is therefore presented in Chapter 24.

Programme

One reason why metallic iron can transform montmorillonite is assumed to be because it can act as a reductant of the montmorillonite's structural iron(III). A change in the oxidation number of the iron leads to a change in the montmorillonite's layer charge and thereby also its properties and potential to be transformed into a non-swelling clay mineral.

There are microbes in nature (iron-reducing bacteria, IRB) that can reduce trivalent iron as a part of their metabolism. The presence of IRB in different bentonites has been investigated in ABM experiments (Svensson et al. 2011). IRB were also studied when the outer section of the Prototype Repository was excavated, where they were observed sporadically, especially near the contact with the rock (to be reported in 2013). There are indications that these microbes can in certain cases reduce the structural iron in montmorillonite and thereby also increase the probability of illitization (Kim et al. 2004). An experimental programme is recommended to examine under what conditions this transformation can occur, and above all if it can also take place in compacted bentonite.

Another potential phenomenon that is being studied is microbial activity in a "water saturation window". The activity of microbes in a compacted water-saturated bentonite is very low, and the same applies in a bentonite with a low degree of water saturation. Theoretically, however, there could be a "window" in water saturation where the water content is high enough to favour microbes, but the swelling pressure is too low to inhibit them. The survival and activity of sulphate-reducing bacteria are being studied in bentonite samples that have been water-saturated and equilibrated at different water activities. Two types of tests are now planned: one with constant water activity/pressure in each individual sample and one with a gradient of water activity/pressure through the individual tests. Equilibration of the latter is planned to be done via an external gas phase where the water activity is controlled by means of saline solutions on the sides of the samples.

25.5.22 Radionuclide transport in the buffer

The transport resistance between the immobile water in the buffer and the limited area on the fractures in the rock is an important barrier function in the KBS-3 concept.

Conclusions in RD&D 2010 and its review

No direct viewpoints emerged from the review of RD&D programme 2010.

Newfound knowledge since RD&D 2010

With the goal of experimentally verifying and visualizing the Q-equivalent model, this project has focused on the mass transport of radioactive material from the bentonite clay into the groundwater flowing in the fractures in the rock, see Figure 25-26. This has been done by first visually determining the aperture in a transparent artificial rock fracture and then measure the concentration of a light-absorbing substance that is permitted to diffuse into the fracture at set intervals, see Figure 25-27. Further experiments are aimed at linking diffusion in the fracture with an imposed perpendicular flow.

Programme

The project for verifying the model for transport between the immobile water in 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 produce results and make parameter variations quickly.

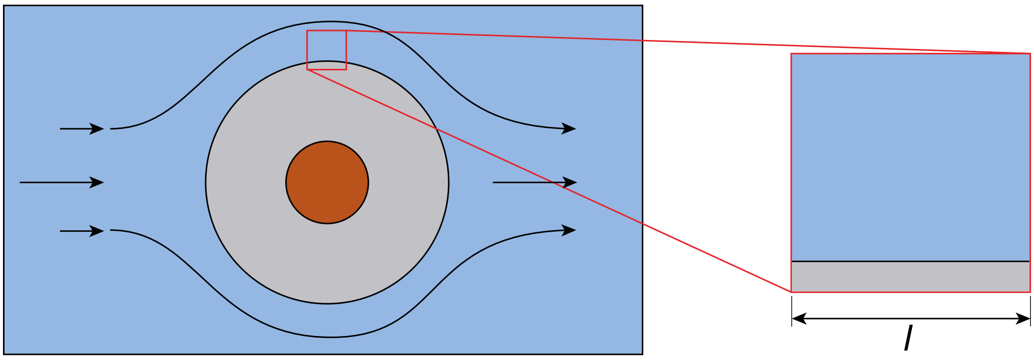


Figure 25-26. *Principle of the Q-equivalent model.*

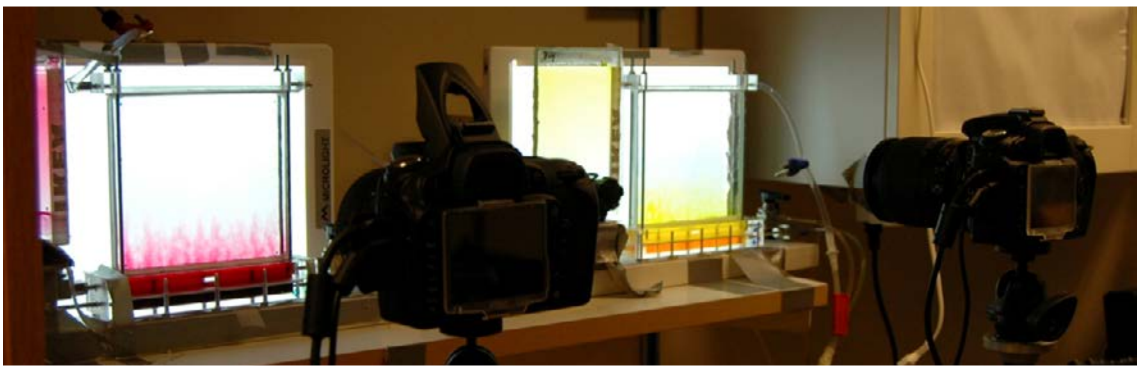


Figure 25-27. *Measurement of diffusion in an artificial fracture.*

26 Geosphere

26.1 Initial state of the geosphere

According to the discussion in SR-Site Section 5.1, the initial state of the geosphere is defined as the natural undisturbed state at the time when the excavation of the repository begins. This means that even short-duration processes or changes in the geosphere caused by excavation and operation of the repository need to be considered. The primary basis for determining this initial state consists of the results from the site investigations (i.e. the site description and its models, see for example Andersson et al. (2013)), the monitoring programme (SKB 2007a), the detailed characterization (see Section 14.4 “Detailed characterization”) and the rock works (see Section 14.5 “Execution methods and construction materials”).

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 during the operation of the facility. The requirements and premises that apply to the underground openings are described in the programme for technology development, rock, see Chapter 14.

The initial state of the geosphere is mainly described with the site model (SDM). Its description (SKB 2011e, summary in Chapter 4) is refined in part with the aid of the methods described in Section 14.4 “Detailed characterization”, 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.

26.2 Overview of processes in the geosphere

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 caused by construction of the repository.

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 (repeated movements in existing fractures) or creep in rock (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. Solutes are carried along with the flowing water by advection. 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 the repository structure, 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 concerned with 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 strongly 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.

Conclusions in RD&D 2010 and its review

In RD&D Programme 2010, SKB presented an overview of the processes that affect the Spent Fuel Repository's barriers and groundwater transport of solutes to and from the repository's near-field. For a more detailed account of the programme associated with these processes, SKB referred to the subsequent chapters in RD&D programme 2010.

SSM complained about the lack of a description of how the selection of processes in this section was linked to Table 17-1 in RD&D Programme 2010, which presented an overview of the processes and the scope of the planned activities. SSM noted the lack of references to the research on erosion-related processes described in the climate section.

Newfound knowledge since RD&D 2010

See relevant process in Sections 26.3–26.23.

Programme

See relevant process in Sections 26.3–26.23.

26.3 Heat transport

Conclusions in RD&D 2010 and its review

RD&D Programme 2010 described the principles for thermal modelling of the Forsmark and Laxemar sites. Spatial variation and upscaling to relevant modelling scales were coupled to the geological site models and analyzed by means of geostatistical methods (Sundberg et al. 2008).

The continued work of quantifying and limiting uncertainties in temperature calculations, as well as development of both field and laboratory methods for determination of thermal properties, was also presented. The focus was on methods for determinations of canister spacings in deposition tunnels, which is dependent on e.g. thermal conditions. This means that it is possible to take into account the distribution and spatial correlation of thermal conductivity properties in the thermal dimensioning calculations that serve as a basis for the canister spacings in the different rock domains for a given layout (Hökmark et al. 2009). An important part of the thermal dimensioning calculations is to determine an adequate, but not excessive, uncertainty margin.

The THM report for the geosphere (Hökmark et al. 2010), which also served as a basis for SR-Site, analyzed the consequences in terms of number of canisters that would be affected if the total uncertainty margin in the dimensioning calculations should for some reason prove to be insufficient.

RD&D Programme 2010 also commented that 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. With the spatial variation of the thermal conductivity on a larger scale, traces of historic climate changes can be analyzed (Sundberg et al. 2009).

In its review, SSM suggested that SKB should refine field methods in order to be able to verify the thermal properties of the rock in an effective manner.

Newfound knowledge since RD&D 2010

Development work on measurement of thermal properties has primarily been pursued within the framework of the detailed characterization programme, see Section 14.4.

Certain model development regarding thermal processes has taken place in conjunction with analysis of the Prototype Repository in the Äspö HRL (see Section 13.1.2 "Backfill and plug"). The most important conclusion is that the thermal evolution can be described well with the methodology used to predict the thermal evolution in Forsmark. This methodology is based on the thermal relationships incorporated in the distinct element code 3DEC. In this code, the cylindrical canisters are represented by specially developed combinations of line sources, the rock is assumed to have homogeneous heat transport properties, and the fact that the tunnel backfill and the bentonite buffer have different heat transport properties than the rock can be disregarded.

With this methodology and with thermal conductivities in agreement with values calculated by means of laboratory determinations performed on samples from the rock mass in the Prototype Repository (Sundberg et al. 2005), the temperature evolution in all measurement points in the experiment's outer section are reproduced with an error of less than two degrees. In the inner section, the calculations systematically overestimate the temperature by 2–4 degrees, but only after drainage of the experiment has begun about 1,000 days after the start of heating. The explanation, or part of the explanation, for the discrepancy is probably the cooling of the inner section caused by drainage. Heat transport in the outer section, where drainage has been considerably less, appears on the other hand to be very well described by the methodology that has also been employed for the Spent Fuel Repository in Forsmark.

Programme

The development work regarding measurement of thermal properties will primarily be pursued in the second phase of the detailed characterization programme, see Section 14.4 "Detailed characterization". Moreover, to further develop the analysis of thermal load, the accuracy with which the decay heat can be determined must be clarified, see Section 11.7 "Decay heat". Others initiatives with a bearing on thermal processes and heat transport are handled within the framework of possible climatological scenarios, see Chapter 19 "Climate evolution".

26.4 Groundwater flow

Compared with previous years' RD&D programmes, the section "Groundwater flow" has a somewhat different structure in this year's programme. The section is divided into five subsections: "Surface hydrology and near-surface hydrogeology", "Hydrogeology in the deep rock", "Specific issues linked to the Spent Fuel Repository", "Specific issues linked to SFR and SFL", and a description of the international cooperation within SKB's "Task Force on Modelling of Groundwater Flow and Transport of Solutes". Each subsection describes in the customary order conclusions in RD&D 2010 and its review, results achieved since the previous RD&D programme, and the programme for the coming RD&D period. This new structure makes it easier for the reader by structuring the contents more clearly. Furthermore, links and references to other chapters in the RD&D programme are clearer. Also noteworthy is that the new structure discusses matters relating to the initial state of groundwater flow in the present and the next section ("Gas flow/dissolution"), and not in Section 26.1 as in previous RD&D programmes.

Furthermore, a change has been made in the subdivision of surface hydrology processes with regard to what is described in this section and what is described in the sections on the biosphere/surface system. Modelling of water flow and solute transport in the uppermost part of the system (usually called the surface system, surface ecosystems or the biosphere) includes flow and transport in near-surface groundwaters and different forms of surface water on the ground surface, as well as interactions between surface water and groundwater, and between water at or near the ground surface and water in the atmosphere. The issues investigated are primarily linked to the safety assessment's modelling of radionuclide transport in surface ecosystems and associated radiation doses to man and the environment (long-term consequences). Moreover, data are needed as a basis for the environmental impact assessment's analyses of the repository's hydrological impact and its consequences during the comparatively brief initial construction and operating period.

The need to integrate model descriptions, and the fact that coordination of modelling efforts has become increasingly into focus as the latest generation of site-descriptive models and safety assessment models have been developed, means that this RD&D programme, to a greater extent than previous ones, describes development work for rock and surface systems jointly. Most aspects of water flow and non-reactive transport in near-surface groundwaters and surface waters are therefore dealt with in this section. In Chapter 27 "Surface ecosystems", the descriptions of flow and transport are restricted primarily to lessons learnt and development efforts directly linked to biosphere objects and associated modelling of radionuclide transport and doses, see Section 27.6 "Hydrology and transport". Retention and reactive transport of radionuclides in the soil layers are also dealt with in the chapter on "Surface ecosystems", see Sections 27.5 "Biogeochemistry" and 27.6 "Hydrology and transport".

26.4.1 Surface hydrology and near-surface hydrogeology

Conclusions in RD&D 2010 and its review

In its review of RD&D Programme 2010, SSM generally concluded that they concur with SKB's view that hydro- and transport modelling in near-surface systems are important components in providing process understanding on the migration of radionuclides in the surface environment and as input to the safety assessment's biosphere transport models. SSM also took a positive view of the fact that SKB has improved the links between the hydromodels, the process-based transport models and the biosphere transport models.

Newfound knowledge since RD&D 2010

As during the preceding programme periods, development of understanding, conceptual models and numerical models since RD&D Programme 2010 has mainly been pursued in project form aiming at producing supporting material for applications to build and operate the Spent Fuel Repository. Concerning surface hydrology and near-surface hydrogeology, the surface hydrological modelling of Forsmark carried out within the SR-Site safety assessment is presented by Bosson et al. (2010). To obtain data as a basis for comparisons between the sites, surface hydrological modelling of Laxemar Simpevarp was also done; this much smaller modelling task is presented by Sassner et al. (2011). All surface hydrological modelling was done with the modelling tool MIKE SHE, which includes calculation routines for e.g. saturated and unsaturated groundwater flow, surface water flow on the ground surface and in watercourses, evaporation processes and the impact of vegetation on hydrology. Density-driven flow is not simulated using this tool, however.

Considerable development efforts have been devoted to modelling of future hydrological conditions, where extensive surface hydrological modelling of Forsmark was done for SR-Site in order to gain a better process understanding and to obtain better data for transport modelling. The importance of different processes and conditions for water balances and other general hydrological characteristics was investigated by means of a series of calculation cases. These included different shoreline locations representing different points in time from the present to 10,000 years AD, climate cases representing today's climate, future climates with a higher temperature and more precipitation, and colder climates with permafrost, as well as different variants of soil layer models and vegetation descriptions. The results show the importance of climatic conditions, compared with other variables, for the hydrology of the area as a whole. These models and modelling results have been further analyzed and published as articles in scientific journals (Bosson et al. 2012a, Berglund et al. 2013a).

Describing periglacial systems with permafrost reaching relatively great depths in the rock has been a matter of particular interest during the past programme period. MIKE SHE modelling was done of a hypothetical future periglacial system in Forsmark as a part of SR-Site. The presence of permafrost was represented by assigning very low permeabilities to rock and soil layers, and the dynamics in the active layer during the year was simulated by a time series where the uppermost part of the ground profile was either frozen, unfrozen or in transition between these states.

The modelling runs were focused to a great extent on taliks, i.e. unfrozen areas that can exist beneath large lakes and can act as flow and transport pathways from repository depth to the surface. Transport pathways in and between recharge and discharge taliks in the model area, and flows in different types of taliks were calculated (Bosson et al. 2010). As expected, the results show that great changes occur in the flow pattern when the climate has changed so much that permafrost is established. Also in this case, in-depth analyses have been performed resulting in scientific publication (Bosson et al. 2012b).

Periglacial hydrology is also the theme of a Ph. D. project at Stockholm University that has been on-going since 2010. This Ph. D. project is being conducted within the GRASP project (Greenland Analogue Surface Project). Field measurements are made within a catchment area characterized by periglacial conditions, a thick permafrost layer and a presumed talik in Kangerlussuaq on Greenland. This is the same area being used in the GAP, see Section 19.7 "Greenland Analogue Project (GAP)". The research has thus far largely been focused on establishing monitoring stations and collecting data, but some initial modelling work has also been performed. Two publications (Bosson et al. 2012a, b) were published during 2012 as part of the Ph. D. project.

In the SR-Site safety assessment, surface hydrological modelling was also used to more closely study flow and waterborne transport to and within selected biosphere objects and to generate input data for the transport and dose calculation model used in biosphere modelling. These activities are described in Section 27.6 "Hydrology and transport", where research and development in modelling of reactive transport is also described, i.e. models that combine transport with flowing water and processes that contribute to transformation and/or retardation of the transported substances.

The activities described above are all related to the evaluation of long-term consequences of possible radionuclide transport from a repository after closure. MIKE SHE was also used for modelling of the impact on groundwater levels during the initial construction and operating phase. The repository remains open during this period, and leakage of groundwater into the tunnels may impact groundwater near the ground surface and various types of surface waters. This impact, which mainly consists of lowering of the groundwater table (drawdown), can in turn lead to negative consequences for ecosystems and natural assets worthy of protection.

An important development task that has now been reported is the actual methodology for combining the modelling of the hydrological impact of the repository with the identification and classification of natural assets in the impact assessment. Summarizing reports that describe methodology and execution have been published by SKB, see Werner et al. (2010a) regarding Forsmark and Werner et al. (2010b) regarding Laxemar-Simpevarp. The work has also been presented in the form of scientific articles (Werner et al. 2011, 2013). The hydrological modelling, which includes a further development of the coupling between the groundwater model MIKE SHE and the calculation routine that handles the inflow to the tunnels, is described by Mårtensson and Gustafsson (2010). Detailed modelling of specific natural assets (ponds and wetlands), including analyses of measures to be taken such as supply of water, is presented by Mårtensson et al. (2010).

All development initiatives described thus far that deal with Forsmark have been carried out within or in direct connection with SKB's efforts to prepare applications for construction and operation of the Spent Fuel Repository. Data from SKB's site investigations and monitoring programmes have, however, also been used in scientific research on more general hydrological topics. SKB's investigation area in Forsmark lacks major watercourses and thereby constitutes an example of a type of area that is not usually the object of extensive collection of hydrological data such as surface water flows. The results from the general scientific research benefit SKB directly and strengthen our knowledge base concerning Forsmark.

Meteorological and hydrological data from Forsmark have been used to quantify the importance of such areas for solute transport to the Baltic Sea. These data have also been used in comparisons between different types of catchments and to investigate the effects of degradation and retardation processes along different transport pathways in areas of the type Forsmark represents. This has resulted in a number of scientific publications produced at Stockholm University, for example journal articles by Darracq et al. (2010), Destouni et al. (2010) and Persson K et al. (2011) and a Ph. D. dissertation (Persson 2011). Time series data from Forsmark have also been used as a part of the supporting material for a Ph. D. dissertation at KTH (Juston 2010, 2012).

Programme

The biggest research effort in surface hydrology will be carried out within the above-mentioned Ph. D. project in surface hydrology with a link to periglacial conditions. Knowledge concerning hydrological processes in periglacial areas is inadequate, and a better conceptual understanding of hydrology and near-surface hydrogeology in an environment dominated by permafrost is needed in order to be able to describe the conditions that will prevail in Forsmark in the future. Hydrology is an important part of the GRASP project, since solute transport within and between different ecosystems is affected to a great extent by the water flows in the area. Accordingly, the Ph. D. project concerning hydrological processes under periglacial climatic conditions is being pursued as a part of GRASP. Hydrological investigations have been underway since 2010 and will continue during the next few years as well.

This hydrological part is aimed at obtaining a better understanding of the water flows within and between different hydrological domains in a landscape dominated by permafrost. The project is focused on the hydrological processes in the active layer, i.e. the uppermost ground layer that thaws each summer on top of the permafrost and where there can be hydrological activity during the summer months. The role of taliks in the overall hydrological water balance in a periglacial system is also one of the main topics being explored in the project. Taliks, which can occur beneath lakes and watercourses in a periglacial landscape, are unfrozen parts of the rock where interaction between deep and shallow groundwater may take place.

The lake in the investigated area on Greenland, Kangerlussuaq, is a talik, according to current indications from measurements. A weather station installed in the catchment area is generating detailed information on local precipitation and evaporation. The meteorological information – in combination with measurements of water pressure, soil water content and ground temperature – constitutes an important basis for calculation and conceptual understanding of the water balance in the area.

Existing conceptual and numerical hydrological models for Forsmark are being used in the project and constitute the basis for developing equivalent models for periglacial climatic conditions. The hydrological models that will be developed are planned to deal only with permafrost conditions without the presence of an ice sheet.

Further model development will take place within the ongoing SR-PSU safety assessment project for the extension of SFR. However, this model development will be reported within the project and not in this RD&D programme. Other activities mainly include minor code development work on MIKE SHE. The possibility of including simulation of hydrogeochemistry in MIKE SHE will be explored.

26.4.2 Hydrogeology in the deep rock

Conclusions in RD&D 2010 and its review

SSM stated that SKB's programme presented within the chapter on groundwater flow was generally appropriate and gave a good overall picture of the issues SKB considers requiring further investigation. The Authority further noted that SKB has taken heed of the comments made on RD&D Programme 2007 regarding testing and verification of hydrogeological models, handling of variability in fracture aperture, supraregional groundwater modelling and connections between groundwater flow in soil and rock. SSM further noted that SKB had dealt with the comments offered in the review of SAR-08 (SKB 2008a).

SSM also took a positive view of the fact that SKB planned to further investigate how DFN models can be generated and parameterized to better reflect actual hydrogeological conditions. The Authority thought that the question of whether flows in DFN models were overestimated as presumed by SKB should be further studied to gain a better understanding of the system. SSM furthermore said that SKB should study modelling of flow through networks of channels in order to determine conceptual uncertainties in flow calculations using fracture network and continuum models.

SSM was further of the opinion that groundwater discharge through the sea bottom should be taken into account by SKB in groundwater modelling.

SSM also found it urgent that SKB realizes its plans to conceptualize flow models during periods of glaciation and permafrost based on results from the GAP.

As in reviews of previous RD&D programmes, SSM wanted SKB to pursue efforts to strengthen the credibility of the hydrogeological models. Examples of important questions are upscaling of parameters from measurement data, transfer of boundary conditions in complex models, and the validity of the hydro-DFN models and the relevance of their results for different users.

Newfound knowledge since RD&D 2010

The preceding RD&D period was strongly influenced by the conclusion of the SR-Site safety assessment project. Thus, most hydrogeological modelling results and reports that later served as a basis for the reporting in SR-Site's final report were presented already in RD&D programme 2010. However, additional newfound knowledge has emerged from supplementary studies that were not published in time for inclusion in RD&D programme 2010.

The main contribution here is a development of the hydro-DFN methodology that was devised late in the site-descriptive modelling of Laxemar (Joyce et al. 2010). New elements in the updated methodology are mainly that single boreholes are simulated (rather than all boreholes being combined) with their actual inclination (rather than a perfectly vertical borehole), that deformation zones are included in the simulation, and that the different fracture sets are calibrated individually (rather than collectively). This new methodology has also served as background material in the planning of a new project for geological and hydrogeological discrete fracture network modelling, see further Section 26.24.1 "DFN" for a presentation of the new project.

During the preceding RD&D period, time and resources were also devoted to summarizing the hydrogeological simulations done within SR-Site. A summarizing report (Selroos and Follin 2010) describes the overall hydrogeological modelling strategy and the model setups used in SR-Site. Further, the report describes the most important results, and how they have been used in other disciplines within the safety assessment. Moreover, the most important hydrogeological analyses and the results from the site-descriptive modelling (SDM-Site) and the safety assessment (SR-Site) have been described in seven scientific articles published in a series in "Hydrogeology Journal".

Great progress has been made in the field of glacial hydrogeology since RD&D programme 2010. Both the GAP project (see Section 19.7) and analyses within SR-Site have contributed to this development. Specifically, the setting of boundary conditions associated with an overlying ice sheet in hydrogeological models has been analyzed (Vidstrand et al. 2012). Large-scale hydrogeological models have also been constructed of the area where GAP is being conducted (Jaquet et al. 2010, 2012). These models have been used to investigate the effect of permafrost distribution in front of (including taliks) and beneath the ice, the effect of different boundary condition assumptions for the ice/rock interface, and the effect of the resolution of the topography in the model. How site-specific data can be incorporated in the models has also been studied. Progress has also been made in transport modelling coupled to glacial flow conditions; this is described in Section 26.24.4 "Integrated modelling – radionuclide transport".

Code development has also been pursued during the period to enable coupled hydrogeological and hydrogeochemical problems to be simulated. The modelling tool ConnectFlow has been further developed. Among other things, the option of including geochemistry and hydrogeochemistry has been added by integrating PHREEQC. This is described in greater detail in Section 26.24.3 "Integrated modelling – hydrogeochemical evolution".

Other code developments have mainly been linked to simulation of surface hydrology, where functionality has been developed for importing various surface features such as watercourses and lakes as CAD objects. This makes it easier to represent important components in the surface hydrological system. Moreover, the Navier-Stokes solver in DarcyTools has undergone further development aimed at handling flow in the surface hydrological system. The other, previously considered possible applications such as injection, matrix flow and subglacial flow have not been further developed since concrete needs have not arisen.

The main code development that has occurred within ConnectFlow is a new methodology for simulating density-driven flow in discrete fracture networks. This is a generalization of the previous methodology, which was used e.g. in SR-Site, where density is time-independent. Furthermore, an alternative methodology has been implemented for describing matrix diffusion based on a finite volume approach that permits the use of variable parameters for different chemical species. Improvements have also been made in the graphical user interface.

New knowledge regarding coupled hydromechanical processes has emerged from non-SKB-funded projects. The effect of mechanical shearing on the distribution of transmissivity and resulting flow in a single fracture has been studied numerically in Vilarrasa et al. (2011). The results indicate that an imposed shear stress results in an anisotropic transmissivity distribution. This in turn causes the flow in the direction perpendicular to the shear stress direction to be higher than the flow in a direction parallel to the shear stress.

Programme

The various initiatives that are planned in the coming programme can be divided into three main categories: Initiatives aimed at achieving greater process understanding; initiatives to integrate hydrogeological calculation tools with other relevant disciplines such as hydrochemistry, geology, rock mechanics and climate studies, and initiatives relating to code maintenance. The various planned initiatives are described below in accordance with this categorization.

Two Ph. D. projects are being conducted to gain a better understanding of important processes and to achieve new knowledge regarding how different types of data can be used in modelling. The first project is concerned with groundwater flow and transport of solutes from a supraregional scale down to a local catchment scale. The purpose is to achieve an internally consistent methodology for simulating groundwater flow and transport over multiple scales, and then to investigate which scales need to be taken into consideration for different applications. This is achieved by describing spatial variability in a consistent manner over all relevant scales. The project builds further on the experience gained by SKB in supraregional groundwater flow in recent years year (Ericsson and Holmén 2010), but may also use other methodologies and other codes than used in previous projects carried out under SKB's auspices. The idea is to study different climatic conditions as well, i.e. both today's climate and future glacial conditions, and relate climatic conditions and scale aspects. How rapid flow transients affect solute transport on different catchment scales will also be studied.

The other Ph. D. project is concerned with how different types of data obtained in connection with underground construction (i.e. construction of tunnels, ramp and shafts for the Spent Fuel Repository) can be utilized in groundwater flow modelling. The channel network code Chan3D, developed at KTH, is being used here. See further Section 26.24.4 "Integrated modelling – radionuclide transport".

A number of projects that are aimed at linking to other disciplines in various ways are planned. The projects with coupled hydrogeological and hydrogeochemical models are continuing. These projects include integration of PHREEQC in ConnectFlow and a coupling of DarcyTools with PFLOTRAN. This is described in greater detail in Section 26.24.3 "Integrated modelling – hydrogeochemical evolution".

As mentioned above, a project is planned for development of a new methodology for geological and hydrogeological discrete fracture network modelling. This project is described in detail in Section 26.24.1 "DFN". Here it can be pointed out that the project has two primary purposes. One is to devise a standard methodology for both geological and hydrogeological fracture network modelling. The other is to devise a methodology for enabling the previously used purely stochastic models to achieve a higher degree of determinism by using locally measured quantities in tunnels and boreholes.

A final report will be produced within the GAP project during the coming RD&D period. But work still remains to be done when it comes to how collected data can be used to gain a better understanding of specifically the role of the subglacial layer for groundwater infiltration under glacial conditions. The subglacial layer comprises the contact between the overlying ice sheet and the rock beneath. By performing modelling with different assumptions regarding the properties and function of this layer, and by comparing simulated and measured data on pressures and flows, a better understanding can hopefully be achieved. In addition, modelling projects are being planned to study permafrost evolution (see Section 19.5 "Permafrost") and the resulting groundwater flow distribution in a realistic catchment area. The site descriptive model for Forsmark that was used in SR-Site may be suitable for this purpose. Questions here are how permafrost and coupled groundwater flow evolve in a transient manner as the climate gradually becomes colder. As an initial study, simulations are being done in a simplified two-dimensional model.

A project is also planned to include a mechanical component in the simulation tool DarcyTools. This will enable coupled hydromechanical processes to be studied. With this version of the code, it would for example be possible to analyze how an ice load affects hydraulic properties, specifically the transmissivity of fractures, and thereby flow, and how tunnel construction might affect hydraulic properties. This development work is being coordinated with the development project concerning conceptual understanding of the hydromechanical properties of the rock mass that is described under "Programme" in Section 26.24.2 "Integrated modelling – thermo-hydro-mechanical evolution".

When it comes to code maintenance and code development, a test is also planned of a new upscaling algorithm for use in DarcyTools. One limitation of the current upscaling algorithm is that it results in isotropic properties in all directions for the calculation cells. If a sufficiently high numerical resolution is not used, the effect is that anisotropy caused by preferential fracture orientation is lost in the upscaling. Alternative methods that preserve the anisotropy will be evaluated and incorporated in DarcyTools if they prove usable.

Another code development project in DarcyTools is to tie together the new functionality for surface hydrological objects with the Navier-Stokes solver. The idea here is to use the Navier-Stokes solver in a separate calculation step for water flow in watercourses, lakes and coastal areas. Separately, the groundwater flow model is used to generate a spatially extended source term along the surface hydrological objects for use in the Navier-Stokes model. The models, which are thus coupled via a source term, must be modified to permit smooth and automatic communication between the different codes.

Some code development is planned for ConnectFlow as well. This mainly involves implementation of the new methodology for fully coupled transient flow in discrete fracture network models. Development is also needed to incorporate matrix diffusion in this type of model, and to handle coupled transient density-driven flow in combined discrete models and continuum models.

26.4.3 Specific issues linked to the Spent Fuel Repository

Conclusions in RD&D 2010 and its review

Specific issues linked to the Spent Fuel Repository were not mentioned in RD&D programme 2010.

Newfound knowledge since RD&D 2010

Within the project on detailed site investigations, SKB has gone through methods that could potentially be used for measurement and modelling in detailed site investigations in order to identify suitable improvements, see Section 14.4.2 "Investigations and modelling". A number of areas in hydrogeological modelling that need improvement have been specified. These are primarily site descriptive modelling and modelling of the near-field of tunnels and deposition holes. In detailed conceptual site modelling it is mainly integration with other disciplines (geology, chemistry, rock mechanics and surface systems) that should be improved, but a general updating of the modelling methodology is also recommended. In modelling of the near-field of tunnels and deposition holes, SKB has identified a need to develop a methodology for judging whether a deposition tunnel or a deposition hole is suitable or not for deposition. The updated methodologies will be documented in manuals for detailed characterization.

SKB has identified a need to follow ongoing or planned development work in the hydrogeological areas that are linked to other disciplines. These areas are: modelling of deformation zones and large fractures, DFN, geochemistry and radionuclide transport.

Programme

In the detailed characterization programme, SKB has identified a need to follow ongoing or planned development work in the hydrogeological areas with regard to how the discipline is linked to other disciplines. As mentioned above, these areas are modelling of large fractures and deformation zones, DFN, geochemistry and radionuclide transport. However, a development project will be carried out in the detailed characterization programme for hydrogeology that entails a link between groundwater flow and geochemistry. This will be done by creating an interface between DarcyTools and the calculation tool PFLOTRAN. PFLOTRAN is a general groundwater flow and transport code that also handles geochemistry. Only the geochemistry functionality will be used in the project in question, however. This development work will give DarcyTools the same functionality as ConnectFlow with respect to the option of simulating coupled hydrogeological-geochemical evolution.

An important question for the Spent Fuel Repository is whether unfavourable positions for deposition holes can be avoided. Geometric criteria (FPC, EFPC) are used in SR-Site, but initial analyses within SR-Site indicated that inflow criteria may be effective to screen deposition holes. Further analyses are planned to investigate whether inflow criteria are applicable, and how they should be defined and applied in practice. To start with, the inflow analyses in SR-Site are planned to be redone with the ConnectFlow code (DarcyTools was used in SR-Site). ConnectFlow is a discrete fracture network model where inflow in single fractures can be simulated more accurately than it was in SR-Site. In a subsequent phase, various hydraulic tests will be simulated in the devised model to see whether unfavourable positions can be identified by means of hydraulic tests. These activities will be coordinated with what is being done in the detailed characterization programme.

26.4.4 Specific issues linked to SFR and SFL

Conclusions in RD&D 2010 and its review

Specific issues linked to SFR and SFL were not mentioned in RD&D programme 2010.

Newfound knowledge since RD&D 2010

During the period 2010–2012, three studies were conducted for the purpose of achieving a better understanding of how measured and modelled geological and hydrogeological conditions in the SFR area affect the groundwater within the regional area for the extension of SFR (PSU):

- Öhman and Follin (2010) present a hydrogeological fracture model (Hydro-DFN) for the rock mass between fracture zones based on a preliminary version of the deformation zone model for PSU (Curtis et al. 2010), collected hydrotest data measured with the Posiva Flow Log, and fracture orientation data from the BIPS borehole camera.
- Öhman et al. (2011a) present a conceptual hydrogeological model for the bedrock based on an update of the deformation zone model (Curtis et al. 2011). The hydraulic parameterization of deformation zones and the Hydro-DFN of the rock mass are based on all available hydrotest data in the area, i.e. both older SFR data and newer PSU data.
- Öhman et al. (2011b) present flow simulations where the degree of detail in the model setup is increased stepwise in order to study how sensitive the groundwater flow is to different interpretations of measured data. The goal of the flow simulations is to evaluate the reliability of the conceptual model for the safety assessment project for the extension of SFR, SR-PSU.

Within the regional model area for PSU, 40 deformation zones with lengths greater than 300 metres have been identified by Curtis et al. (2011). All zones except three have been interpreted as steeply dipping to vertical. Curtis et al. (2011) also presents 31 borehole intervals with properties similar to deformation zones (for example intervals with elevated fracture frequency). These have not been included as deformation zones in the geological model because they cannot be modelled in 3D according to the methodology used, see Curtis et al. (2011), where the intervals in question are designated as PDZs (Possible Deformation Zones).

Öhman et al. (2011a) note that most hydrotest data measured in the PSU project in the area for the extension of SFR coincide with subhorizontal fractures, and that several of the most transmissive subhorizontal fractures correlate well with the aforementioned PDZ intervals. Öhman et al. (2011a) analyze all PDZ intervals from a hydrogeological perspective and note that hydraulic interferences have occasionally been detected between different boreholes and that the interferences can in some cases be interpreted as indicating the existence of subhorizontal structures between different PDZ intervals. Eight of the most centrally located subhorizontal structures within the regional model area have been modelled deterministically by Öhman et al. (2011a) in the conceptual hydrogeological model. The structures are designated SBA1-SBA8 (SBA = Shallow Bedrock Aquifer), and four of them coincide with aforementioned PDZ intervals. In addition to the SBA structures, Öhman et al. (2011a) propose that seven PDZ intervals be modelled by conditional stochastic modelling, and that data that coincide with SBA structures and PDZ intervals not be integrated in the Hydro-DFN model of the rock mass.

Earlier groundwater models developed for the SAFE and SAR-08 safety assessments (Holmén and Stigsson 2001, Holmén 2005) were calibrated against measured inflow to the existing SFR facility. In the SFR extension, the groundwater model has not been calibrated against the inflow to SFR. There are two reasons for this change in methodology:

- The area envisioned for the extension of SFR contains geological structures that are not in direct contact with the existing SFR facility, i.e. structures that make little or no contribution to the measured tunnel inflow to SFR.
- The decline in the inflow to SFR over time (approximately 60 percent decrease in 22 years), in combination with increasing drawdown, indicates a transient change in the hydraulic properties around the open SFR facility. This raises questions regarding how representative measured data (pressure and inflow) are for calibrating the groundwater model in view of the fact that it is water-saturated conditions that are analyzed in the safety assessment.

Instead of a calibration against measured inflow to SFR, Öhman et al. (2011b) performed a sensitivity analysis, see above.

The hydrogeological modelling that was performed within the framework of the SR-PSU safety assessment project during the period in question is included in the supporting material for the application for an extended SFR. A new methodology for modelling future landscape evolution has been used in the biosphere modelling; future soil layer conditions and the situation for future lakes and rivers (including depth and threshold values) have been calculated for different shoreline positions. As a result, a more accurate description is obtained of the surface hydrology, which is particularly advantageous when analyzing shallow repositories. A number of studies have been conducted to find a suitable site and design for an extension of SFR. Moreover, different parameterizations of the hydraulic properties of the rock have been studied to handle uncertainties in the description of the rock.

Programme

As a follow-up of the safety assessment of SFR (SR-PSU), the modelling methodology used will be evaluated and compared with the corresponding methodology used in SR-Site. Different codes were used for groundwater modelling in the different projects (DarcyTools or ConnectFlow). (MIKE SHE was used in both projects). The purpose is to evaluate how the codes worked, make comparative analyses and test runs, and if possible formulate a common strategy for future assessments.

A study of different repository concepts for SFL, including a qualitative assessment of their long-term safety function, is under way. The goal of the study is to choose one or two repository concepts to proceed with, see further under "Current situation" in Chapter 6.

26.4.5 Task Force on Modelling of Groundwater Flow and Transport of Solutes

Conclusions in RD&D 2010 and its review

SSM took a positive view of SKB's plans to study more closely the hydraulic interaction between rock and bentonite at a deposition hole.

Newfound knowledge since RD&D 2010

In the SKB Task Force on Modelling of Groundwater Flow and Transport of Solutes, Task 7 concerning site modelling on different scales with data from flow measurements has been concluded. The conclusions have been published in a number of modelling reports and as articles in scientific publications (for example Frampton 2013).

In Task 8, modelling was linked to the experiment BRIE (Bentonite Rock Interaction Experiment) in the Äspö HRL. The hydraulic interaction between rock and bentonite is being studied in two downscaled deposition holes 30 centimetres in diameter. The resaturation process is at the centre of attention here. The effect of rock stresses on specifically the permeability of the rock and thereby on the function of the repository is being studied with coupled hydro-mechanical models.

Programme

Within Task 7, the modelling reports will be evaluated by an external reviewer.

Task 8 will continue with modelling of BRIE and possibly also modelling of an experiment with a similar problem but on a larger scale. Task 8 will be reported by the modellers and evaluated by an external reviewer.

The work of starting a new modelling task, Task 9, has been initiated. It is possible that it will include modelling of experiments that investigate the transport properties of the rock such as matrix diffusion and sorption. A suitable candidate for the modelling exercise could be the REPRO experiment (Aalto et al. 2009) which is being conducted in Onkalo in Finland.

26.5 Gas flow/dissolution

Conclusions in RD&D 2010 and its review

The discussion of the discipline gas flow/dissolution in RD&D Programme 2010 coincided with what was said about the resaturation process in the discussion of Task 8 in the Task Force on Modelling of Groundwater Flow and Transport of Solutes.

Newfound knowledge since RD&D 2010

The programme regarding gas flow/dissolution coincides with the one described in Section 26.4.5 “Task Force on Modelling of Groundwater Flow and Transport of Solutes” concerning resaturation processes and specifically Task 8. Task 8 uses data from the BRIE experiment at the Äspö HRL. The resaturation results from the experiment are being simulated both within the Task Force on Modelling of Groundwater Flow and Transport of Solutes and within the Task Force on Modelling of Engineered Barriers.

Resaturation of bentonite in a deposition tunnel has been modelled by Svensson (2009), while resaturation of deposition holes has been modelled within Task 8. The modelling results in Task 8 indicate that the time for resaturation differs widely depending on whether the hydraulic connection is discrete through fractures or evenly distributed over a contact surface between two hypothetically homogeneous materials. An analysis of the resaturation time for bentonite answers the question of how quickly the repository goes from the initial state after closure to the saturated state; saturation is usually assumed in groundwater flow calculations for long-term safety.

The laboratory results indicate that the reported model for bentonite is well suited to represent the water uptake process in the field experiment. In situ-results from the field experiment indicate that water uptake takes place primarily via fracture flow. A final assessment of the scope of the matrix flow cannot be made until after excavation of the experiment.

Programme

The BRIE experiment will be concluded in 2013, but the modelling work and analysis of the results will continue within the Task Force on Modelling of Groundwater Flow and Transport of Solutes. Rewetting of both bentonite and rock will be modelled. In addition, the interaction between these processes will be investigated, along with the influence of whether the hydraulic connection is evenly distributed or discrete, i.e. limited to those places where fractures intersect deposition holes.

26.6 Movements in intact rock

Conclusions in RD&D 2010 and its review

RD&D Programme 2010 described a project concerned with the possible spalling effect in canister holes. In an experiment at the Äspö HRL, the confinement effect was investigated in a number boreholes about 500 millimetres in diameter. The field experiment CAPS (Counterforce Applied to

Prevent Spalling) used LECA pellets to simulate the confining effect of unsaturated bentonite pellets. The experiments indicate that the small confining force 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 (Glamheden et al. 2010). A programme had 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 (see for example Jacobsson 2010a, b).

In its review, SSM took a positive view of SKB's continued development work in cooperation with Posiva regarding spalling and stressed that SKB should consider the site-specific situation in Forsmark in this work. SSM further found it encouraging that SKB was prepared for the possibility that the results of SR-Site may lead to the need for further studies. In addition, SSM considered that SKB should initiate research to show that the results of the CAPS experiment can be applied to full-sized deposition holes in the rock conditions that prevail in Forsmark.

SSM was moreover of the opinion that SKB should study the influence of the geometry of the fracture network on spalling and its consequences for the safety assessment. An SSM-funded study suggests that tensile fractures could arise around the canister holes at full bentonite swelling pressure as long as the temperature stresses are not fully developed (Backers and Stephansson 2008). SSM thought that SKB should take these results into account in its continued work. SSM noted that SKB did not present any plans to try to reduce the uncertainties in the rock stress model for Forsmark and contended that SKB should continue its efforts to reduce these uncertainties. The Authority further wanted SKB to follow up the previous initiatives that appear to have yielded promising results for Laxemar (Mas Ivars and Hakami 2005, Hakami 2006, Hakami and Min 2009). Moreover, SSM thought that SKB should study how the rock stress models for the near-field of the Spent Fuel Repository are related to the large-scale stress models that are used to calculate the stresses during a glacial cycle (Lund et al. 2009).

Newfound knowledge since RD&D 2010

SKB has participated in the development work on spalling within the framework of the Posiva Spalling Experiment (POSE) conducted by Posiva. Preliminary results show that fracture propagation is very sensitive to changes in the anisotropy direction of the rock strength (Siren 2011). Friction angle and cohesion are important parameters in fracture initiation, while fracture toughness is not of particularly great importance, although it controls fracture growth once it has been initiated. The final phase of the experiment will be concluded during 2013 (Christiansson et al. 2012).

The buffer's maximum swelling pressure is 15 megapascals (SKB 2011e, Figure 8-2). Analyses of the near-field rock in Forsmark for different assumptions concerning the magnitude and orientation of the in situ stresses show that the tangential stresses after excavation of deposition holes are not less than about 20 megapascals anywhere (Hökmark et al. 2010). This means that there is a margin of about five megapascals against tensile failure in the deposition walls, even if credit is not taken for the thermally induced stress addition. The THM analyses of the near-field rock further show that the thermally induced stress addition grows to about 15 megapascals already during the first month after deposition (Hökmark et al. 2010). Meanwhile, THM analyses of the buffer-rock system done for different assumptions concerning the hydraulic conductivity of the rock, occurrence of intersecting water-bearing fractures etc show that it takes at least seven years or so to achieve full water saturation (Åkesson et al. 2010). The much faster growth of the thermally induced stresses thus means that the margin to tensile failure will be much greater than five megapascals during the entire thermal phase. After about 10,000 years, the thermally induced stresses are down to about five megapascals, which means that the margin to tensile failure in the depositionhole walls has then declined to about 10 megapascals.

Programme

The work of explaining, managing and damping thermally induced spalling will continue, since it affects one of the safety functions. Any additional calculation work will, however, be of limited scope, since thermally induced spalling, according to the conclusions in SR-Site (SKB 2011e, Section 15.7.4), is not critical for long-term safety.

SKB has initiated a Ph. D. project entitled "Stress-induced spalling, management and measures". The project involves development of numerical and laboratory methods for assessing the potential for spalling and for gaining a better understanding of fracture propagation in crystalline rock. Methods should be developed to characterize sources of uncertainty by statistical means and estimate how sensitive the results are to these uncertainties. Uncertainty assessments should take into account the natural variability of the rock and sources of error in testing methods. The results of the extensive rock mechanical testing that has been done mainly in the site investigations, as well as collaboration with the Commission on Rock Spalling at the International Society for Rock Mechanics (ISRM), will constitute the basis of the work. Study of alternative testing methods for determining the stress level where spalling can occur will be a principal activity. Different rock types should be tested, for example from Äspö, Forsmark and Onkalo. Collaboration with the Posiva Spalling Experiment (POSE) is planned.

26.7 Thermal movement

Conclusions in RD&D 2010 and its review

It was reported in RD&D Programme 2010 that the analytical thermomechanical solution for the Spent Fuel Repository had been further developed so that cases with several different deposition areas can be analyzed. The original equations have been coded to user-friendly calculation sheets and have been used in sensitivity analyses and analyses of how the deposition sequence can affect the near-field evolution (Hökmark et al. 2010).

Preliminary results existed for the thermomechanical evolution in the Prototype Repository at the Äspö HRL. The analysis has now been completed, see below. Further, a literature study about the pressure dependence of the thermal expansion coefficient of typical rock types in the repository rock had been initiated.

SSM said that SKB should conduct model studies concerning propagation of existing fractures and initiation of new fractures in the surrounding rock due to thermal stresses around the deposition holes and the deposition tunnels. For example, the SKI-funded study by Rutqvist and Tsang (2008) indicates that there is a risk of continuous tensile failure in the walls of the deposition tunnels during the thermal phase. SSM further considered that SKB should make additional efforts to reduce the uncertainties in the calculations of the thermal evolution in the near-field.

Newfound knowledge since RD&D 2010

The analysis of the thermomechanical evolution in the Prototype Repository at the Äspö HRL has now been completed. It shows that the method, or hypothesis, that is applied in the modelling of the thermomechanical evolution in Forsmark gives credible results. For example, it shows that temperatures measured at different distances and in different directions from the Prototype Repository's heaters can be accurately reproduced with the aid of the thermal relationships incorporated in the numerical tool 3DEC. The calculated stresses around the deposition holes, the tangential stresses, lie just below the spalling strength determined in the APSE and CAPS experiments. On inspection of the hole walls after the conclusion of the experiment, it was also found that no spalling had occurred, which shows that the calculations had not significantly underestimated the stresses (see also Section 13.11 "Prototype Repository", subsection "Results").

Comparison with acoustic emission data confirms the calculated location of the stress concentrations during the heating phase. Even though these observations support the calculation results, they do not suffice as a direct quantitative verification of the stress calculations. The direct stress measurements made in several positions around the outer two deposition holes confirm the magnitude of the stresses during the excavation phase, but have proved to be unreliable during the following thermal phase. The instruments installed for deformation measurements have not worked as intended and have not been able to be used for comparison with the calculated deformations. In summary, the analysis of the Prototype Repository has confirmed the methodology for thermomechanical modelling that is applied for Forsmark in principle and when it comes to the order of magnitude of the stresses, but not when it comes to exact stresses and deformations.

The possible pressure dependence of the thermal expansion coefficient of typical repository rock types has been determined via a literature study. The purpose has been 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. Briefly, the literature study shows that such underestimation, if it is a real possibility at all, would be due to the fact that the mineral grains, during testing of unloaded specimens, expand into existing microfractures instead of fully contributing to the volume expansion of the specimen. In tests of loaded specimens, the microporosity would be blocked in advance so that the volume expansion of the grains would contribute more to expansion of the specimen. The idea that this process is possible is based on observed differences between theoretically calculated and actually measured coefficients of volume expansion. In cases where the measured coefficient has been smaller than the calculated one, it turns out that the microporosity of the rock type in question has been substantially greater than for the rock types that dominate in Forsmark. Furthermore, these specimens, compared with the specimens that have shown better agreement between calculated and measured volume expansion, have a pronounced anisotropy in their microfracture structure. There is therefore no evidence that this process, i.e. that the mineral grains in unloaded specimens expand at the expense of the microporosity, can have led to an underestimation of the coefficients of volume expansion that will apply when the constrained rock in Forsmark is heated. It is rather likely that the reported values are slightly overestimated. There is a small, but fundamental. pressure dependence at the crystal lattice level and, additionally, measurements have been performed for temperature increases up to 80°C, which is about 20°C above the maximum temperatures in the repository rock.

General development of codes for modelling has been initiated in order to gain a better understanding of large-scale thermomechanical processes, see further Section 26.24.2 “Integrated modelling – thermo-hydro-mechanical evolution”. However, current knowledge is judged to be sufficient to bound the importance of the thermally induced movement for long-term safety.

Programme

The thermal stresses could, at least theoretically, cause loads on deformation zones in the repository rock that could possibly trigger seismic movements. Calculations that quantify the possible effects of such thermally induced earthquakes on the Spent Fuel Repository will be performed. The methodology is the same as is also used to quantify effects of postglacial earthquakes.

The risk of under/overestimating the coefficient of thermal expansion will be further studied.

26.8 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 impact and effects of glacially induced faults.

Conclusions in RD&D 2010 and its review

RD&D Programme 2010 described the work of developing and analyzing models of the evolution of stability and stress in the Earth’s crust during a glacial cycle. The work had mainly been done within the framework of the SR-Site safety assessment. The overall goal was to determine boundary conditions in a regional perspective for the evolution of rock stresses in Laxemar and Forsmark during a glacial cycle. The stability of faults in Laxemar and Forsmark had been analyzed under the assumption of a seismogenic depth of 9.5 kilometres (Lund et al. 2009).

A study had been conducted of the potential for hydraulic jacking in conjunction with a glaciation (Lönnqvist and Hökmark 2010). The analysis concluded that hydraulic jacking cannot reach depths below about 250 metres. Furthermore, an expanded numerical analysis was presented of the effects of earthquakes on fractures in the repository rock (Munier and Hökmark 2004, Fälth and Hökmark 2006).

The transmissivity changes in near-field fractures during construction, thermal loading and a glacial cycle had evolved according to thermomechanical near-field models (Hökmark et al. 2006). Stress data obtained from new three-dimensional analyses of the latest glaciation are used for the stress evolution during the glacial cycle (Lund et al. 2009). A more realistic stress-transmissivity model had been constructed using parameter values from fractures in Forsmark (Glamheden et al. 2008) and based on a literature study (Fransson 2009).

A stability analysis of a plane with a system of tunnels with dimensions similar to those of the deposition tunnels in the Spent Fuel Repository had been carried out (Lönnqvist et al. 2010). The analysis had been done with the two-dimensional distinct element code UDEC and for a number of different load cases. In summary, the analysis showed that the repository will not act as a plane of weakness.

SSM took a positive view of SKB's plans for in-depth studies of the magnitude, variability and uncertainty of the rock stresses at different depths and scales that take into account the thermal phase, glacial cycles and earthquakes. SSM noted that SKB had not explained how most of the viewpoints voiced by the regulatory authorities regarding earthquakes in the review of RD&D Programme 2007 had been addressed. Regarding the impact of earthquakes on the final repository, SKB refers, according to SSM, to SR-Site without further explanation of the purpose of the studies. The Authority therefore shared the view of the Swedish Society for Nature Conservation and the Swedish NGO Office for Nuclear Waste Review that the issues relating to risks of earthquakes associated with magnitude and frequency are not addressed in the research programme. Since the Authority could not evaluate the background material for SR-Site either, the recommendations for studies presented in the review of RD&D programme 2007 were reiterated. SSM thought that SKB should conduct additional studies regarding material properties such as strength and stiffness of large fractures, fracture zones and faults. SSM shared the opinion of Uppsala University that a survey of the regional structures should be performed by SKB.

SSM expressed a wish that SKB would study the safety-related importance of repeated earthquakes of small magnitudes as well as earthquakes of magnitude greater than six. Moreover, SSM thought that SKB should present more studies regarding respect distance. SSM also noted that SKB does not refer to relevant studies that have been done, for example concerning work with criteria for choice of deposition positions (Munier 2010). SSM was of the opinion that SKB should explore the probability that fractures, fracture zones or faults situated outside of the Forsmark site will grow and penetrate into the final repository during an earthquake or the thermal phase. In this context, the consequence of heterogeneity in fault properties and stresses should also be taken into consideration. SSM said that SKB should study more closely the possibility of distinguishing between stable and unstable faults from an earthquake point of view. Beyond this, the Authority wanted SKB to investigate whether earthquakes of importance to the safety of the repository can be triggered by stresses that arise in the thermal phase.

Newfound knowledge since RD&D 2010

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 tunnel scale, see Section 14.4 "Detailed characterization". To further improve the state of knowledge concerning future earthquakes and to better quantify remaining uncertainties, research has been conducted within structural geology, seismology and tectonics.

Many of the viewpoints and questions expressed by SSM in connection with the review of RD&D programme 2010 were addressed in SR-Site. For example, multiple earthquakes were addressed in the Geosphere Process Report (SKB 2010r, Section 4.3). The methodology of calculating the number of critically positioned canisters that was developed by Munier (2010) was applied in SR-Site and is described in the main report (SKB 2011e, Section 10.4.5), along with the idea of distinguishing stable from unstable zones, which was described in detail in Fälth et al. (2010). See also "Programme" below.

The Swedish National Seismic Network (SNSN) has, since the start of the automatic system in 2002, registered, localized and calculated focal mechanisms for more than 5,000 earthquakes, see Figure 26-1. The earthquakes have magnitudes of between about –2 and 5.3 (Kaliningrad). The last large expansion of the network, in southwestern Sweden, was instrumented in 2008, and since then SNSN has relatively good coverage in the seismically most active areas of the country. There are 66 stations today. SKB has financed a considerable part of the expansion of the national seismic network. Since 2008, SNSN continuously gathers data in real time from all stations, which means that the quantity of data for analysis has increased significantly compared with previous years, when only data segments from detected events were collected. The new, refined seismic network has fundamentally improved the potential for interpretation of earthquake activity in Sweden.

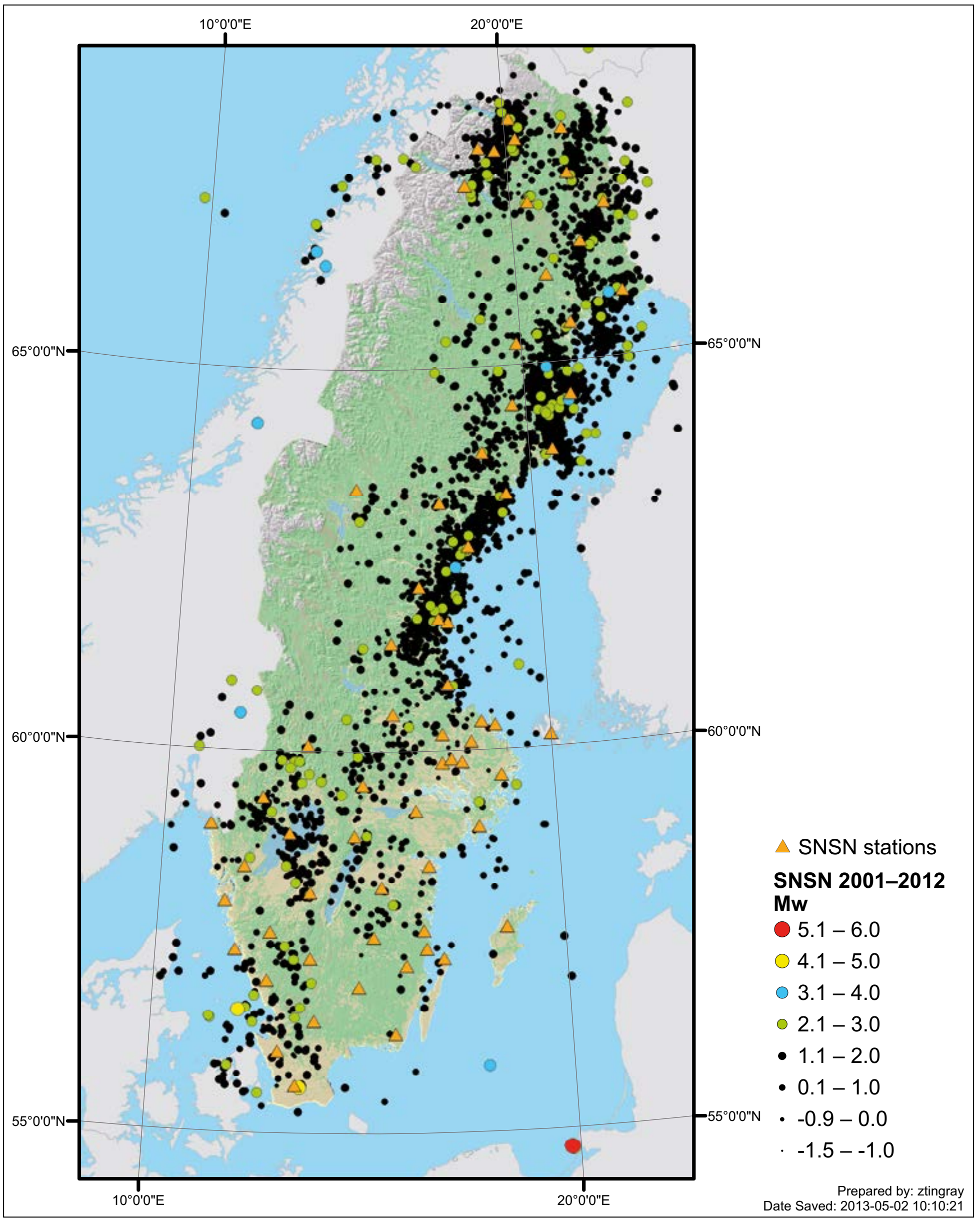


Figure 26-1. *Earthquakes registered by the Swedish National Seismic Network (SNSN) during the period 2001–2012 (Mw = moment magnitude scale). Data from SNSN (for example Bödvarsson 2012).*

SKB has recently initiated a tomographic analysis (see for example Tryggvason and Linde 2006), which yields a three-dimensional velocity model, and structure, for Sweden and new locations for the earthquakes (see further under "Programme").

Estimates of the stresses that caused the great postglacial faults in northern Fennocandia require knowledge of the geometry of the earthquake planes. Investigations have been conducted at the Pärvie fault of microearthquakes to see whether they define a major earthquake plane below a depth of three kilometres. Seismicity at Pärvie is relative low, however, which makes the analysis difficult (Lindblom 2011). 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. During 2012, a summarizing excursion was made in northern Sweden on the initiative of SKB and SGU (an Excursion Report is planned).

Deformation measurements using GPS technology were initiated in the Forsmark area in 2005 and continue today. 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. Data have been analyzed in campaigns, and work is under way to summarize the experience gained with this technique (Ekman et al. 2013).

Programme

A large-scale integrated analysis of data from the Swedish national seismic network has not been done yet. There is now enough data to permit an evaluation based on simultaneous analysis of large numbers of earthquakes. Sweden's crust is not homogeneous, so that the seismic velocities vary widely in Sweden. It is probable that the earthquakes differ between different regions with regard to depth distribution and mechanisms. An integrated analysis of the Swedish earthquakes, including the seismic velocity distribution in the crust, could lead to a better understanding of stress build-up in the Swedish bedrock. SKB intends to carry out a number of sub-projects connected to the national seismic network as follows:

Three-dimensional structure and relocation of Swedish earthquakes: Both earthquakes and seismic events caused by explosions are used for velocity determination, since the latter have a more homogeneous geographic distribution over Sweden. The tomographic analysis (for example Tryggvason and Linde 2006) will yield a three-dimensional velocity model, and structure, for Sweden as well as new locations for the earthquakes. Major earthquakes with good station coverage are used for supplementary studies of depth with the aid of depth phases. Earthquakes in areas with many earthquakes are relocated by means of relative methods for the highest possible resolution of structures. The earthquakes are compared with geological structures and geophysical data in the form of magnetic and gravity maps.

Swedish focal mechanisms and earthquake-derived stresses in Sweden: New focal mechanisms are calculated for the relocated earthquakes (see above). Earthquakes with determined mechanisms are included in calculations of the stresses that caused the earthquakes, see for example Rögnvaldsson and Slunga (1993) and Keiding et al. (2009). Here as well, the earthquake mechanisms are compared with geological structures and geophysical data. The stresses are related to other data, for example measurements in boreholes and geological indicators. The stresses are also compared with models of the stress state in the inner parts of tectonic plates. The study is expected to yield a significant increase in the number of well-determined observations of stresses at depth in the Swedish crust.

Altogether, the above projects are expected to provide greater knowledge concerning active structures in the Earth's crust, the earthquake-generating mechanisms and the stress field in the Swedish crust.

Study of postglacial earthquakes in the Skellefteå area: The geometry and earthquake activity of one of the Swedish glacially induced faults (Pärvie) has been studied in detail (for example Lindblom 2011). Unfortunately, earthquake activity during the project's measurement period did not permit any definite conclusions to be drawn about the geometry of the whole fault. However, there are two glacially induced faults in the Skellefteå area, and the area exhibits the highest earthquake activity in Sweden, see Figure 26-1. Reflection seismic studies (Juhlin and Lund 2011) show that the Burträsk Fault dips steeply and may have a parallel fault at depth. The results of the earthquake analysis of the Pärvie Fault (Lindblom 2011) show, however, that more stations are required so that as many of the minor earthquakes as possible can be analyzed. Six temporary and four permanent seismic stations are now (2013) established in the area. The project is expected to yield in-depth knowledge of glacial faults, their geometry and their current activity and enable better models to be made of the conditions that prevailed when the great earthquakes occurred based on current stress conditions.

Method development for aftershock studies: The great earthquakes on Iceland in 2000 and 2008, with magnitudes of between 6.3 and 6.5, caused many aftershocks. The Icelandic seismic network recorded nearly 1,800 aftershocks during the first 24 hours after the earthquake of 17 June 2000. The aftershocks then continued for several years. The other earthquakes with magnitudes greater than 6 have caused similar sequences. The quality of the determination of the aftershock's focal mechanisms varies with the number of stations that registered the shocks. These data can be used to study the location and intensity of the earthquake activity (see for example Lindman et al. 2010) as well as which fractures have been activated and the size distribution of the displacements. The goal is to initiate a method study with aftershock analysis for data from Iceland in order to develop earthquake modelling (see for example Fälth et al. 2010).

Local seismic network in Forsmark: As a part of SKB's research on natural and induced seismicity, SKB is preparing the installation of a local microseismic network in Forsmark (see Section 14.4.2 "Investigations and modelling").

Further development and earthquake modelling: The modelling technique described in Fälth et al. (2010) will be applied to Forsmark with site-specific in situ stresses, glacial stresses and site-specific structural geology, i.e. the geometry of deformation zones and fractures in the deposition areas. A Ph. D. project has also been initiated with the general objective of refining and validating the methodology for numerical simulation of earthquakes that is described in Fälth et al. (2010) and developing the coupling between numerical modelling and seismology/geophysical science. Examples of points/questions that will be included are: the importance of the width (thickness, with or without damage zone) and geometry of the fault; the importance of the variation of the strength properties over the fault plane; techniques to simulate the spontaneous evolution of the failure (in contrast to the programmed progression that has been used up to now); the importance of how the edges of the fault against the host rock are modelled and requirements on (and scope of) the empirical comparison base (see also aftershock studies described above).

Shear movements of major fractures and possible propagation: In the numerical studies and analytical observations made for the SR-Site safety assessment (Fälth et al. 2010, Hökmark et al. 2010), the maximum fracture slip is directly proportional to the size (radius) of the fracture. The slip (shear displacement) of a large fracture is therefore by definition large. The consequences of the pessimistic assumptions that have been made in the assessment entail an excessive loss of deposition holes and a less efficient use of the available disposal volume. Furthermore, the number of canisters that may be damaged in conjunction with earthquakes is overestimated. To reduce the pessimism in our predictions, a Ph. D. project has been initiated with the goal of gaining a better understanding of how a more realistic description of large rock fractures – with regard to the scale dependence and distribution of the strength properties over the fracture surface and the geometry of the fracture surface – affects the shear displacements in the models. Another question is whether the stress concentrations that are built up at the edges of the fracture during shearing can lead to propagation of the fracture, and how this affects the total shear displacement.

Different initiatives are under way in Sweden within the Swedish Deep Drilling Programme (SDDP). The Swedish Research Council has granted funds for the purchase of a drilling rig able to perform different types of scientific investigations of conditions down to about three kilometres in the Fennocandian Shield. The drilling rig has now been purchased and is being administered by Lund University. 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. For example, drilling in the Åre district within the framework of the COSC project (Collisional Orogeny in the Scandinavian Caledonides, SDDP 2013a) is planned to be initiated in 2013. SKB is involved in relevant aspects of the SDDP pertaining to general geoscientific knowledge accumulation regarding conditions in the Swedish bedrock. Furthermore, SKB has become involved in the preparations for drilling investigation holes through glacially induced faults within the framework of the DAFNE project (SDDP 2013b).

Lidar: The National Land Survey of Sweden has since 2009 been laser-scanning Sweden for the purpose of constructing by 2015 a nationwide digital elevation model with a mean error in elevation better than 0.5 metre for a 2 metre grid (see e.g. Petersen and Rost 2011). As this RD&D programme is being written, large parts of the country have been scanned and digital elevation models have been constructed for about 75 percent of Sweden's surface area (Lysell 2013). Compared with previous techniques for construction of digital elevation models, laser scanning is superior in detecting topographical lineaments. Preliminary studies have already shown that the previously mapped glacially induced faults exhibit a more complex geometry than previously known. Moreover, a large number of new topographical lineaments have been detected. It is possible that some of these are expressions of glacially induced fault tectonics, and SKB therefore intends to closely follow the developments in this field as a part of its ongoing work with seismology linked to long-term safety.

26.9 Fracturing

Conclusions in RD&D 2010 and its review

In RD&D Programme 2010, SKB described its activities concerning the properties and safety-related aspects of the excavation-damaged zone (EDZ). SKB's involvement had mainly been within the framework of the international programme DECOVALEX 2011 and a targeted programme with several sub-projects called ZUSE (Swedish acronym for "Mechanical and Hydraulic Properties of the Excavation-Damaged Zone") (Bäckblom 2008, Olsson et al. 2008, 2009, Ericsson et al. 2009, Hudson et al. 2009, Christiansson et al. 2009, Neretnieks and Andersson 2009, Glamheden and Hökmark 2010).

SSM took a positive view of SKB's plans to evaluate the experiments in the parts of the Prototype Repository that were planned to be opened. SSM expressed additional viewpoints regarding the excavation of long-term experiments. SSM's viewpoints on the continued work with the excavation-damaged zone can be found in Section 14.5.2 "Rock excavation". SSM also offered viewpoints on the need for further studies of fracturing.

Newfound knowledge since RD&D 2010

Measurement data (acoustic emission data, stress measurements, thermal data) from the Prototype Repository in the Äspö HRL, which was opened in 2011, have been evaluated along with results from thermomechanical calculations to investigate the extent to which fracture propagation can occur around a deposition hole. The conclusion is that based on the calculations results, no fracturing is expected to occur. The calculated movements along existing fractures are small and primarily elastic and do therefore not give rise to critical stress concentrations in the fracture tips. The fact that the movements are small is in turn due partly to the fact that the fractures are small, and partly to the fact that the thermal stresses are only significant on parts of the fracture plane situated close to the tunnel and the canister holes. The acoustic emissions that cannot be linked directly to stress concentrations in the walls of deposition holes do not appear to be associated with any fracture plane. Hence, no observations contradict the calculation results (see also Section 13.11 "Prototype Repository", subsection "Results").

Programme

A research programme has been initiated in collaboration with the University of Alberta to develop a synthetic rock mass for crystalline rock types, and based on this synthetic rock mass (SRM) to model strength properties. The criteria for using the SRM methodology to predict the long-term strength of the rock mass within a timeframe of a million years will be studied. Two alternatives will be investigated for simulating crystalline rock types: (i) spherical grains and (ii) polyhedral grains (Voronoi tessellation). The research will develop a strategy for generating polyhedral grains, which resemble the grain size distribution and the statistical property distribution in intact rock and its grain boundaries. The programme will study how best to tie together the grains in a logical manner to be able to imitate the microstructure of the rock. The programme also aims to determine the influence of the properties of the material associated with grains and grain joints on the behaviour of the intact rock.

Preliminary results based on modelling of the Äspö Pillar Stability Experiment (Lan et al. 2013) showed a promising possibility to shed light on the importance of heterogeneity on the mineral grain scale on the failure process. The two-dimensional modelling of the Pillar Stability Experiment suggests that the initiation of failure in crystalline rock begins with microstructural failures (on the grain scale) and that the heterogeneity on that scale is of importance for the failure process, which is initiated to a high degree as tensile failure between the mineral grains. The induced fractures lead to a redistribution of the stress field on the grain scale as the failure propagates. One of the questions intended to be studied more closely is the failure criteria that apply to grain boundaries and interfaces. Brittle failures cannot be represented by traditional Mohr-Coulomb shear failure criteria. The project is also studying whether this also applies to the small grains used in a GBM (Grain-Based Model) concept. Another question that must be resolved is the effect of grain size distribution on the GBM and handling of the upscaling that is needed to obtain realistically manageable models.

The intention is also to use the synthetic rock mass to study how the GSI (Geological Strength Index) method can be linked to typical fracture frequencies (based on DFN models) in crystalline rock. See Section 14.5.2 "Rock excavation" for programme concerning EDZ.

26.10 Time-dependent deformations

Conclusions in RD&D 2010 and its review

RD&D Programme 2010 presented a coordinated study dealing with the behaviour of microfractures, subcritical fracturing and creep (Damjanac and Fairhurst 2010). The investigative work was 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. One conclusion of the article was that a stress threshold exists for crystalline rock types (40–60 percent of the uniaxial compressive strength). Another conclusion was that an exponential extrapolation of the results of short-term creep tests shows a time-independent strength equivalent to a driving stress ratio of about 0.45. This means that a linear extrapolation to a final zero strength is unwarranted (Potyondy 2007).

SSM took a positive view of the fact that SKB has studied very long-term processes for the rock's strength evolution. However, SSM said that SKB should also study dynamic processes in fractures and faults that are activated during very short periods of time and where the strength of the material is exceeded, for example in connection with earthquakes. In this context the influence of temperature should also be taken into account. SSM was further of the opinion that SKB should carry out studies confirming that time-dependent fracture growth does not affect stability and permeability in the final repository's near-field during the different repository phases. In such studies, SKB should take into account stress corrosion cracking (SCC) in all loading cases (tension, shearing, tearing) since fracture growth takes place not only under tension, but also under shearing or tearing at high confining pressure (Backers 2005, Backers and Stephansson 2012).

Newfound knowledge since RD&D 2010

For development of transient rock mechanics modelling, see Section 26.24.2. "Integrated modelling – thermo-hydro-mechanical evolution".

Programme

For development of transient rock mechanics modelling, see Section 26.24.2. "Integrated modelling – thermo-hydro-mechanical evolution". Some of these issues are also dealt with in Section 26.8 "Reactivation – movement along existing fractures".

26.11 Advection/mixing – hydrogeochemistry

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 26.12 deals with the importance of advection and dispersion for radionuclide transport.

Conclusions in RD&D 2010 and its review

RD&D Programme 2010 noted that the code M3 had been finally documented. The mixing model had been used in the site descriptive models and advection calculations had been applied in SR-Can. There were also plans to begin an improvement of the methodology for studying hydrochemical changes near a tunnel in the Äspö HRL.

SSM found studies in open tunnels to be urgent, especially for studying the possible presence of oxygen in unsaturated zones in the near-tunnel rock.

Newfound knowledge since RD&D 2010

Models that include advection and mixing, coupled with chemical reactions, have been used in the SR-Site safety assessment (Salas et al. 2010). In the consultation responses received by SSM in connection with its review, external reviewers have however commented that the scope and the uncertainties of the chemical model have not been sufficiently well studied (Bath 2012, McMurry and Bertetti 2012).

Hydrochemical changes caused by a tunnel have been evaluated. The measurements, which span over a 25-year period from both SFR (Nilsson A-C et al. 2010, 2011) and the Äspö HRL (Mathurin et al. 2012), have revealed great differences between the sites. For example, the penetration of modern Baltic Sea water differs depending on the orientation and properties of the water-bearing structures.

Development of a methodology for so-called "passive sampling" of dissolved metals has been initiated and is being applied in the Äspö HRL. The principle is based on adsorption of dissolved metals on a gel in a pressurized test chamber. The focus in the sampling is on rare earth metals to permit comparisons of the analogues europium and gadolinium with the actinides americium(III) and curium(III).

Equipment to conduct online measurements of the parameters pH, Eh, temperature, electrical conductivity and dissolved oxygen is being developed at the Äspö HRL. The system will be able to record changes in these parameters continuously over time and will be computer-monitored. The results of the measurements are expected to furnish important information and input to design of equipment for groundwater sampling in packered boreholes underground during the construction of the Spent Fuel Repository.

Programme

The monitoring programmes in Forsmark and Äspö will continue. The equipment for online measurements will continue to be evaluated and developed at the Äspö HRL. Planning of the detailed characterization of Forsmark is described in Section 14.4 "Detailed characterization". Development of numerical models is described below in Section 26.24.3 "Integrated modelling – hydrogeochemical evolution".

26.12 Advection/dispersion – radionuclide transport

SKB's research on advection and dispersion is mainly concerned with channelling in single fractures and fracture networks, as well as other mechanisms that can contribute to large-scale dispersion of a contamination plume.

Conclusions in RD&D 2010 and its review

SSM found it positive that SKB is continuing to work for a better understanding and handling of channelling. Similarly, the Authority found it positive that SKB wants to examine and assess the use of the porosity concept within different areas.

Newfound knowledge since RD&D 2010

A comprehensive and fundamental modelling effort has been made to gain a better understanding of channelling in single fractures. One of the great challenges in simulations of this type is to solve the flow problem with high accuracy, since great differences exist in aperture between different parts of the fracture plane. A new methodology, based on so-called Fup basis functions, has been used to solve the flow problem with high accuracy. Fup basis functions is a numerical methodology for describing different signals, functions, materials and/or solutions to differential equations with an adaptive multi-resolution of all space and time scales with given accuracy. The two first publications will come out during 2013 and present the results for different assumptions of aperture distribution and relationships between aperture and transmissivity. The results indicate that higher aperture variability can lead to either unaffected, lower or higher flow-related transport resistance (F-factor), depending on whether the underlying aperture variability is multi-Gaussian, strongly connected non-Gaussian (high connection between areas with large aperture), or unconnected non-Gaussian (high connection between areas with small aperture). This result indicates the difficulties of finding a general correction factor for channelling. A statistical method for defining a reduction factor for a given aperture variability is proposed in one of the coming publications.

It is noted that the planned application according to RD&D Programme 2010 with the Navier-Stokes solver in DarcyTools has not been used in this project, since the Fup methodology proved to be a better alternative. Another study has been done where advective transport and dispersion in conductivity fields that deviate from a multi-Gaussian normal distribution have been studied with the aid of semi-analytical transport modelling (Fiori et al. 2010). The study indicates that dispersion increases if the spatial correlation in low-permeability zones increases, while dispersion decreases if the correlation increases in high-permeability zones. This result indicates that structural-geological differences have a great impact on resulting transport characteristics.

An SKB-independent research study (Larsson 2012) concerning channelling in fractures and fracture networks has been conducted as a Ph. D. project at Uppsala University. The results arrived at by Larsson (2012) support the results summarized above that will be presented during 2013. However, Larsson (2012) only investigates a subset of the different variability structures reported in the SKB study, but applies the presented methodology not only to single fractures but also to fracture networks.

RD&D Programme 2010 discussed a possible porosity project aimed at sorting out different porosity concepts and definitions. This project has been delayed and is now planned to be carried out during the coming three-year period or possibly during the subsequent three-year period.

Programme

The numerical simulations with the Fup technique will continue. During the coming RD&D period, a project equivalent to the one presented above will be conducted, but instead of individual particle trajectories being analyzed, a streamtube approach will be used. This study thereby provides supplementary information on how the flow-related transport resistance should be formulated and possibly be reduced in safety assessment applications where groundwater flow is typically simulated in fractures with a homogeneous aperture.

A study will be carried out to determine whether channelling, specifically channel width and frequency, can be estimated on tunnel walls. The hypothesis is that the temperature of the apparently dry surface can be estimated by means of infrared photography. Channels that flow out on the tunnel wall, with flows that are virtually invisible to the naked eye since the water evaporates, will lead to a decline in temperature on the wall. From the temperature data, it may then be possible to identify how large the cooled areas are, and how many channels there are per unit area. This study is linked to and can furnish important information for the DFN-R project described in Section 26.24.1 “DFN”.

26.13 Diffusion – groundwater chemistry

This section is about the effects of molecular diffusion of groundwater components. The importance of this process for nuclide transport is dealt with in the next section. Diffusion of the groundwater’s gas components is discussed in Section 26.21 “Gas formation/dissolution”.

The interaction between the matrix pore water 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 2010 and its review

The investigations done on drill cores from Forsmark and Laxemar showed that components of waters with widely differing ages and origins are represented in the matrix: glacial meltwaters, post-glacial sea water and meteoric water from a warmer climate from the time before the latest glaciation. The results also showed that matrix diffusion takes place over distances of several tens of metres.

There was nothing in the results to indicate that the matrix pore water might have extremely high salinities, which, if this were the case, could have an adverse effect on the properties of structural or backfill materials.

Newfound knowledge since RD&D 2010

Improvements have been made in the analytical quality of the diffusion experiments for both conservative (bromine, chlorine, delta-deuterium and delta-oxygen-18) and non-conservative components (with corrections for possible reactions with minerals). This strengthens confidence in the use of e.g. magnesium as an indicator of water of marine origin.

The diffusive isotope equilibrium technique has been used to analyze the stable isotopes in the pore water, and the results agree with conventional knowledge. At present, the possibility cannot be ruled out that these results are influenced by fractionation effects and/or that not all moisture or all water has been completely extracted from the rock specimen. Reliable values for stable isotopes are of importance in determining the origin of the pore water. Further analytical studies are required to increase confidence in the interpretation of these pore waters.

Programme

Laboratory experiments (for example anion exclusion or studies of through-electromigration) are planned to permit comparison with the results of out-diffusion experiments that have been performed with chloride ions and comparison of various methods for pore water extraction. The possibility of performing these experiments jointly with Posiva will be evaluated. To increase confidence, analyses are planned to be done at several laboratories. See also the programme described in the next section, 26.14.

26.14 Diffusion – radionuclide transport

Conclusions in RD&D 2010 and its review

Prior to RD&D Programme 2010, studies had been conducted to evaluate uncertainties associated with the electrical resistivity method for measurement of the formation factor. The long-term diffusion experiment LTDE-SD had also been carried out at repository depth to determine diffusion and sorption properties in situ. Several small test cores had been taken and penetration depths had been measured.

SSM was in favour of the fact that SKB has conducted a study to evaluate the uncertainties associated with the electrical resistivity method for measurement of the formation factor. However, SSM commented on the absence of formation factors below 10^{-5} in the evaluation.

Newfound knowledge since RD&D 2010

In the site investigations, measurements of matrix diffusivity have been performed both under atmospheric pressure in the laboratory and in situ by means of electrical methods. A statistical analysis of the data suggests that samples taken to the laboratory are disturbed (SKB 2010q, r). The disturbance is presumably due to the fact that the samples are taken from their natural confined state, and that drilling in addition to other sample treatments cause mechanical disturbance. On average, matrix diffusivities obtained in the laboratory are one order of magnitude higher than those obtained in situ. This suggests that up to 90 percent of the transport capacity seen in the laboratory may be induced, which would mean that laboratory measurements are not suitable for investigating natural variability (but they are still invaluable for investigating processes). This in turn imposes stricter requirements on in situ measurements, which could entail that further validation and method development are necessary.

This also requires better evaluation of processes associated with an electrical potential gradient, but not necessarily a concentration gradient. Surface conductivity is such a process, as well as electrical frequency dependence (Löfgren et al. 2009). Anion exclusion and other processes could possibly be affected differently by an electrical field than by a concentration field. The electrical in situ measurements have also been used to justify the assumption of large-scale pore connectivity (SKB 2010q, r). They have further been used in a new method to evaluate the volumetric fracture aperture of flowing fractures associated with PFL (Posiva Flow Log) anomalies (SKB 2010q). Recently, an evaluation was done of the in situ-matrix diffusivity of the rock in the volume surrounding SFR, similar to the ones done previously in connection with the site investigations in Forsmark and Oskarshamn.

During the preceding period, the diffusion and sorption experiment in LTDE-SD was formally concluded and reported (Widestrand 2010a, b, Nilsson K et al. 2010). An interesting observation is that the penetration profiles cannot be explained with simple one-dimensional diffusion models. It remains to evaluate whether this is an artefact of the experimental setup or indicates an inherent property in the rock. It has been suggested that a better fit can be achieved if the rock matrix is modelled as a heterogeneous medium. Such modelling would require input data in the form of an increased parameterization of the rock matrix. Data and findings from the ongoing Finnish programme REPRO (Rock Matrix Retention Properties, Aalto et al. 2009) may be useful for modelling in, for example, the Task Force on Modelling of Groundwater Flow and Transport of Solutes (see Section 26.4.5).

A study showing that geogases – helium, methane, radon etc – diffuse through the matrix at the velocity and the flow rate predicted by transport models has been published (Neretnieks 2013). The analysis is based on data from a depth of 1,000 metres at Forsmark, Laxemar and Olkiluoto, as well as numerous other observations and measurements. The theoretical transport model obtained in this study has been used in modified form to show that K_d for radium used in SR-Site (Crawford 2010) is consistent with measured natural radon-222 and radium-226 activities in Forsmark's groundwater.

Another current study shows that matrix diffusion has an influence on stress-induced mineral dissolution and re-precipitation. This may in turn have an effect on fractures formed due to spalling around deposition holes when they are squeezed by the swelling bentonite. At elevated temperature in particular, they can shrink together due to dissolution on the pressure side and precipitation in the surrounding porous rock matrix.

An analytical solution has been developed for diffusion into stagnant zones, followed by diffusion into a rock matrix consisting of several matrix layers (Mahmoudzadeh et al. 2012).

Methods based on electromigration have shown promising results (Löfgren and Neretnieks 2006, Löfgren et al. 2009, Vecernik et al. 2012). The method permits measurements over longer diffusion distances than traditional methods.

Programme

The analytical solution for diffusion into stagnant zones, followed by diffusion into a rock matrix with several layers, that has been developed (Mahmoudzadeh et al. 2012) is intended to be implemented in Chan3D. Moreover, methods must be developed to measure channel widths. This quantity is essential for assessing the importance of stagnant zones. There are suggestions as to how this quantity can be estimated, based on evaporation measured with an infrared camera, see Section 26.12 "Advection/dispersion – radionuclide transport".

A modelling project is planned to study the conceptual assumptions behind the matrix diffusion model that is traditionally used both in the safety assessment and in evaluation of experiments. The model says that groundwater flow takes place in open fractures, while diffusion into the immobile water in the matrix takes place according to a one-dimensional model with usually homogeneous properties. In reality, however, the matrix is intersected by microfractures that are more or less water-bearing. In the project, the fractures in the matrix will be included and discretized finely enough so that even possible flow in the matrix can be simulated. The purpose is to see whether a simple advection-diffusion model can reproduce the modelled transport processes, or whether more complex models are needed. The calculations will be supported by X-ray tomography measurements of rock material from Forsmark for the purpose of identifying the relevant size of the microfractures.

Methods based on electromigration have shown promising results (Löfgren and Neretnieks 2006, Löfgren et al. 2009, Vecernik et al. 2012). In parallel with development of the method and applications in the laboratory environment, it is judged that potential exists to develop this method for application in the field.

It can be noted that the planned simpler setup of LTDE-SD (LTDE Light) that was discussed in RD&D Programme 2010 has not been realized. The detailed characterization programme (see Section 14.4 "Detailed characterization") will initiate this project so that results are available in time for establishment of the experiment when the Spent Fuel Repository has got under way.

26.15 Reactions with the rock – hydrogeochemistry

Reactions between the rock, in particular the fracture-filling minerals, and the groundwater are a constantly ongoing process, albeit with highly variable intensity and scope. Most of the fracture-filling minerals have been formed under hydrothermal conditions (which means they are older than 400 million years), but low-temperature minerals such as calcite, pyrite and (closer to the ground surface) iron oxyhydroxides and manganese (oxyhydr)oxides, also occur. They can be utilized to understand the evolution of the groundwater when it comes to e.g. redox conditions and microbial activity. Reactions with the minerals in the rock matrix occur to a limited extent, and the processes in a low-temperature environment are slow. In the long term, however, these reactions with the primary minerals in the rock matrix may take on increased importance for the chemistry of the groundwater.

Redox reactions are important for assessing the long-term safety of final repositories, both for spent fuel and for low- and intermediate-level waste. 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.
- Oxygenated water could penetrate down to repository depth during periods of greatly changed hydrogeological conditions, for example in conjunction with a glaciation.

Conclusions in RD&D 2010 and its review

Prior to RD&D Programme 2010, quite a few studies of fracture-filling minerals had been conducted in Laxemar, Äspö and Forsmark. The studies of the redox front in the other parts of the rock in Laxemar and Äspö, as well as of the reddish coloration of the wall rock, indicated stable anoxic conditions at repository depth and that oxygen is consumed in the upper 150 metres of the rock, if not already in the soil layer. These studies were based on mineral and element distributions and have been supplemented by uranium-series analyses, which have helped to detect processes that are geologically recent, i.e. processes that were active during and after the last ice age. SSM took a positive view of the field measurements that had been done, but pointed out the need for supplementary mathematical models.

Newfound knowledge since RD&D 2010

One of the questions that constantly comes up is how far down in the rock oxidizing conditions can be seen as a result of penetration of glacial meltwater. The same methodology as was used in the analysis of drill cores from Laxemar and Äspö was therefore tested (Tullborg et al. 2008, Drake et al. 2009) on drill core samples from the Greenland Analogue Project (see Section 19.7 "Greenland Analogue Project"). All analyses and interpretations are not finished yet, but the results from the drill cores DH-GAP-01 and DH-GAP-03 indicate that the transition from oxidizing to reducing conditions takes place in the uppermost 150 metres here as well. The results from uranium-series analyses (Uranium-Series Disequilibrium, USD) indicate that the oxidation noted in the minerals has probably not occurred during the Holocene but is probably considerably older, and that the water flux has presumably been greatly reduced due to permafrost for long periods that include the Holocene, up to the present. This may be one of the reasons there is not as good agreement as there was in Laxemar and Äspö between the redox zone detected in the minerals and the results of the uranium-series analyses, which indicate that the oxidation seen from the Holocene has reached the same depth as previous oxidations during the Quaternary. Supplementary analyses will be done of the drill core DH-GAP-04.

The lessons learned from the uranium-series analyses, along with determination of the degree of oxidation of fracture-filling minerals, have been applied to uranium-bearing minerals from fractures in Forsmark for the purpose of gaining a better understanding of which processes have caused mineral precipitation and which processes dissolve the precipitates and cause the elevated uranium concentrations in the groundwater. Mineralogy and trace element chemistry are also analyzed on these samples (Sandström et al. 2011, Sandström and Tullborg 2011), and different uranium phases (e.g. phosphate and silicate) have been identified on fracture surfaces from the Forsmark area (SFR). Moreover, groundwater from several sections in Forsmark is being analyzed in order to follow the evolution of the uranium concentrations during the monitoring period.

In order to gain a better understanding of conditions for sulphate reduction during initial (undisturbed) conditions in the bedrock, isotope studies of low-temperature precipitates of sulphide minerals in the fracture system in Laxemar have been analyzed by Secondary Ion Mass Spectrometry (SIMS) and the results published in Drake et al. (2013). These studies show that iron sulphides of biogenic origin have been formed at depths of nearly a kilometre. The large sulphur isotope fractionation and the variation of isotope ratios within the same pyrite grains indicate that these sulphides have been precipitated as a result of microbial activity in situ. Equivalent studies of sulphide and calcite precipitated in boreholes (on equipment) are being pursued as a part of the investigations to gain a better understanding of sulphide formation in the borehole sections.

Models of the consumption of dissolved oxygen in glacial meltwater were also developed during SR-Site (Sidborn et al. 2010). The results show that dissolved oxygen can only reach deposited canisters of spent fuel under a series of pessimistic and extreme conditions. Experimental studies of abiotic reactions between dissolved oxygen and various minerals, kept in sterile flasks, have been conducted and evaluated at Stockholm University, and the results will be published shortly.

The EU project ReCosy was concluded during the period and the results have been presented in some ten or so scientific publications and reports written by the project participants. The project examined redox effects on radionuclide chemistry and the difficulties of measuring redox potential in both natural and laboratory environments.

Programme

In order to learn more about how groundwater composition changes with time, studies of ion exchange processes have recently been started at Stockholm University. These studies should result in a quantification of the exchange of sodium and calcium that occurred when rain or sea water infiltrated the areas investigated by SKB. Isotope data on strontium-90 taken from the Äspö HRL may also contribute knowledge on these processes. The project will be pursued in cooperation with Posiva.

The investigations of uranium in groundwater and fracture-filling minerals in the Forsmark area are continuing. Supplementary samples will be taken of both drill cores and groundwater. The focus is on the mechanisms for redistribution (mobilization/deposition) of uranium during different time periods and under different redox conditions.

Studies of the redox zone in the way done on samples from Laxemar/Äspö and within the GAP project may be done for Forsmark as well when new boreholes are drilled.

Isotope studies of precipitates of sulphide mineral and calcite at low temperature in the fracture system in Forsmark will be carried out with SIMS, as a comparison with the isotope study in Laxemar (Drake et al. 2013). These studies will also look for signs of anaerobic microbial methane oxidation at great depth (cf. Drake et al. 2012). This will be done by means of SIMS analyses of carbon isotope ratios in calcite crystals from fracture surfaces at different depths.

After the experimental studies of abiotic reactions between dissolved oxygen and different minerals at Stockholm University have been published, it may be necessary to collect supplementary data. Development of efficient numerical models for the penetration of dissolved oxygen in fractures will be initiated, since the numerical models used in SR-Site were extremely time-consuming.

26.16 Reactions with the rock – sorption of radionuclides

Conclusions in RD&D 2010 and its review

A number of measurements had been carried out prior to RD&D Programme 2010, for example sorption of radionuclides on crushed material and on whole pieces of rock (quantified as the distribution coefficient K_d), Cation Exchange Capacity (CEC) and specific surface area (expressed by the BET sorption isotherm). A method for measurement of K_d values on whole pieces of rock, where dissolved ions are transported to the sorption surface via electromigration, had been developed in a Ph. D. project that has now been concluded. The method is several orders of magnitude faster than the one used to evaluate K_d for sorbing substances in a through-diffusion test. Another Ph. D. project had been started for development of process-oriented sorption modelling. The background to the project was the need to reduce conceptual uncertainties associated with the use of K_d values. Characterization and definition of the effective surface area for sorption has proved to be of special importance.

SSM was in favour of the fact that SKB had partially succeeded in explaining the large spread in published K_d values, which can vary by several orders of magnitude for one and the same radionuclide. It appears likely that normalization to a representative surface area is the most important explanation. It was pointed out that SKB should continue these investigations for the purpose of reducing the uncertainty margins in the use of K_d values in the safety assessment. It was noted that useful information could perhaps be obtained from research on weathering kinetics for rock-forming minerals, since experience relating to the influence of reactive surfaces is similar in this area. The proportion of secondary minerals with significantly larger available surface areas than the rock-forming minerals may be an important factor. It was also commented that the sensitivity of the methods can vary widely depending on the experimental design and the liquid/solid ratio.

SSM also pointed out that since K_d values measured in the laboratory are affected by many conditions, such as hydrochemistry (pH, Eh, ionic strength, complexation, competing substances) and the properties of the sorption surfaces, supplementary studies involving fundamental sorption models are needed. Thermodynamic sorption models have been developed and are being used increasingly to gain a better understanding of sorption mechanisms.

Newfound knowledge since RD&D 2010

The transport modelling work, using data from the Äspö HRL, has progressed more or less according to plan in a Ph. D. project at KTH in cooperation with Chalmers University of Technology. A lot of measurement data has been collected on the minerals present in the rock at Äspö. It can be mentioned that measurement data has been collected on porosity and the specific surface area of minerals, and a survey has been made of which minerals are important for sorption. The distribution of sorption on a surface has also been investigated.

In SR-Site, K_d values were used to quantify radionuclide sorption in the safety assessment calculations. Crawford (2010) presents a methodology for how K_d values for use in the safety assessment can be obtained from experimentally measured K_d values in combination with different types of modelling. In brief, the method involves multiplying the experimentally determined value by a number of transfer factors to adjust for differences between different conditions in laboratory measurements and in situ. The different transfer factors represent differences in surface area available for sorption, differences in cation exchange capacity, and differences in hydrogeochemical conditions. Mechanistic hydrogeochemical modelling (for example with PHREEQC) is used to obtain the factor that quantifies differences in hydrogeochemical conditions. This step is very similar to the use of smart K_d values described in Section 26.24.4. Aside from other factors, the irreversibility of sorption may be a source of variations and uncertainties in K_d values.

Other progress within the sorption field can be summarized as follows:

- A site-specific mechanistic model for surface complexation in granite is being developed in the EU project CROCK (Rabung et al. 2012). Hopefully, this will result in a generalized methodology for handling of sorption properties for substances where there is no laboratory data. A well-functioning mechanistic model for sorption can be used directly in codes for reactive transport modelling, or to derive K_d values for site- and scenario-specific geological and hydrogeochemical conditions (according to the so-called smart K_d concept).

- A program (PATHTRAC) has been developed where non-linear sorption processes are coupled with a model for transport and matrix diffusion in such a way it that permits considerably faster calculations for safety assessment purposes. The methodology is being developed and verified in the CROCK project. Verification is being done by means of fully coupled modelling using a well-known reactive transport model (CrunchFlow). In principle, the programme can handle all types of non-linear sorption arising from spatial and temporal variations in hydrogeochemical parameters, such as classical non-linearity, for example Langmuir and other isotherms. There are also plans to implement the methodology in COMSOL Multiphysics to permit modelling of more complex 2D/3D flow systems with matrix interactions. It is possible that in the future the methodology can be introduced in a modelling tool that is specialized in handling fractures, such as DarcyTools or ConnectFlow, see Section 26.4.2 “Hydrogeology in the deep rock”.
- Calculations have been done in SR-Site to try to quantify the influence of diffusion in stagnant zones in the safety assessment. The retardation of conservative substances, compared with sorbing substances as observed e.g. in the SWIW tests during the site investigation, shows signs of non-Fickian dispersion.
- A development project has been carried out aimed at correcting in situ electrical measurements of formation factors for artefacts related to surface conductivity that have a significant influence on the measurement results (SKB 2010q, Rabung et al. 2012 and planned scientific publication).

Results from an extensive mapping and quantification of fracture-filling minerals have been analyzed statistically and reported (Löfgren and Sidborn 2010a, b). An article on the same topic is being prepared.

Methods including electromigration have also shown promising results for sorption of radionuclides on surfaces and in the rock matrix (André 2009). It should be possible to develop and apply electromigration in conjunction with measurements for many more nuclides.

A number of different methodological parameters of batch sorption experiments have been investigated (Holgersson 2012). These are: total metal concentration, solid-to-liquid ratio, phase separation method, sorption on walls of the experimental equipment, measurement method and pre-treatment method. Sorption of europium-152 on crushed Kivetty granite with different particle size distributions was studied. A synthetic, oxygen-free and saline groundwater of the type found in Olkiluoto was used with different initial pHs.

Programme

Research and development of codes that can handle retention processes, such as sorption, will continue. The focus will be on understanding and mechanistic modelling. In safety assessment, it is likely that the K_d concept will still be used. The intention is, however, to continue developing the concept of smart K_d values, since the K_d value is dependent on local hydrogeochemical properties. This work is expected to improve the tools that are used in safety assessment.

The transport modelling work, with the aid of data from the Äspö HRL, will continue with supplementary measurements and interpretation of the experimental results. A lot of measurement data has been collected on the minerals present in the rock at Äspö. For example, measurement data have been gathered on porosity and specific surface area. A survey of which minerals are important for sorption has also been performed. The whole study is expected to be described in a Ph. D. dissertation.

Continued development is planned of mechanistic sorption models, smart K_d values, the program PATHTRAC and implementation in e.g. COMSOL Multiphysics. CROCK (Rabung et al. 2012) was concluded in 2013, but it is possible that other international fora will be established for this development.

Reinterpretation of data from SWIW tests is planned to gain a better understanding of the connection between channelling and diffusion in slow-flowing or immobile water. This will be done with the aid of transport models that take such effects into consideration.

Further development and application of methods that include electromigration is planned for determination of radionuclide sorption on surfaces and in the rock matrix for more nuclides than has been done so far (André 2009). The use of electromigration methods in situ has potential advantages over laboratory measurements, in part because it is difficult to create a reducing hydrochemistry in the laboratory, and in part because rock samples taken to the laboratory are mechanically disturbed. In general, it is difficult to translate data from crushed rock in the laboratory to intact rock in situ. It has also been suggested that electrical or electromigration methods can be used for pore water analyses.

Another area where there is a need to strengthen future competence is solution chemistry, which includes leaching and sorption processes.

26.17 Microbial processes

Microbes affect the chemistry of the groundwater by accelerating reactions that would occur very slowly or not at all in their absence, such as formation of sulphide by sulphate reduction at the temperatures that prevail in and around the repositories. Since microbes also need nutrients which they obtain from the surrounding environment, they affect the chemical composition of the groundwater by consumption and transformation of nutrients and energy sources. Redox reactions are particularly affected, but precipitation and weathering reactions, as well as processes that influence radionuclide transport, can also be affected (Pedersen 2002).

Conclusions in RD&D 2010 and its review

The RD&D Programme 2010 stated that the focus should be on the capacity of the microbes for sulphide formation and oxygen reduction in the near- and far-field.

SSM judged that SKB should strive for a well-substantiated analysis of which factors influence the long-term evolution of the sulphide concentrations over one climate cycle. Further, SSM was of the opinion that since acetogenic bacteria can both consume and, in some cases, produce hydrogen at great depths in the bedrock, they may be of importance for hydrogen-producing corrosion even in the absence of dissolved sulphide. They can therefore affect the life of the canisters, which is why studies of acetogenic bacteria are needed. It must also be determined whether limiting factors for the metabolism of sulphate-reducing bacteria in compacted bentonite also apply to acetogenic bacteria.

Few measurements of dissolved hydrogen gas in groundwater at repository depth have come to the attention of SSM, and in most cases the concentrations are below the detection limit (Hallbeck and Pedersen 2008), which means that both sampling technique and analysis methods need to be improved, if possible.

Newfound knowledge since RD&D 2010

The question of representative sulphide concentrations arose when highly variable and in some cases elevated sulphide concentrations were measured in boreholes with stationary monitoring equipment, while the initial measurements in conjunction with the chemical characterization generally showed much lower values. In a field study in Laxemar and at the Äspö HRL (Rosdahl et al. 2011), problems that can be attributed to the monitoring equipment were investigated. A compilation has been made of all sulphide analyses carried out in the site investigations (Tullborg et al. 2010 a, b) and different sampling methods have been compared. In general, water that has been standing in the borehole section before sampling shows the highest values. The length of the section (between packers) and the number of water-bearing structures and their location in the section are factors of great importance in determining how much water needs to be pumped out to get water that represents the situation in the rock. To be able to better understand and describe the processes that favour sulphide production in the borehole section, several selected sections in the Äspö HRL have been studied with respect to water and mineral chemistry, microbes and corrosion. These results are currently being evaluated and will be reported soon.

Additional publications (Hallbeck and Pedersen 2012, Pedersen 2012a, b, c) present research on microbial processes that lead to the formation of sulphide, and on the ability of microbes to buffer the redox potential at a low and favourable level for the repository. The work has mainly been carried out at repository depth in the Äspö HRL under in situ conditions with naturally occurring microbes. Indications that anaerobic methane oxidation can occur in deep groundwaters have been observed in Finland (Pedersen 2012c), and there are fracture-filling mineral data that are explained by the process (Drake et al. 2012).

New methods have been developed for sampling microbial biofilms on surfaces in water-bearing fractures (Jägevall et al. 2011). However, it was difficult to gather sufficient quantities of DNA from the surfaces using available methods. A method development project was carried out in 2012 where the previous investigations were supplemented with several different cultivation methods and measurements of biomass that have higher sensitivity than biomolecular methods. The purpose was to arrive at a combination of molecular and cultivation-based methods that can be used for analyses of fracture surfaces from boreholes in the Äspö HRL and in Forsmark. Methodology for sampling and quantification of the activity of microorganisms on fracture surfaces was tested and developed. Tests were performed on natural as well as artificial fracture surfaces in drill cores from the Äspö HRL. The project showed that it is possible to detect microorganisms on natural fracture surfaces in a drill core retrieved from a depth of 400 metres in the Äspö HRL. The project also showed that it is very important to perform sampling of fracture surfaces immediately after drill cores are retrieved to prevent reduction of the number of surviving cells and deterioration of the quality of the sample material, which in turn ensures higher certainty in the analysis results. To permit enrichment of biofilms, a circulation system with two flow cells was set up, see Figure 26-2.

Programme

The research on the relationship between hydrogen, methane, sulphide and microorganisms will continue. Another goal is to use a more sensitive method for hydrogen analyses or to send samples for hydrogen analysis to a laboratory with a lower detection limit than the one that has been used in the past. Field experiments in several selected sections in the Äspö HRL will be carried out to learn more about the processes that occur in borehole sections between the sampling occasions.

The relationship between content and distribution of gases and microbial processes in deep groundwaters will also be explored. Efforts are also planned for method development and to learn more about gases and their origin in groundwater and rock matrix.

A literature review regarding recent research findings concerning acetogens in the deep geosphere will be made before deciding whether to make the experiment as these are very difficult to culture grows very slowly and may have generation times often hundreds to thousands of years (see e.g. Lever 2012). Methods for analyzing biofilms on fracture surfaces will be improved to reduce interference in ATP analysis and DNA extraction of biofilms from fracture surfaces. A good method for enrichment of biofilms on rock surfaces must be developed in the laboratory for this purpose. Enrichment is considered necessary since biofilms in natural environments contain very little DNA, often below the detection level for molecular analysis methods. The method will be designed so that it will be possible to enrich biofilms in situ in the Äspö HRL or in Forsmark.

Figure 26-2. *Circulation system with two flow cells (stainless steel cylinders at left) containing material for enrichment of biofilms.*

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.

Isotope studies of low-temperature precipitates in the fracture system in Forsmark will be carried out to investigate whether there are signs of anaerobic microbial methane oxidation at great depth, see Section 26.15 "Reactions with the rock – hydrogeochemistry".

26.18 Decomposition of inorganic engineering material

Decomposition of inorganic engineering material is a process that is of importance for the hydrogeochemistry in the geosphere in an initial phase and in the vicinity of the repository, when conditions are affected by the construction process and by steel and cement in the repository.

Conclusions in RD&D 2010 and its review

Field experiments in Grimsel, Switzerland, yielded data on the interaction between cement pore water and rock minerals. 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. SSM asserted that the properties of low-pH cement have not been adequately evaluated and pointed out shortcomings in SKB's modelling of the long-term evolution of these materials.

Newfound knowledge since RD&D 2010

Revised models of cementitious grouts were developed and used in SR-Site (Grandia et al. 2010). The effect of cement additives and of steel corrosion was evaluated by Hallbeck (2010). Studies of how cement additives can affect radionuclide sorption in the rock (Andersson et al. 2008) resulted in a supplementary study of how different laboratory methods affect the obtained sorption data. The study will be published shortly.

Programme

SKB will monitor the literature in the field, but no further experimental work is planned on the effects of engineering materials on the geosphere.

26.19 Colloid formation – colloids in groundwater

Colloids are particles on the nanometre to micrometre scale and therefore have a high specific surface area (surface area per unit mass). 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, salinity is relatively high and natural 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 or during a prolonged period of global warming, low-salinity water could penetrate down to repository depth and come into contact with the bentonite buffer, creating a potential for montmorillonite colloids to be released for further transport. Release of montmorillonite colloids could cause mass losses from the buffer, whose functionality would thereby be reduced. If a canister is damaged, radionuclides sorbed to montmorillonite particles could also be transported out from the buffer.

Conclusions in RD&D 2010 and its review

Measurements of natural colloids were carried out during the site investigations in Laxemar and Forsmark. Laboratory experiments were carried out regarding bentonite erosion in artificial fractures. Transport tests in columns of fracture-filling materials from Äspö showed that montmorillonite colloids sorb even under unfavourable conditions. SKB planned to participate in the joint Colloid Formation and Migration (CFM) project in Grimsel, Switzerland, organized by Nagra. SKB also planned to study how colloid size affects stability and radionuclide sorption. The need for studies of gamma radiation effects on montmorillonite colloids was to be investigated.

SSM expressed a positive view of SKB's plans, but said that there was a great need for additional studies.

Newfound knowledge since RD&D 2010

SKB has participated in the joint CFM project aimed at investigating the release and further transport of montmorillonite colloids. This is being done in the field experiment in Grimsel and in laboratory experiments and modelling work at KTH. Several parallel modelling activities are being pursued in CFM to interpret the field experiments and lay a foundation for future safety assessments. These results are available internally to the project participants and as working reports, and will be published as reports and scientific articles.

Montmorillonite colloids exposed to gamma radiation have proved to be more stable in low-salinity water than unirradiated ones (Holmboe et al. 2009, 2012). Neither irradiated nor unirradiated montmorillonite colloids are stable in high-salinity water. It has been found that significantly less cobalt(II) sorbs to the surface of the irradiated montmorillonite colloids (Holmboe et al. 2011) to undergo surface complexation. The sorption of positive cesium ions, which are cation-exchanged, is not affected by the irradiation.

A method has been developed for size fractionation of montmorillonite colloids. Moreover, the different size fractions are being studied to see whether the surfaces look the same on small and large colloids, and what implications this has for stability and sorption properties.

Programme

SKB intends to continue its collaboration with CFM. Installation of a field experiment is planned for 2014, and the experiment and its evaluation are expected to continue until 2018. CFM posts updated information on planned experiments on the Web (Grimsel 2013). Transport modelling in the CFM project will be pursued in four lines of development: i) continued development of multi-phase transport code where kinetic exchange between colloids, liquid phase and solid phase is included for evaluation of the transport experiments in Grimsel; ii) continued work with the COMSOL flow simulation model for the shear zone in Grimsel, which is being used in CFM to gain a better understanding of the effects of heterogeneities in hydraulic conditions in the shear zone; iii) model for tracer transport in transient flows will be used for evaluation of multiple tracer tests where oscillations in breakthrough curves probably derive from transient conditions; and iv) correlation of laboratory results with field data. Much of the work will be done at KTH.

Detailed studies of the surface properties of the colloids associated with sorption and stability are being conducted at KTH. The goal is to identify whether the small colloids are miniatures of the large ones and thereby have the same surface properties, which would affect modelling of stability, transport and sorption properties in the different size fractions.

26.20 Colloid formation – radionuclide transport with colloids

Conclusions in RD&D 2010 and its review

SSM was in favour of the fact that SKB planned to continue its development of colloid transport models and was trying to gain a better understanding of the controlling processes. However, SSM noted that SKB had not provided a clear picture of how data and knowledge gaps are to be filled and how this is linked to the safety assessments.

Newfound knowledge since RD&D 2010

A Ph. D. project that has addressed the formation, stability and transport of colloids has been concluded (see Section 26.19). Development of colloid modelling has continued, and studies have been conducted to gain a better understanding of the controlling processes associated with bentonite erosion.

The code MARFA was used in SR-Site to simulate transport with colloids (SKB 2010k). Two different limiting cases were studied. A reduction factor for the material input parameters can be defined for reversible equilibrium sorption of radionuclides on colloids. For this case, MARFA can be used with only one redefined input parameter. The other case is irreversible sorption on colloids. For this case, the proportion of nuclides sorbed on colloids was calculated in an initial calculation step. Then MARFA was used in two parallel runs, one for the nuclides irreversibly sorbed on colloids and one for nuclides in aqueous solution. Thus, MARFA could be used for all colloid cases studied in SR-Site.

In the CFM experiment at Grimsel, tests were conducted in 2012 where transport of clay colloids, conservative tracers and the radionuclides sodium-22, barium-133, cesium-137, neptunium-237, americium-243, plutonium-242 and thorium-232 have been investigated. The breakthrough curves were evaluated to see how fast the clay colloids are transported and how large a portion of the radionuclides accompany the clay colloids. Calculations of the sorption dynamics for sorption of radionuclides on colloids have been started.

Programme

SKB intends to continue its participation in the CMF project with laboratory experiments, one Ph. D. candidate and modelling. The CFM project's main experiment will start in early 2014, according to the plan. The main experiment will begin with the installation of a ring-shaped bentonite unit around a cylinder with packers installed in a borehole that intersects a deformation zone. The bentonite contains ampoules with radionuclides that will be released when the ampoules rupture due to the bentonite's swelling pressure. Migration of colloids and nuclides can then be detected at an analysis point.

26.21 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 nitrogen condensed with other materials when the planet was formed, and nitrogen has been seeping through the geosphere into the atmosphere ever since. Most other gases can have several sources.

Conclusions in RD&D 2010 and its review

After compilation of data on dissolved gases in Forsmark and Laxemar, both SKB and SSM concluded that there was a need for more data and development of equipment and analysis methods.

Newfound knowledge since RD&D 2010

Sampling and analysis of gases and of isotopes of gases have been carried out in conjunction with water sampling in the sulphide field studies described above in Section 26.17 "Microbial processes". The purpose has been to distinguish the origin of the different gases and thereby gain a better understanding of the sulphate reduction in the groundwater. The greatest interest has been focused on carbon dioxide, methane and hydrogen. Other gases that have been sampled are noble gases, nitrogen, ethane and ethene.

An analysis of the long-term gas flow in the deep rock was performed during SR-Site (Delos et al. 2010), where parameters were determined to assess how these gases could participate in processes such as sulphate reduction and methane ice formation.

Programme

Dating of groundwater is one of many important questions that need answering in conjunction with a final repository. This is true both for old waters (for example deep-lying saline waters or paleowater that has formed in conjunction with glaciation), but also young waters such as postglacial penetrations of rainwater in the bedrock. Data on dissolved gases in groundwater are used for this purpose. When it comes to younger groundwaters, tritium analyses have been a good tool for distinguishing modern from old groundwater. The lower the atmospheric concentrations of tritium become, the more difficult it is to use tritium. Carbon-14 has been used for postglacial waters. However, it is important to remember that carbon-14 in principle provides information on the carbon system (both organic and inorganic), and a translation to the water's residence time requires that the reactions that have affected this system along the flow paths be taken into account.

The isotopes helium-4, krypton-81, argon-39 and chlorine-36 (which is not a gas) are best suited for determining long residence times. The method for sampling and analysis of helium and its isotopes, helium-3 and helium-4, will therefore be developed. Helium-4 is a mantle-derived gas, but is also produced by alpha disintegration in the bedrock. It is known that the longer the residence time a water has, the higher is its concentration of helium-4.

As described in Section 26.17 "Microbial processes", the research on the relationship between hydrogen, methane, sulphide and microorganisms will continue. To study the relationships between content and distribution of gases and microbial processes in deep groundwaters, work is planned to analyze delta-carbon-13 and delta-deuterium in methane and hydrogen, which, along with other hydrogeochemical data, will furnish information on the origin of these gases.

26.22 Methane ice formation

At low temperature and high pressure, water and methane form a solid phase called methane ice. Methane ice can form for example under permafrost, see e.g. Gascoyne (2000). Methane ice can affect the repositories in different ways, see for example Tohidi et al. (2010).

Conclusions in RD&D 2010 and its review

The studies in Lupin and in High Lake had been concluded, but no methane ice was encountered at these locations.

Newfound knowledge since RD&D 2010

The studies in Canada's permafrost areas were analyzed in Stotler et al. (2010). So far no sign of methane ice has been found in the GAP project, which is described in Section 19.7 "Greenland Analogue Project".

The potential for methane ice formation was investigated during SR-Site (Tohidi et al. 2010), where it was concluded that with the methane concentrations and flows measured in Laxemar, Forsmark and Olkiluoto, there is no risk of formation of methane hydrates during permafrost periods.

Programme

SKB will monitor the literature in the field, but no further research is planned.

26.23 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 2010 and its review

It was observed in RD&D Programme 2010 that even though there is some support for the contention that freezing-out processes may have affected the groundwater chemistry in Laxemar, and to a lesser extent in Forsmark, it has not been possible to quantify the degree to which freezing has modified the groundwater chemistry. The plans for RD&D Programme 2010 included further studies of freezing-out processes in the permafrost areas within the GAP project. SSM considered these studies to be urgent.

Newfound knowledge since RD&D 2010

The groundwater sampling in the GAP project (which is described in Section 19.7 "Greenland Analogue Project") has been difficult, and all data are not available yet. Based on available data, no evidence has been found of salt exclusion (Harper et al. 2011), and the reported salinity in some of the groundwaters, as well as lake water from Greenland, may be the result of many processes, such as evaporation, eolian transport of nearby marine deposits, or geological sources. The geochemistry of the local bedrock may also be an important controlling factor for the salinity of the groundwater in Greenland.

Programme

A new sampling will be attempted in the deep borehole in Greenland, and the results will be interpreted and reported, see also Section 19.7. Aside from this, SKB will monitor the literature in the field and will remain vigilant to indications of salt exclusion within the ongoing programmes for hydrogeochemistry in Äspö HRL and Forsmark.

26.24 Modelling

26.24.1 DFN

Conclusions in RD&D 2010 and its review

SSM's main conclusion regarding fracture network models was that SKB should devote more effort to studying conceptual uncertainties in fracture network modelling. Specifically, the Authority points out:

- That spatial and structural relationships between fractures can be important for calculated groundwater flows and thereby radionuclide transport, which the Authority has studied (Geier 2010).
- The use of the Äspö HRL as a way to reduce the parameter span.
- Studies of fracture plane geometry to investigate its effect on groundwater flow and radionuclide transport, on both a fracture scale and a regional scale.

SKB's efforts to gain a better understanding of fracture networks in crystalline bedrock were viewed positively by SSM. Moreover, the Authority and Östhammar Municipality took a positive view of SKB's efforts to integrate the fracture network models of the different disciplines.

Newfound knowledge since RD&D 2010

Itasca in France, together with Geosciénce Rennes, has constructed a new model for generation of fractures with truncations called UFM 2013 (Davy et al. 2013). The model, which is based on a Pareto distribution of the fracture sizes, results in a size distribution that has different slopes for small and large fractures, depending on generation rate and applied truncation rules.

The uncertainties in the orientation of objects measured in boreholes influence the evaluation of data and can result in incorrect fracture network models. This mainly pertains to the concentration parameter, choice of orientation model and, to a lesser extent, the mean pole for the fracture set (Stigsson and Munier 2013). This in turn may affect the connectivity of the fracture network.

A methodology for defining fracture domains on a purely statistical basis has been developed and is presented in Darcel et al. (2012). A more exhaustive report is under review and will be printed shortly.

Programme

In RD&D Programme 2010, six points are identified to gain a better understanding of discrete fracture networks in crystalline bedrock:

1. Investigate how estimated intensity depends on the measurement method.
2. 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.
3. Develop a method for better evaluation of differences between different models.
4. Further explore the aperture distribution of the fractures over the fracture plane.
5. Study effects of interconnection of fractures, for example via channelling, truncation against other fractures, or alternative methods of generating fracture networks, and.
6. Continue to evaluate alternative concepts of fracture generation.

These six points will continue to be cornerstones in the development work on fracture network modelling. In addition, a seventh point concerning the impact of orientation uncertainty on objects measured in boreholes and their consequence for the DFN models will be studied more closely.

It is noted that issues linked to the aperture distribution of fractures (point 4 above) and channelling (point 5 above) will be handled during the coming three-year period in projects described in Section 26.12 “Advection/dispersion – radionuclide transport”.

Based on the results of the UFM model, a project has been started where the concept is applied to data from Äspö in order to test its capacity to reflect known data. This project touches upon points 5 and 6 above, and to some extent point 2 as well.

Point 6 above is handled in a specific project, DFN-R, which is described below. Other points on the list will not be further explored during the coming three-year period.

The method for evaluating the uncertainties for objects measured in boreholes will be refined in the future. The work is expected to make it possible to estimate the underlying orientation parameters for the fracture tip by inverse modelling. The effects of taking into account or neglecting the orientation uncertainty will also be studied by evaluation of e.g. degree of connectivity and flow.

Discrete fracture network modelling is a central part of the safety assessment for the KBS-3 concept in fractured rock. The DFN modelling in SR-Site was based on stochastic methods, since available information was mainly available in statistical form, i.e. the fracture properties of the rock were, with few exceptions, not deterministically known in space. The goal of the DFN-R project (where R stands for repository) is to develop a new methodology for the DFN modelling to be used in conjunction with the construction of the Spent Fuel Repository. Important parts of the DFN-R project are to:

1. Develop a discipline-wide DFN that simultaneously takes into account geological, rock mechanical and hydrogeological issues.
2. Develop a methodology for conditional stochastic simulation of measured information in pilot holes, deposition tunnels and deposition holes at repository level.
3. Identify the investigation strategy required by the developed methodology.
4. Critically review the implications of fundamental assumptions in the proposed methodology.

The third point underscores the need for coordination between the DFN project and the detailed characterization programme (see Section 14.4 “Detailed characterization”). It is estimated that the DFN-R project will go on for three years, between 2013 and 2015, and will be coordinated with similar activities initiated by Posiva. Where applicable, data from Posiva’s Onkalo facility will also be analyzed.

26.24.2 Integrated modelling – thermo-hydro-mechanical evolution

Conclusions in RD&D 2010 and its review

RD&D Programme 2010 emphasized that an important point of departure in the design philosophy for the Nuclear Fuel Repository in Forsmark will be the application of the Observational Method. In order to make better predictions of the expected thermal, mechanical, and hydraulic behaviour, it is necessary to develop models that can calculate parameters that will be monitored as the construction of the repository progresses. This is not a trivial problem, since it is often possible to measure what cannot be calculated, 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 ongoing task for SKB, see Section 14.4.2.

SKB described its participation in the project DECOVALEX 2011, which consisted of three separate parts: i) ventilation experiment in a clay tunnel in Mont Terri, ii) the Äspö Pillar Stability Experiment (APSE) and iii) flow transport and hydrogeological modelling of a water tunnel in the Czech Republic. The APSE experiment was analyzed by seven different modelling teams, including one from SKB. The first part of the work focused on developing models that can simulate uniaxial compressive strength tests on drill cores. Great success was achieved by using models that simulate the mineral composition of the rock (Lan et al. 2013).

SSM noted that certain conceptual issues from the review of RD&D Programme 2007 had not been taken into account sufficiently in RD&D programme 2010. SSM particularly emphasized issues relating to fracture formation, fracture propagation and interconnection of different fractures in the vicinity of the deposition holes. The processes and scenarios which SSM particularly wished to be analyzed included reactivation of deformation zones due to thermally induced stresses in the near-field of the final repository or near the surface (Rutqvist and Tsang 2008) and the influence of an ice sheet on fracture propagation and short-circuiting of the fracture network between nearby deposition holes (Backers and Stephansson 2012).

Regarding technical modelling-related aspects, SSM contended that when it comes to rock mechanics and coupled processes, SKB should strive in its modelling to quantify the influence of realistic fracture geometries on states and processes relevant to safety. Furthermore, SSM asserted that SKB should strive for a coupling between the rock stress models on a repository and regional scale and that the models should study fault stability during a glacial cycle.

Newfound knowledge since RD&D 2010

SKB has initiated a development programme with new calculation codes and result verification based on the unique databases that have been established in different studies in Sweden and internationally. 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 has continued to refine the conceptual assumptions regarding stress-transmissivity relationships for fractures and deformation zones, which are now being used to evaluate results from mechanical and thermomechanical simulations of repository evolution. An initial analysis has been done of stress-transmissivity relationships in data from Forsmark. The study analyzed PFL-determined fracture transmissivities and normal stresses over known deformation zones. In summary, the study does not provide any support for the view that the magnitude of the flow along fractures in Forsmark is determined solely by the normal stress that acts over deformation zones or persistent fractures. This is not surprising, since the majority of the fractures/zones were formed over a billion years ago and the current stress state has only prevailed for the past 12 million years or so. It is more likely that the transmissivity values are controlled by fracture roughness, open channels, fracture stiffness and fracture-filling materials (Martin and Follin 2011, Martin et al. 2010).

SKB has initiated studies to take into account the discontinuous distribution of transmissivity and storage coefficient that exists in the bedrock. The work is a continuation of the theoretical and laboratory studies that have been focused on hydromechanical couplings in single fractures (Ericsson et al. 2009, Fransson 2009).

The focus of the modelling of the APSE experiment within the DECOVALEX programme has been to adapt coupled thermomechanical models in order to be able to calculate how the stress in the pillar changes when it is warmed up. The goal has been to determine the spalling strength in the pillar. In a subsequent phase, elasto-plastic modelling was carried out to simulate the geometry of the failure that occurred in the pillar. In summary, the modelling efforts (Lan et al. 2012) show promise in being able to shed light on the importance of heterogeneity on the mineral grain scale for the failure process, see Section 26.9 "Fracturing".

Programme

The development programme initiated by SKB for new calculation codes and result verification will continue in the form of several coordinated sub-projects. 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. This means that certain conceptual thermo-hydro-mechanical issues are also included in the development programme, which in turn also touches upon the described processes: movements in intact rock, thermal movement, reactivation – movement along existing fractures, fracturing and time-dependent deformations. 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. The most important sub-projects are:

Thermomechanical modelling of the Prototype Repository in the Äspö HRL: An important step in the modelling work is to test the modelling methodology (continuum representation of the rock with homogeneous properties) used in SKB's safety assessment by comparing measurement data with observations. Based on this, a discussion is being held of requirements on the level of detail in the material model and the initial state, the importance of local/large-scale variations (scale dependence) in material properties, and possible qualitative and quantitative discrepancies between measured values/observations and modelling.

Pore pressure evolution during the different phases of the final repository: The work is primarily focused on describing the evolution of the pore pressure during the different phases of the final repository. The focus has also been on compiling results from the studies done within SR-Site (Hökmark et al. 2010, Lönnqvist and Hökmark 2010) regarding hydromechanical couplings and maximum depth for hydraulic jacking during a glacial cycle.

Development of DFN for simulation of fracturing in crystalline rock: SKB has used a DFN strategy to simulate the groundwater flow (hydro-DFN) and natural fracture systems (geo-DFN). However, it is not known what properties are needed in this representation of natural fracture systems to simulate the strength of the rock mass, which SKB intends to study more closely along with the possibility of simulating flow in a discontinuous excavation-damaged zone. The EDZ model from Äspö (Ericsson et al. 2009, Olsson et al. 2009) is considered to be a suitable point of departure.

Geo-DFN modelling includes all fractures – open, closed and partially sealed. Closed fractures can also cause zones of weakness in intact rock, and their influence on the strength of rock mass should be less than the influence of open fractures. Sensitivity studies will be performed to assess the importance of open versus closed fractures and their connectivity (see also Section 26.24.1 "DFN").

Conceptual understanding of the hydromechanical properties of the rock mass: Conceptual models for hydromechanical coupling will be further developed, in particular with regard to fracture geometry and geological history. Based on laboratory and field tests, the conceptual models will be verified and provide input data to numerical modelling. Furthermore, the sub-programme will develop and improve characterization methods for describing hydromechanical relationships which will in turn serve as a basis for design and construction of the Spent Fuel Repository.

26.24.3 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 (Laaksoharju et al. 1999). The calculated mixing proportions and the actual measured composition comprise the basis for calculating the scope of reactions. By coupling the reactions with groundwater flow and solute transport, it is possible to obtain powerful tools that can provide a deeper understanding of possible long-term changes during future glacial cycles.

Conclusions in RD&D 2010 and its review

By RD&D Programme 2010, the site models for Laxemar and Forsmark were ready and the EU project FUNMIG had been concluded. It was planned to start development of coupled reactive transport models based on streamline simulators.

SSM noted that there was a need for development of coupled geochemistry-transport codes and pointed out the need for an increased focus on the evolution of salinity in the Forsmark lens during an entire climate cycle.

Newfound knowledge since RD&D 2010

In the SR-Site safety assessment (Salas et al. 2010), results from groundwater flow and advection calculations were coupled with a chemical model, see also Section 26.11 "Advection/mixing – hydrogeochemistry". Models of consumption of dissolved oxygen in glacial meltwater were also dealt with in SR-Site (Sidborn et al. 2010), see Section 26.15 "Reactions with the rock – hydrogeochemistry".

Models where transport and chemical reactions have been welded together are associated with enormous challenges in the form of calculation efficiency and feasibility due to the size of the regional systems and the difficulty of integrating different processes that take place over widely disparate temporal and spatial scales. These problems have normally been overcome by means of a compromise: either (a) a representation of the site hydrodynamics with an oversimplified geometry (e.g. two-dimensional cross-sections), but with a detailed description of geochemical processes, or (b) a complex three-dimensional. hydrogeological model where solute transport is given a simplified solution with K_d-based expressions.

An effective geometric solution is the use of streamline models. The methodology, which has been given the name FASTREACT (Framework for Stochastic Reactive Transport), is based on the theory of stochastic-convective models (Shapiro and Cvetkovic 1988, Simmons et al. 1995), which says that, under certain conditions, a complex transport problem can be divided into a number of independent streamlines (particle paths). The numerical framework has been adapted to geochemical reactions: the whole streamline set is represented by a single PHREEQC one-dimensional reactive transport process (Parkhurst and Appelo 1999). Simulation where the longitudinal coordinate, for example the distance from a contamination source, is interpreted in terms of advective travel time. FASTREACT has been tested against a synthetic case that represents the flow and transport conditions in mildly heterogeneous porous media, and the report is currently undergoing review (Trinchero et al. 2013), see also other applications described in Section 26.4.4.

Powerful numerical tools have been developed in recent years, based on massively parallel high-performance computing systems in combination with efficient numerical methods. These new models can avoid the compromises that were previously necessary.

The modelling tool ConnectFlow has been further developed. Among other things, the option of including geochemistry and hydrogeochemistry has been added by integrating PHREEQC, which is also mentioned in Section 26.4.2 "Hydrogeology in the deep rock". This development has benefited from the technical development of computers and an increased parallelization of the code. This permits more realistic paleohydrogeological simulations and simulation of transport of reactive solutes from e.g. dissolution of grouting cement, as well as monitoring changes in groundwater pH. It is also possible to represent a spatial variation of minerals. Ion exchange reactions are included, and it is possible to model clogging of pores.

Programme

When the modelling becomes more advanced by integrating hydrogeology with geochemistry, the need for tests and confirmation of results also increases. The results should if possible be tested against field experiments and other modelling concepts. It is likely that the traditional concepts such as phase K_d will be used for a while longer within e.g. safety assessment. Results from simplified models should, however, be compared with results from tools that use the newly-developed concepts.

FASTREACT will be used to simulate the hydrogeochemical evolution of the Forsmark site. The results will be compared with those calculated by Salas et al. (2010), based on a reactive mixing strategy.

In a separate track, efficient numerical models need to be employed for "stiff" systems of ordinary differential equations, which apply to modelling of the penetration of dissolved oxygen in fractures. The experience from SR-Site was that the numerical methods that were then used for non-stiff systems were extremely time-consuming.

In the detailed characterization programme there are plans to develop a new numerical interface, Bridge-iDP, to couple DarcyTools (Svensson 2010) with the powerful parallelized modelling tool PFLOTRAN (Hammond et al. 2011), which is mentioned in Section 26.4.2. Due to its numerical efficiency, the coupled model is expected to be a good platform for models on a regional scale with fully coupled reactive transport models in granitic rock. Forsmark will also be used as a test case for iDP. The goal is to improve site understanding, test conceptual models and determine long-term changes in the hydrogeochemical conditions, especially at repository depth.

The development of ConnectFlow with respect to integration of PHREEQC, which is also mentioned in Section 26.4.2, will continue. The updated modelling tool should be tested against more comparable codes. Furthermore, there are plans to apply the expanded modelling tool to problems where it used to be necessary to simplify simulations of e.g. the paleohydrogeological evolution of a site and the dissolution of cement used for grouting.

26.24.4 Integrated modelling – radionuclide transport

Conclusions in RD&D 2010 and its review

SSM was positive to the fact that SKB had developed an alternative calculation code, MARFA, for radionuclide transport.

SSM was positive to the fact that SKB had carried out tracer tests in the field. The field tests provide a powerful tool that contributes to increased process understanding and can improve confidence in the parameterization of the transport models in the safety assessment.

SSM was in favour of the planned activities such as further development of MARFA and additional SWIW tests. However, SSM noted that SKB did not clearly explain the role of integrated transport modelling in the safety assessment. SSM considered that SKB should better integrate the consequence analyses for the biosphere and the geosphere, since SKB's current consequence analysis is still separated into two parts, so that certain processes cannot be fully handled with the current method, for example radionuclide decay chains.

Newfound knowledge since RD&D 2010

A modelling study has been carried out to evaluate the possibility of using a streamtube methodology for calculating geochemical evolution and transport of reactive solutes along flow paths. The modelling tool FASTREACT has been used in this study (Trinchero et al. 2013), where the geometry of the flow paths and advective travel times come from an independent groundwater flow simulation.

The results indicate that this could be a powerful tool for simulation of geochemical evolution (since the solutions are one-dimensional along the flow paths), but mixing effects are neglected. The coupled geochemical-hydrogeological model development projects described in Section 26.24.3 are judged to be more promising in this respect. When it comes to transport of solutes such as radionuclides, the use of smart K_d values (see "Programme" below) may be an alternative to the methodology offered by FASTREACT.

In SR-Site, the radionuclide transport code FARF31 (Norman and Kjellbert 1990) was used for production calculations. The new code MARFA (Painter and Mancillas 2009) was used for supporting calculations, specifically for simulation of radionuclide transport during glacial cycles and for analysis of the geosphere's retention function.

A simplified methodology was used in the glacial calculations where the effect of changed flow conditions was incorporated by only changing the advective transport velocity for different climate regimes, but not changing the geometry of the flow paths. A more detailed analysis of radionuclide transport during glacial cycles was done in a study conducted after SR-Site (Selroos et al. 2012). Specifically, an updated version of MARFA (version 3.3.1) was used where both changes in flow magnitude and direction can be handled. The results indicate that the method used in SR-Site is usually very conservative. The methodology used in SR-Site misses much of the spatial variation and dilution implied by variable flow fields during a glacial cycle. Moreover, the method used in SR-Site overestimates the breakthrough curve under permafrost conditions.

A more detailed analysis of retention in the geosphere has also been carried out with MARFA (Selroos and Painter 2012). Compared to SR-Site, the effect of retention in deformation zones, tunnels and soil layers was studied more thoroughly (the focus in SR-Site was on the effect of the EDZ and the crown space under the tunnel roof). The new study shows that deformation zones with realistically assigned properties can make a great contribution to radionuclide retention (greater effect than was taken credit for in SR-Site). Moreover, the study shows that retention in tunnels and soil layers can also be substantial; in tunnels it is advective delay that contributes to retention, while in soil layers it is sorption of sorbing nuclides.

A number of studies involving tracer tests were carried out during the site investigations. Nordqvist et al. (2012) describe the design, execution and experimental results of the various tracer tests conducted in Forsmark and Laxemar-Simpevarp. Single-Well Injection-Withdrawal (SWIW) tests and sorbing cross-hole tests over a relatively short scale to verify retention are discussed, along with large-scale tests over several hundred metres in deformation zones and fracture networks to study connectivity.

Numerical model evaluation of a selection of the SWIW tests with uranin and cesium tracers is described in Cvetkovic and Cheng (2011). The study shows that retention in Laxemar-Simpevarp is greater than in Forsmark. This agrees with what was previously presented in SDM-Site (Crawford 2008, Crawford and Sidborn 2009). Moreover, the results indicate that certain fractures in Forsmark may have a thin layer on the fracture surface with a higher porosity than the underlying matrix rock.

The Tracer Retention Understanding Experiment (TRUE-1 and TRUE Block Scale) has been concluded, but publication of the results in scientific journals continues. Tracer tests done within the TRUE Block Scale were evaluated in Cvetkovic et al. (2010); the results show that the retention is diffusion-controlled. Cvetkovic and Frampton (2010) show how discrete fracture network modelling can be used in the evaluation. The two articles together offer a coherent methodology for evaluation of tracer tests. The possibility of predicting diffusion-controlled tracer tests on different scales is further studied in Cvetkovic (2010a). The study shows that prediction on scales of up to 200 metres is possible by means of a simple model and the use of independently measured information. The effect of alterations of the matrix nearest the fracture surface (the rim zone) is studied more closely in Cvetkovic (2010b). The study shows that this type of altered zone, which has been observed in fractures in the Äspö HRL, may be of great importance for additional retention of sorbing nuclides on long time scales.

In the TRUE-1 Completion project, additional studies have been conducted on the experimental site for TRUE-1 in order to verify previous assumptions and results, if possible. The primary overall goal of TRUE-1 Completion was to learn more about the internal structure of the dominant structure in the experiment (the fracture). Another goal was to update the conceptual model of this structure. The work was done in three main parts consisting of 1) tracer tests and hydraulic tests, 2) epoxy injection with overcoring and core mapping, and 3) analysis of core material. One of the general conclusions is that the structure in question is heterogeneous with respect to conductivity and/or connectivity within the borehole area and is undulating and/or stair-case shaped. Finally, it is concluded that the rock matrix plays an important role for retention, but that the interaction between cesium and rock is influenced by irreversible or slowly reversible sorption. Hence, a simple cation exchange model is not sufficient for a satisfactory description of the results. The results achieved in TRUE-1 Completion do not contradict previous results, but rather clarify and strengthen them.

The planned SWIW experiment with synthetic groundwater that was presented in RD&D Programme 2010 has been carried out and will be reported. Two experiments with several different tracers have been carried out (one with a waiting period between injection and withdrawal, and one without a waiting period). The tracers were either added to or removed from the synthetic groundwater. The experiments were successful with well-defined breakthrough curves for all tracers. Clear differences could be seen for all tracers between the cases with and without a waiting period. The differences are qualitatively consistent with what is expected, given the tracers' diffusion properties. Analysis of the added tracers indicates that matrix diffusion is active during the experiment, but a simple model using a constant diffusion coefficient cannot explain both the early and the late part of the breakthrough curve.

Generic modelling studies aimed at improving simulation of transport of solutes such as radionuclides in fractures and fracture networks have continued. In Frampton and Cvetkovic (2011) and Cvetkovic and Frampton (2012), advective travel time and the flow-related transport resistance in discrete fracture network models are studied based on fracture data from Laxemar. The studies use the tempered one-sided stable density (TOSS) distribution to analytically describe advective travel time and flow-related transport resistance. In Cvetkovic (2011), the TOSS model is further generalized and related to other existing transport models such as the advection-dispersion equation. In Cvetkovic (2012), a general model for retention processes in groundwater transport is presented. The model can, exactly or approximatively, recreate most retention processes known in the literature that may be of importance for the transport rate.

Programme

Further development of the MARFA code will continue. Some functionality will be added during the coming period, for example diffusion in stagnant water with further diffusion into the matrix. Furthermore, the code will be made generally available via the Open Source concept.

When the interface between DarcyTools and PFLOTRAN is finished (see Section 26.24.3), the coupled tool could be used to obtain more realistic K_d values (smart K_d values) for radionuclide transport simulation in MARFA. The K_d values could then be calculated with the aid of the locally and temporally prevailing geo- and hydrochemistry. The model chain DarcyTools-PFLOTRAN-MARFA will thereby have been created, which could be used in e.g. the safety assessment. The actual code development work here lies in creating the interfaces between DarcyTools-PFLOTRAN and MARFA to ensure simple and efficient use. As a first step in this development work, however, an interface will be created between the codes FASTREACT and MARFA, both of which are based on one-dimensional transport pathways. FASTREACT calculates K_d values (which can vary along the transport pathway and in time) for a given hydrogeochemistry; these K_d values are then used in MARFA.

MARFA will also be used in a study where transport in the geosphere and biosphere are coupled in a transient manner. Achieving this coupling requires some code modification in input/output between the codes, but no changes are needed in the way transport processes are conceptualized by the codes. The purpose of the study is to see whether and when the normally used dose conversion factors are conservative. Thus, the actual release from the geosphere is used as time-varying input in the biosphere model in the planned study. In the methodology used in SR-Site, a dose conversion factor based on a unit release from the geosphere is calculated in the biosphere model. This dose conversion factor is defined for the maximum dose in the biosphere model, and is then used as a constant to calculate dose from the geosphere release. This approach is deemed to be conservative, but a formal quantification remains to be done.

Further, more general research linked to evaluation of tracer tests and increased process understanding via integrated modelling of groundwater flow and/or transport of solutes will continue. Specifically, the SWIW experiments with synthetic groundwater will be further analyzed with more complex models than the ones currently used.

Development of the channel network model Chan3D (Gylling 1997) will continue. Chan3D is based on the assumption that groundwater flow and transport of solutes takes place in a network of channels in fractured rock. The transport part includes matrix diffusion, sorption on fracture surfaces and in the rock matrix, and decay. Diffusion into stagnant zones with subsequent diffusion into a rock matrix consisting of different minerals will be implemented. The hydrodynamic dispersion is built into the channel network and does not need to be included by means of assumed parameters. The modelling tool can be coupled to the near-field model Near3D and can thereby handle most of the calculations normally included in the safety assessment. Improved methods for incorporating data from boreholes and tunnels are also planned.

27 Surface ecosystems

Surface ecosystems (the biosphere) include the regolith (loose deposits including soil layers and sediments), superficial groundwater, surface water, the land surface and the lower part of the atmosphere, plus all organisms present in these environments. The consequences of a release from a final repository for spent nuclear fuel or other radioactive waste – in the form of a radiation dose to humans, animals and plants – arise when radionuclides are cycled in the ecosystems. Calculations of the accumulation and cycling of radionuclides in surface ecosystems, and of the risks associated with a possible 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 state of the surface ecosystems also comprises 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 radiologically most important processes and phenomena in the ecosystems. Another goal is to develop methodology and models that can be used to determine long-term radiological risks for humans and other organisms in conjunction with the final disposal of pent nuclear fuel and other radioactive waste. In the following sections we give a detailed account of the background to the current programme, present new results, and formulate guidelines for the future research programme for surface ecosystems.

27.1 Summary

Since RD&D Programme 2010, 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 of describing the ecosystems on the sites, the SR-Site safety assessment and the ongoing safety assessment of SFR (SR-PSU). SKB has developed conceptual and numerical models that describe the ecosystems and the transport of waterborne elements in surface ecosystems. Models of the extent of vegetation on land and in the sea and of surface-hydrological flows have 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 have continued to be published at a rapid pace, with more than 40 SKB reports and around 30 scientific articles since RD&D programme 2010. Several of the publications summarize the multi-year programmes for site investigations and safety assessment. A special issue of the scientific journal Ambio (2013, vol. 42:4) carried 14 scientific articles dealing with SKB's work in surface ecosystems. In addition, the results have been presented at conferences, seminars and university courses.

Research and development during the coming programme period will build further on the current activities and include studies to improve process understanding and development of calculation methodology. A great emphasis will be placed on questions and uncertainties identified in ongoing safety assessments and preparations for assessment of the future repository for long-lived waste, SFL. New efforts are planned to improve the description of aquatic and terrestrial ecosystems and the transition between them in an evolving landscape with human impact.

27.2 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 (Lindborg 2008, Söderbäck and Lindborg 2009, SKB 2010t). 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 (sea, lake, wetland) 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 other organisms, the model can be simplified and the simplifications justified. One way to do this is to set up an interaction matrix (Hudson 1992). In an interaction matrix, the system in question is divided into different elements. Surface ecosystems are thereby divided into both abiotic (for example Quaternary deposits, hydrochemistry and surface water) and biotic components (for example 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 at the beginning of the 21st century (Kautsky 2001). The biosphere matrix has comprised an important tool in the design of the site investigation programmes and the ecosystem models. In reviews of previous RD&D programmes, however, SSM (or then-SSI) has noted the lack of an integrated description of processes that are relevant for constructing the models that are used to calculate doses in the safety assessment.

Based on the general interaction matrix, ecosystem-specific interaction matrices have been compiled since RD&D Programme 2010 for limnic (Andersson 2010), marine (Aquilonius 2010) and terrestrial (Löfgren 2010) ecosystems. Knowledge from the site has been used in this work, together with scientific literature, in order to make sure that all processes identified as important in the interaction matrix have been included in the radionuclide transport models. Definitions and descriptions of the processes included in the matrix have been presented in an SKB report (SKB 2010u).

The three different components – site description, mechanistic ecosystem models and models of radionuclide transport – together comprise, along with the results of various research projects, the most important elements in the long-term iterative work of improving process understanding, see Figure 27-1.

In order to be able to understand and describe transport of radionuclides from a repository, knowledge of flows of materials that can serve as carriers of radionuclides is needed. The dominant carrier of radionuclides is water, in the repository as well as in the rock and in the loose deposits. In surface ecosystems as well, water is of great importance for transport and dispersal of radionuclides. Moreover, radionuclides can be spread via gas exchange and by the uptake of substances in organisms, which can in turn be eaten by other organisms and ultimately by man. For most radionuclides that could escape from a geological repository, the dominant dose source for humans is intake via food or, in some cases, water.

A common method in ecology and physiology is to generalize the food chain by describing flows between organisms in the form of organic carbon (kgC) or energy (kJ or kcal). This makes it easy to set up ecosystem budgets, where total consumption is balanced against the sum of growth and losses in faeces or metabolism. SKB has chosen the same approach to illustrate flows of organic matter in ecosystems, including the flows that end in food for humans or other organisms.

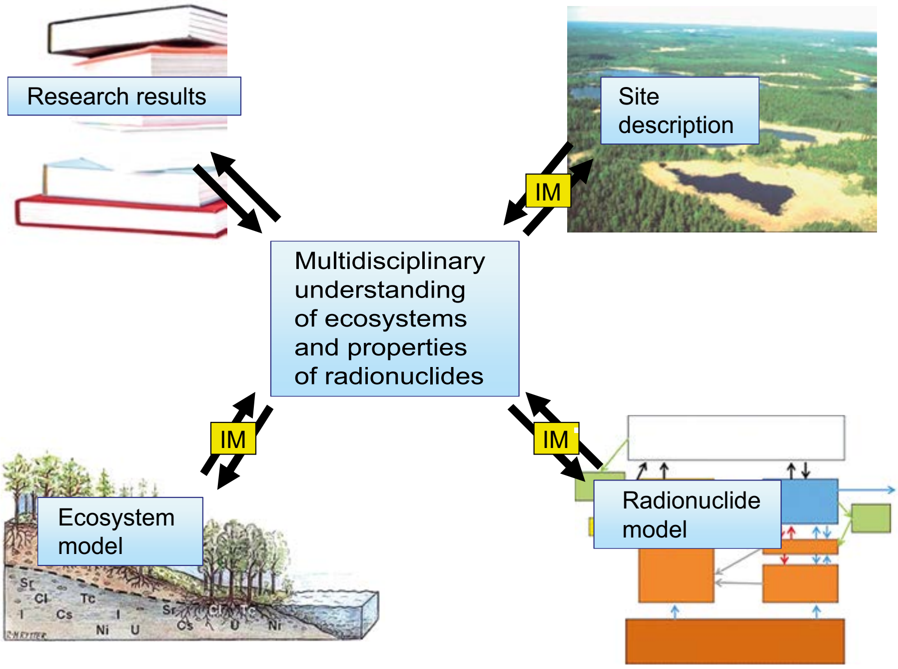


__Figure 27-1.__ Illustration of the long-term iterative work of identifying and improving the understanding of important processes in surface ecosystems. An initial version of the interaction matrix (IM) served as an important basis for the design of site investigation programmes and of ecosystem models and models of radionuclide transport. Each of the different components generates new knowledge and new questions that are used to develop the other components. The accumulated body of knowledge (middle of illustration) can be found in the reports that describe the respective ecosystems (Andersson 2010, Aquilonius 2010, Löfgren 2010).

The organic matter comprises radionuclide carriers and is normally expressed in kilograms of organic carbon (kgC). This means that the organic carbon is primarily a carrier of radionuclides and not an analogue of radionuclides. Secondarily, carbon is also a good analogue of many substances that are readily bound in organic molecules (such as carbon-14, iodine, nitrogen and phosphorus) or sorb to organic matter. Other advantages are that the results of the site investigations are much easier to compare with other ecological studies, and that variability decreases when the unit kgC is used instead of e.g. fresh weight.

Conclusions in RD&D 2010 and its review

The emphasis in the description of surface ecosystems in RD&D Programme 2010 was on applying radiological process knowledge in the SR-Site safety assessment based primarily on the results of the site investigations in Forsmark and Laxemar-Simpevarp. 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 how man and the environment are exposed to all relevant radionuclides. The methodology should also reflect current 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 and dissemination of newfound knowledge via scientific publication and active participation in conferences and seminars.

In its review of RD&D Programme 2010, SSM concluded that there was a great need for further work in the safety assessment for low- and intermediate-level waste. Further, it was unclear to the Authority how collected data on carbon pools and fluxes in various terrestrial ecosystems can be used for non-carbon-related radionuclides. SSM was of the opinion that SKB should gather site data not only from today’s selected site, but also from sites where data can represent properties of ecosystems that are tens of thousands of years old. Moreover, SSM called for a more systematic methodology for selecting K_d values (which describe the distribution of substances between solid and liquid phase) for use in the safety assessment.

SSM took a positive view of the ongoing work on Greenland and considered it important that the results of this work be applied to existing and planned repositories in Forsmark, for example to conceptualize flow models during periods of glaciation and permafrost. SSM also thought that groundwater flow beneath the ocean surface should be taken into consideration, and that a better substantiated extrapolation of the present-day hydrological model to future conditions is needed. As far as modelling tools are concerned, SSM thought that SKB should have access to several modelling tools so that the purpose of the model in the safety assessment can be achieved in the best possible way. The Authority also called for an integration of the consequence analyses for the biosphere and the geosphere.

In its review of RD&D Programme 2010, the Swedish National Council for Nuclear Waste called for a more comprehensive account of the content of the planned research programmes concerning terrestrial and aquatic ecosystems.

The Royal Swedish Academy of Sciences found that the research programme presented in RD&D Programme 2010 generally exhibits higher scientific quality than previous programmes, but that there is still room for improvement.

Programme

SKB has based its planning of research and development regarding surface ecosystems during the coming programme period on the conclusions in RD&D Programme 2010 and the regulatory authorities' comments on the programme, as well as on insights from the site description, the SR-Site safety assessment, the ongoing review of SR-Site, and the current work with SR-PSU and the SFL programme. The research programme for surface ecosystems has been designed with a view to the requirements which future safety assessments of both commissioned and planned repositories will make on the description of transport and accumulation of radionuclides in surface ecosystems. The programme is presented in the following sections for the different disciplines.

27.3 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 mainly in the low-lying points of the landscape and therefore usually reaches wetlands.

Radionuclides transported with the groundwater can be taken up in plants, mainly via their roots, and can thereby be accumulated in biomass. For most radionuclides, 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 passes through them, see Figure 27-2.

The work with site investigations and site modelling in Forsmark and Laxemar-Simpevarp has resulted in a number of new insights regarding fluxes and accumulation of organic matter and distribution of elements in the terrestrial ecosystems (summarized in Löfgren 2010). Figure 27-2 shows the most important processes for transport and accumulation in the mire and in the adjacent cultivated mire. The results of continued work have since been published in separate reports, and additional reports are planned in conjunction with the safety assessment for the planned extension of SFR, SR-PSU.

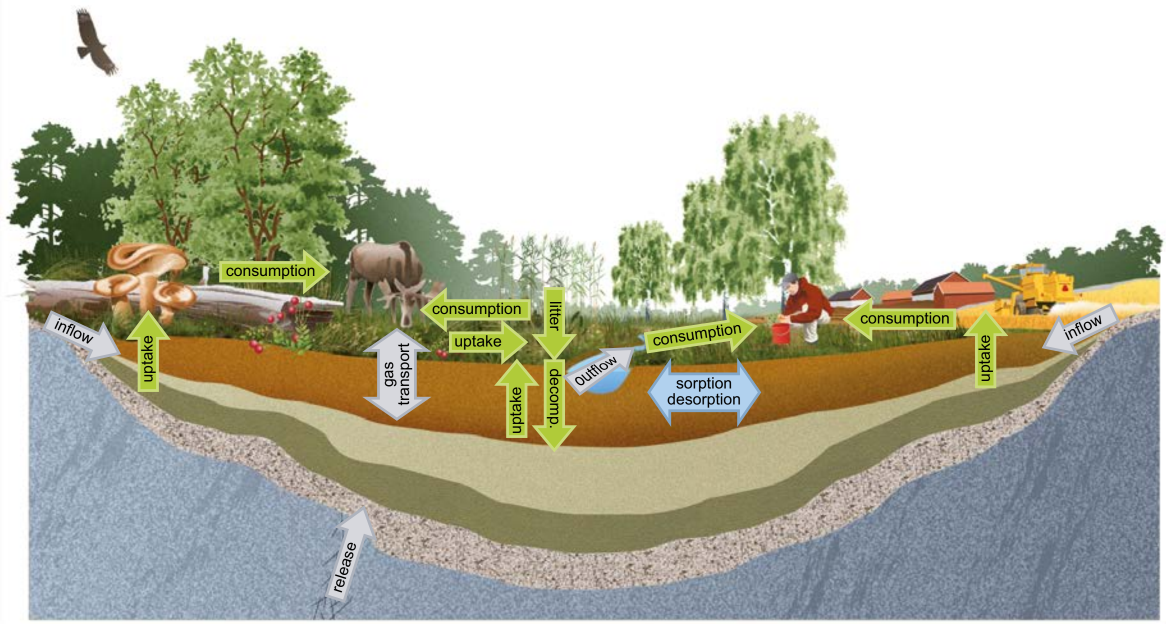


Figure 27-2. A conceptual description of important fluxes and processes affecting the accumulation of elements in a wetland and in arable land on a drained part of the wetland, where human exposure is in focus (Löfgren 2010). Green arrows indicate fluxes mediated by biota, grey arrows are water and gas fluxes, and blue arrows represent sorption/desorption processes. Consumption includes drinking water. The mire was preceded by marine stages followed by a lake stage, which is indicated by the presence of gyttja clay and postglacial clay beneath the peat deposits.

Conclusions in RD&D 2010 and its review

RD&D Programme 2010 identified the need to broaden the description of properties that characterize wetlands over a hypothetical succession gradient from coast to an inland position. The programme also expressed an ambition regarding targeted studies of wetlands bordering on lakes. Furthermore, there was an ambition to investigate how radionuclides are taken up in crops, since ingestion of contaminated farm crops comprises an important source of human exposure to many radionuclides, see Section 27.9 "Radionuclide modelling". In its review, 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 streams, lakes and sea, and human use of wetlands.

Newfound knowledge since RD&D 2010

Knowledge of the function of the terrestrial ecosystems on the investigated sites was summarized in Löfgren (2010). This knowledge is based on a synthesis of results from site investigations and site modelling, along with an application of a general scientific understanding of terrestrial ecosystems gained from the scientific literature to conditions on the investigated sites, see Figure 27-1. Conceptual and quantitative terrestrial ecosystem models were presented as well, along with important data and a description of parameters used in the SR-Site safety assessment. The report comprises the most important process report when it comes to description of terrestrial ecosystem processes and will be a cornerstone for updates of the state of knowledge for future safety assessments.

Several studies with a focus on carbon-14 have been conducted in the ongoing safety assessment, SR-PSU. In one study, concentrations of dissolved inorganic carbon (DIC) and dissolved organic carbon (DOC) in mires situated above sea level near Forsmark were investigated (Löfgren 2011). The investigated mires thereby represent an age sequence where the DIC concentration decrease with the increasing age of the mire, with the lowest concentration in the highest situated mire. The DIC concentration was also positively correlated to pH and electrical conductivity, where a high conductivity indicates a higher concentration of calcium cations (Ca^{+2}). Higher concentrations of calcium cations are typical of calcareous groundwater. Conversely, the DOC concentration was highest in the highest-situated, oldest mire. Both of these patterns can be explained by low water exchange and leaching processes in surrounding catchments. Sampling also yielded concentrations of naturally occurring inorganic carbon (DIC) in the mire water, which is essential for calculating the concentration of carbon-14 in the radionuclide model.

In order to be able to assess the safety of present-day and future repositories, where carbon-14 is expected to constitute a great risk to man and the environment, it is important to understand how carbon is cycled in discharge areas for deep groundwater. Carbon dioxide is taken up by plants via photosynthesis and is thereby accumulated in the aquatic and terrestrial food webs, while methane and less degradable organic carbon compounds are less susceptible to biological accumulation. In order to survey the current state of knowledge with respect to carbon pools and carbon fluxes in wetlands, a literature review was initiated in cooperation with the Swedish University of Agricultural Sciences in Uppsala. The results of this literature review will be presented in the reports from SR-PSU.

Several radionuclides exhibit the highest dose conversion factors for food that is grown locally. To shed further light on different aspects of the importance of wetlands as potential agricultural land, a joint project was started with the Geological Survey of Sweden, SGU, in 2012. The succession, horizontal expansion and historical use of mires were described from existing peat profiles in northern Uppland. Preliminary results show that half of the mires have sediments from a previous lake stage. The other half have sediments from the sea phase without a lake stage. None of the cultivated mires were directly underlain by till, which supports the current landscape evolution model, see Section 27.8 "Landscape evolution and deposits". The study also calculates how quickly drained peat is compacted and oxidized, which is of importance for release and cycling of radionuclides in crops and also crucial for the ability of these soils to serve as arable land for limited periods of time.

At sites where carbon is accumulated in the landscape, there is also a potential for accumulation of many other substances. A study within the Krycklan project (a big project for integrating studies of water quality, hydrology and aquatic ecology in a catchment area in Västerbotten) examined how naturally occurring uranium and thorium are accumulated in wetlands in catchments (Lidman et al. 2012), and more specifically how uranium, thorium and radium are accumulated in a mire in a forest landscape (Lidman et al. 2013). The studies show that about 60–70 percent of the uranium and thorium that reach the mire is retained there, and differences in concentrations among the three radionuclides in the mire depended on groundwater flows into the mire. Other substances that are strongly bound to organic matter probably behave similarly.

A degree project at Stockholm University has studied how concentration ratios (soil-to-plant transfer factors) are affected by the plants and the soil they grow in (Raguž et al. 2013).

To understand how volatile elements (carbon-14, iodine-129) are transported in the atmosphere, a review of turbulent transport in the atmosphere has been done in cooperation with Lund University, see Figure 27-3 (Tagesson 2012). Besides a process description, it was also shown how certain important parameters, such as diffusivity in the atmospheric layers, can be estimated based on measurements of temperature profile and wind speed. The study served as a basis for the revised description of the ground-level atmosphere in the SR-PSU safety assessment, where concentrations of carbon-14 are used to calculate exposure of humans and other organisms, see Section 27.9 "Radionuclide modelling".

When doses to non-human organisms are calculated, data on the range and numbers of different animals and plant groups are needed. These data are taken from the site investigations, as well as from ongoing environmental monitoring. The environmental monitoring has continued to follow mammal and bird populations in Forsmark and Laxemar-Simpevarp (Green 2010, 2011, Truvé 2012). Most populations have been stable between the survey occasions. However, a number of predators (such as lynx, fox and otter) have increased in number, as has the wild boar population. Hunting statistics have been used to monitor the development of the moose population, and a large body of data is now available to follow the effects of, for example, changes in hunting pressure or establishment of wolf (Cederlund and Broman 2010a, b, 2011a, b, Cederlund et al. 2012a, b).

During construction of the Spent Fuel Repository in Forsmark, it may be decided to create or recreate wetlands in the area. In order gain a better understanding of important processes in wetland restoration, a study has been carried out which includes previously drained peatlands in the vicinity of Forsmark (Hedberg et al. 2012).

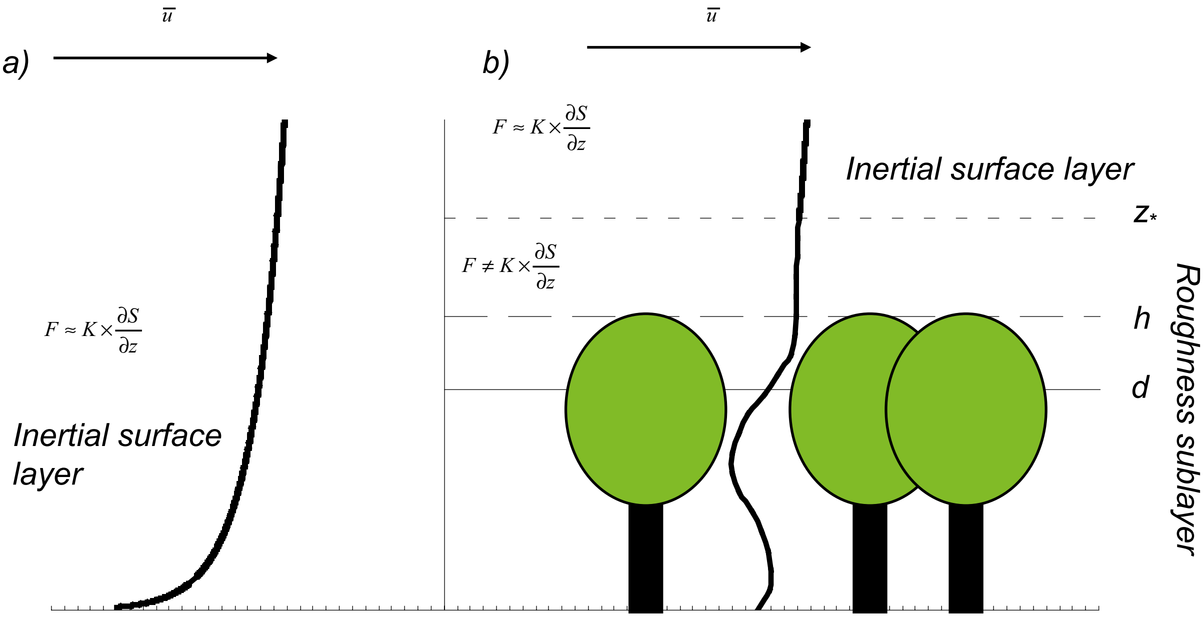


Figure 27-3. *A conceptual description of gas flux in the inertial surface layer (constant flux layer) and the roughness sublayer, where F describes a flux as a function of a state variable (S) and the height above ground (z), as well as the eddy diffusion coefficient (K). ū is the mean wind profile above the short vegetation (a), and above and within the roughness sublayer of a forest canopy (b). (Tagesson 2012, Figure 4-1).*

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Newfound knowledge in recent years confirms that if releases of radionuclides should occur to the terrestrial ecosystems, low points in the terrain where wetlands are located are the most likely recipient, see Section 27.6 "Hydrology and transport". SKB will therefore carry out additional studies to gain a deeper understanding of transport and accumulation of radionuclides in wetlands and agricultural lands. This knowledge will be used to update SKB's radionuclide model and to reduce uncertainties in the description of key factors in these calculations.

SKB will continue its long-term efforts to replace or supplement the concentration ratios for organisms with mechanistic models. In the case of terrestrial plants, the cycling of organic matter (primary production and degradation), nutrients (analogues of radionuclides) and water (transpiration or uptake in roots and leaves) controls the uptake of radionuclides. Process understanding and modelling of the processes in terrestrial ecosystems will be combined with biochemical site data (see Section 27.5 "Biogeochemistry") to modify the radionuclide models, which are described in Section 27.9 "Radionuclide modelling". A special focus will be placed on root uptake, where knowledge gained from CoupModel (Gärdenäs et al. 2009) and previous forest models (Avila 2006) will be applied, supplemented with newfound knowledge on crop irrigation. In addition, the importance of gas transport, for example methane and carbon dioxide, in soil and atmosphere will be elucidated, along with gas uptake processes via primary production and microbial metabolism.

In terrestrial ecosystems, soil chemical properties change as soils evolve over long periods of time. Certain substances are leached out while others are enriched, and important substances in the ground are gradually leached out, which can change the conditions for sorption of radionuclides in the ground. To shed light on the importance of succession and age for different soil chemical properties, existing site data, supplemented with field collections in an age gradient of suitable catchments from coast to inland, will be analyzed. This will be combined with studies of solute transport between mire and lake (see Section 27.4 "Aquatic ecosystems"), where mobile substances such as calcium ions, magnesium ions, DOC, POC (particulate organic carbon), DIC and inert elements such as rubidium, zirconium and titanium, will be studied in profiles through the landscape. These studies will provide a deeper understanding of solute transport and thereby transport of radionuclides from terrestrial environments to lakes. The profiles provide important input data to a more detailed description of transport in the surface systems. This work is planned to include tests of other tools for modelling of solute transport, for example COMSOL, see Section 27.9 "Radionuclide modelling".

The long-lived radionuclide chlorine-36 is expected to be important in the assessment of safety in a future SFL. However, the view of chlorine in nature is changing. Chloride, which has previously been regarded as the dominant form, has proved to be reactive, and the quantity of organic chlorine is much higher in many environments than the quantity of mobile chloride. The cycling of chlorine in terrestrial ecosystems is to a high degree influenced by biological processes such as uptake in vegetation and chlorination of organic matter in, for example, superficial soil layers. However, little is known about how chlorination and dechlorination processes are regulated and how these processes affect transport under different environmental conditions. SKB is therefore planning an in-depth evaluation of the distribution pattern of chlorine in terrestrial ecosystems on the investigated sites, with the objective of relating the observed pattern to processes and properties in soil and vegetation. The work may require supplementary site data (see Section 27.5 "Biogeochemistry") and modelling.

Wetlands can expand from former lake basins, resulting in waterlogging and further peat growth in the surrounding area. The expansion of fen and bogs is primarily determined by precipitation and topography, and the horizontal expansion of peatland in a flat landscape can thereby be affected by changes in climatic conditions. Peatland expansion in turn affects where arable land is located, which is of importance for the dose calculations. Peatland expansion in a colder climate (such as Greenland) will therefore be compared with peatland expansion in Forsmark to understand how peatland expansion will change in a future landscape in Forsmark.

27.4 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, see Section 27.9 "Radionuclide modelling". 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 vital when it comes to assessing the effects of a 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. Figure 27-4 shows the most important processes for transport and accumulation in lakes.

In many cases, potential discharge areas for radionuclides from a 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 may 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, previously accumulated radionuclides may be released in connection with e.g. resuspension, land uplift or land cultivation and give rise to elevated doses during limited periods.

Conclusions in RD&D 2010 and its review

In RD&D Programme 2010, SKB planned to continue development of models for transport and accumulation of radionuclides in aquatic systems in a Ph. D. study, and to further develop the succession model for transition from marine to limnic systems. 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, see Sections 27.3 "Terrestrial ecosystems" and 27.6 "Hydrology and transport" for newfound knowledge.

In its comments on RD&D Programme 2010, SSM pointed out that SKB should expand the studies of geochemical behaviour in the aquatic systems to include more substances than carbon.

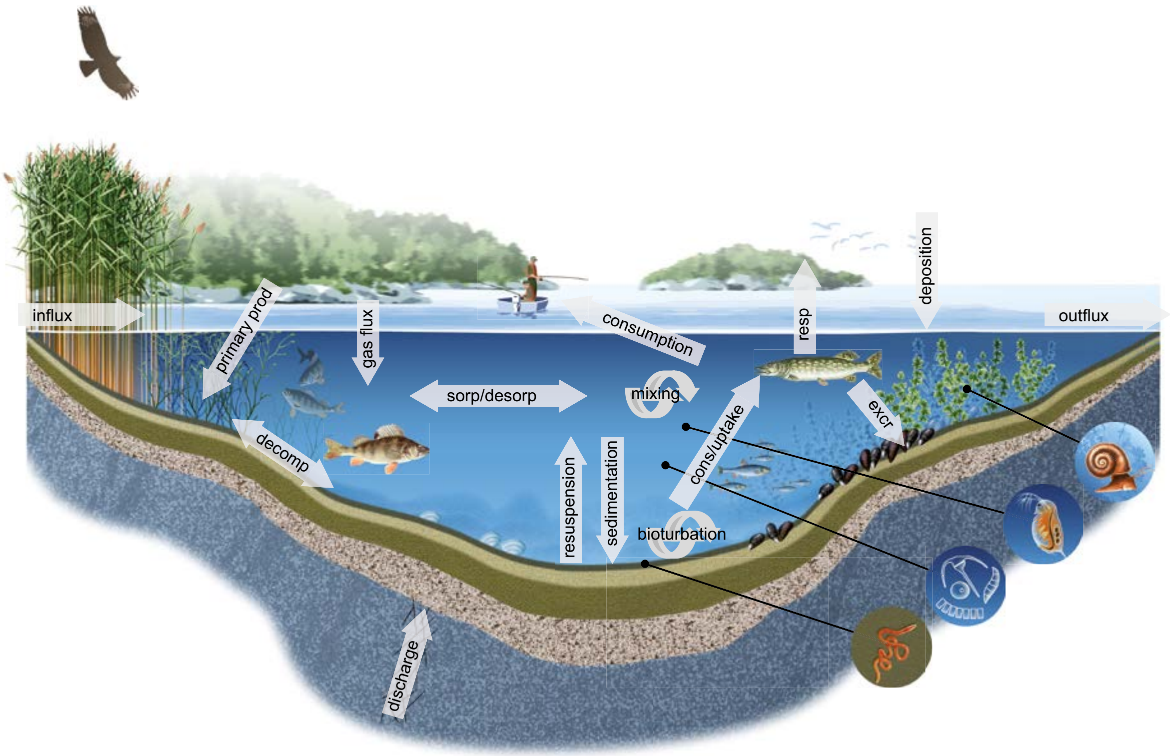

Figure 27-4. *Important processes for radionuclide cycling in aquatic systems, with an example from a lake ecosystem (Andersson 2010). The arrows indicate different processes and their dominant direction. The processes in the sea off Forsmark are similar, but in some cases other organisms dominate. All processes are important in running waters.*

Newfound knowledge since RD&D 2010

An important milestone in SKB's efforts to gain a better understanding of important processes in aquatic ecosystems was passed with the publication of descriptions of marine ecosystems (Aquilonius 2010) and limnic ecosystems (lakes and streams) (Andersson 2010). The reports summarize the results of the site investigations and compare them with the results of other studies in Sweden and the world. In addition, models have been developed to describe important fluxes of organic carbon (see Section 27.2 "Basic principles for description and modelling of surface ecosystems") and a number of other elements.

Sea currents are of crucial importance for fluxes and accumulation of various elements during the marine period. For this reason, a supplementary hydrodynamic model has been developed to describe water exchange in Öregrundsgrepen during the period from 6,500 BC to 9,000 AD (Karlsson et al. 2010, Eriksson and Engqvist 2013).

Dispersion and uptake of different elements during the marine period has been simulated for Öregrundsgrepen by means of two mechanistic models: a model with high spatial and temporal resolution, Ecolab (Erichsen et al. 2010), and the previously developed SKB model (Kumblad et al. 2003, Kumblad and Kautsky 2004, Konovalenko et al. 2013). In the Ecolab model, a detailed description of ecological processes is coupled with the hydrodynamic model in order to mechanistically describe fluxes of carbon, nitrogen and phosphorus in three dimensions. In the simulations, concentration ratios for phytoplankton varied during the year and in different parts of the water mass. A comparative study of the two models (Erichsen et al. 2013) shows that the previous SKB model is coarser but faster, but that the results of the two models are in reasonable agreement. Both models provided confidence intervals for modelled values of concentration ratios, which is a clear improvement over previously used models. In the coming work, the models will be able to be used as tools to investigate the extent to which different organisms are affected by radionuclide releases to marine ecosystems.

A better understanding has been achieved of how stable elements and radionuclides are distributed in the marine basins in the Forsmark area by modelling done in a Ph. D. study (Konovalenko 2012, Erichsen et al. 2013, Konovalenko et al. 2013). A study of the chemical composition (stoichiometry) in different organisms and organic matter in the coastal ecosystem in Forsmark has been carried out by analysis of 48 different elements (Bradshaw et al. 2012). This study shows, by means of modelling, the strength of using stoichiometry to model transport of elements in food webs. Aside from this, mass balances on the landscape level have been calculated for a large number of elements within SR-Site. A discussion of what influences the transport of these elements (including timescales) can be found in Tröjbom and Grolander (2010). Mass balances for carbon and other elements in individual lakes (Andersson 2010) and marine basins (Aquilonius 2010) have also been described.

Studies in the Krycklan catchment have provided information on fluxes of elements in streams and mire systems. Water flows and the composition of the catchments control the input of terrestrial carbon to the streams in the Krycklan catchment (Laudon et al. 2011). Dissolved organic matter is taken up in food webs via bacteria, and fluxes of organic carbon from terrestrial to aquatic ecosystems can be of great importance for the aquatic systems (Laudon et al. 2011). Other important elements studied in Krycklan are selenium, uranium and thorium, whose dynamics in streams is discussed in Lidman et al. (2011, 2012).

Knowledge concerning landscape succession and the effects on the ecosystems when they transition from marine to limnic systems due to land uplift has been increased by a compilation of existing knowledge (Hansen 2012). This study covers effects on vegetation and animals as well as effects on water temperature, water exchange and salinity. 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 (Brydsten 2006). The model, which is based on empirical relationships, has since been refined during the work with SR-Site. Establishment of the common reed has been found to be of great importance for when and at what rate the basins become infilled. A supplementary study of reed has been conducted to enable the model to be updated so that it can better handle infilling (Strömgren and Lindgren 2011).

In addition to the fact that reed serves as a step in the infilling of marine basins and lakes, aquatic plants (such as stoneworts and perfoliate pondweed) are important in SKB's radionuclide transport models since uptake of certain radionuclides, such as carbon-14, is assumed to occur via photosynthesis. During the site investigations in the Forsmark area, SKB carried out several studies concerning the occurrence, biomass and production of aquatic plants in aquatic systems in the Forsmark area, which have been summarized in the limnic and marine ecosystem reports (Andersson 2010, Aquilonius 2010). Since then, field investigations have been conducted to validate the marine vegetation model (Aquilonius et al. 2011) and to supplement biomass estimates in streams (Andersson et al. 2011).

In addition to previous investigations of lakes in the Forsmark, supplementary studies of four ponds have been conducted. The ponds exhibit a hydrochemistry similar to that of the lakes, which means that they are well-buffered and exhibit a high pH and high calcium concentrations (Qvarfordt et al. 2010, 2011).

A better understanding of important processes in lakes in a permafrost landscape is being obtained by the investigations of lakes in Kangerlussuaq in Greenland. A data compilation is under way and will be reported in the coming years, see further Section 27.7 "Effects of long-term variations".

Programme

As mentioned above, SKB has developed mechanistic ecosystem models for the aquatic environment, where the fluxes are controlled by primary production, consumption, respiration and degradation. The unit for the description is organic carbon, which represents the energy that is cycled in e.g. food. Cycling 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 different elements. SKB will support the continued development of these models, with the goal of describing cycling and accumulation of radionuclides in organisms and organic matter in seas and lakes as well as in bordering wetlands. Model descriptions will be validated with already collected data from Forsmark. These models will be further developed in the ongoing Ph. D. study at Stockholm University (Konovalenko 2012), and the work will focus on the development of the lake models.

The studies in the Krycklan catchment, together with other scientific literature, suggest that terrestrial organic carbon may be of great importance in the aquatic food web when it is taken up by bacteria and carried further up in the food web (Hessen and Tranvik 1998, Jansson et al. 2000, Berggren et al. 2010, Laudon et al. 2011). This entails a dilution factor with respect to carbon-14 in the aquatic food web, since today's models are based solely on the production of aquatic plants (including algae and bacteria), where inorganic carbon-14 in the water is taken up by the primary producers and transported further in the food web. Supplementary field investigations and modelling are planned to investigate the potential importance of this dilution of carbon-14 in lakes in the Forsmark area.

In the previous safety assessment for SFR, SAR-08, the highest dose to humans was obtained from fish in lakes. Test fishing has been carried out in both lakes and coastal areas in the Forsmark area, and fish migration has been measured in streams (Andersson 2010, Aquilonius 2010). In order to be used in the modelling of radionuclide transport, the test fishing results must be converted to biomass per unit area and production per unit area, which requires a number of conversion factors. These conversion factors have been associated with great uncertainties. However, new studies show that echo sounding can provide a more certain estimate of biomass per unit area in certain types of lakes (Emmrich et al. 2012). A review of the state of knowledge in this field is planned, and depending on the results, supplementary field work may be undertaken in the Forsmark area. The review should also cover the increasing body of knowledge regarding fish migration between lakes and coastal basins, since this can also influence the ecosystem models and mass balances used by SKB, which in turn influence dose calculations.

27.5 Biogeochemistry

Studies of concentrations of different element in the surface systems, i.e. biogeochemistry, can furnish 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. Most elements are present in relatively constant proportions in organisms. Comparing the stoichiometric distribution of the elements in organisms with their distribution in the water and the soil can provide good guidance regarding the origin of the elements and the processes that control their large-scale distribution pattern 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.

Distributions of elements in surface ecosystems can also be used to understand how radionuclides are transported and accumulated in the environment. Stable or long-lived isotopes of many radionuclides occur naturally, for example carbon, iodine, thorium, radium, uranium and nickel. In the case of other radionuclides, the properties of a similar stable element, for example potassium compared with cesium-137, or the properties of an entire group of elements, for example 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. Partition coefficients (K_d) describe the distribution of elements between solid and aqueous phases, for example in deposits or on particles in surface water systems. Bioconcentration 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.

Estimating element ratios for a specific site based on general literature reviews often involves great uncertainty, since it is difficult or impossible to relate literature data to site-specific biological, chemical, and physical conditions. In SR-Site, site-specific estimates of element concentrations were used wherever possible to calculate concentration ratios and K_d (Nordén et al. 2010).

Conclusions in RD&D 2010 and its review

It was stated in RD&D Programme 2010 that collected data would be further processed and evaluated and supplemented with some additional sampling in Forsmark. SSM said in its review that SKB should develop a more systematic methodology for selecting K_d values. Furthermore, SSM said that SKB should study geochemical behaviour for more elements than carbon, for example iodine, uranium and thorium.

Newfound knowledge since RD&D 2010

In the reporting of the results of SR-Site, SKB published a compilation of knowledge concerning different elements based on results from the site investigations and recently published literature (Nordén et al. 2010).

During the site investigation programme, SKB characterized concentrations of different elements for regolith, pore water and surface water, as well as for a large number of terrestrial and aquatic organisms (Johanson et al. 2004, Engdahl et al. 2006, 2008, Hannu and Karlsson 2006, Roos et al. 2007, Brun 2008, Kumblad and Bradshaw 2008, Grolander and Roos 2009, Sheppard et al. 2009, Tröjbom and Nordén 2010). The resulting database is unique, both in the quantity of element-specific information that has been collected from two distinct geographic areas and in the systematic and synchronized sampling that lies behind the site investigations. This biochemical knowledge was summarized in the reports published within the SR-Site safety assessment (Tröjbom and Nordén 2010).

Site data was used wherever possible to calculate concentration ratios for SR-Site. SKB updated earlier literature compilations (Karlsson and Bergström 2002) with current data, including from the IAEA's database (IAEA 2010). Site and literature data were 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). All available information was used to estimate concentration ratios, but data from the site was weighted more heavily than data from other sites.

The chemical composition of abiotic and biotic material from a sea bay in the Forsmark area has been studied in detail (Kumblad and Bradshaw 2008). This study investigated 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. It was shown in a scientific article (Bradshaw et al. 2012) that the relative abundance of macronutrients, such as nitrogen and phosphorus, increased upward in the food web, at the same time as the number of elements contributing substantially to biomass decreased. By coupling the ratio between carbon and other elements to an ecosystem model based on carbon fluxes, it was shown that stoichiometric relationships can predict how different elements are transferred in the food web. Most stoichiometric imbalances could be explained by ecological or biological processes.

Additional sampling and analyses of e.g. agricultural land and crops were performed prior to the safety assessment for the extension of SFR, SR-PSU (Sheppard et al. 2011, Sohlenius et al. 2013). Pore water and solid material from five different soil types that represent the main regolith types that can be used as arable land in the area (clayey till, glacial clay, clay-gyttja, and cultivated and undisturbed peat) were analyzed for element content and the results were then used to calculate K_d values. The chemical composition of the grain, stems and roots of cereals from several of these localities was also analyzed and used together with soil chemical data to estimate concentration ratios. The biochemical information from Sheppard et al. (2011) and Sohlenius et al. (2013) has been included in the geochemistry database compiled within SR-PSU and will be used to determine the K_d values and concentration ratios that will be used in the safety assessment for an extended SFR, SR-PSU.

Site data has continued to be given top priority in the determination of element ratios in SR-PSU. Site data from Forsmark have been regarded as most relevant for making a best estimate of both K_d values and concentration ratios. In addition to sampling at the sites, laboratory measurements have been made of the sorption properties of soil samples. Sorption was studied for seven elements (iodine, cesium, strontium, nickel, europium, uranium and neptunium) in till, sand, clay, clay-gyttja, gyttja and peat (Holgersson 2009). Evaluation of the results of these studies is continuing.

In parallel with the development of how site-specific information is used to estimate element concentrations, SKB is conducting long-term research to gain a better process understanding of retention and biological uptake for different elements. An example is the ongoing Ph. D. study in the Krycklan catchment, where the turnover of selenium, uranium and thorium in boreal systems has been investigated and where process-based modelling of radionuclide retention has been developed in a series of studies, see further Section 27.6 "Hydrology and transport". SKB has also used mechanistic descriptions of the biological processes that control uptake and accumulation in modelling studies, among other things to study spatial and seasonal variation of element concentrations in a marine food chain, see Section 27.4 "Aquatic ecosystems".

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The information that has been used to describe element concentrations on SKB's investigation sites is unique. In the programme for biogeochemistry in the surface system, collected data will continue to be processed to gain a better understanding and improve the description of retention and biological uptake on different spatial scales.

Despite the large quantity of data, uncertainties still exist in many cases when it comes to partition coefficients and bioaccumulation for a number of elements that may be important from a dose viewpoint in safety assessments. An example is radium, which has been measured in Forsmark but has only yielded detectable results in a limited number of samples from organisms. An evaluation is expected to provide an answer to whether existing data can be used to support parameterization of models, for example by the use of analogous elements and/or organisms. After such an evaluation, supplementary sampling may have to be carried out or the monitoring programme may have to be modified. Important elements for future safety assessments are deemed to be radium (including the uranium decay chain), carbon, iodine, cesium, nickel, molybdenum and chlorine. All of these elements except cesium are biologically active, and in the case of carbon, iodine and chlorine, biological processes affect the form in which the elements occur. After evaluation, targeted field work will be done to measure processes that are deemed to be quantitatively important.

Site data for concentration ratios sometimes differ from corresponding values reported in the literature (Torudd 2010). In general, the biggest differences appear to exist among those elements where the literature data is based on only a few observations. In the case of terrestrial and limnic ecosystems, there is a tendency for site data to give lower concentration ratios than literature data; the reason for this will be explored in future programmes. SKB intends to compare concentration ratios in a salinity gradient (lakes, sea bays, open sea) on the investigated sites to see if there are significant differences between the different biotopes.

In addition to the fact that site information provides input data for calculation of concentration ratios and K_d values, the concentrations of different elements are important for an understanding of different processes or interactions between different elements. The process-based modelling comprises a complement to or substitute for concentration ratios and K_d values, see further Section 27.6 "Hydrology and transport". The further evaluation of site data will therefore also include important control variables – such as abiotic environment, concentration of nutrients and species composition – to furnish quality-assured data for the process modelling, but also to evaluate/modify the environmental monitoring programme, see Section 27.10 "Environmental monitoring".

27.6 Hydrology and transport

This section describes research and development activities in the field of surface hydrology and transport, with a focus on activities related to identification and description of biosphere objects in the safety assessment's biosphere models. Thus, only part of the development work in surface hydrology and transport is presented here. Other development work, along with equivalent activities focused on the rock, is described in Section 26.4 "Groundwater flow". Development related to description of retention processes and reactive transport is also described here, while activities aimed directly at determining K_d values have been presented above, in Section 27.5 "Biogeochemistry".

Transport modelling is a central component in both dose modelling and ecosystem modelling. However, 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, activities associated with site description as well as safety assessment and environmental impact assessment (EIA) are discussed in Section 26.4 "Groundwater flow", while generation of hydrological input data to biosphere modelling is described below.

Conclusions in RD&D 2010 and its review

It was noted in RD&D Programme 2010 that initiated efforts to link newfound knowledge from the Krycklan study to SKB's investigation areas would continue, as would the process-based modelling of advective-reactive radionuclide transport for the purpose of including more parts of the surface system.

Similarly, a continuation was planned of the work with mass balances as a way to describe and predict the distribution of elements in the landscape, along with the work of characterizing the size and locations of discharge areas and the processes that drive transport. Further development of the calculation tools used for modelling of hydrology (water flows) and advective transport of solutes was also planned.

In its review of RD&D Programme 2010, SSM recommended that SKB integrate the consequence analysis for the biosphere and the geosphere. Further, SSM shared SGU's view that groundwater flow beneath the sea surface should be taken into account in modelling of groundwater flow and also wanted SKB to strive for a better-substantiated extrapolation of the current hydrological model to future conditions.

Newfound knowledge since RD&D 2010

During the past programme period, development of models for flow and transport in the surface system has mainly been pursued within the SR-Site safety assessment and via modelling activities in support of this. Particular mention should be made here of the modelling runs with MIKE SHE that were carried out for the purpose of investigating the discharge of groundwater from the rock and transport conditions within potential biosphere objects in Forsmark. The modelling was done with a relatively high resolution (cell size 20 metres in the horizontal plane) and consisted primarily of particle tracking and modelling of transport with advection and dispersion of solutes, without taking retention process into account. The focus of the modelling was on conditions in 10,000 AD in subareas with objects included in the biosphere modelling's landscape and radionuclide transport models. Models and modelling results are described in the reporting of SKB's surface hydrological modelling in SR-Site (Bosson et al. 2010) and have also been presented in articles in scientific journals (Berglund et al. 2011, 2013a, b).

In the safety assessment, particle tracking is used in the hydrogeological models used in geosphere modelling to calculate where groundwater that has passed through the repository volume, and might therefore contain radionuclides in the future, reaches the surface system, where exposure to radiation can take place. These discharge points serve as a basis for the identification and delimitation of the biosphere objects that are used to model radionuclide transport and its consequences. The uncertainties in the locations of the discharge points have been investigated by parallel modelling with MIKE SHE, which permits a more detailed representation of hydrological processes near and on the ground surface. The results reveal certain differences between the models, but also show that these differences are not of any great importance for the description in terms of biosphere objects (Lindborg 2010, Berglund et al. 2013b).

The spread of solutes in the groundwater when they reach the surface system has been studied by advection-dispersion modelling with MIKE SHE. In this modelling, sources were placed at a depth of about 40 metres beneath the ground surface between the repository and the surface, after which transport in the upper part of the rock and in the soil layers was studied. The results show that transport often takes place vertically up to the top soil layer, where horizontal spreading can then occur (Bosson et al. 2010), see Figure 27-5. Exceptions with e.g. more extensive spreading in the uppermost part of the rock do occur, however, and it is difficult to identify typical migration patterns or plume sizes from these results. The analysis also illustrates that delimitations of plume sizes and affected areas are strongly dependent on what concentration is assumed as a boundary towards surrounding "pure" water (Lindborg 2010).

A new, or at least considerably improved, feature of the safety assessment is the coupling between surface hydrological modelling and the model used to model radionuclide transport and doses, see Section 27.9 "Radionuclide modelling". In SR-Site, detailed hydrological calculations were done for today's conditions for a number of selected areas with lakes surrounded by wetlands (Bosson et al. 2010). Water balances and fluxes between different units, for example geological layers, that correspond to "boxes" in the radionuclide transport model, were calculated for an average object and transformed to flow parameters in the transport model. In this calculation, the fluxes in the average object were converted to parameters that could be scaled with the sizes of biosphere objects and their catchments. This was done so that all biosphere objects could be modelled with their respective geometric data (see Löfgren 2010 for details). Calculations of flow components for use in the radionuclide model were done for both tempered and periglacial domains, where data in the latter case were obtained from a talik in the MIKE SHE model for periglacial conditions in Forsmark, see Section 26.4 "Aquatic ecosystems".

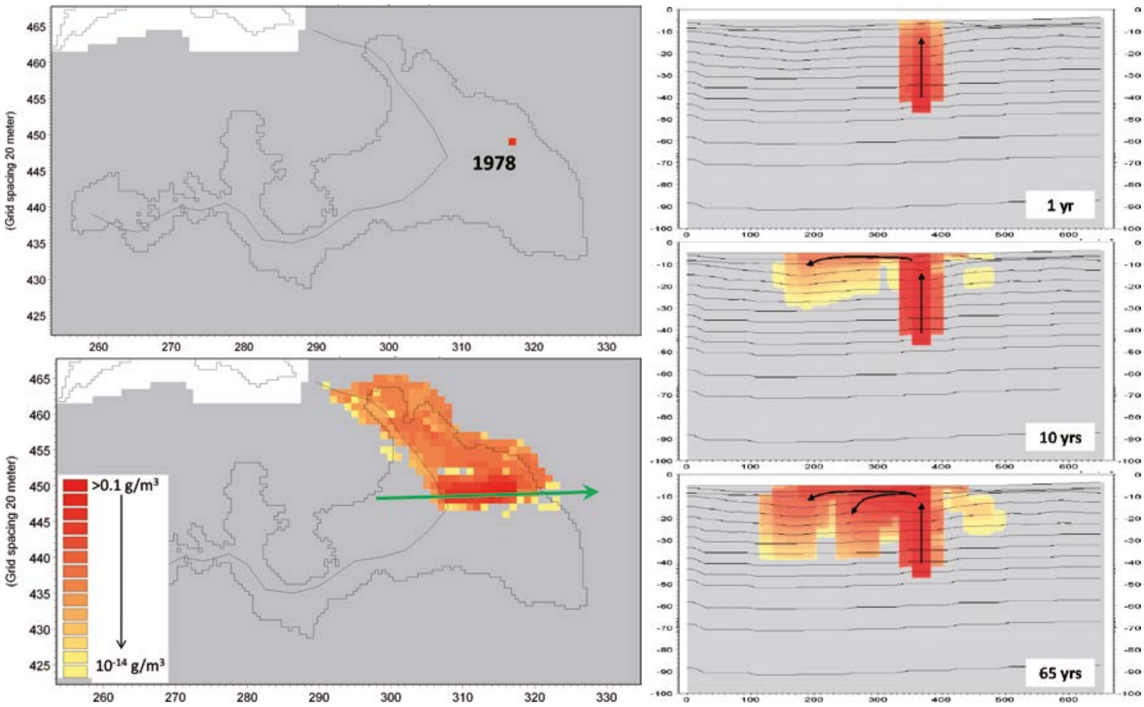

Figure 27-5. *The left side shows a terrestrialized lake from above (line indicates original shoreline), with calculated concentrations at the surface. Darker colour indicates higher concentrations. On the left side, the upper figure shows the discharge point (the source) at a depth of 40 metres, while the lower figure shows the concentration at the surface after 65 years. The right side shows concentrations after 1, 10 and 65 years along the profile marked with a green arrow in the left-hand map (Bosson et al. 2010).*

During the past few years, development of conceptual and numerical models for reactive transport in different types of soil material has been pursued in a series of projects. The projects have primarily been carried out within the framework of site investigations and safety assessments. The modelling done in conjunction with SR-Site is presented by Piqué et al. (2010). Based on site data from Forsmark, the objective has been to identify and describe retention processes, for example different kinds of sorption and precipitation processes, that could affect radionuclide transport in the soil layers on the site. The identified processes have thereafter been quantified as far as possible by numerical modelling of reactive transport, whereby advection and dispersion have been combined with so-called process-based or mechanistic modelling of retention. The intention has been to support the description in terms of primarily empirically grounded partition coefficients (K_d values) used in the safety assessment. The results of this development work have also been presented in scientific publications (Grandia et al. 2011, Piqué et al. 2013a).

Comparisons between measured K_d values and calculated values from the reactive transport modelling show qualitative agreement, which means that the same elements are retarded to an equal extent. The quantitative differences are in some cases great, and measured K_d values are usually greater than the calculated ones. This indicates that some work remains to be done with process identification and quantification of parameter values before quantitative agreement can be achieved. The modelling has increased progressively in terms of complexity and the number of elements included. In the most recent stage, decay chains have been introduced in the numerical transport model and the effects of uncertainties in parameter values have been investigated by probabilistic sensitivity analysis (Piqué et al. 2013b). In the reporting of this modelling, it is concluded that retention and decay of thorium under certain conditions can be of importance for the generation and transport of radium.

SKB has for a number of years funded research in the Krycklan project, which is a research programme that is investigating numerous questions in hydrology, hydrogeochemistry and element transport by extensive field measurements. The research is based on a large body of data that includes hydrological data – regarding e.g. surface water flows from catchments of various sizes and variations in other properties – and concentration data from long-term measurements of elements and radionuclides of interest to SKB. Measurements are being conducted in an area with lakes, wetlands, streams and forest areas similar to those in Forsmark. SKB's support has mainly gone to a Ph. D. study and a postdoc study. The Ph. D. study, which will be concluded during 2013 as planned, has resulted in a number of publications describing element transport on different scales in the studied landscape (Lidman et al. 2011, 2012, 2013).

Lidman's research has investigated the differences in the export of elements such as uranium, thorium and selenium from catchments with different proportions of wetlands (mires) and forest (Lidman et al. 2011, 2012). The results show the great influence of the wetlands on the transport of elements on the landscape level. The studied elements are accumulated to a much greater extent in wetland-dominated catchments than in forest-dominated catchments. The differences between different catchments were found to be systematic and predictable. In a detailed study of a selected mire, significant differences in activity concentrations of radioactive isotopes were observed between different parts of the mire. Comparatively high concentrations were noted in areas where inflow of groundwater could conceivably occur, while concentrations were significantly lower nearer the outlet of the mire. It was concluded that it was probable that only a small part of the mire's capacity for accumulation of radionuclides had been utilized and that it would therefore continue to act as a sink for a long time to come.

Programme

The details in the continued programme for hydrology and transport can be found in Chapter 26 "Geosphere". The application of this knowledge is described in Sections 27.3 and 27.4 for the different ecosystems, as well as in connection with the studies of biogeochemistry (Section 27.5), landscape evolution (Section 27.8) and radionuclide modelling (Section 27.9). Studies of the transition between geosphere and biosphere are of great importance for surface ecosystems. This includes investigations and modelling of discharge beneath sea and lake, as well as process-based studies of K_d. Furthermore, the ongoing research programme in the Krycklan catchment will continue with feedback to equivalent phenomena in the Forsmark area and comparative studies on Greenland under permafrost conditions.

27.7 Effects of long-term variations

SKB's understanding of how elements are transported and accumulated in surface ecosystems is mainly based on descriptions of phenomena and processes that can be observed today in Forsmark and Laxemar-Simpevarp. Many processes are climate-dependent and vary in intensity as the climate varies. 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 rebound to a new state of equilibrium after depression of the bedrock by ice sheets. Salinity changes in the Baltic Sea offer another example where 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. SKB's climate programme is described in Chapter 19 "Climate evolution". The programme for surface ecosystems describes the long-term effects that are of great importance for an understanding of the surface ecosystems.

Conclusions in RD&D 2010 and its review

RD&D Programme 2010 described plans for a compilation of the function of the surface systems during a permafrost domain. SSM took a positive view of the work being pursued within the Greenland Analogue Project, GAP, and considered it important that the results be interpreted appropriately with a view to the conditions for existing and planned repositories in Forsmark. SSM also found it urgent that SKB realize its plans to conceptualize flow models during periods of glaciation and permafrost based on results from the GAP. SSM further pointed out that SKB should gather site data not only from today's selected site, but also from sites where data can represent properties of ecosystems that are tens of thousands of years old. 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 2010

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 (Kjellström et al. 2009) (see Chapter 19 "Climate evolution"). Based on this description and the relevant scientific literature, SKB has 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 (Andersson 2010, Aquilonius 2010, Löfgren 2010). Surface hydrology and water transport have been modelled for climates that are both colder and warmer than today (Bosson et al. 2010). The effect of a colder climate on terrestrialization of lakes and formation of taliks (areas of unfrozen soil under e.g. lakes) has also been modelled (Brydsten and Strömgren 2010, Hartikainen et al. 2010). The results have been used in SR-Site to describe surface ecosystems during hypothetical future climate domains (Lindborg 2010).

A new model for shoreline displacement during the entire reference evolution has also been developed in conjunction with the SR-Site safety assessment (Lindborg 2010). The new model consists of a combination of the previous model, which described shoreline displacement from the last deglaciation until today (Söderbäck 2008), and a model for future shoreline displacement (Whitehouse 2009).

The Greenland Analogue Surface Project, GRASP, was started in 2011 after a field visit and a literature review of the prospects of a climate study as an analogue of a colder climate (Engels and Helmens 2010, Nielsen 2010, Clarhäll 2011). GRASP began with data collection, descriptions and modelling of a catchment on Greenland. The project is focusing on solute transport and mass balances at an ecosystem and catchment level, see Figure 27-6. The results will also be used to describe how the coupling between surface ecosystems and the hydrology in the rock is affected by periglacial conditions. The understanding gained from these investigations will be used in the next step for comparative analyses with Forsmark.

Programme

During the coming programme period, SKB will describe currently existing periglacial environments by continuing to work in GRASP. The understanding gained from these investigations will be used in the next step for comparative analyses with Forsmark. The results of the GRASP 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. Studies and modelling of selected catchments in a temperature gradient from temperate to periglacial conditions are planned during the programme period. The results will be used to gain a better understanding of how processes and phenomena that drive transport and accumulation of radionuclides may be affected by changed climatic conditions.

Figure 27-6. *Picture of a permafrost landscape at SKB's investigation area in Kangerlussuaq, Greenland. Currently existing environments are used as an example of Forsmark in the future to understand long-term changes of the landscape in Forsmark. Here an area with constantly frozen ground and other conditions that can be expected to prevail during colder periods is being studied.*

27.8 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 (Söderbäck 2008). The historical description has been combined with an understanding of the present-day appearance and function of the landscape (SKB 2008c, 2009c), and 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 and SR-PSU safety assessments, SKB has 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 (Lindborg et al. 2006, 2013, Lindborg 2008, 2010).

The principal phenomena that drive landscape evolution in Forsmark are climate variation and shoreline displacement, see Chapter 19 "Climate evolution". During the Holocene, natural climate changes has led to a succession from ice-age marine conditions to a temperate climate where the landscape is ultimately dominated by terrestrial ecosystems (Söderbäck 2008, Lindborg 2010). On top of these large-scale and slow changes, the landscape is affected by the deposition and reworking of sediment, which entails that lakes become shallower and infilled (Brydsten and Strömgren 2010). As the ecosystems succeed one another, the chemical properties of the soil and water change, along with the species composition of plants and animals.

Hydrological discharge areas for deep groundwater are located in topographical low-lying points in the landscape, for example around lakes, streams and wetlands, see Section 27.6 "Hydrology and transport". 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 27.6.

Conclusions in RD&D 2010 and its review

In RD&D Programme 2010, the objective was to further develop methodology used to describe landscape evolution so that more process-driven parameters could be included in the landscape evolution model. In its review of RD&D Programme 2010, SSM recommended that SKB use information from not just today's selected site, but also from sites where data can represent properties and increase process understanding of other types of potential future ecosystems.

Newfound knowledge since RD&D 2010

Since RD&D Programme 2010, SKB has developed a new version of the landscape evolution model for Forsmark (Lindborg 2010), see Figure 27-7. As a part of the SR-Site safety assessment, the understanding gained from the historical landscape evolution and from models of future landscape evolution has been used to devise descriptions of possible future ecosystems and landscape in Forsmark (Andersson 2010, Aquilonius 2010, Bosson et al. 2010, Löfgren 2010, Piqué et al. 2010, Tröjbom and Grolander 2010).

These descriptions are based on the new model for shoreline displacement (Brydsten and Strömgren 2010, Lindborg 2010) and cover the entire current interglacial (with periods of permafrost) as well as climate variants that could arise due to global warming. The process understanding gained from the site has been translated into numerical models that describe the hydrodynamics in the sea (Karlsson et al. 2010) (see Section 27.4), the future accumulation and erosion of sediments in the sea and on land (Brydsten 2009) for an entire interglacial, and how near-shore marine ecosystem are affected by shoreline displacement (Hansen 2012). The hydrology associated with future landscapes and climates has been linked to this, see Section 27.6. 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 (Saetre et al. 2013, Sohlenius et al. 2013).

Programme

During the coming programme period, SKB will work to gain a deeper 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 a conceptual scientific understanding and on improving the description of processes and parameters that are of crucial importance for safety assessments.

Efforts aimed at gaining a better understanding of how phenomena and processes in the landscape change with different climates will continue. By utilizing analogous sites with a different climate than the one in Forsmark today, the conceptual understanding of radionuclide transport at a landscape level will be strengthened. An example is the current work in a catchment at Kangerlussuaq on Greenland and in the Krycklan catchment in Västerbotten.

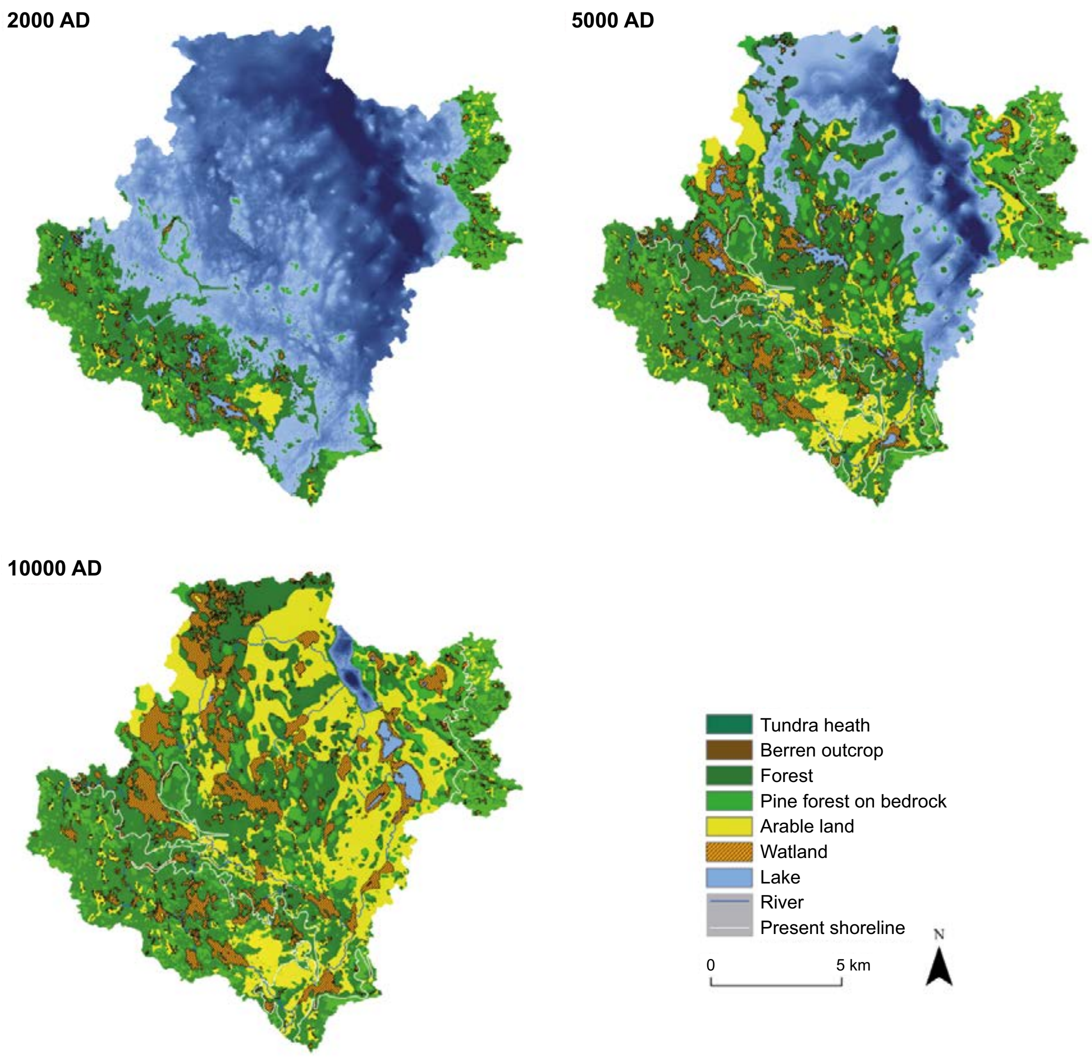

__Figure 27-7.__ Possible future evolution of the landscape in Forsmark.

Development of the conceptual understanding of how radionuclides are transported and accumulated in a changing landscape will also be based on results from simulation studies and planned empirical studies of mass balances at the landscape level in Forsmark, in the Krycklan catchment and on Greenland. According to the plan, the knowledge will be compiled in a conceptual model of the landscape that will cover several possible types of land use and future climates. Studies will be carried out of possible future discharge areas for deep groundwater and deep gases to permit a better description of radionuclide transport to and within these areas in cooperation with the geochemistry programme and the hydrology programme.

Preliminary analyses show that the estimated infilling rate for lakes, which plays an important role in the dynamics of landscape evolution, is associated with uncertainties. Knowledge from the planned biosphere programme will therefore constitute an important basis for refining models that describe the ecosystem succession from sea bay to wetland on the landscape level and how this succession is affected by climate change, see Sections 27.3 "Terrestrial ecosystems" and 27.4 "Aquatic ecosystems".

The calculated radiation dose to future inhabitants is based on assumptions and simplifications regarding how people will use the landscape and its natural resources. The primitive society that has been analyzed has been chosen to give the highest individual doses. SKB will continue to work to learn more about historical and contemporary self-sufficient cultures and human nutritional needs as a basis for an accurate description of future human living habits.

27.9 Radionuclide modelling

The research programme for surface ecosystems is aimed at gathering data for a reasonable assessment of the radiological risks to humans and other organisms associated with the final disposal of spent nuclear fuel and other radioactive waste. A central part of the programme is therefore using our understanding of processes and phenomena to simplify the description of the future evolution of the site 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. Since radiological risk is assessed for a time that lies thousands or tens of thousands of years in the future, it is not self-evident that consequence calculations based on knowledge of today's humans can be directly translated into actual health risk to future inhabitants of the site. The radionuclide concentrations in the environment (soil, sediments and water) are also combined with dose conversion factors and transfer factors to calculate exposure to other organisms than man.

Conclusions in RD&D 2010 and its review

Our objective in RD&D Programme 2010 was to further develop the methodology for dose calculations for application in safety assessments. The representation of processes that drive the cycling of carbon, along with underlying assumptions, was to be revised in detail prior to the safety assessment of an extended SFR. It was also a long-term ambition to supplement empirical transfer factors between the environment and human food with a mechanistic description of plant uptake and bioaccumulation.

In its review of RD&D Programme 2010, SSM took a positive view of the fact that SKB will take dilution effects into account in the coming SR-Site safety assessment. SSM thought that SKB should integrate the consequence analysis for the biosphere and the geosphere in order to improve the handling of certain processes in the transport modelling, for example radioactive decay. Furthermore, SSM recommended that SKB explore the possibility of improving the modelling of carbon-14 in terrestrial ecosystems based on model comparisons done in BIOPROTA, and that SKB should improve the validation of the carbon-14-model that is intended to be used in the safety assessment for an extended SFR. In the initial review of SR-Site, SSM identified important issues in the consequence analysis that partially concern SKB's biosphere programme. Among other things, SSM judged that there is a need for additional information when it comes to the representation of radionuclide distributions as homogeneous in biosphere objects, as well as the verification of models and calculation codes used.

The Swedish National Council for Nuclear Waste found in its review of RD&D Programme 2010 that viewpoints put forth by the Council in earlier reviews had been partially heeded. The Council observed that the model systems used by SKB are complex, and that it is therefore important that the model calculations be validated with field data from existing surface ecosystems in Forsmark. The Council found that the issues addressed by SKB in RD&D Programme 2010 are relevant in this context. However, the Council called for a more detailed description of SKB's projects, and an account of how SKB intends to utilize the models that will be developed in the continued work.

Newfound knowledge since RD&D 2010

Development work in the area of radionuclide modelling in surface ecosystems has been dominated during the past three-year period by the two safety assessments SR-Site and SR-PSU. The results of SR-Site were reported in an extensive compilation and application of site data and newly developed models (see summaries in SKB 2010t and Kautsky et al. 2013a). In SR-PSU, the modelling methodology has been further developed with a focus on a description of transport and accumulation of carbon-14.

In SR-Site, the consequences for man and environment of a possible canister failure were calculated, under the assumption of constant radionuclide release. To assess the safety of future human beings, calculations were also carried out for a pulse release of radionuclides. The calculations for both types of releases were summarized in landscape dose conversion factors (LDFs). The unit for LDFs is annual dose (sieverts) per release rate (bequerels per year) or per pulse (bequerels). The final dose is thus calculated by multiplying the release that reaches the surface by the appropriate dose conversion factor, and then summing over all groundwater-borne radionuclides that reach the surface. A constant release was also assumed for the assessment of dose to other organisms, but instead of reporting dose conversion factors, dose rates (grays per hour) were calculated directly (Torudd 2010). Altogether, transport and accumulation were simulated for 40 radionuclides in 18 potential discharge areas in the future landscape at Forsmark for the different climate scenarios, see Figure 27-9. The simulations were done with the tools Pandora and Ecolego.

The most important updates of the modelling methodology that were done in SR-Site had to do with the calculations of the activity concentration in the environment (Avila et al. 2010, 2013). The model that was used describes how the entire assumed release of radionuclides reaches a single discharge area. After having passed through the deeper till layers are radionuclides cycled in superficial soil layers, surface water and atmosphere, see Figure 27-8. The model describes the succession of ecosystems continuously over time, as a function of land uplift, sedimentation and lake infilling. The transition between sea and lake is described with an isolation time of around 500 years, after which the size of the lake gradually diminishes. Aquatic and terrestrial ecosystems occur simultaneously in the same discharge area, but the size relationships change over time (Lindborg 2010). It was assumed in the calculations that the entire release reaches the future discharge area where the consequences are greatest.

The methodology for calculating exposure of future humans inhabiting a given landscape object was also further developed in SR-Site. During the time a discharge area is just below or above sea level, humans in the landscape object are assumed to support themselves as hunter-fisher-gatherers. When the area is located far enough above sea level, it was assumed that most of the wetland is drained and used as agricultural land, or for grazing. The most exposed individuals were assumed to inhabit and get all of their food from the contaminated area, as far as possible. The individuals that got the highest exposure in the calculations, over all times and discharge areas, comprised the most exposed group in SR-Site (Avila et al. 2010).

In order to judge whether a potential release of radioactivity can pose a risk to the environment, dose rates to a broad spectrum of organisms were calculated in SR-Site with the Erica tool (Beresford et al. 2007, Brown et al. 2008). The calculated dose rates were several orders of magnitude below the level that poses a risk that radiation could affect reproduction and/or survival. Analyses with generic and site-specific data for biological uptake and/or morphology gave similar results, and the effect of parameter uncertainty did not affect the conclusions. From these results, SKB draws the conclusion that there is no reason to assume that any of the species on the site (Forsmark) would be harmed by the increased radiation exposure caused by a release (Torudd 2010, Torudd and Saetre 2013).

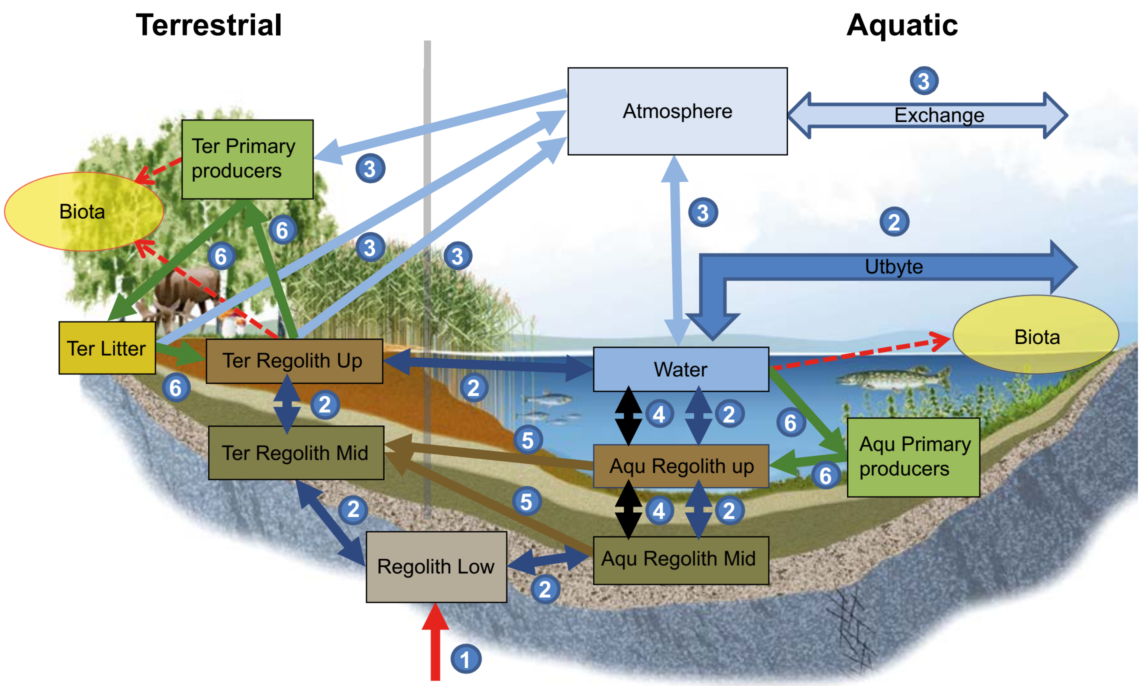

Figure 27-8. Conceptual illustration of the radionuclide model in SR-Site. Boxes represent compartments for radionuclides, thick arrows radionuclide fluxes linked to material fluxes, and dashed arrows uptake by organisms calculated with concentration ratios. The model consists of an aquatic part (at right) and a terrestrial part (at left), which rest on a common base of lower regolith and share a common atmosphere. The terrestrial and aquatic regoliths consist of an upper part and a middle part. There are plants (primary producers) in both environments. The terrestrial environment also contains a compartment that represents litter. The red arrow (1) represents the release from the geosphere. The type of material fluxes that carry radionuclides are indicated by colour: Radionuclide transport in dark blue indicates water flows, light blue gas flows, black sedimentation/resuspension, brown peat growth (terrestrialization) and green biological uptake/decomposition. Exchange with the environment (2 and 3) is illustrated by double-tipped arrows. A more detailed description is provided in Andersson (2010) and Avila et al. (2010).

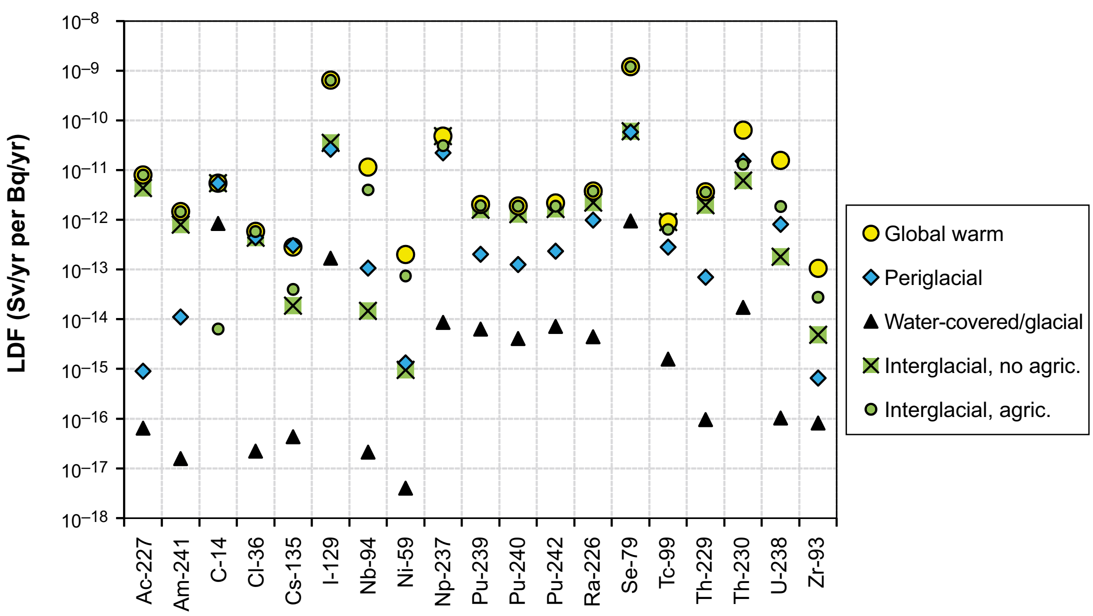

Figure 27-9. Maximum landscape dose conversion factors (LDFs) for 19 radionuclides calculated for different climate and land use cases. The LDFs for the initial marine period were used to represent a submerged landscape in the vicinity of the ice front (submerged/glacial). Figure modified from SKB (2010t).

In the ongoing safety assessment SR-PSU, SKB has further developed the modelling methodology. Model development has primarily been focused on improving the description of transport and accumulation of carbon, since carbon-14 has dominated the dose in previous safety assessments for SFR (main reports for SAR-08 and SAFE).

Plant uptake of carbon dioxide from the atmosphere is a potential exposure pathway for carbon-14. SKB has followed the comparison and evaluation of agricultural models for carbon-14 done in recent years in the organization BIOPROTA (Limer et al. 2012). As a natural extension of this work, Tagesson (2012) has compiled micrometeorological literature dealing with turbulent transport in atmospheric layers near or just above the vegetation, see Section 27.3 "Terrestrial ecosystems". SKB will apply this knowledge in the radionuclide model for SR-PSU.

The biggest carbon pools in the landscape are organic sediments and peat that have been accumulated under oxygen-free conditions in lake bottoms and wetlands. Carbon stored in these pools will be released when the land is drained and cultivated. The radionuclide model will be updated so that it better describes processes effecting accumulation and oxidation of radionuclides in organic matter. Assumptions regarding the land use and habits of future inhabitants have been modified based on historic self-sustaining cultures. In this way the limitations of the landscape and human nutritional needs can be taken into account in the safety assessments' stylized calculation cases (Saetre et al. 2013).

In SR-PSU, the exposure calculations for non-human organisms will continue to be based on the methods in the Erica tool, but the calculations will be integrated with the modelling of transport and accumulation in the surface ecosystems. This permits time-dependent calculations of exposure and a direct handling of uncertainties in calculated environmental concentrations. Great importance has also been attached to estimating realistic transfer factors for the organisms in question, with an emphasis on using site-specific information, see Section 27.5 "Biogeochemistry".

Programme

SKB will also continue its development of the methodology for dose calculations based on newfound knowledge from the biosphere programme and applications of the methodology to safety assessments. The review of SR-Site is under way, and during the coming three-year period the licensing process for the extension of SFR will also get under way. Furthermore, preparations are being made to calculate dose consequences from SFL. This means that supplementary model development, tests of models and summaries of consequences may be necessary to answer reviewers' questions or clarify previous work in the safety assessments.

Even though the models that represent the surface ecosystems are similar for the different repositories, requirements regarding resolution of processes and time periods vary with the type of radionuclides in, and the timing of, possible releases from the different repositories. The models may therefore need to be adapted to the specific questions.

The development work that has been done in SR-PSU and that includes e.g. carbon-14 cycling will be presented during the coming three-year period. When it comes to SFL, efforts are required to better describe the cycling of chlorine and accumulation of chlorine-36, molybdenum-93 and different nickel isotopes. Model development will primarily be based on collected data, see Section 27.5 "Biogeochemistry". The description of atmospheric exchange will be broadened from carbon dioxide to include other compounds that can occur in the gas phase, for example radon, iodine, selenium and chlorine. These models can also be used to study radiation releases during the operation of the facilities.

A long-term project to obtain a more realistic representation of the uptake of radionuclides in aquatic and terrestrial food webs is under way. The objective is to identify the dominant fluxes of organic matter, nutrients and water within the different ecosystems and the factors that regulate these fluxes, see Section 27.3 "Terrestrial ecosystems" and 27.4 "Aquatic ecosystems", in order to update the representation of biological uptake in the radionuclide modelling. It can replace or supplement the concentration ratios for organisms with mechanistic models. The work in each ecosystem identifies the dominant fluxes and uptake factors, at the same time as the radionuclide models are modified. The advantage of a more process-oriented description of biological uptake is that many processes are common to many organisms and that they are radionuclide-dependent. Moreover, it is possible to explicitly take into account how accumulation in the food web is affected by e.g. climate and other changes in the environment, which can reduce uncertainties.

A revision of the handling of the uranium decay chain, including radon transport, in the modelling of the surface ecosystems will be initiated. The revision is planned to include transport of and exposure to radon, and aims at shedding light on the consequences of different transport rates for the calculated dose. Furthermore, the review of site data (see Section 27.5 "Biogeochemistry") is expected to generate new insights that can be used to contrast measured concentrations with calculations, or validate limited parts of the radionuclide models.

SKB plans to reconsider how the temporal and spatial delimitation of the most contaminated areas affects the exposure of non-human organisms. SKB will therefore collaborate in a BIOPROTA project called SPACE (Scales for Postclosure Assessment Scenarios), see further Section 27.11 "National collaborations, international work and dissemination of information."

The Pandora tool that was used for dose calculations in SR-Site has been developed in cooperation with Posiva (Åstrand et al. 2005). The functionality of the tool will be evaluated in relation to SKB's long-term needs. It will also be compared with alternative tools, such as Ecolego and COMSOL. Today, dose calculations for non-human organisms are carried out with the Erica tool. SKB will also evaluate the possibilities of integrating the dose calculations for all organisms (including humans) in a common tool.

27.10 Environmental monitoring

A final repository can affect the environment in different ways, both during construction and operation and after closure. The potential impact can be both radiological and non-radiological. The potential radiological impact is evaluated in the safety assessments, whereas it is assumed that the non-radiological impact can take place during construction and operation and is addressed in the EIA process.

Prior to the start of the site investigations, a thorough survey was done of which variables in the surface system need to be described in a site investigation for a final repository (Lindborg and Kautsky 2000). The survey also identified which parameters need to be described with respect to variation in time. This served as a basis for the extensive programme for collection of both abiotic and biotic parameters that was carried out during the site investigations and that has been reported in connection with the site descriptions (Lindborg 2008, Söderbäck and Lindborg 2009). A reduced sampling programme has also continued after the conclusion of the site investigations in 2007.

Conclusions in RD&D 2010 and its review

In the review of the RD&D programmes 2007 and 2010, the Swedish National Council for Nuclear Waste took up issues related to environmental monitoring in several places. Among other things, the importance of studying the variation of the biosphere parameters in time was pointed out. It was further pointed out that it is urgent that the continued research and development work shed light on the selection of measurable parameters that can provide a picture of conditions in and around the final repository.

Programme

SKB plans to continue with the monitoring programme currently in place for Forsmark. If new knowledge from the biosphere programme shows that additional variables or measurement points are needed to support the biosphere modelling or to distinguish natural variation from effects of a future repository, the programme will be expanded accordingly. In order to distinguish natural variations from repository impact, time series from reference areas will also be needed for a number of key parameters. In the detailed characterization programme that will be designed prior to the start of construction of the repository, the need for reference data will therefore also be met. SKB judges that there is no need for targeted research efforts in this area, but an evaluation of the completed investigations and collected data needs to be done.

27.11 National collaborations, international work and dissemination of information

SKB is participating actively in a number of international cooperation fora concerned with radiological safety for man and the environment, including within IAEA 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 symposia and seminars.

Activities

SKB has participated in the IAEA's programme for modelling of radiological risk in the environment (EMRAS), and in 2010–2011 SKB chaired Working Group 3 (WG3) in EMRAS II. The working group's goal has been to support the updating of the IAEA's recommendations for biosphere modelling coupled to climate and environmental change for assessing safety in the final disposal of radioactive waste. From 2012, SKB's involvement in EMRAS has continued in MODARIA (Modelling and Data for Radiological Impact Assessments), where SKB is chairing WG6. SKB is also participating in WG8 and WG9, which deal with non-human organisms. In addition, SKB has participated in marine modelling in WG10.

SKB is also participating in the IAEA project HIDRA (Human Intrusion in the context of Disposal of RAdioactive waste), which started in 2012 and is planned to be concluded in 2015. SKB is chairing WG2, which deals with social aspects in the generation of scenarios for human intrusion.

SKB has acted as a reference group for the EU project STAR (Strategy for Allied Radioecology, http://wiki.ceh.ac.uk/display/star), where it has expressed the need to secure access to radioecological competence in the future.

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 that perform assessments of radiological risk such as SKB, Posiva, EDF and Nagra, and regulatory authorities such as SSM and NDA (Nuclear Decommissioning Authority). SKB has actively participated in the task groups for carbon-14, integration of geosphere and biosphere, and exposure of non-human organisms. SKB has also participated in the task 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 (see Section 27.7 "Effects of long-term variations" and Chapter 19 "Climate evolution") as well as with a number of other organizations: NMWO (Canada), NDA (UK) and GEUS (Denmark). The work on surface ecosystems has involved field sampling under permafrost conditions, and SKB has arranged a number of workshops on this subject.

An account of SKB's work on surface ecosystems and climate has been published in a special issue of the scientific journal Ambio (Kautsky et al. 2013b). Various issues of scientific interest are dealt with there in 14 articles aimed at exemplifying the work of depicting ecosystems and man in the future. The first article is a summary and overview of the special issue (Kautsky et al. 2013a). The work with climate is described in Näslund et al. (2013) and the work with landscape evolution in Lindborg et al. (2013). Water exchange is described from various aspects (Berglund et al. 2013a, b) including the sea (Corell and Döös 2013, Eriksson and Engqvist 2013) and management of groundwater lowering (Werner et al. 2013). The retention properties of regolith deposits are shown by field measurements (Sohlenius et al. 2013) and by models (Piqué et al. 2013a). Modelled uptake in organisms is compared between different models, as well as with site data (Erichsen et al. 2013). Dose calculations are presented for man (Avila et al. 2013) and for other organisms (Torudd and Saetre 2013). A review has been done of how man has historically cultivated land or utilized natural resources such as food, together with a compilation of reasonable ways to obtain food in the future (Saetre et al. 2013).

SKB has moreover presented data and scientific results at a number of international and national symposiums. Among other things, SKB gave a keynote address on SKB's marine research at the International Conference on Marine and Coastal Ecosystems (MarCoastEcos 2012). SKB has also contributed with presentations and posters at a number of different symposia: the International Conference on Radioecology and Environmental Radioactivity (ICRER-2011) in Canada (Berglund et al. 2011, Ikonen et al. 2011, Lindborg et al. 2012, Werner et al. 2011), Swedish OIKOS 2013 and the Swedish Society for Marine Sciences (SHF2011). Members of SKB's biosphere task force are regularly engaged to review articles to be published in scientific journals. SKB's has also been consulted by such organizations as the Svealand Coastal Water Management Association for its expertise on the biosphere.

SKB gives annual lectures on its biosphere programme at several of Sweden's universities and is supporting research projects at the Swedish University of Agricultural Sciences, Umeå University, Stockholm University and Linköping University. SKB is supporting the seminar programme at the Centre for Radiation Protection Research at Stockholm University. SKB is funding five Ph. D. candidates associated with the biosphere programme at Stockholm University, Umeå University and the Swedish University of Agricultural Sciences in Umeå. An SKB-funded Ph. D. thesis in oceanography (Corell 2012) and a licenciate thesis in radioecology (Konovalenko 2012) were presented in 2012.

Programme

During the coming programme period, SKB will remain active in BIOPROTA and continue to participate in the various working groups in the IAEA's MODARIA project. SKB's involvement in the IAEA's HIDRA project on human intrusion in the repository 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. SKB will also continue to monitor international development and evaluation of existing models for calculation of activity concentrations of e.g. carbon-14 and radium-226 in BIOPROTA.

The collaboration with Posiva will continue with a joint compilation of site data, in addition to BIOPROTA and MODARIA. SKB is also planning to participate in the EU project GHG-Aquaflux in the area of methane cycling in lakes.

28 Other methods

There are two possible main approaches for managing the spent nuclear fuel. One entails regarding the nuclear 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 clearly 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 28-1 illustrates the principles and strategies for final disposal of spent nuclear fuel that have been considered at different times. The strategies that are of interest today are:

- Reprocessing, partitioning and transmutation.
- Geological disposal.

In reprocessing, fissile material is recovered and the fission products are separated (partitioned) as waste. Partitioning and transmutation (P&T) can offer opportunities to reduce the quantity of certain long-lived radionuclides in the spent nuclear fuel. Compared with spent nuclear fuel, the decay products from transmutation have shorter half-lives or are even stable in some cases. However, radioactive waste still remains after partitioning and transmutation and must be disposed of. P&T should therefore not be seen as a strategy for final disposal, but rather as a possible means of making more efficient use of the fuel in new types of reactors in the future (von Lenza et al. 2007).

Using transmutation solely to reduce the quantity of high-level, long-lived waste is not efficient, in terms of either costs or resources. For this reason, partitioning and transmutation will probably be only considered if new types of reactors, so-called fast reactors, are developed that can utilize uranium and plutonium more efficiently. These reactors could also be used for transmutation, but at a higher cost (CEA 2012).

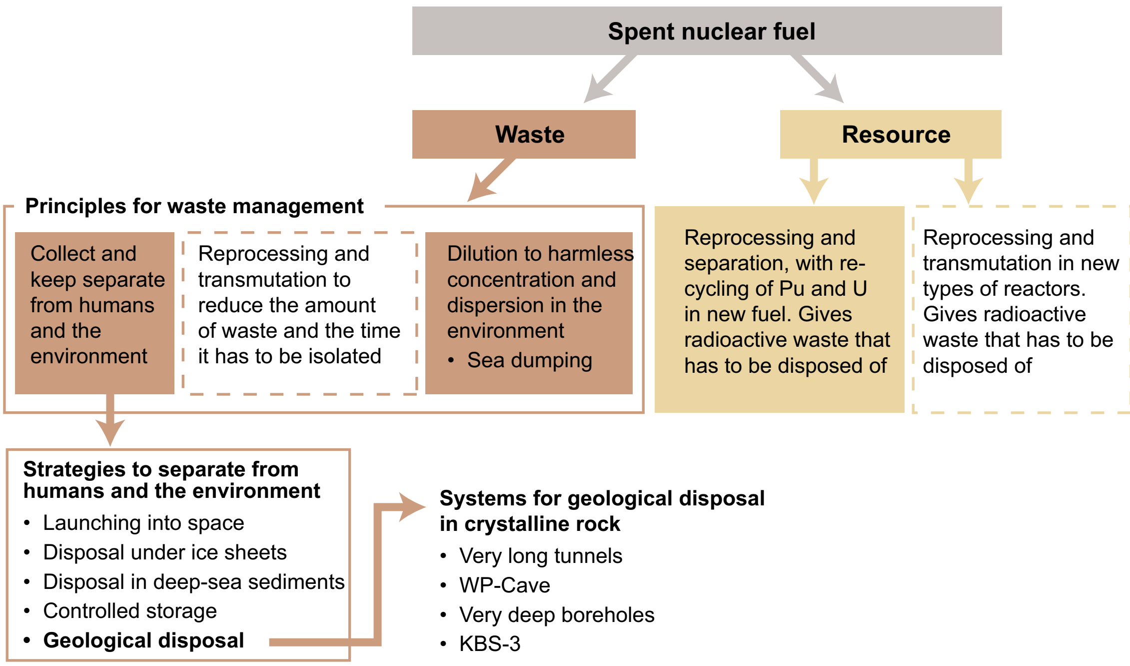

Figure 28-1. *Principles, strategies and systems for disposal of spent nuclear fuel. The principles in the dashed boxes are based on technology that is not available today.*

By unanimous judgement, P&T is a research area that will require several decades of development before full-scale 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 (CEA 2012).

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 besides the KBS-3 method that is held up by some actors as being of interest for Sweden is disposal in deep boreholes.

P&T and disposal in deep boreholes are described in Sections 28.1 and 28.2, respectively.

28.1 Partitioning and transmutation (P&T)

Research and development on methods for partitioning and transmutation of long-lived radionuclides in spent nuclear fuel has attracted growing interest in the past decade. The main purpose of P&T is to remove, or at least greatly reduce, the quantity of long-lived radionuclides that have to be deposited in a final repository.

The most important radionuclides in this respect are the so-called transuranics, or transuranium elements, which are elements heavier than uranium. Transuranic elements are formed in nuclear reactors by the capture of one or more neutrons by uranium atoms, are then transformed by radioactive decay to neptunium, plutonium, americium or curium. 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 (technetium-99, iodine-129) may also be of some interest for transmutation.

Today there are 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. Plutonium comprises about 90 percent of the total quantity of transuranics in spent nuclear fuel from today's light water reactors.

The goal of the current research on partitioning is to find and develop processes that are suitable for separation of heavier transuranics (and possibly certain fission products as well) on an industrial scale, while the goal of the current research on transmutation is to define, investigate and develop facilities that are suitable for transmutation of the aforementioned long-lived radionuclides on an industrial scale.

Research on partitioning and transmutation started back in the 1950s when nuclear power was a growing industry. During the following decades, P&T was mainly associated with the development of breeder reactors. When interest in this development fell to a very low level in the early 1980s, interest in partitioning and transmutation more or less died.

This interest was reawakened in the 1990s, with a focus on accelerator-driven systems (ADS) for transmutation of long-lived radionuclides in spent nuclear fuel. Interest in fast reactors for transmutation has been rekindled in recent years. As an example, 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 P&T has been given highest priority. Design of such a reactor is in full swing. Interest in fast reactors is now greater than interest in accelerator-driven systems, and research in this field has largely been focused on questions of a non-technology-specific nature, i.e. areas where the results can be used for both critical and accelerator-driven systems.

A survey of the situation for research in partitioning and transmutation was done on behalf of SKB in 1997–1998 and the findings were reported in Enarsson et al. (1998). Similar situation reports were published by the reference group for P&T research in 2004, 2007, 2010 and 2013 (Ahlström et al. 2007, Blomgren et al. 2010). The assessments presented below are based on the 2013 report.

Conclusions in RD&D 2010 and its review

SSM maintained that partitioning and transmutation is not currently a realistic alternative to direct disposal of the spent nuclear fuel. It is, however, very important to maintain and develop competence in the field. In a similar vein to what was said by SKI in its review of RD&D Programme 2007, SSM wished to cite the following reasons for its standpoint:

- According to the Nuclear Activities Act, the RD&D programme shall be comprehensive, which means that even strategies and methods that may be currently regarded as less realistic need to be included. This particularly applies to methods that are currently the subject of considerable development efforts in the world.
- To maintain the competence needed to makes its own judgements of developments in the field, Sweden must participate actively in this development work. Following passively is not enough when it comes to such a complex area as partitioning and transmutation.
- An even more important aspect is that SKB's programme in the P&T area contributes to maintaining a sufficiently high level of research and education in the areas of nuclear technology that are also of crucial importance for radiologically safe management of nuclear material and nuclear waste within the current programme.
- Similarly, an active participation in the joint international research in this field provides better opportunities to assimilate knowledge of importance for the current nuclear waste programme as well as other areas of nuclear technology.

SSM therefore contended that SKB needs to continue investing in research on partitioning and transmutation at the level proposed in SKB's programme, i.e. between six and seven million kronor per year.

Like SKB, SSM was of the opinion that the partitioning studies should continue to be focused on liquid-liquid extraction in aqueous systems, not least because this provides greater breadth in knowledge accumulation of importance for other parts of the nuclear fuel cycle as well, including final disposal.

Newfound knowledge since RD&D 2010

In 2009, the Swedish Research Council awarded the largest research grant by far in 30 years to new nuclear technology in Sweden, in response to a direct instruction from the Government. As a result, for the first time since the national referendum in 1980, considerable state funds were available for P&T research in the country. SKB was thereby no longer the sole outside funding body in the field; in fact, SKB was not longer the largest funding body.

A commitment to P&T would enable Sweden to operate new fast reactors for over 100 years by reprocessing the fuel from today's nuclear power plants. It would also reduce the quantity of long-lived radionuclides. A successful development of P&T within the framework of advanced fuel cycles will, however, not eliminate the need for final repositories for high-level and long-lived waste. The complex processes will inevitably generate waste flows containing long-lived radionuclides, even though the quantities will be less. Successful development of P&T may, however, reduce the requirements on the engineered barriers. It may also reduce the need of disposal volume for high-level waste. The volumes of low- and intermediate-level waste will probably increase due to the partitioning processes, however.

It is important for Sweden to participate in international development efforts while maintaining a reasonable level of competence within the country, at least as long as a large proportion of the country's electricity is based on nuclear energy. Competence developed in conjunction with research on partitioning and transmutation is valuable, not just for assessing development and potential in this field, but also for development of safety and fuel supply at existing reactors.

Recently, a generation change has taken place in the Swedish university research groups active in nuclear-power-related research, and at present these activities are growing rapidly, both due to increased interest in research and a greater need for education. The leading scientists in the new generation have all established themselves in projects supported by SKB and the Swedish Centre for Nuclear Technology at KTH, and most of them have been involved in P&T research. This research has thereby already played a crucial role in supporting Swedish competence in the field of nuclear power.

Programme

As before, the goals of SKB's research on partitioning and transmutation of long-lived radionuclides are to:

- Examine how this technology is developing and how it will influence waste streams from nuclear installations and their nuclide content.
- Support Swedish universities so that they can participate in large international projects in the area and thereby build competence in knowledge areas of importance for SKB's core activity.
- Judge whether, and if so how, this can be utilized to simplify, improve or develop a system for disposal of the nuclear fuel waste from the Swedish nuclear power plants.

SKB has limited its involvement to partitioning technology and new fuels. Research is being pursued in accordance with annual activity plans. Overall assessments are made prior to important decisions in the nuclear waste programme. Synergies with the research project Genius funded by the Swedish Research Council concerning the fourth generation of reactors, as well as the joint project with France in the field, have been exploited and will continue to be exploited in the future.

Previously, the official line in Sweden was that nuclear power was to be phased out, and research on P&T was then pursued as monitoring of alternative technology to direct disposal. Today it is permissible to build new nuclear power plants in Sweden, which opens up the possibility of nuclear power for another century. It is possible that advances in technology over this timespan could be considerable, and P&T research could conceivably have consequences far beyond being a possible alternative to direct disposal spent nuclear fuel. SKB's annual spending in this area has in recent years been on the level of six to seven million kronor, but has now been halved since research and development of new reactors is not SKB's task.

The prospect of transmutation of the waste is in conflict with the principle that Swedish nuclear fuel may not be reprocessed. Any decision regarding transmutation of the Swedish nuclear fuel waste is therefore outside of SKB's sphere of responsibility, and at present SKB's goal is to commence disposal of spent nuclear fuel in a geological repository during the next two decades.

28.2 Deep Boreholes

The concept of disposal in deep boreholes entails that canisters with spent nuclear fuel are emplaced in boreholes at a depth of 2–5 kilometres. The top part of the borehole is sealed.

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 are stagnant at great depths. The reason for the stagnant conditions is that the groundwater has high salinity (and thereby high density), resulting in a stable stratification that counteracts remixing with the lighter fresh water above. Figure 28-2 illustrates schematically how properties such as water flux, salinity, temperature and rock stresses change with depth.

Figure 28-3 shows a proposal for a conceptual model of the uppermost five kilometres of the bedrock in Sweden (Juhlin et al. 1998). 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.

In the concept previously presented by SKB, the spent fuel is encapsulated in canisters with an outside diameter of 0.5 metre, each of which contains four BWR assemblies or one PWR assembly (SKB 2000b). This requires boreholes with a diameter of 800 millimetres at repository depth. The canisters are stacked on top of one another in a deposition zone at a depth of between two and four kilometres. In this way, a deposition hole can hold around 300 canisters, which means that about 60 deposition holes are needed to accommodate the spent fuel in the operating scenario that produces 6,000 copper canisters according to the KBS-3 method.

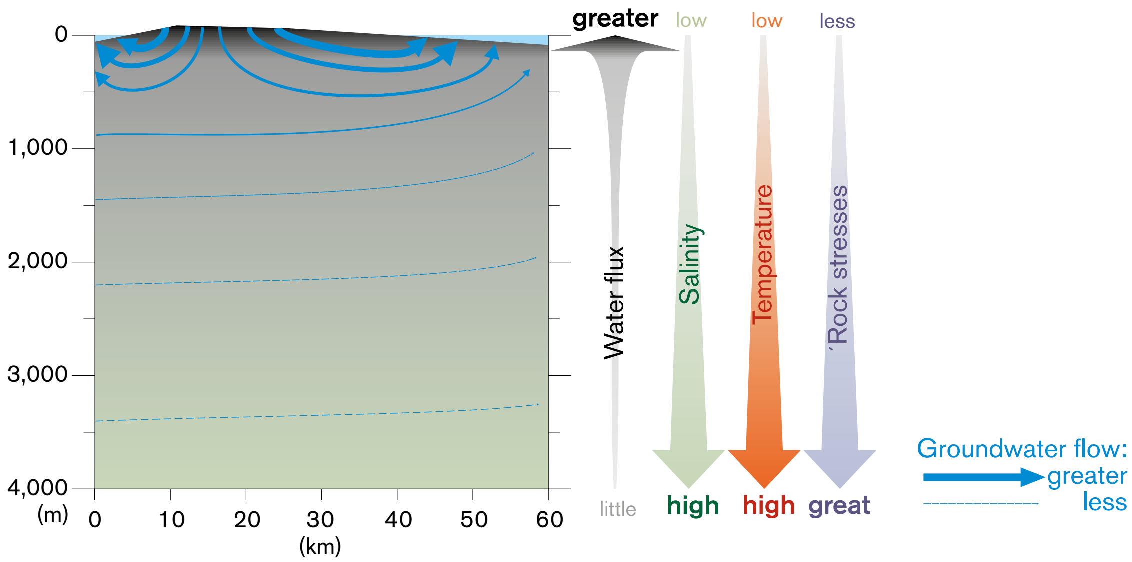

Figure 28-2. *Schematic illustration of how the properties of the Swedish bedrock change with depth.*

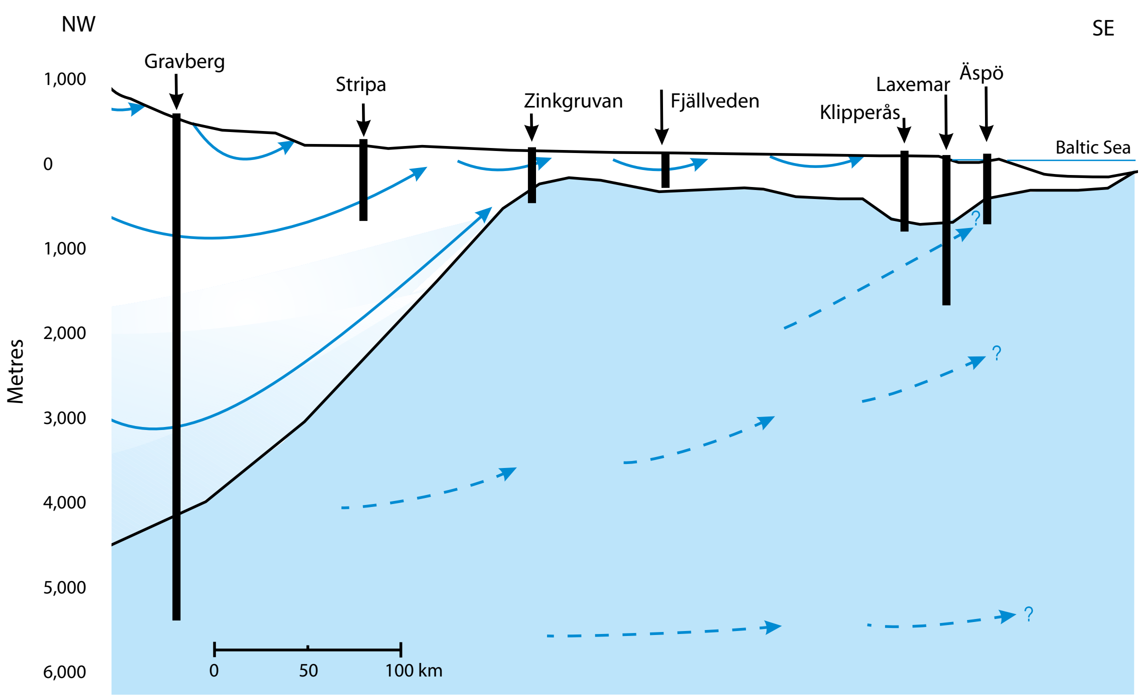

Figure 28-3. *Schematic illustration of water circulation and variations in salinity along a profile from northern Dalarna to eastern Småland (Juhlin et al. 1998).*

Sandia National Laboratories, one of the US Department of Energy's laboratories, published reports in 2009 and 2011 on disposal of radioactive waste in deep boreholes (Brady et al. 2009, Arnold et al. 2011). Brady et al. (2009) proposes that the holes should be five kilometres deep with a diameter at repository depth of 445 millimetres. According to the report, such holes can be drilled with existing drilling equipment of the type used to drill geothermal wells. The proposal in Arnold et al. (2011) is based on boreholes with a slightly smaller diameter at repository depth, 432 millimetres.

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 could take place at multiple locations near local storage facilities and nuclear reactors. This reduces the transportation need.

Arnold et al. (2011) proposes that the canisters be joined together to a 200-metre long string of 40 canisters that is lowered into the borehole. When the canister string has reached the bottom, a mechanical plug is installed that seals the hole about 0.5 metre above the top canister. A 10-metre-long concrete plug is cast on top of the plug for the next canister string to land on. The procedure is repeated 10 times until the deposition zone (3,000–5,000 metres) is filled with a total of 400 canisters.

After the deposition zone has been filled with canisters and intervening plugs, the deposition work is concluded by casting a concrete plug. After the casing has been taken up, the section from 3,000 to 1,500 metres is bare and is plugged with concrete plugs, ballast and bentonite in alternating layers. The top part of the hole, which is still lined with casing, is plugged mainly with concrete plugs.

SKB's assessment is that the disposal method proposed in Arnold et al. (2011) is more developed and is also judged to be safer than the method previously described by SKB (SKB 2000b) and that the method could possibly be used for disposal of canisters in deep boreholes.

The proposed distance between the holes is 200 metres. According to Brady et al. (2009), the canisters are made of steel casing with an outside diameter of 340 millimetres, each of which holds one PWR or two BWR assemblies. Approximately 80 deposition holes of this type would be required to dispose the spent fuel from the reference scenario with 50–60 years' operation of the Swedish nuclear power plants.

According to Brady et al. (2009) and Arnold et al. (2011), 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 reason, according to the authors, is that no long-lived corrosion protection of the canister is needed down in the deposition zone, since it is the geological barrier, with the stagnant saline water, that will prevent radionuclides from escaping into the bedrock. The authors propose that the canisters be fabricated of steel casing tube of types and dimensions used by and developed for the drilling industry.

Conclusions in RD&D 2010 and its review

SKB made the same judgement in RD&D Programme 2010 as in RD&D programmes 2001, 2004 and 2007. There is no reason to believe that disposal in deep boreholes would increase the safety of the final disposal of the spent nuclear fuel. A KBS-3 repository can, in contrast to deep boreholes, be constructed, operated and sealed in a manner that can be controlled and verified at every step. Moreover, a repository according to the deep boreholes concept is associated with great uncertainties regarding the evolution of the repository after closure. Furthermore, SKB notes that no country in the world recommends disposal in deep boreholes as a preferred alternative for disposal of spent nuclear fuel.

In RD&D Programme 2010, 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 Nuclear Fuel Repository. Conclusions from the report are presented below in the section "Newfound knowledge since RD&D 2010".

As far as continued work is concerned, SKB said that they would 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.

In its review statement on RD&D Programme 2010, SSM said that they did not share SKB's view that a more detailed analysis of the stability of the groundwater at great depths can only be made in qualitative terms. SSM contended that SKB has previously carried out quantitative studies to investigate whether the heating of the water at great depths by the canisters could serve as a driving force to counteract the tendency of the saline water (brine) not to flow upward (Marsic et al. 2006) and that the modelling results indicated that this is not the case.

SSM was of the opinion that SKB should further study uncertainties associated with the stability of the brine at great depth. Besides relevant conceptual and parameter uncertainties, which were addressed in part in the earlier study, questions concerning the influence of large gradients caused by a retreating ice sheet or wells should be investigated, for example.

SSM took a positive view of SKB's plans to present an in-depth expert assessment concerning the feasibility of disposal in deep boreholes in the application.

SSM further noted that SKB had not reported any connections between participation in the Swedish Deep Drilling Program and the work with the alternative method of deep boreholes.

SSM shared Uppsala University's view that a concept whose only protective barrier is the rock has a serious deficiency, since it does not comply with SSM's regulatory requirement on a multiple-barrier system. The Authority nevertheless thought that the initiatives described above should be carried out by SKB, since the possibility cannot be ruled out that multiple barriers can be maintained in a deep boreholes repository.

In its examination of the application for the final repository for spent nuclear fuel, SSM will judge whether the account of alternative methods presented in the application is sufficient to permit an assessment of the choice of method.

Newfound knowledge since RD&D 2010

A report presenting a comparison between final disposal of spent nuclear fuel by the KBS-3 method and disposal in deep boreholes has been published (Grundfelt 2010). The purpose has been to highlight method-distinguishing factors. The ambition has been to make the comparison as fair as possible, even though there are great differences between the two methods in terms of both quantity and quality of the data.

The comparison has been made from the perspective of several crucial aspects:

- site selection criteria, premises for construction, deposition and sealing, and nuclear safety in the handling of encapsulated spent nuclear fuel,
- long-term safety of a closed repository, physical protection and nuclear safeguards, as well as planning premises in the form of lead times, development needs and costs.

An overall conclusion from the report is that there is nothing today to suggest that a switch to planning for disposal in deep boreholes would result in a safer final disposal of the spent nuclear fuel than is offered by the KBS-3 method. Further, it is concluded that the KBS-3 method provides a final disposal of the spent nuclear fuel that can be controlled and verified at every step, while this is not possible in the case of disposal in deep boreholes.

Technical premises for drilling and deposition

A report describing the technical premises for drilling of and disposal in deep boreholes has been published (Odén 2013). The report is based on a relatively broad survey of today's drilling technology from project descriptions, articles and contacts in the industry.

Drilled boreholes show that it is possible to drill 5,000-metre deep holes in crystalline rock. However, none of the holes has the diameter recommended by Sandia (445 millimetres) at the proposed repository depth (3,000–5,000 metres) (Brady et al. 2009). The hole that comes closest is a hole drilled in southern Germany with a diameter of 375 millimetres down to a depth of 6,000 metres.

It has proved difficult to drill straight holes. In most holes it was necessary to drill "sidetracks" due to drilling problems. The exception is the borehole in southern Germany, which was drilled to a depth of 7,000 metres without sidetracks. Moreover, the hole was drilled straight (± 0.5 degree) and vertical thanks to active vertical control equipment. Furthermore, it has been found that problems with spalling and breakout are greatest during the actual drilling procedure. Once the borehole is cased it is normally stable.

With regard to drilling and disposal in deep boreholes, the following principal conclusions can be drawn from these studies:

- The holes should be vertical and straight to make it easy to install casing and later to lower the canister strings.
- Branched holes or sidetracks should be avoided, since they will interfere with deposition.
- The holes should be drilled by directional drilling, with a maximum deviation of 0.5 degree.

The combination of drilling both deep and with large diameter in crystalline rock is not so common and has mainly been used in different research projects. Most of these holes were drilled a number of years ago. Due to the recent increased interest in geothermal energy, the drilling technology has been further developed and equipment exists today for drilling down to five kilometres with a hole diameter of 445 millimetres in crystalline rock with a maximum deviation of 0.5 degree. This requires that the holes are drilled with active vertical control.

Experience from drilling five-kilometre-deep holes in crystalline rock is understandably very limited. The technical problems are considerable, and it must be expected that a large proportion of the boreholes will not be suitable for deposition, at least not to start with. This is a factor of great economic importance.

Distance between boreholes

The area that would be required for a repository according to the deep boreholes concept is determined by the spacing between the boreholes, the amount of fuel to be deposited and the size of the canisters. 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 (SKB 2000b).

With active vertical control, the boreholes could, from a drilling point of view, be a few tens of metres from each other without risk of "collision" (Odén 2013). Calculations of heat output indicate that the distance between boreholes could be reduced at least to about 100 metres (Marsic and Grundfelt 2013a). If the holes are located closer to each other, they could have a greater thermal impact on each other.

Groundwater flow and gas formation

In the calculations with heat output, sensitivity with respect to hydraulic properties in the borehole was also studied (Marsic and Grundfelt 2013a). An increased hydraulic conductivity proved to be of considerable importance for the water flow in and near the boreholes, which leads to some remixing at the interface between saline and fresh groundwater. The calculated travel times for groundwater from the deposition zone to the overlying layer are, however, longer than the duration of the heating.

There will be large quantities of steel in a deep borehole. Not only is it assumed that the canisters will be made of steel, the casings will also be of steel. Corrosion will therefore occur with associated hydrogen gas formation. The hydrogen gas will rise and create a driving force for upward flow in the bentonite slurry in the borehole and in the groundwater around the hole. The consequences of this have not been calculated quantitatively but may be considerable, since large amounts of hydrogen gas can form.

Consequences of handling mishaps

In view of the actual execution procedure, drilling and deposition, and the far-reaching requirements on nuclear safety and radiation protection, an Achilles heel for deep boreholes is judged to be the risk that already deposited canisters will be damaged or that the canister string or parts of it will get irrevocably stuck above the deposition zone. However, the risk of this happening is judged to be much lower with the procedure described in Arnold et al. (2011) than with previously studied deposition procedures, such as SKB (2000b), since the diameter has been reduced and the canisters are lowered in strings of 40 connected canisters that do not risk damaging already deposited canisters.

The accident scenario can otherwise result in having to plug and seal the hole above the stuck canister. If the canister is stuck in a section with mobile groundwater, this will result in a leakage of radionuclides if the canister is damaged or when it has corroded.

The consequences of such a scenario have been analyzed in Grundfelt (2013). The analysis shows that the probability of such an accident scenario must be lower than on the order of 10^{-5}–10^{-4} per hole in order that SSM's risk criterion is not exceeded.

Geoscientific data from deep boreholes

A report has been published on geoscientific data from deep boreholes (Marsic and Grundfelt 2013b). The work is an update of the compilation from 2004 done by SKB regarding geoscientific information for assessing the possibility of disposing of radioactive waste in four- to five-kilometre-deep boreholes (Smellie 2004). In relation to previous surveys, only one borehole of relevance to disposal in deep boreholes in the Fennocandian Shield has been encountered. Data from this hole, which is situated at Outokumpu in eastern Finland, show that there are groundwater-filled fracture zones at great depth as well (down to 2,500 metres). The water in these fractures has a relatively high salinity, and the groundwater chemistry in other respects shows that this groundwater has not been in contact with more superficial water in a very long time.

Stability of the saline layer

The changes in climate-related processes such as permafrost formation, ice sheet retreat and shoreline displacement that are described for the 100,000-year reference evolution of a KBS-3 repository also apply in an analysis of disposal in deep boreholes. The prospects that the repository will cope with the various stresses associated with a glaciation are, however, essentially different.

In the case of disposal in deep boreholes, retardation due to the very slow flow of groundwater is the repository's main safety function, while the engineered barriers – canister, buffer and closure – cannot be expected to offer any real protection. In conjunction with future glaciations, the primary safety function may be affected by an expected increased groundwater flow (including penetration of meltwater with low salinity), by permafrost and by glacially-induced earthquakes. With today's knowledge, certain things can be said about the initiating processes, but very little about the effects on the long-term safety of the final repository.

Large changes in groundwater flow in a 100,000-year perspective are expected in conjunction with glaciations. The largest impact on the hydraulic gradient and thereby on groundwater movement occurs when the steep ice front advances or retreats over the repository area, see e.g. Vidstrand et al. (2010a, b). However, the effects are dampened by the fact that the movement of the ice front takes place over a permafrost landscape and that the retreat takes place with a less steep ice front as well as with an elevated sea level (Vidstrand et al. 2010a, b, SKB 2006c, 2010w).

When the ice front passes over a repository area, it is likely that the transition zone between the superficial groundwater with low salinity and the deeper-lying saline groundwater will be affected, and both uplift and depression can be expected during the passage of the ice front. As a result, the stagnant groundwater conditions that prevail at great depths risk becoming less stable. The effects can be expected to be less in the lower part of a repository than in the upper part due to the greater distance to the transition zone. The results of model calculations of the groundwater situation during the advance of the ice sheet show qualitatively that effects on the groundwater in the form of changes in pressure, salinity and flow rates are possible at depths that correspond to the deposition zone (Vidstrand and Rhén 2010). However, the effects at these depths are heavily dependent on what properties and salinities have been assumed in the calculations. It should at the same time be noted that no calculations of radionuclide transport from the depths relevant for disposal in deep boreholes have been carried out, since sufficient data are lacking to quantify the processes that influence such transport.

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. A KBS-3 repository can, in contrast to deep boreholes, be constructed, operated and sealed in a manner that can be controlled and verified at every step.

SKB nevertheless intends to continue to monitor developments in the areas of drilling and disposal in deep boreholes. SKB intends to make use of any results that emerge from the Swedish Deep Drilling Program that could be of relevance to the deep boreholes concept.

Part V

Social science research

29 SKB's programme for social science research

29.1 Overview

Considerable interest was devoted to societal aspects of nuclear waste disposal already in the feasibility studies carried out by SKB during the period 1993–2000 in eight municipalities. The feasibility study reports contain descriptions and analyses of demographic trends, 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. However, this material has more the character of qualified survey results than of research.

During the period 2002–2009, when site investigations were being conducted in the municipalities of Oskarshamn and Östhammar, SKB devised, in close collaboration and dialogue with the two municipalities, a programme for so-called "societal studies", to be undertaken as a part of the site investigations. This material also has more the character of qualified survey results than of research.

In parallel with these societal studies, SKB carried out a social science research programme between 2004 and 2011. The purpose was to:

- Broaden the perspective on the societal aspects of the Nuclear Fuel Programme. This would 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 would thereby provide a sounder basis for various decisions.
- Contribute background data and analyses for research on the societal aspects of large industrial and infrastructure projects. In this way, experience gained from the Nuclear Fuel Programme could be of benefit in other similar projects.

The programme was divided into four research areas:

- Socioeconomic impact – macroeconomic effects.
- Decision processes.
- Public opinion and attitudes – psychosocial effects.
- Global changes.

The boundaries between these research areas have not been strictly drawn. All in all, SKB's social science research programme encompassed research in the social sciences as well as the humanities and jurisprudence. It dealt with issues in disciplines such as the behavioural sciences, economics, philosophy, politics/political science and Swedish and international law.

An initial call for proposals in the four research areas was sent out to a number of universities and institutes of technology in the spring of 2004. Additional calls for proposals went out during the period 2005–2009. The proposals and project descriptions received were assessed by a Social Science Advisory Group appointed by SKB. The Group consisted of renowned researchers with an interest in the social sciences. A total of 18 projects were selected based on proposals from different researchers, funded and carried out. The projects concerned 15 different scientific disciplines and engaged 23 different researchers from ten universities and institutes of technology, see Figures 29-1 and 29-2.

The costs of the social science research programme have amounted to about SEK 25 million.

Reports of what has been achieved within the framework of the programme have been published in various forms. During 2005–2010, SKB published popular scientific books containing descriptions of ongoing and recently concluded projects (SKB 2005, 2006g, 2007b, 2008d, 2009d, 2010v).

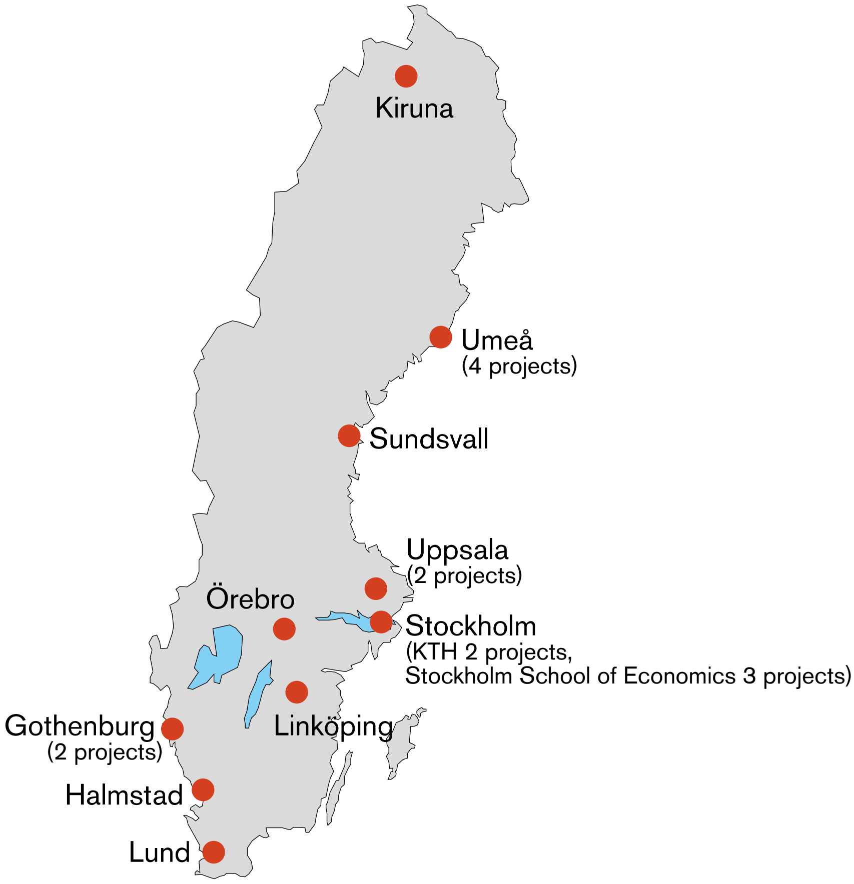

Figure 29-1. *Geographic breadth of social science research programme.*

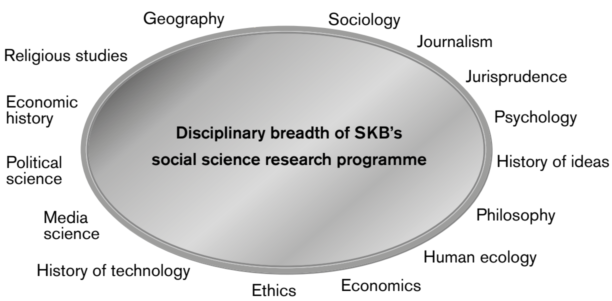

Figure 29-2. *Disciplinary breadth of social science research programme.*

Final reports from the various projects were published as SKB reports, and in one case in book form (Andersson-Skog 2007, Anshelm 2006, Cramér et al. 2007, 2010, Egan Sjölander 2007, Frostenson 2008, 2010, Hansson 2010, 2012, Johansson and Lisberg Jensen 2006, Kaijser and Högselius 2007, Keskitalo et al. 2009, Lindgren and Strömgren 2007, Nord and Stúr 2010, Pettersson 2008, Sandberg 2008, Sjöberg 2006, 2008, Soneryd and Lidskog 2006). In addition, RD&D Programme 2007 and RD&D programme 2010 contained presentations of the final reports from research projects that were available when the respective RD&D programme was written. The final reports published after this are presented in Section 29.3.

In 2009, SKB published a booklet containing the Social Science Advisory Group's summarizing comments on the research programme as well as general presentations of the projects (SKB 2009e). An updated version was published in the spring of 2011 (SKB 2011f), along with an English translation (SKB 2011g), for an international conference marking the conclusion of SKB's social science research programme.

Since then SKB has had an evaluation done of the programme. The results of the evaluation were published in 2012 (Söderberg 2012), see Section 29.4.2.

SKB believes that the research results have contributed to a deeper understanding of historical, economic and public opinion aspects of issues related to the final disposal of spent nuclear fuel and have thereby contributed to an increase in the general knowledge base for SKB's work. The results have also been of use in SKB's practical work, see further Section 29.5.

29.2 Conclusions in RD&D 2010 and its review

29.2.1 SKB's account

SKB presented its view on a number of general questions related to the social science research programme in RD&D programme 2010, Chapter 28. Chapters 29–32 presented results from the research projects that had been concluded or were in progress within each of the four research areas.

The discussions of general questions were based on the viewpoints that had been expressed in the review of RD&D Programme 2007 and can be summarized as follows.

The relationship between research and other documents

In its review statement on RD&D Programme 2007, the Swedish Nuclear Power Inspectorate had called for further clarification of what the relationship between social science research, licence applications, the environmental impact statement and social science studies looks like. In RD&D programme 2010 SKB stressed that the EIS should describe the consequences of the applied-for activity for man and environment. The purpose of the social science research programme is, on the other hand, 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 Nuclear Fuel Repository. The main target group for the research programme is decision-makers, local and national. The programme is independent of both the EIS and the licence applications. SKB described the difference between the social science research programme and societal studies in roughly the same way as in RD&D programme 2007.

In its review of RD&D Programme 2007, the Swedish National Council for Nuclear Waste had proposed that the research programme should be supplemented by studies of future economic consequences of the handling of the nuclear waste issue and a field of research that sheds light on global changes and safety culture. Further, the Council had stated that the social science research programme should continue after the applications for licences under the Environmental Code and the Nuclear Activities Act to establish the final repository for spent nuclear fuel had been submitted. SKB explained in RD&D programme 2010 that the intention was to publish a report (in Swedish and English versions) that provides a summarizing evaluation of the activities of the social science research programme to date and the needs that may exist for future research. This report was published as planned (SKB 2011f and SKB 2011g). SKB also announced an international conference in the spring of 2011 for discussion of the results that had been obtained. Only after that would SKB make an assessment of future research needs based on the premises that apply during and after licensing and decision.

Work forms

SKB recalled that the first call for proposals for funding within the framework of the social science research programme was carried out in 2004 and that the programme had since been built up gradually in dialogue with the concerned municipalities. The Swedish National Council for Nuclear Waste had previously pointed out the importance of having actors other than SKB fund social science research around the nuclear waste issue. In RD&D programme 2010, SKB emphasized that it would benefit the entire research field if other actors than SKB also took the initiative in this direction and continued (see RD&D programme 2010, page 399): "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".

SKB further repeated what they had stated in RD&D Programme 2007 regarding the fact that the researchers themselves formulate their own research topics and take responsibility for methodology, results and conclusions.

Further, SKB described the procedure for calls for proposals, the most important criteria for obtaining funding, and the disciplinary and geographic spread between different universities of the 18 projects funded by the programme.

29.2.2 Review comments

Both the Swedish Radiation Safety Authority and the Swedish National Council for Nuclear Waste commented in their reviews of RD&D programme 2010 on what SKB had said about the programme for social science research. Viewpoints were also expressed to the Swedish Radiation Safety Authority in statements of comment from the concerned municipalities, government agencies and other organizations. The following main viewpoints emerged.

In summary, the Swedish Radiation Safety Authority (SSM 2011) found it positive that SKB is conducting social science research within the framework of its nuclear fuel programme. The research provides a better understanding of the economic and social dimensions of the final repository for spent nuclear fuel and thereby contributes to an overall picture of the final disposal process. The Authority noted that this opinion was shared by nearly all reviewing bodies who have submitted comments on the matter. However, the Authority thought that SKB should give a clearer description of the role of the social science research programme in relation to SKB's other activities within the framework of the nuclear fuel programme, as well as what benefit SKB derives and has derived from the social science research.

The Swedish National Council for Nuclear Waste recalled in its review statement (Swedish National Council for Nuclear Waste 2011) the discussion concerning the importance of research in the social sciences and humanities for the final disposal issue that has been going on in Sweden since the 1980s. In the late 1990s and early 2000s, this discussion had stimulated independent university researchers to undertake studies in these areas. The Council exemplified this with three publications (Lidskog 1998, Sundqvist 2002, Andrén and Strandberg 2005)[1]. Further, the Swedish National Council for Nuclear Waste recalled its 2002 proposal for a programme independent of SKB for social science and humanistic nuclear waste research and the fact that the programme funded by SKB since 2004 was now approaching its conclusion.

The viewpoints presented by the Council led to the following five conclusions (Swedish National Council for Nuclear Waste 2011, p 105):

- "It is the Council's opinion that SKB has detached its social science research programme from its fundamental final disposal mission in an unsatisfactory manner.
- There is still a great need for social science research around nuclear waste, which should be (1) as independent as possible of economic and political interests but still (2) of relevance to Swedish nuclear waste management.
- The future research should study the consequences of increased competition on the global market for raw materials (for example copper), the consequences of crucial changes in the ownership of nuclear power, and conditions for societal planning and decision-making.
- A number of the changed premises that could warrant a far-reaching re-examination of the execution of the nuclear waste programme and SKB's main timetable, are associated with different types of societal changes, which could be made the subject of social science research.
- In the light of SKB's evaluation and the Council's upcoming review, the Government should (1) study the forms for how independent social science and humanistic research on the nuclear waste issue is to be conducted in the future and (2) in the upcoming research bill, make sure that money from the Nuclear Waste Fund is allocated in the coming decades to social science research."

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[1] For a list of social science studies relating to final disposal of spent nuclear fuel outside of SKB's research programme during the period 2007–2012, see Söderberg (2012, Appendix 16).

The decision taken by the Government on 27 October 2011 in response to the review of RD&D programme 2010 does not contain any particular statements regarding SKB's social science research programme (Regeringsbeslut 2011-10-27).

At SKB's invitation, representatives of the Swedish National Council for Nuclear Waste elaborated on some of the Council's criticism at a meeting in November 2011. It emerged that the Council did not intend to take any further initiatives in the matter before the Council had been given an opportunity to consider the results of the evaluation of the social science research programme that was then being done by SKB, see Section 29.4.2.

SKB's conclusions from the review of the social science research programme in RD&D programme 2010 are presented in Section 29.5.

29.3 Newfound knowledge since RD&D Programme 2010

Chapters 30–33 of RD&D Programme 2007 presented final reports from the eight projects that had been finished at that time. Eight other final reports from seven projects were presented in RD&D programme 2010, Chapters 29–32. Since RD&D programme 2010 was written, three more final reports have been published, which are presented here in Sections 29.3.3 and 29.3.5.

29.3.1 Four research areas and eighteen projects

A total of 18 projects have been carried out within the four research areas covered by the social science research programme. The following list (based on SKB 2011f) shows which final reports are attributable to these 18 projects (the title of the final report corresponds for the most part with the name of the project). Two of the listed reports (Sandberg 2008 and Pettersson 2008) describe the same research project. It should be emphasized that several of the projects are of relevance in more than just the research area they have been attributed to.

- Socioeconomic impact – macroeconomic effects
  - Växtkraft av kärnkraft? Kärnkraftetableringens socioekonomiska effekter i Oskarshamn och Östhammar 1960–2000 ("Growth power from nuclear power? – The socioeconomic effects of the nuclear power investments in Oskarshamn and Östhammar 1960–2000"), Andersson-Skog 2007.
  - Slutförvarets lokala effekter på befolkning och sysselsättning i Östhammar och Oskarshamn ("Local effects of the final repository on population and employment in Östhammar and Oskarshamn"), Lindgren and Strömgren 2007.
- Decision processes.
  - Allmänhet, expertis och deliberation – samråd om slutförvar av kärnavfall ("Public, experts and deliberation – Consultations on the final repository for nuclear waste"), Soneryd and Lidskog 2006.
  - Grunden för beslut i kärnavfallsfrågan. Upplevelser av lagstiftningsgrund och MKB-process ("Creating the basis for decisions in the nuclear waste issue. Experiences of parties and participants of the legislative basis and the EIA process."), Keskitalo et al. 2009.
  - Etiska och filosofiska perspektiv på kärnavfallsfrågan ("Ethical and philosophical perspectives on nuclear waste"), Hansson 2010.
  - Ansvarstagande i kärnbränslecykelns slutsteg – ett rättsligt perspektiv ("Responsibility at the back end of the nuclear fuel cycle – a legal perspective"), Cramér et al. 2010.
  - Från ödesfråga till övrig fråga – en studie av den politiska debatten om kärnavfallet i Sverige 1976–2009 ("From make-or-break issue to back-burner issue – a study of the political debate about nuclear waste in Sweden 1976–2009"), Nord and Stúr 2010.

- Public opinion and attitudes – psychosocial effects.
  - Opinion och attityder till förvaring av använt kärnbränsle ("Public opinion and attitudes to disposal of spent nuclear fuel"), Sjöberg 2006.
  - Identitet och trygghet i tid och rum – Kulturteoretiska perspektiv på kärnavfallsfrågans existentiella dimensioner ("Identity and security in time and space – cultural theoretical perspectives on the existential dimensions of the nuclear waste issue"), Johansson and Lisberg Jensen 2006.
  - Som natt och dag trots samma kärnas ursprung? Om (o)likheter och opinioner i nationella och lokala/regionala mediers hantering av kärnavfallsfrågan ("Like night and day – Concerning (dis)similarities and opinions in the treatment of the nuclear waste issue in national and locial/ regional media"), Egan Sjölander 2007.
  - Etisk argumentation i slutförvarsfrågan – Etiska värderingskonflikter i diskussionen om det svenska kärnavfallet ("Ethical argumentation in the final repository issue – Ethical value conflicts in the discussion of the Swedish nuclear waste"), Frostenson 2008.
  - Ungdomars syn på kärnkraft och demokrati sedan 1980-talet – Attitydepidemier, stigberoenden och teknisk kulturrevolution ("Young people's view of nuclear power and democracy since the 1980s – Attitude epidemics, path dependencies and technical-political cultural revolution"), Sandberg 2008.
  - Attityd till slutförvar av använt kärnbränsle – Struktur och orsaker ("Attitudes to final repository for spent nuclear fuel – Structure and reasons"), Sjöberg 2008.
- Global changes.
  - Från energiresurs till kvittblivningsproblem – Frågan om kärnavfallets hantering i det offentliga samtalet i Sverige 1950–2002 ("From energy resource to disposal problem – The issue of nuclear waste management in the public discourse in Sweden, 1950–2002"), Anshelm 2006.
  - Nationellt ansvar för använt kärnbränsle i en utvidgad europeisk union? ("National responsibility for spent nuclear fuel in an enlarged European Union?"), Cramér et al. 2007.
  - Resurs eller avfall? Politiska beslutsprocesser kring använt kärnbränsle ("Resource or waste? The politics surrounding the management of spent nuclear fuel"), Kaiser and Högselius 2007).
  - Unga sjunga med de gamla! En jämförande analys av grundläggande värderingar och uppfattningar om demokrati och politik bland blivande vuxna från 24 länder ("Keeping the faith – A comparative analysis of basic values and attitudes towards democracy and politics among young adults from 24 countries"), Pettersson 2008.
  - Slutförvarets industriella organisering – Fallgrop eller följdriktighet ("The Industrial organization of the final repository – Pitfall or consistency?"), Frostenson 2010.
  - Tidsperspektiven i svenska samhällsbeslut ("Time perspectives in societal decisions in Sweden"), Hansson 2012.

29.3.2 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 Nuclear Fuel Repository will affect the community's economy and demographic development.

Socioeconomic development includes both narrow economic aspects such as employment, industrial establishment, entrepreneurship, property values, municipal finances and tourism, and broader socioeconomic effects such as travel to and from the community, in- and out-migration to or from the community, and the community's public image and attractiveness.

The two projects that have been carried out in this research area were described in RD&D programme 2007. Calls for proposals in 2008 and 2009 did not result in any additional project applications in the area, something which SKB regretted in RD&D programme 2010 (page 403).

Conclusions in RD&D 2010 and its review

In their reviews of RD&D Programme 2010, neither the Swedish Radiation Safety Authority nor the Swedish National Council for Nuclear Waste have discussed questions related to the research area "Socioeconomic impact – macroeconomic effects". No additional projects within the research area have been carried out. SKB's view of the need for further research in this area is presented in Sections 29.4.1 and 29.5.

29.3.3 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-making 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 processes, for example to what extent decisions are perceived to be legitimate, fair and effective.

Conclusions in RD&D 2010 and its review

Three of the research projects in the area "Decision processes" were presented in RD&D programme 2010 and a fourth project was described in RD&D programme 2007. In their reviews of RD&D Programme 2010, neither the Swedish Radiation Safety Authority nor the Swedish National Council for Nuclear Waste has commented on the reporting of these projects. SKB's view of the need for further research in this area is presented in Sections 29.4.1 and 29.5.

Newfound knowledge since RD&D Programme 2010

The results of the fifth research project in the area "Decision processes" are summarized in the following. The summary is based on the final report published in the autumn of 2010 (Nord and Stúr 2010).

Från ödesfråga till övrig fråga – en studie av den politiska debatten om kärnavfallet i Sverige ("From make-or-break issue to back-burner issue – a study of the political debate about nuclear waste in Sweden"), Mid Sweden University

The overall aim of the project "From make-or-break issue to back-burner issue – a study of the political debate about nuclear waste in Sweden" was to investigate how the premises for the political decision processes in the issue of final disposal of the Swedish nuclear waste are affected by changes in public opinion and global events. The paper therefore focused on questions concerning how the national political debate leading up to decisions interacts with the media debate, and the public opinion dynamic that arises when the two debates relate to each other. Particular interest was devoted to the arguments and standpoints that occur in politics and media and how they refer to the prevailing public opinion situation and to conditions in the surrounding world of a political, legal, economic, ecological and technical nature. An analysis of the arenas and actors involved in the debates was judged to provide valuable knowledge regarding how the political agenda has been formed at different points in time.

The following questions were of central interest to the study:

- What characterizes the parliamentary and 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?

The course of the debate at four different times and in connection with four parliamentary elections during the period 1976–2009 was analyzed in the study. The subject of the analysis included four parliamentary parties: the Centre Party, the Liberal Party, the Moderates and the Social Democrats. These four parties were chosen because they were represented in the Swedish Riksdag during the entire period studied. The study was mainly conducted as a qualitative textual analysis of public print, party documents and media content. The qualitative text analysis was complemented to some extent by other existing written documentation such as political memoirs and debate books.

The results of the study show that the nuclear waste issue was at the centre of the domestic political debate during the 1970s, but then gradually lost ground in political importance during the following decades. What can most accurately be described as a make-or-break political issue at the start of the studied period was, according to the authors, by the end of the same period at most a back-burner issue. The authors characterize it as a residual issue that was of limited interest and had virtually vanished from the political sphere.

According to the authors, the nuclear waste debate serves as a good illustration of the life cycle of a political issue and illustrates the circumstances under which an issue can be politicized at one stage only to be marginalized and struck from the political agenda at a later stage. For example, six times as much was written on this subject in the four leading daily newspapers during the last three weeks of the 1976 election campaign as during the 1998 election campaign. When it comes to the number of parliamentary motions pertaining to nuclear power and nuclear waste, there were less than half as many during the parliamentary year 2009/2010 compared with ten years earlier. In other words, this is an issue that is gradually declining in opinion-related importance, both in the political sphere and in media coverage.

The status of nuclear waste in the Swedish political debate can, according to the authors, generally be explained by reference to an opinion-related interaction between politics and media. According to the authors, this interaction entails that political actors make rational judgements and take standpoints to maximize their own influence and to win voters, keep their party together or promote cooperation with other parties. Such considerations can explain why nuclear power and its environmental consequences were such a big issue in the 1976 election (when the Centre Party wanted to politicize the issue), as well as why it was such a minor issue in the 1998 election (when all political parties contributed to depoliticizing the issue). The interesting point is how the media coverage during these elections reflects the political positions and how compliantly the media's interpretations of the political positions are formed.

The nuclear waste issue is, also according to the authors, characterized not only by the willingness of the parties to politicize or depoliticize it during different periods, but also by the existence of other environment-related issues that compete for the attention of the parties, the media and the voters. The 1980s debate about algal blooms and mass seal deaths as well as the 2000s debate about the global climate threat are issues that have probably influenced the stand taken by public opinion-makers on the waste issue.

The authors also believe that the nuclear waste issue in Sweden has a different character, compared with most other political issues. The difference is that the importance of the issue waxes and wanes, from being extremely important in the 1970s in the political debate and the formation of governments, to playing a marginal role in the first decade of this century as a political issue. The first period's favourable circumstances and mutually reinforcing opinion stances between politics and media could hardly pose a more striking contrast to the latter period's mutual silence and waning importance.

29.3.4 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 the Nuclear Fuel Repository is moreover a drawn-out process, involving different actors during different phases. The purpose of research within this area was 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 planning of 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 and worries about risks, as well as self-interest are some examples of other important factors. It is therefore also important to shed light on the symbolism surrounding the Nuclear Fuel Repository and its activities.

Conclusions in RD&D 2010 and its review

Three completed projects in the research area "Public opinion and attitudes – psychosocial effects" were reported in RD&D programme 2007. Three more final reports from projects in the area were presented in RD&D programme 2010. In their reviews, neither the Swedish Radiation Safety Authority nor the Swedish National Council for Nuclear Waste commented on these reports. No additional projects within the research area in question have been carried out. SKB's assessment of the need for further research in this area is presented in Sections 29.4.1 and 29.5.

29.3.5 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 Nuclear Fuel Repository. At the same time, the question of establishment of a repository is very clearly related to changes in the world around us. The purpose of the research with this area 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 licence applications. The knowledge may also be important for the future operation of the final repository. The economic situation and development in the local community is dependent on a variety of circumstances in the surrounding world. What will the Swedish state, bearing 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 factor 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 in 30 years? What impact will future European integration have on nuclear waste management, and to what extent will this affect Sweden's waste management programme?

Conclusions in RD&D 2010 and its review

Two projects in the research area "Global changes" were reported in RD&D programme 2007. Two more final reports from projects in the area were presented in RD&D programme 2010. In their reviews, neither the Swedish Radiation Safety Authority nor the Swedish National Council for Nuclear Waste has commented on these reports. SKB's view of the need for further research in this area is presented in Sections 29.4.1 and 29.5.

Newfound knowledge since RD&D Programme 2010

Final reports from a fifth and a sixth research project in the area "Global changes" are summarized in the following. The summary is based on final reports published in the autumn of 2010 (Frostenson 2010) and the autumn of 2012 (Hansson 2012).

Slutförvarets industriella organisering – fallgrop eller följdriktighet ("Industrial organization of the final repository – pitfall or logical consequence?"), Uppsala University

An overall presentation of the research project "Industrial organization of the final repository – pitfall or logical consequence?" was included in RD&D programme 2010, Chapter 30. Three questions had been posed (Frostenson 2010):

- How will the final repository project be organized operationally and structurally over time?
- Why does SKB choose to organize the final repository project in this manner?
- What contextual organization is taking place in the final repository project and what will the consequences of this be?

The overriding ambition of the research project was thus to explore and clarify organizational factors for the units included in the upcoming final repository project and to identify and analyze the associated organizational and corporate governance problems. By "contextual organization", the author referred to the fact that deeper relationships are established between SKB's owners and SKB on the one hand and the municipalities of Östhammar and Oskarshamn on the other hand, and that SKB's own organization is affected by this situation.

The report discusses how the different industrial units in the final repository project can be run and what structure is chosen, for example with regard to ownership and integration of units. The reason why SKB chooses to organize the final repository project in a certain way is also examined. Active organizing narrows the final repository arena, and the issue of the final repository is transformed in many ways to a local matter, according to the author. There is a clear tendency for SKB's roles to be multiplied in order to be able to handle the demands made by key stakeholders – in particular the municipalities – on the organization of the final repository project.

Tidsperspektiven i svenska samhällsbeslut ("Time perspectives in societal decisions in Sweden"), Royal Institute of Technology, Stockholm

The final report Tidsperspektiven i svenska samhällsbeslut ("Time perspectives in societal decisions in Sweden", Hansson 2012) is an overview of problems and principles associated with decision-making that have effects in the future. It is divided into a theoretical part and an empirical part.

The theoretical part discusses several different methods and principles for evaluating future outcomes of the decisions we make today. One common method is to discount the future values using a discount rate, just as in the case of ordinary cost calculations. To do this, all values must be translated into money terms, including losses of human life and environmental values. One of the difficulties with discounting is that we do not have any generally accepted methods for determining such monetary values. Another difficulty is that discounting leads to absurd results when it is applied over long time periods. If we use a discount rate of three percent, for example, an action that leads to the death of ten billion people in the year 2800 will be regarded as less serious than an action that leads to the immediate death of one person.

Another principle for the assessment of future effects is sustainable development. While discounting is a very precise concept, it is much more difficult to clarify what sustainable development entails in exact terms. A distinction is usually made between two main interpretations: weak and strong sustainability. By "weak sustainability" is meant that we should meet the needs of the present generation without making it more difficult for future generations to meet theirs. If so, it is permitted to deplete a natural resource, as long it is replaced with something else, for example a new technology those future generations can use instead. The concept of "strong sustainability" entails considering each resource as unique and requiring that each one be preserved.

The report recommends that a distinction be made between different types of natural resources. Purely technical natural resources can be handled with the concept of weak sustainability and therefore with the ordinary discounting model, given that the time spans are so short that we can assume an interest economy similar to the present one. Other resources, in particular those related to biodiversity, should instead be treated with the concept of strong sustainability.

Two other principles are discussed more briefly. One is the moral philosophical discussion, which has mainly been concerned with the decisions people make about their own future. Most philosophers who have dealt with this issue have contended – that since different time periods belong to a human life to an equal extent – it is not reasonable to prefer a benefit now over the same benefit at some future point in time. The second principle is uncertainty discounting, which is based on the principle that we usually find it reasonable to attach less importance to uncertain benefits and costs than to certain ones. As a rule, the further in the future an event is expected to occur, the more uncertain it is. In practice, uncertainty discounting will therefore to some extent have the same effect as time discounting.

In the empirical part, interviews have been done with Swedish authorities and sector representatives regarding their time horizon, i.e. how far in the future they analyze and plan, as well as their economic valuation tools, in particular discount rates.

The interviews revealed great differences between the time horizons applied in different societal sectors. In most areas, the planning horizon is shorter than 30 years. In some environmental issues – particularly those related to climate change, forests and the survival of species – the time horizon is 100 years. Management of spent nuclear fuel proved to be the only area where detailed analyses are regularly made with a time horizon of more than 100 years. Here the perspective is as long as 1,000,000 years or more. This long time horizon is warranted by reliable scientific information concerning how long the fuel remains radiotoxic.

To some extent, these differences in time horizon are due to differences in knowledge bases and planning needs, but some differences between the time horizons in the different sectors are difficult to explain. The lack of systematics is striking, and according to the author of the report, it would be wise in many cases to consider the possibility of using a longer time perspective than is now done. This is particularly true when dealing with serious and irreversible changes, for example relating to the survival of species and future climate change.

The interviews also revealed a remarkable lack of systematics in the choice of discount rate. The discount rates used in the transport sector have traditionally had a great influence far beyond the transport sector. The Swedish Transport Administration's economists are very thorough in their analyses, but they naturally have a heavy focus on the transport sector. The four percent discount rate used in the transport sector is considered by the author to be remarkably high when applied, for example, to environmental detriment in a hundred-year perspective. For example, a negative effect of climate change that occurs in 100 years would only be counted as 1/50th as serious as if it had occurred today.

The use of discount rates in government agencies appears to be ill-considered and unwarranted, according to the report. There is good reason to consider alternative approaches, including the aforementioned proposal to apply traditional economic discounting only to economic benefits and use other computational methods for quantities that cannot be assigned a market price, such as human life, health and environmental values. Such an approach is in practice already applied in the nuclear waste field. The economic costs of the Spent Fuel Repository are discounted in the customary fashion, while future environmental and health effects are not discounted but regarded as just as serious as if they had occurred today. This approach appears to correspond well with the ideal of sustainability and with a generally accepted view of our responsibility to future generations. There are very strong reasons for continuing to apply it in the nuclear waste field. Moreover, there are strong reasons for trying this approach in other areas as well where long-term decisions have to be made, for example in decisions relating to climate change and biodiversity.

29.4 Comments and evaluation

29.4.1 Social Science Advisory Group's comments

A first summarizing report on the research carried out in the social science research programme was published in late 2009 (SKB 2009e) and was presented briefly in RD&D programme 2010. An updated version, which was also published in an English translation, (SKB 2011f and SKB 2011g, respectively), was produced in the spring of 2011 for an international conference marking the conclusion of SKB's social science research programme.

The persons responsible for these summarizing reports were the three researchers forming the Social Science Advisory Group for the research programme, see Section 29.1. The reports contain a presentation of all completed projects and the most important results, along with comments on the social science research programme's activities and the need for future research.

The Social Science Advisory Group's comments were formulated around different themes and are summarized as follows.

Character of the programme – applied research with breadth and depth

The premise for SKB's social science research programme has been that the knowledge yielded by the programme should contribute to a broadened knowledge base regarding the possible effects and societal problems entailed by the nuclear waste. The programme has been an example of sectoral research and has therefore had a clearer focus than any programme within e.g. the Swedish Research Council or the Riksbank Fund could have.

A question posed by the Social Science Advisory Group was why state research councils and regulatory authorities have not created major programmes to fund research projects in such an important field as the social and societal dimensions of nuclear waste. The Group believes that such programmes would have complemented the research initiated by SKB.

"This has not, however, been the case... As a consequence, much of our knowledge concerning social science and humanities perspectives on the nuclear waste issue in Sweden has been obtained through SKB's Social Science Research Programme, even though there are a few research projects funded also by other sources" (SKB 2011g, page 73).

According to the Social Science Advisory Group, SKB's social science research programme has involved applied research of relevance to the disposal of Swedish spent nuclear fuel. But applied research is not necessarily uncritical research, the Group stressed and noted that projects and contributions to the programme's yearbooks include both results that can be perceived as outspokenly critical to phenomena, decisions, actions and central actors in the field (including SKB) and results that are in line with current plans to develop a final repository for spent nuclear fuel.

Quality and effects of the programme

The Social Science Advisory Group emphasized that there has been no shortage of good project proposals, but due to constraining circumstances it has only been possible to grant funding to the best and most important research areas. The following criteria have guided the assessment of applications:

- The research projects must focus on topics linked to SKB's task of managing and disposing of Sweden's spent nuclear fuel. They must enhance the quality of the data underlying SKB's and the concerned municipalities' future decisions relating to the siting of a final repository for spent nuclear fuel. The research projects must broaden perspectives on and enhance knowledge of the nuclear waste issue.
- Purpose, problem, work plan, method and expected results must be clearly formulated. The expected results must be relevant for both the research area and SKB. The costs of the projects must be reasonable and realistic. Applied research is prioritized. The research must be of high quality and based on the scientific state-of-the-art in each discipline.
- The participants in the research projects must be well-reputed in their fields, be well-acquainted with the background and content of the siting process and be updated on the current situation in SKB's programme. In contrast to the research being conducted with funds from public research councils and foundations, the projects supported by SKB thus have a clear character of applied research. To be relevant they must also naturally be of high scientific quality.

The Social Science Advisory Group's conclusions concerning the effects of the programme can be summarized as follows:

- A contact network of all participating researchers at different universities has been built up, which has resulted in an accumulation of knowledge in the Swedish research community and facilitated future research efforts in the area.
- The nuclear waste issue has gained greater visibility and more and more researchers can therefore see how their special expertise can contribute new knowledge.
- The 18 projects within the framework of the programme have in different ways contributed knowledge of relevance for decision-making in the final repository issue. The projects cover a number of research fields, including the behavioural sciences, the social sciences, economics, law and history.
- The programme has attracted prominent and widely published researchers with large international contact networks.
- The findings of the projects regarding societal processes, attitudes and decision-making should be able to be generalized to other activities and be used in other research, for example environmental and energy research with ties to other industrial activities.

Need for continued research

The Social Science Advisory Group posed the question as to whether the programme had from the start missed some central dimension or important theme and whether any new research areas or research problems had emerged during the course of the projects. No clear answer was given to this question; instead, the Group looked at each of the four research areas and identified a number of questions that were deemed to be of interest to explore further in the future, in research funded either by SKB or by other funding bodies. In summary, the Group arrived at the following conclusions:

Socioeconomic impact – macroeconomic effects

The two projects that have been carried out in the area "Socioeconomic impact – macroeconomic effects" have been narrowly focused on the local effects of the establishment of a final repository for spent nuclear fuel on population and employment. Other questions may also need to be explored by means of research, such as: When faced with the construction of a final repository that takes a relatively long time, how should local planning be done so that a) future educational needs are met, b) local capabilities to deliver input materials are strengthened, c) opportunities to establish complementary businesses are exploited, d) the housing requirements of the local population are met (so that they choose to stay), and e) the local culture sector is made more attractive?

There are also other issues with broader implications. Different projects have shown that the primary socioeconomic effects of a repository occur outside of the municipality in which the repository is built. What are the consequences of this for the region and the nation and how can this development be influenced? One question with even broader implications concerns how the final repository can be related in terms of resources to the country's overall energy supply and was formulated as follows (SKB 2011g, page 79):

"A potentially relevant question is then: if one applies the same criteria for risks, benefits, costs and environmental impacts to all parts of the energy supply system, what scenarios would then appear to be nationally and globally suitable?"

Decision processes

The results of the projects that have been carried out in the area "Decision processes" pointed towards great complexity already in the closely-related work of handling essential decision-making dimensions. The Social Science Advisory Group said that the problem is in part of a formal nature and has to do with perceived ambiguities in regulations, roles and processes. Another aspect is social and concerns which persons participate and their contributions, while a third aspect is theoretical and concerns what different principles and methods can be utilized as point of departure. More research that discusses different points of departure in relation to the final repository is therefore needed, along with different ways of evaluating consequences over extremely long periods of time.

There is also a need to shed more light, in both breadth and depth, on different kinds of existing formal decision structures and decision processes at the national, regional and municipal levels, by public authorities and other bodies, as well as the content of and principles behind the work that is done. Questions concerning relationships between national and international decision-making, and how it is regulated and works, also need to be further explored. At the individual and group level, there is also reason to take a closer look at what factors determine the positions that are taken on an issue, and how they are presented and negotiated before a decision is made.

Public opinion and attitudes

The projects that have been carried out in the area "Public opinion and attitudes" have provided an accurate picture of the public opinion situation in the concerned municipalities and the country as a whole, however during a relatively brief period. The Social Science Advisory Group indicated the need to follow attitudes of private individuals over a longer period of time to see to what extent changes occur, i.e. to use longitudinal data.

Studies of the following type were called for:

- Before-and-after surveys where reactions to events and decisions are systematically studied with various data collection instruments.
- An international study (in cooperation with one or more international research organizations) of attitudes towards final disposal, nuclear waste and nuclear power in different nuclear power municipalities in the world.
- More detailed studies of how people (experts, the general public, young and old, etc) make various kinds of decisions.
- Who constitute "the silent majority", how do they reason and what does it entail to refrain from becoming involved?
- The importance of new or different life styles and communication styles for attitude formation.
- Do attitudes change in the course of the progressing from a newly awakened interest to personal participation in discussions or decision processes?

Global changes

The Social Science Advisory Group recalled that the calls for proposals had called for research on global change processes, the nuclear waste issue in Europe, international legislative developments and the importance of new threats and risks as well as new technology. The projects in the area "Global changes" have explored some of these issues. But at the same time these issues are constantly changing and therefore remain ever topical.

The Social Science Advisory Group judged that changing perceptions and realities connected to threats and risks, as well as new ownership conditions and technological breakthroughs, will entail new starting points for the decisions that may or will be made regarding the nuclear waste. Research on conceivable and actual long-term changes in the way societies are governed, including the rules and forms governing participation, is therefore also relevant both to people's perceptions and to decisions on the management of nuclear waste. The Group particularly pointed to changes such as the fact that during the period the social science research programme has existed, there has been a shift of political power in Sweden, the EU has undergone enlargement and further integration, increasingly grim reports on future environmental threats have been presented, and, not least, a daunting international economic crisis has afflicted our part of the world. In retrospect, the nuclear waste programme has remained within its given framework, while the external global environment has changed. Increased use of new forms of media and information technology, changing life-style ideals and sources of identification, perceived economic insecurity, and an internationalization of finance and industry as well as of many professions and courses of education were viewed by the Social Science Advisory Group as examples of global societal changes that have direct or indirect repercussions on perceptions and decisions. Long time series of data and constant research efforts are often required to capture and comprehend these different kinds of processes. Such research efforts lay beyond the scope of this programme.

29.4.2 Independent evaluation

A consultant was engaged by SKB to conduct an evaluation of SKB's social science research programme. His report (Söderberg 2012) was presented at a seminar arranged by SKB in September 2012.

The evaluation focused a number of general questions concerning the purpose and effects of the programme, plus a number of questions concerning its execution. The assignment was not to undertake some kind of scrutiny of the quality of the scientific results reported by the different projects. Responsibility for that assessment has rested with the Social Science Advisory Group.

The results of the evaluation have been summarized below.

Origin of the programme

SKB's programme for social science research was designed during the period 2002–2003 in response to the discussions of the need for such research on issues relating to the final disposal of spent nuclear fuel that had been going in SKB's world for 15–20 years. Since the late 1980s, the Swedish National Council for Nuclear Waste had held a number of seminars attended by researchers and decision-makers dealing with such issues as democracy, decision-making in complex matters and, not least, the ethical aspects of alternative courses of action. In connection with the review of RD&D programme 2001, the Swedish National Council for Nuclear Waste (called KASAM in Swedish at that time) pointed out the need for qualified social science research in the nuclear waste field. The fact that leading politicians in the municipalities where SKB was conducting site investigations saw a need for this research was probably a particularly important factor in the initiation of SKB's social science research programme. Since the Government was not prepared to earmark state research funds for these purposes, SKB had no alternative but to establish its own research programme.

Purpose of the programme

SKB had formulated the purpose of its programme for social science research in terms that reflect both an uncertainty regarding the possible results of the research and openness to such results. However, the point of departure was that one of the company's main tasks within the foreseeable future would be to submit applications for licences to build a final repository for spent nuclear fuel on a suitable site in Sweden. The most essential aspects of the purpose of the social science research programme is expressed in these formulations: "to broaden the perspective on the societal aspects of the Nuclear Fuel Programme" and "to provide deeper knowledge and a better body of data as a basis for site- and project-related studies and analyses."

Some of the researchers who have been interviewed for the evaluation expressed the viewpoint that SKB's social science research programme ought to have had broader purposes than those that were formulated.

Such a purpose should, according to this reasoning, have been that the results of this research would be carefully considered within SKB with a view to the possibility that the results could lead to *essential* changes in the direction of the scientific and technical development work concerning the management and final disposal of spent nuclear fuel being pursued by SKB.

This reasoning is rejected in the evaluation report, which says that it is not realistic to expect that SKB should, at the beginning of the 21st century, have formulated a purpose for its social science research programme implying that the results of this research could lead to a questioning of the direction of the technical and scientific development work that had been conducted for the past quarter century.

In the interviews conducted for the evaluation, it had also been suggested that research on societal aspects of nuclear waste management should include issues that more properly lie within the sphere of responsibility of national energy policy than within SKB's and its owners' responsibility for the management of spent nuclear fuel under current legislation. According to the evaluation report, the most obvious conclusion is that if the national political bodies (the Government and the Riksdag) see a need for further research on such broader topics, they should seek to initiate and fund such research and formulate its purpose.

Results and effects

A contact network of all participating researchers at different universities has been built up, which has resulted in an ongoing and accelerating accumulation of knowledge in the Swedish research community and has facilitated future research efforts in the area. At least one university (Linnaeus University) is considering ways to devote further research to the societal aspects of nuclear waste management.

Otherwise, the findings of the evaluation are in line with the Social Science Advisory Group's comments as regards the effects of the programme (see Section 29.4.1).

Strengths and weaknesses

The strength of the completed programme of social science research lies primarily in the fact that it came about at all, and that it was carried out with clear goals and in forms that ensured high scientific quality of the reports from the different projects. A weakness was that SKB's involvement simultaneously placed the legitimacy of the research under suspicion. The evaluator notes that this weakness is by no means unique to SKB's social science research programme, but characterizes all research funded by industrial enterprises on issues of direct interest to the activities of the enterprise.

Forms for reporting of results

SKB has made an effort to achieve a broad dissemination of the results of the research projects in the social science area, for example by publishing popular annual reports on current and recently completed projects. The participating researchers have been encouraged to publish their research in the international scientific literature, but the research grants do not seem to be generous enough to facilitate such publication.

Continued social science research in the nuclear waste area?

The evaluation report asserts that as a result of SKB's social science research programme, both SKB and the Swedish National Council for Nuclear Waste, as well as individual researchers and others, have come up with many other ideas for future research. Regardless of the outcome of the ongoing review of SKB's applications for licences under the Environmental Code and the Nuclear Activities Act to construct a final repository for spent nuclear fuel in Forsmark, there will probably be a need for further research on nuclear-waste-related topics of a humanistic and social-scientific nature. The scope and forms for a continuation of SKB's social science research programme should be considered by the company management.

Should SKB contribute financially to further social science research in the nuclear waste area?

The evaluator says that it is clearly in the interests of society that research of a humanistic and social-scientific nature be conducted on nuclear-waste-related topics. With its social science research programme 2004–2011, SKB has shown awareness of the need to broaden the scope of its research beyond the technical and scientific boundaries that confined it prior to the 1990s. SKB has also demonstrated that it is able to carry out a programme with good quality research on such topics. There are no fundamental objections to SKB's continuing to fund such research. But the lessons gained from the discussions of the completed social science research programme should lead to consideration of other forms of support.

Forms for possible grants from SKB

The evaluation report says that if SKB wants to support research of a humanistic social-scientific nature on nuclear-waste-related topics in the future, SKB should consider models that better avoid the risk that different stakeholders will question the legitimacy of such SKB-sponsored research. An example of such a model is that SKB pledges certain sums for research within certain disciplines for a given period of time, but arranges for another funding body to handle calls for proposals and project selection. Another model may entail that SKB provides long-term funding of chairs at a university that intend to focus on humanistic and social science topics associated with nuclear waste management. The pros and cons of these models, along with certain others, were discussed at the aforementioned seminar where the evaluation was presented.

29.5 How SKB views the need for continued social science research around the final repository

The necessity of formulating and investigating other issues than purely technical and scientific ones in connection with the work of designing systems for the management and final disposal of the residual products of nuclear power – often called nuclear waste – has long been clear. Back in the late 1980s, the predecessor of the Swedish National Council for Nuclear Waste (KASAM), as well as the former National Board for Spent Nuclear Fuel, called for studies of these matters of a humanistic and social-scientific nature. During the 1990s feasibility studies for siting of a final repository for spent nuclear fuel, SKB devoted considerable interest to societal aspects. Individual researchers in the humanities and social sciences at universities and colleges also began to devote interest to these matters. Interesting results were published in a number of articles in scientific publications and in some large studies, but do not seem to have attracted much notice beyond the immediate circle of interested scientists. Proposals by the Swedish National Council for Nuclear Waste to the Government to allocate earmarked financial resources for broader-based research of a humanistic and social-scientific nature on the problems surrounding the waste from nuclear power did not lead to any concrete measures.

SKB's initiative in the early 2000s to fund and carry out a social science research programme was more immediately occasioned by, and therefore focused on, issues associated with one of SKB's main tasks. This task is to develop and deploy safe systems for the final disposal of the most hazardous residual product of nuclear power: spent nuclear fuel. Eighteen different research projects, covering four areas of research which overlap each other in some respects, have been carried out within the framework of the social science research programme 2004–2011. These areas are entitled "Socioeconomic impact – macroeconomic effects", "Decision processes", "Public opinion and attitudes – psychosocial effects" and "Global changes".

The results have contributed knowledge of relevance to final disposal and have been disseminated in particular to interested citizens in the concerned municipalities. The contact network that was built up between participating researchers at different universities and technical institutes resulted in an accumulation of knowledge and facilitated future research efforts in the area. Another effect is that the nuclear waste issue has achieved greater visibility. More researchers can therefore see how their special expertise can contribute new knowledge. Researchers who have been in charge of or participated in the projects have been encouraged to publish the results internationally.

In their reviews of RD&D Programme 2007 and RD&D Programme 2010, both the Swedish National Council for Nuclear Waste and the Swedish Radiation Safety Authority have expressed appreciation of the social science research programme. There has, however, been some criticism, chiefly that the programme should have included studies of other problems as well. At the same time, both the Swedish National Council for Nuclear Waste and the Swedish Radiation Safety Authority have emphasized that research of a humanistic and social-scientific nature on the nuclear waste issues should receive financial support from others besides SKB. In its review statement concerning RD&D programme 2010, the Swedish National Council for Nuclear Waste urged the Government to study the forms for independent research of social and humanistic aspects on the nuclear waste issue and to allocate earmarked funds for such research in the coming decades.

It can be observed that the results of SKB's social science research programme are also of interest for SKB's work with the final disposal of the low- intermediate-level nuclear waste. The research projects' survey of societal processes, attitudes and decision-making can in many cases be generalized to these activities and should also be able to be utilized in other research, for example environmental and energy research with ties to various industrial activities.

The reasons that led SKB in the early 2000s to initiate a social science research programme in the form in which it existed no longer carry the same strength. At the same time, the programme has contributed to an accumulation of knowledge that would probably not have occurred without this initiative. An increasing number of social scientists and humanists have begun to take an interest in these matters. The scientific community, as well as private citizens and political leaders, have increasingly come to appreciate the need for and value of this research. In SKB's view, further research in this area should primarily be funded in the manner that is customary in the academic world, i.e. by having researchers apply for funds to various research funding bodies, for example state and private research councils or the like.

SKB therefore does not intend at present to initiate a new research programme of the type that has now been concluded. One reason for this is the fact that the processing of SKB's applications under the Environmental Code and the Nuclear Activities Act for licences to build a final repository for spent nuclear fuel in Forsmark is currently under way and is not expected to be finished before 2015 at the earliest. However, SKB is prepared to fund research projects of a social-scientific nature within areas deemed to be of importance for SKB's activities, particularly in the municipalities of Oskarshamn and Östhammar.

30 Information preservation across generations

30.1 Conclusions in the review of RD&D programme 2010

In its review of RD&D Programme 2010, the Swedish Radiation Safety Authority (SSM) commented on the lack of an account of SKB's work on information preservation linked to the final disposal of spent nuclear fuel, but observed at the same time that SKB has previously conducted a study in the area.

Since RD&D programme 2010, SKB has initiated a project dealing with the issue of the preservation of records and knowledge on the final repository for spent nuclear fuel.

30.2 Background and previous work

SKB's method of final disposal of the spent nuclear fuel entails that no active measures will be necessary, in the form of for example maintenance or improvements, for the repository to remain safe. The existence of the Spent Fuel Repository should not constitute a burden on future generations. Knowledge and information on the final repository for spent nuclear fuel should, however, be passed on to future generations, among other things in order to avoid inadvertent intrusion.

The project NKS CAN-1.2 – "Conservation and retrieval of nuclear information" (Jensen 1993) was carried out within the framework of Nordic Nuclear Safety Research (NKS). The purpose of the CAN project was to answer the following questions:

- Which type of information should be preserved for future generations?
- In what form should the information be preserved?
- What quality should the information have as regards type and form?
- How can the information be retrieved even after a very long time?

The CAN project attracted a great deal of attention in international circles. In 2003 the IAEA arranged an international conference in Rome to address questions relating to the preservation of knowledge and information for a long time, and the results of the CAN project were reported there.

In 2007 SKB compiled information on what has been done in the area in Sweden and certain selected countries (Bowen-Schrire et al. 2007). In 2008, essential aspects of information preservation were presented, along with risks that can lead to the loss of important information. Based on this, a proposed plan of action for SKB was also described (Bowen-Schrire et al. 2008).

The question of how information can be preserved and who bears responsibility has come up in the consultations for Clink and the Spent Fuel Repository and in connection with the licence applications under the Environmental Code and the Nuclear Activities Act for these facilities. The subject has received attention in the media, and wishes have also been expressed by the regulatory authorities, in particular by the former Swedish Radiation Protection Authority in the consultations, that SKB should propose a plan of action for how information and knowledge can be preserved for a very long time.

30.3 Forms for transfer of information

There are two fundamental principles for how information can be passed on to future generations: successive information transfer and information transfer aimed directly at a distant future. Successive information transfer entails that people transfer information to each other, for example by creating an archive that can be accessed by future generations. In the case of information transfer directly to a distant future, physical markers can be used to indicate the presence of a final repository in the landscape in a way that will endure for a very long time. The marker should convey a message that entails some kind of warning. The countries that are working with long-term information preservation are all focused on successive transfer. Some countries are also working with information transfer directly to a distant future.

The future target group for this knowledge and information may vary. It could, for example, be the next generation of construction project administrators, regulatory authorities, politicians, prospectors for natural resources, researchers or private citizens. Different target groups have different needs for information and knowledge.

One important stakeholder in the area of information and records preservation is SSM. Sweden has acceded to the Nuclear Non-Proliferation Treaty, whereby it has undertaken to submit the country's entire nuclear energy programme – including final repositories – to international inspection. According to an agreement with the UN's nuclear energy agency, the IAEA, Sweden must have a system for nuclear safeguards that enables the IAEA to carry out international inspections. In Sweden, nuclear safeguards are implemented by SSM, which issues regulations regarding what information is to be preserved and how it is to be preserved. The question of how safeguards for a final repository for spent nuclear fuel are to be applied and what information is to be preserved must first be dealt with by the international bodies that carry out the inspections, after which it is SSM's responsibility to incorporate international treaties into Swedish regulations.

Regardless of who the future recipients are, it is important that the information be readily accessible and easily comprehensible. It must be able to be read physically, but its content must also be understandable for the readers. Preserving knowledge and information on the Spent Fuel Repository far into the future thus makes demands on more than just archiving.

30.4 Current work

The important questions regarding information preservation need not to be resolved before the final repository for spent nuclear fuel is to be closed and sealed, probably no earlier than around 2085. It is not possible, either for SKB or the concerned authorities, to know today how best to proceed so far in the future. The overall goal of SKB's current work is therefore to find ways and means to continue to keep the question of how to preserve information and knowledge on a final repository for radioactive waste after closure alive and updated. Furthermore, an analysis is needed of the extent to which long-term preservation of data emerging from the work of developing the method for final disposal of spent nuclear fuel and building the final repository is necessary.

SKB's current efforts include its own research projects, participation in an OECD-NEA project and cooperation with Andra, SKB's French counterpart. The goal of SKB's own research is to learn more about how information and knowledge can be preserved for a long time. This is done by considering how we have acquired the knowledge we have today about historical and ancient phenomena as well as how language develops and changes over time.

In early 2012, SKB initiated its first research project concerning information preservation, "One hundred thousand years back and forth", at Linnaeus University in Kalmar. This research examines three areas and considers the following issues:

- Time and the distant future. How can an archaeological understanding of human development contribute to an understanding of both preservation and communication a hundred thousand years in the future? Is there any real human continuity over such a long time?
- Remembering, learning and understanding history within the framework of man's historical consciousness and collective memory. What can we learn from current debates on the didactics and pedagogy of history to aid in communication with future generations? What can a discussion of historical consciousness, different concepts of time and cultural/collective memory contribute to decisions on strategies for information preservation far into the future after closure of a final repository for spent nuclear fuel?
- Nuclear waste, cultural heritage and the future. What is known (or can be assumed) about the future within the cultural heritage sector? What is known (or can be assumed) about the future within the nuclear power sector? How is this knowledge in each case translated into research assignments, cultural heritage management methods and guidelines, and how do they match one another?

The assignment to Linnaeus University is planned to take three years, which means it will be finished in 2014. The primary results will be reflections on the questions in scientific articles. Another important effect of the work is to spread interest in the question of how information can be preserved for long periods of time to other fields of expertise and other countries via the participation of researchers in international conferences within other areas than radioactive waste management.

SKB has also initiated another research project in the field of information preservation. This project is planned to start at the Centre for Theology and Religious Studies at Lund University at the end of 2013. The project will examine language-related questions. How should we think and work in order to make it easier for people in the future to understand the language we use to pass on information? What is needed to reconstruct an extinct language? How should a technology be described so that it is understood, even after it has been abandoned and forgotten?

The assignment is for one year and involves examining the above questions in the context of the question of how to preserve information about the Spent Fuel Repository and publishing articles in scientific journals. This assignment will also have the effect of spreading interest in the question to new research fields, since the articles will be submitted to journals outside of SKB's technical and scientific research areas.

Twelve countries, including Sweden, are participating in the OECD-NEA project "Preservation of Records, Knowledge and Memory (RK&M)" with efforts in the following areas:

- Assessment of different technical and administrative means for transferring information from one generation to another and directly to a distant future.
- Evaluation of measures to mitigate and adjust potential information losses.
- Analysis of technical and organizational perspectives for information preservation against the background of previous societal development.
- Initiative for internationally harmonized methods in order to avoid unnecessary differences between different countries' actions.
- Analysis of the economic challenges of long-term information preservation and suggestions for how to include them in future programmes.
- Establishment of effective forms for all concerned parties to work together both nationally and internationally.

The project started in September 2011 and will continue until 2014 (OECD/NEA 2011). The primary concrete end product is planned to be a menu-driven document that will allow people to identify the elements of a strategic action plan for preservation of records, knowledge and memory of a final repository for radioactive waste. The document is planned to take the form of an electronic database.

Furthermore, SKB is involved in a joint project with its French counterpart, Andra, concerning information preservation. Andra has an extensive programme for the next few years, for example concerning cultural heritage and social issues:

- The general public's perception of long-term timescales (ethics, philosophy, sociology etc).
- Archaeology in the landscape.
- Preservation of historical sites and industrial memory.
- Continuity of institutional organizations.
- Consequences of social breakdowns (war, natural disasters etc).
- Memory and the history of science.

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SKB's (Svensk Kärnbränslehantering AB) publications can be found at www.skb.se/publications. References to SKB's unpublished documents are listed separately at the end of the reference list. Unpublished documents will be submitted upon request to document@skb.se.

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SKBdoc id, version	Title	Issuer, year
1091554 ver 3.0	Säkerhetsredovisning för drift av slutförvarsanläggning för använt kärnbränsle (SR-Drift) kapitel 3 – Krav och konstruktionsförutsättningar. (In Swedish.)	SKB, 2010
1179234 ver 1.0	Referensrapport till SAR allmän del kapitel 6 – Källtermer. (In Swedish.)	SKB, 2009
1199888 ver 1.0	Verksamhet, ledning och styrning – Uppförande. (In Swedish.)	SKB, 2011
1323062 ver 1.0	Begäran om förtydligande information (In Swedish.)	Strålsäkerhets-myndigheten, 2011
1333256 ver 2.0	Svar på begäran om kompletteringar angående kapselfrågor. (In Swedish.)	SKB, 2012
1339709 ver 1.0	C-14 accumulated in ion exchange resins in Swedish nuclear power plants	SKB, 2012
1359832 ver 1.0	Avveckling och rivning av kärnkraftblock (In Swedish.)	SKB, 2012
1378692 ver 1.0	Report to SKB on the rational synthesis of the isosacarinic acids.	SKB, 2013
1416968 ver 1.0	Low and Intermediate Level Waste in SFL 3-5 - Reference Inventory	SKB, 1998

Appendix

Abbreviations

Ab initio	Latin for “from the beginning”.
ABM	Alternative Buffer Materials. Experiment at Äspö HRL where possible buffer materials are investigated.
ALARA	As Low As Reasonably Achievable. Keeping the radiation doses as low as reasonably achievable, economic and social factors taken into account.
ANDRA	Agence National Pour la Gestion des Dechets Radioactifs, France.
ANU	Model for ice sheet reconstructions.
APSE	Äspö Pillar Stability Experiment. Concluded experiment at Äspö HRL for studies of how large load the rock can take.
Asha	Indian bentonite from the Kutch region.
ASIED	Alpha Self-Irradiation Enhanced Diffusion.
ATB	Waste transport container.
ATB 1T	A new container for transport of long-lived low- and intermediate-level waste in BFA tanks.
BA	Burnable Absorber.
BAT	Best Available Technology.
BELBAR	Bentonite erosion: effects on the long term performance of the engineered barrier and radionuclide transport. EU project on buffer stability and buffer erosion.
BFA	Rock cavern on Simpevarp Peninsula for dry interim storage of operational waste.
BFA tank	Tank for dry interim storage of long-lived low- and intermediate-level waste.
BIOPROTA	International collaborative project on key issues for assessment of long-term radiological safety in the biosphere.
BIPS	Borehole Image Processing System. Video photography of the borehole wall.
BKAB	Barsebäck Kraft AB.
BLA	Rock cavern for low-level waste in SFR.
BMA	Rock cavern for intermediate-level waste in SFR.
BRIE	Bentonite Rock Interaction Experiment. Experiment at Äspö HRL to gain a better understanding of how water moves from the rock to the bentonite buffer in the Spent Fuel Repository.
BRT	Rock cavern for whole reactor pressure vessels.
BTF	Concrete tank repository in SFR, mainly intended for dewatered ion exchange resins.
BWR	Boiling water reactor.
CAD	Computer Aided Design.
CAPS	Counterforce Applied to Prevent Spalling. Concluded experiment at Äspö HRL to investigate ways to reduce the risk of rock breakout.
CIEMAT	Centro de Investigaciones Energéticas, Medioambientales y Tecnológicas, Spain.
Clab	Central interim storage facility for spent nuclear fuel.
Clink	Central facility for handling, interim storage and encapsulation of spent nuclear fuel.
CRT	Canister Retrieval Test. Concluded experiment at Äspö HRL to investigate ways to retrieve an already deposited canister from a deposition hole.
CSH	Calcium silicate hydrate.

DETUM	Project for further development of methods, tools and programmes for investigations and modelling prior to start of construction of the Spent Fuel Repository.
DFN	Discrete Fracture Network model.
DIC	Dissolved Inorganic Carbon.
DOC	Dissolved Organic Carbon.
DOMPLU	Dome Plug Experiment. Full-scale test at the Äspö HRL to test and demonstrate the complete plug system. The test is a part of the joint EU project DOPAS (Demonstration of Plugging and Sealing).
EDZ	Excavation Damaged/Disturbed Zone. The rock around a rock excavation where irreversible changes have taken place.
EFPC	Expanded Full Perimeter Criterion. A criterion indicating that a canister position in the Spent Fuel Repository may not be intersected by a fracture that also intersects the full perimeter of the deposition tunnel.
EPRI	Electric Power Research Institute, USA.
ERICA	Environmental Risk from Ionizing Contaminants. Tool for analyzing biological effects of ionizing radiation in habitats and ecosystems.
EU	European Union.
FEM	Finite Element Method.
FEP	Features, Events and Processes. The factors that could affect the safety of different types of repositories. They are documented in the international FEP database administered by the OECD/NEA.
FKA	Forsmarks Kraftgrupp AB.
FPC	Full Perimeter Criterion. A criterion indicating that if a fracture is observed around the full perimeter of a deposition tunnel in the Spent Fuel Repository, no deposition hole may be positioned so that it would intersect the extrapolated extension of the fracture.
FSW	Friction Stir Welding.
Gadd	Common waste database for registration and reporting of the low- and intermediate-level short- and long-lived waste that is managed or disposed of at SFR, Clab, OKG, BKAB, and FKA.
GAP	Greenland Analogue Project. Project in cooperation with Posiva and NWMO for the purpose of learning, by observations at an existing inland ice sheet, more about how groundwater flow and groundwater chemistry in crystalline bedrock is affected by an ice sheet. The results are used for assessment of long-term safety at the final repository for spent nuclear fuel.
GEUS	De Nationale Geologiske Undersøgelser for Danmark og Grønland (Geological Survey of Denmark and Greenland).
GIA	Glacial Isostatic Adjustment. Model for analyzing the isostatic response to a given ice load history.
GIS	Geographic Information System.
GRASP	Greenland Analogue Surface Project. SKB programme aimed at identifying differences in long-term change processes in near-surface systems between a cold and a temperate climate, and investigating how the hydrological properties and the properties of the ecosystems vary depending on climatic conditions.
HMS	Hydro Monitoring System.
HRL	Hard Rock Laboratory.
IRF	Instant Release Fraction. Term in safety assessment that defines the radionuclides in spent nuclear fuel that are assumed to be immediately soluble.
ISO containers	Containers of sizes standardized by the International Organization for Standardization (ISO) that can be loaded onto railway cars, trucks and ships.

ITU	Institute for Transuranium Elements, Karlsruhe, Germany.
JLH	Joint Line Hooking. A defect occurring in the welds produced by FSW.
KBS-3H	The KBS-3 method with horizontal deposition.
K_d	Sorption coefficient, partition coefficient.
KTB	Canister Transport Cask.
KTH	Kungliga Tekniska Högskolan (Royal Institute of Technology, Stockholm).
KTL	Nuclear Activities Act.
Lasgit	Large Scale Gas Injection Test. Experiment at Äspö HRL to investigate what happens with the gas formed when the canister insert corrodes.
LILW	Low- and intermediate-level waste.
LOT	Long Term Test of Buffer Material. Experiment at Äspö HRL aimed at finding out how bentonite clay behaves under conditions similar to those in a final repository for spent nuclear fuel.
LTDE-SD	Long Term Diffusion Experiment – Sorption-Diffusion. Concluded experiment at Äspö HRL to study to what extent different radionuclides enter the rock matrix.
MB	Environmental Code.
MIS	Marine Isotope Stage.
MMD	Land and Environment Court.
MOX	Mixed Oxide Fuel.
MTO	Man, technology, organization (human factors engineering).
MX-80	Sodium bentonite from Wyoming. Possible buffer material.
Nagra	Nationale Genossenschaft für die Lagerung von Radioaktiver Abfälle, Switzerland.
NDA	Nuclear Decommissioning Authority, UK.
NEA	Nuclear Energy Agency. A cooperation organization for nuclear energy matters within the OECD.
NPP	Nuclear power plant.
NUMO	Waste management organization of Japan.
NWMO	Nuclear Waste Management Organization, Canada.
OECD	Organization for Economic Cooperation and Development.
OKG	OKG Aktiebolag.
PEBS	Long-term Performance of Engineered Barrier Systems. EU project for the purpose of evaluating the sealing and barrier performance of a geological repository as well as how the function of engineered barriers changes with time.
PFL	Posiva Flow Log. Method for measuring permeability to water.
POC	Particulate Organic Carbon.
Posiva	Posiva Oy, Finland.
P-PSAR	Preparatory Preliminary Safety Analysis Report.
PRECCI	Programme de recherche sur l'évolution à long terme des colis de combustibles irradiés. Research programme in France concerning the long-term evolution of irradiated fuel.
PSAR	Preliminary Safety Analysis Report.
PSE	Project SFR Extension.
PSI	Paul Scherrer Institute, Switzerland.
PWR	Pressurized Water Reactor.
QA	Quality Assurance.
R&D	Research and Development.

RAB	Ringhals AB.
RAWRA	Radioactive Waste Repository Authority, Czech Republic.
RD&D	Research, Development and Demonstration.
RH	Relative humidity.
RNR	Radionuclide Retention Experiment. Experiment at Äspö HRL to investigate how the rock retards and filters radionuclides.
RPV	Reactor pressure vessel.
SAR	Safety Analysis Report.
SDM	Site descriptive model.
SFL	Final repository for long-lived waste.
SFR	Final repository for short-lived radioactive waste.
SGU	Geological Survey of Sweden.
SICADA	Site Characterization Database System. Database system for storing and managing data from the different types of geoscientific investigations conducted by SKB. Data from the experiments carried out at the Äspö HRL are also stored in the database.
SNR	Signal-to-Noise Ratio.
SNSN	Swedish National Seismic Network.
SR-Can	Preliminary assessment of safety for a KBS-3 repository at Forsmark and Laxemar with canisters according to the application for the encapsulation plant, published by SKB in November 2006. Can stands for canister.
SR-PSU	Safety Analysis Report for the SFR Extension Project.
SR-Site	Account of long-term post-closure safety of the final repository for spent nuclear fuel, published in March 2011. Site refers to the site of the final repository.
SSM	Swedish Radiation Safety Authority.
Suus	Safety under construction of the final repository.
SVAFO	AB SVAFO. Owned by Ringhals AB, Forsmarks Kraftgrupp AB, OKG Aktiebolag and Barsebäck Kraft AB.
SWIW	Single Well Injection Withdrawal tracer test. Concluded experiment at Äspö HRL to study how the water in the pores in the rock and areas with stagnant conditions affects the groundwater.
TASS	Tunnel for experiments in the Äspö HRL.
TBM	1) Tunnel boring machine. 2) Bored tunnel for experiments in Äspö HRL.
TBT	Temperature Buffer Test. Concluded experiment at Äspö HRL with French steel canisters to study whether two canisters can be deposited on top of one another and how the bentonite clay is affected by high temperatures.
TF EBS	Task Force on Engineered Barrier Systems. International collaboration between specialists and modelling groups on questions concerning the engineered barriers in the future final repository.
THM	Thermal-Hydro-Mechanical.
TRL	Transmitter Receiver Longitudinal technology.
TRUE	Tracer Retention Understanding Experiments. Concluded experiments in the Äspö HRL. Tracer tests on different scales to see to what degree results achieved on one scale are also valid on another.
TVO	Teollisuuden Voima Oyj. Finnish nuclear power company that is building a new reactor in Olkiluoto.