Technical Report
TR-22-11
December 2022

RD&D Programme 2022

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

SVENSK KÄRNBRÄNSLEHANTERING AB

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ISSN 1404-0344
SKB TR-22-11
ID 1997197
December 2022

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Svensk Kärnbränslehantering AB

This report is the English version of SKB RD&D programme 2022. In the event of any discrepancy between the Swedish and English versions, the Swedish original will take precedence.

This report is published on www.skb.se

Preface

Svensk Kärnbränslehantering AB (SKB), which is owned by the companies which operate the Swedish nuclear power plants, is responsible for the management and disposal of nuclear waste and spent nuclear fuel from the nuclear power reactors. Under the Act on Nuclear Activities, the licensees for the reactors must every three years prepare a programme for the comprehensive research and development work and other measures necessary to safely manage and dispose of the nuclear waste and spent nuclear fuel and to decommission the nuclear power plants. The licensees have entered into an agreement to delegate responsibility to SKB to prepare these RD&D programmes and submit them to the Swedish Radiation Safety Authority. SKB here presents the RD&D Programme 2022, which has been prepared in collaboration with the nuclear power companies.

A number of important milestones were reached in the previous RD&D period. In December 2021, the Swedish Government decided to grant SKB a licence to extend the Final Repository for Short-lived Radioactive Waste (SFR) in Forsmark in Östhammar Municipality. The facility needs to be extended to create space for decommissioning waste from the Swedish nuclear power plants. In January 2022, the Government decided to grant SKB a licence to build, possess and operate a final repository for spent nuclear fuel in Forsmark and an encapsulation plant in Oskarshamn Municipality, and determined that the activities covered by the application, involving a coherent system for final disposal of spent nuclear fuel, met the requirements with respect to choice of method and Best Available Technology. The latter was a historic decision that will enable SKB to dispose of the nuclear waste that our generation has created. The Government took the decision to increase the quantity of spent nuclear fuel for interim storage in Clab from 8 000 to 11 000 tonnes in August 2021, after it had detached SKB's application for this from the submitted application for a coherent system for final disposal of spent nuclear fuel. The Government's decisions have now created the conditions necessary for Sweden to implement comprehensive management and final disposal of Sweden's spent nuclear fuel and radioactive waste.

In the RD&D Programme 2022, SKB presents a system-wide perspective of the activities and the planned research and technology development. This RD&D programme takes the strategic direction one step further than the previous RD&D programme. For the establishment of the entire waste system, it is important that activities are carried out in a logical sequence which takes needs and dependencies into account, which is reflected in the schedule of activities and milestones set out in the plan of action, and which focuses on the relationships between activities and milestones for the facilities. Furthermore, SKB considers that the focus will shift and the scope of the RD&D programme will be reduced when the planned facilities transition to licensed facilities under the supervision of the Swedish Radiation Safety Authority. At the same time, the report should also include an overview of all activities and operations necessary to provide a sufficient understanding of the overall system. The Government's decisions mean that the Land and Environment Court can issue environmental permits and set conditions under the Swedish Environmental Code, and that the Swedish Radiation Safety Authority will continue a stepwise review process.

SKB's facilities for research, development and demonstration include the Äspö Hard Rock Laboratory (Äspö HRL). SKB plans to complete the tests at the underground facility of the Äspö HRL during the current RD&D period and it will be possible to complete the long-term experiments that are still under way during 2024. It is assumed that it will be possible to carry out remaining experiments and testing in the underground environment in conjunction with the construction of the final repository for spent nuclear fuel (Spent Fuel Repository) in Forsmark.

SKB has been involved in research and technology development for more than 40 years, and SKB's extensive knowledge exchange within the scientific community in general and between sister organisations in particular has created an invaluable international knowledge bank on the management and disposal of radioactive waste and spent nuclear fuel. SKB's open, knowledge-oriented and collaborative approach has contributed to international recognition of the KBS-3 method for geological final disposal of nuclear fuel. In addition to Sweden, the research and technology development results are now also being used in other countries, primarily in Finland.

SKB is celebrating its 50th anniversary and is also entering a new phase, with a strong purchasing organisation that will implement the government decisions and build new facilities. SFR will be extended to be able to receive operational waste and decommissioning waste from nuclear power reactors. After this, an encapsulation plant will be built for encapsulation of the spent nuclear fuel and the Spent Fuel Repository will be built for final disposal of the copper canisters. This will also require production facilities for copper canisters and buffer materials. The last nuclear facility which will be built by SKB is the Final Repository for Long-lived Waste (SFL), which is needed in order to be able to complete the decommissioning of the Swedish nuclear power plants. Since the publication of the RD&D Programme 2019, two reactors have been shut down: Ringhals 1 and Ringhals 2. Decommissioning of seven reactors is now under way, including the Ågesta reactor, decommissioning of which began in 2020. The extension of SFR has top priority among the final repositories that need to be built. Negotiations on terms and conditions will take place at the Land and Environment Court as early as the end of 2022, after which an application for construction will be submitted to the Swedish Radiation Safety Authority. When a judgement has been announced and approval of the preliminary safety analysis report has been obtained, work on the extension of SFR can begin.

SKB is now continuing the work of realising Sweden's largest environmental protection project and, through the task assigned to us, we are contributing to a fossil-free society within one generation.

Stockholm, September 2022

Svensk Kärnbränslehantering AB

Johan Dasht	Anders Ström	Robert Havel
CEO	Project Sponsor	Project Manager

Summary

The Swedish power industry has been generating electricity from nuclear power for more than half a century. Svensk Kärnbränslehantering AB (SKB), on behalf of its owners, is responsible for the management and final disposal of the nuclear waste and spent nuclear fuel. The purpose of the RD&D Programme 2022 is to satisfy the requirement to present the comprehensive research and development activities that are needed to develop and implement the remaining activities set out in the Act on Nuclear Activities (KTL). The programme presents the plans for implementing the remaining parts of the waste system for nuclear waste and spent nuclear fuel and for decommissioning of the nuclear power reactors. The activities pursued by SKB within the framework of a given licence are presented more comprehensively.

The Swedish waste system

The purpose of the following section is to provide a system-wide understanding of the Swedish waste system. The facilities and transport system needed for the reactor owners to be able to fulfil their obligations under current legislation are described in the programme, as is the planning that is under way to complete parts of the waste system that have not yet been commissioned.

It is the operation and decommissioning of the nuclear power plants that, in addition to conventional waste, gives rise to nuclear waste and spent nuclear fuel, which needs to be managed in such a way that humans and the environment are protected from the harmful effects of ionising radiation. Depending on how harmful the radiation is, the waste needs to be managed in different ways.

Extensive radiation protection measures must be taken for the management of the spent nuclear fuel and, until it is disposed of, it is placed in the Central Interim Storage Facility for Spent Nuclear Fuel (Clab), where it is stored in water, which both shields against radiation and cools decay heat. The spent fuel will be encapsulated in copper canisters in an encapsulation plant that will be built adjacent to Clab. The encapsulation plant and Clab will then be operated as the Central facility for interim storage and encapsulation of spent nuclear fuel (Clink). The copper canisters will be placed deep in the bedrock in a final repository, the Spent Fuel Repository, with natural and engineered barriers and conditions that meet the requirements for post-closure safety.

The nuclear waste, which consists of both short-lived and long-lived radioactive waste, will also be disposed of in geological final repositories. The short-lived waste will be placed in the Final Repository for Short-lived Radioactive Waste (SFR) and the long-lived waste in the planned Final Repository for Long-lived Waste (SFL). In addition to the geological final repositories, a subset of the waste with a very low activity level can be disposed of in near-surface repositories located directly adjacent to nuclear facilities. Some waste can be treated by separating the radioactive parts from the rest, which can be granted clearance from regulatory control. For example, when a nuclear power plant is decommissioned, metal that has undergone certain treatment can be melted down, granted clearance and re-used. What remains will be a smaller volume of radioactive waste that needs to be disposed of in SFR or SFL. Before the final repositories are commissioned, interim storage of the waste takes place at different sites where other nuclear activities are conducted.

All management of radioactive waste and spent nuclear fuel, both within a facility and between facilities, requires handling stations and a transport system. The transport system is crucial for the functioning of the waste system and includes vehicles for land transport, ships and various types of transport casks. Equipment in the facilities must be designed so that all waste containers containing waste, which are transported in adapted waste transport casks, and which are to be placed in final repositories, can be managed in a manner that is safe from a radiation perspective. The design of a waste container therefore considers material choice, the radiation environment in interim storage and final repository facilities, leak tightness, manageability in the facilities concerned, transportability, the impact on the long-term safety of the final repository in question and the waste and other material that the container is to be filled with. When all the acceptance criteria are fulfilled, the waste container can be disposed of in the designated final repository.

The copper canisters in which the spent nuclear fuel will be placed must be sealed to prevent the migration of radioactive substances. The copper canisters consist of a copper shell and an insert with compartments for each individual fuel assembly. In order to achieve a tight seal, they are sealed by welding on copper lids. A special transport cask with radiation shielding will be needed for the transportation of copper canisters between Clink in Oskarshamn and the Spent Fuel Repository in Forsmark.

RD&D Programme 2022

In the RD&D Programme 2022, SKB and the licensees of the Swedish nuclear reactors present their plans for research, development and demonstration for the period 2023–2028. The programme consists of three parts:

- Part I Activities and plan of action.
- Part II Waste and final disposal.
- Part III Decommissioning of nuclear facilities.

Part I Activities and plan of action

Part I describes the activities and plan of action for management and disposal of nuclear waste and spent nuclear fuel from the operation and decommissioning of the Swedish nuclear power reactors. It also describes the priorities and reasons for the research, development and demonstration needed in order to construct and commission new facilities.

The facilities in the waste system that are currently in operation are Clab, SFR and near-surface repositories at the nuclear power plants. In addition, SKB owns a ship with associated transport casks and vehicles. For final disposal of the spent nuclear fuel, the KBS-3 system, it remains to build and commission a new facility adjacent to Clab for encapsulation of the spent nuclear fuel (together referred to as Clink), to develop casks for the transport of canisters of spent nuclear fuel and to construct a final repository for copper canisters of spent nuclear fuel, the Spent Fuel Repository. In addition to these facilities, a system for production of copper canisters and a system for production of buffers and backfilling components is needed.

The capacity of interim storage facilities needs to be increased and SFR needs to be extended to facilitate disposal of low-level and intermediate level operational waste and decommissioning waste. Development of transport casks is also under way. SFL needs to be located at a suitable site and built, and casks for transport of long-lived waste to this repository need to be manufactured and procured.

Decommissioning of nuclear power reactors is under way or planned at most of the Swedish nuclear power plants, and the RD&D programme describes the planning for the reactors that will be decommissioned in the 2020s.

Plan of action

This RD&D programme takes the strategic direction one step further than the previous RD&D programme. It is of strategic importance for the establishment of the entire waste system that activities are carried out in a logical sequence, in which needs and dependencies are considered. In order to enable the nuclear power reactors to operate, space must be available for interim storage of the spent nuclear fuel. For facilities that are decommissioned, interim storage facilities need to be in place for the waste before the necessary final repositories have been built. These and other dependencies are reflected in the activity and milestone schedule, where the focus is on providing a system-wide perspective and highlighting relationships between activities and milestones for the facilities.

SKB takes a long-term approach to the planning of activities based on the plan of action in the RD&D programme, which includes all facilities until they are decommissioned, and on the basis of operational five-year plans, which are updated every year. Planning directives based on known conditions and strategic objectives form the basis of these plans. Safe operation of SKB's facilities has top priority, and in the event of any resource conflicts this always takes priority.

Because SKB's activities are dependent on external influences, such as decisions by reactor owners or regulatory authorities, flexibility is required for replanning. It must be possible to adapt the operations to events in the world around us. The reactors in Ringhals and Oskarshamn that have been shut down have entailed an increased need for transportation of spent nuclear fuel from these reactors to Clab. This, together with the reactors previously shut down in Barsebäck, means that the need for management and disposal of radioactive waste from decommissioning has also increased. The need to be able to commission an extended SFR as soon as possible has therefore increased.

In the previous RD&D period, SKB received important Government decisions on extended storage in Clab and the extension of SFR, as well as on the KBS-3 system, which means that significant milestones have been passed. The licensing cases related to the Nuclear Activities Act (KTL) and the Swedish Environmental Code (MB) are continuing and will continue for several years to come. How long each case will take before an enforcement decision is made or a judgment is announced cannot be specified in advance, and this requires flexibility in SKB's planning.

During the RD&D period, increased storage in Clab and the extension of SFR will have high priority, primarily to ensure continued operation of reactors and to enable management of waste from the reactors that are undergoing decommissioning.

The continued work on SFL has been given lower priority. In the RD&D period, the focus will be on work on the inventory, preliminary acceptance criteria and studies relating to waste containers. When all the necessary data are available for the inventory, including data on the legacy waste to be disposed of there, they will serve as the basis for an assessment of post-closure safety, creating the conditions necessary to be able to apply for a licence and permissibility for the final repository. All in all, this, together with the fact that the work on SFL has been given lower priority, means that the construction start for SFL will be postponed for several years compared with what was reported in the RD&D Programme 2019.

Research and development

SKB 's and the licensees' planning of future research and development activities for the final repositories is based on the plan of action, in which the stepwise decision process forms the basis for important milestones. The milestones relating to the decision steps in the form of applications and safety analysis reports dictate when knowledge and development of technology needs to have reached a certain level, while the approvals of the Swedish Radiation Safety Authority (SSM) dictate when SKB will be able to commence construction and operation of the facilities. When SKB submits the applications for construction of a facility, the aim is to show that necessary knowledge and ability exists to construct the facility in such a way that it meets regulatory requirements.

SKB is continuing its research and technology development efforts in order to facilitate optimisation of, above all, the design of the Spent Fuel Repository, but also to reduce uncertainties in the assessments of post-closure safety by contributing to a better understanding of the processes that affect the final repositories and their barriers and by means of improved models. As regards technological development, work on developing methods and technological solutions are continuing in each area in order to achieve the required level of maturity prior to implementation. Part I, Activities and plan of action, provides an overview of the most important remaining research and development issues. The planned activities during the RD&D period are described in greater detail in Part II, Waste and final disposal.

Part I also describes the remaining work that is planned to be carried out at the Äspö Hard Rock Laboratory (Äspö HRL) before the underground part of the facility is closed. Additionally, the planning work regarding implementation of monitoring of the Spent Fuel Repository during construction and operation, work on issues concerning preservation of knowledge and information about the final repositories after closure, as well as SKB's view of the role of the RD&D programme for openness and insight into research and development issues, are described.

Part II Waste and final disposal

Part II presents the need for continued research, development and demonstration during the RD&D period of the issues identified by SKB as prioritised for the continued management and final disposal of radioactive waste and spent nuclear fuel. The current situation and programme are presented for the following areas: the low-level and intermediate-level waste, the spent nuclear fuel, the repositories' engineered and natural barriers, the surface ecosystem and the climate. Some parts of the research and technology development concern all three repositories and are described in an integrated manner for all of them. The current situation sections on the state of knowledge contain references to background reports with more detailed information. The research and technology development efforts prioritised for the RD&D period are presented as bullet points in the programme sections. For licensing activities, the results of ongoing research and technology development will be presented in more detail in the stepwise licensing process according to KTL for each facility.

The need for research and development activities can be divided into three areas:

- Increased process understanding, i.e. the scientific understanding of the processes that affect the final repository system and thereby the basis for assessing their importance for post-closure safety.
- Knowledge and competence relating to the design, structure, manufacture and installation of the barriers and components to be used in the facilities.
- Knowledge and competence concerning inspection and testing to verify that the system barriers and components satisfy the requirements.

The most important research and development activities within each area for each of the three repositories are described below.

Extension of the Final Repository for Short-lived Radioactive Waste

The extension of SFR is based on a licence issued by the Government pursuant to KTL and permissibility pursuant to MB. The Land and Environment Court (MMD) issues environmental permits and decides on terms and conditions in accordance with MB. SSM will conduct monitoring during the stepwise licensing process. During the RD&D period, SKB will complete the preliminary safety analysis report (PSAR) and submit it to SSM for review and approval. SKB's research and technology development efforts will focus on compiling data for the Safety Analysis Report (SAR) for trial operation of SFR.

Process understanding and modelling

- Updating of the radionuclide inventory in low-level and intermediate-level waste (also relevant for SFL).
- Increasing process understanding of sorption of radionuclides on cementitious materials and degradation of selected organic materials to potential complexing agents (also relevant for SFL).
- Further development of modelling tools for transport of solutes, above all regarding matrix diffusion and sorption in both geosphere and surface ecosystem (also relevant for the Spent Fuel Repository).

Manufacturing and installation

- Some technology development of the barriers and closure components in order to verify requirements and technical design requirements.
- Adaptation of closure technology for investigation boreholes at SFR.
- Design of concrete structures focusing on production methods.

Inspection and testing

- Preparation of preliminary acceptance criteria.

Spent Fuel Repository and Clink

Construction of the Spent Fuel Repository, which, like Clink, is included in the KBS-3 system, is based on a Government licence under KTL and permissibility under MB. MMD issues environmental permits and decides on terms and conditions in accordance with MB. SSM will conduct monitoring during the stepwise licensing process. During the RD&D period, SKB will complete the PSAR for the Spent Fuel Repository, the PSAR for Clink (which is a prerequisite for being allowed to begin construction) and submit these to SSM for review and approval. SKB's research and technology development efforts during the RD&D period are focused on gathering data for a SAR prior to trial operation of the Spent Fuel Repository and Clink.

Process understanding and modelling

- Verify and further develop the site-specific model for Forsmark by means of a detailed site investigation programme to identify necessary activities and measures.
- Further development of a model that can be used to calculate the dissolution rate of the spent nuclear fuel.
- Study concerning the sulphide corrosion process for copper in an unsaturated bentonite buffer.
- Further studies of bentonite losses following colloid release/erosion.
- Development of a model for secondary movements in fractures in the rock during earthquakes.
- Further development of modelling tools for transport of solutes, particularly when it comes to matrix diffusion and sorption in both geosphere and surface ecosystems (also relevant for SFR).

Manufacturing and installation

- Further development of technical systems for final disposal and backfilling.
- Continued development and optimisation of manufacturing methods and structure of the copper canister and its components.
- Develop the method for sealing of investigation boreholes for gently dipping and horizontal holes.

Inspection and testing

- Preparation of a monitoring programme for construction and operation of the Spent Fuel Repository.

Final Repository for Long-lived Waste

SFL is the last final repository SKB is planning to build. Applications for permits under KTL and MB are planned to be submitted during the 2030s and will include a preparatory PSAR. In addition to the issues common to all the different repositories, SKB's research and technology development activities in respect of SFL during the RD&D period are focused on work on the inventory, development of waste containers and preparation of preliminary acceptance criteria.

Process understanding and modelling

- Update of the radionuclide inventory in low-level and intermediate-level waste (also relevant for SFR).
- Increase process understanding of sorption of radionuclides on cementitious materials and degradation of selected organic materials to potential complexing agents (also relevant for SFR).

Manufacturing and installation

- Development of waste containers.

Inspection and testing

- Preparation of preliminary acceptance criteria.

Part III Decommissioning of nuclear facilities

Part III presents the planning for the decommissioning of the Swedish nuclear power reactors and SKB ‘s nuclear facilities.

The licensee for a nuclear facility is responsible for decommissioning in accordance with KTL, the Radiation Protection Act, the Financing Act and SSM’s regulations. The licensee is responsible for the radioactive waste until it has been released from regulatory control or until the Government has granted exemption from responsibility under KTL.

Prior to decommissioning, the necessary licences must be in place. Pursuant to KTL, the Radiation Protection Act and the relevant ordinances, licence conditions and regulations, the following documents must be prepared prior to and in some cases continuously during operation and decommissioning: decommissioning plan and decommissioning strategy, waste management plan, SAR, documentation pursuant to Article 37 of the Euratom Treaty, notification of steps or sub-projects, decommissioning report and inspection programme for clearance.

When the facility/parts of the facility have been released from regulatory control, conventional demolition and restoration of land can be carried out.

The schedules for decommissioning of the Swedish nuclear power plants are governed by the planned operating times of the nuclear power plants. The reactor owners plan to start dismantling and demolition as soon as possible after final shutdown, with planning dependent on the availability of interim storage facilities and final repositories for decommissioning waste.

To streamline the work on decommissioning and waste issues, areas of work have been divided up between the different actors at both company level and Group level. The licensees are decommissioning their nuclear power reactors and SKB ‘s principal task is to establish a final repository for decommissioning waste according to the needs of the licensees. SKB also carries out transportation of radioactive waste and spent nuclear fuel from the nuclear power plants to interim storage facilities and final repositories.

Decommissioning of the nuclear facilities in Sweden will continue until the mid-2070s, when the facilities for management and disposal of spent nuclear fuel and radioactive waste will be decommissioned. The decommissioning activities will be carried out in three main stages; one in the 2020s, one in the 2040s and the final one in the 2070s. The first stage covers Barsebäck 1, Barsebäck 2, Oskarshamn 1, Oskarshamn 2, Ringhals 1, Ringhals 2 and Ågesta.

Contents

Part I

Activities and plan of action

1 Introduction

2 Description of the waste system

3 Plan of action

4 Continued research and development

5 Procedures, competence and resources

Part I of the RD&D Programme 2022 describes the activities and plan of action for management and disposal of radioactive waste and spent nuclear fuel from the operation and decommissioning of the Swedish nuclear power reactors. From a strategic perspective, it is explained why activities are carried out or are planned to be carried out, and background is provided for the order in which they will be carried out. SKB's activities are affected by external factors, and this requires flexible operations, planning and organisation, and the ability to take decisions to work according to changing priorities in the planning of activities and milestones for the nuclear waste programme. In the same way, the reactor owners need to be flexible to respond to changing conditions, which may result in new planning requirements for SKB.

The plan of action forms the basis for a rationale for, and summary of, the planned activities in research and development needed for implementation of the remaining parts of the nuclear waste system and to decommission the nuclear power reactors and SKB's other nuclear facilities. The action plan also describes the procedure SKB has developed to conduct the research and development that is needed to realise the plan and manage the radioactive waste and the spent nuclear fuel in a safe and cost-effective manner.

Part I also includes a summary report on the state of knowledge and planned activities in the area of monitoring that are of interest to SKB, based on the monitoring programme for the Spent Fuel Repository that is under development. The programme aims to provide an overall picture of SKB 's monitoring activities during construction and operation of the repository. In addition, SKB's premise and work on knowledge and information transfer relating to the final repositories are described, both during their operating time and after closure.

1 Introduction

The Swedish power industry has been generating electricity by means of nuclear power for almost 60 years. The country's first commercial nuclear power reactor, the Ågesta reactor, was commissioned in 1964. Since then, a system has been built up, in order to safely manage and dispose of the spent nuclear fuel and nuclear waste from the operation and decommissioning of the Swedish nuclear power reactors. The facilities in operation are the Central Interim Storage Facility for Spent Nuclear Fuel (Clab), the Final Repository for Short-lived Radioactive Waste (SFR), near-surface repositories and interim storage facilities adjacent to the nuclear facilities, as well as a ship and transport casks.

For safe management and disposal of the spent nuclear fuel in the long term, what remains is to construct and commission the system of facilities needed for final disposal, the KBS-3 system. This will be followed by operation of the system and, when all spent nuclear fuel has been disposed of, the facilities can be decommissioned and the final repository can be closed. The KBS-3 system includes Clab and an encapsulation plant for the spent nuclear fuel (together known as Clink), casks for transport of copper canisters containing spent nuclear fuel and a final repository spent nuclear fuel (Spent Fuel Repository). In addition to these facilities, a system for the production of canisters and a system for the production of buffer and backfill components are required.

The capacity of interim storage facilities must be increased and SFR needs to be extended to facilitate disposal of short-lived low- and intermediate-level operational- and decommissioning-waste. In addition, a site needs to be found for an additional final repository, the Final Repository for Long-lived Waste (SFL), the repository needs to be constructed and transport casks for long-lived waste need to be procured.

SKB's plan of action in this RD&D Programme describes the overall plans for implementing the remaining parts of the waste system and adapting the existing facilities in such a manner that human health and the environment are protected – today and in the future. A number of nuclear reactors are currently being decommissioned. The RD&D programme therefore describes the planning for the reactors that will be decommissioned in the 2020s.

1.1 Prerequisites

Important prerequisites for the safe management and disposal of the spent nuclear fuel and nuclear waste from the Swedish nuclear power reactors consist of applicable regulatory frameworks, fundamental principles for execution and political policy decisions, the planned operating times of the reactors and the principles for the allocation of the residual products from nuclear activities into different waste categories.

1.1.1 Relevant regulatory frameworks and SKB's mission

Pursuant to Section 10 of the Nuclear Activities Act (KTL, SFS 1984:3), a party that conducts nuclear activities shall be responsible for ensuring the safe decommissioning of facilities and for the management and disposal of spent nuclear fuel and nuclear waste. Section 11 prescribes that the holder of a licence to operate a nuclear power reactor shall be responsible for the comprehensive research and development work required to fulfil the obligations in Section 10.

The licensees for the nuclear power reactors in Forsmark, Oskarshamn, Ringhals, Barsebäck and Ågesta are Forsmarks Kraftgrupp AB, OKG Aktiebolag, Ringhals AB, Barsebäck Kraft AB and Vattenfall AB. These companies are hereinafter referred to as the reactor owners.

Svensk Kärnbränslehantering AB (SKB) is owned by the reactor owners and their owners, Vattenfall AB, OKG Aktiebolag, Forsmarks Kraftgrupp AB and Sydkraft Nuclear Power AB. On behalf of its owners, SKB is responsible for the management and final disposal of the nuclear waste and spent nuclear fuel from the Swedish nuclear power plants. For this purpose, SKB owns and operates a transport system and facilities for waste management.

The reactor owners are responsible for decommissioning of their respective nuclear power reactors, and SKB has been given the task to participate in the planning of decommissioning. SKB collaborates with the different decommissioning projects on central issues related to waste management, for example acceptance criteria and type descriptions for decommissioning waste and also transports.

Pursuant to Section 12 of KTL, a party that holds a licence to possess or operate a nuclear power reactor shall, in consultation with other reactor owners, prepare or arrange for a programme for the comprehensive research and development work and other measures needed for the safe management of nuclear waste and spent nuclear fuel, as well as the safe dismantling and demolition of nuclear facilities. Such a programme for research, development and demonstration (RD&D programme) shall be submitted to the Swedish Radiation Safety Authority (SSM) every three years. SKB, on behalf of and in cooperation with the reactor owners, prepares the RD&D programmes and submits them to SSM.

Pursuant to the Act on the Financing of the Residual Products of Nuclear Power (the Financing Act, SFS 2006:647), the reactor owners are obliged to fund the costs of the future measures required for the management and final disposal of residual nuclear products, the dismantling and demolition of the facilities, and the research required to enable this. The Swedish National Debt Office has overall responsibility for safeguarding the nuclear power industry's payment liability and for ensuring that the financing system is working as it should. On behalf of the reactor owners and pursuant to the Financing Act, SKB shall present cost calculations every three years – see Section 1.3.

In addition to the nuclear waste received from the reactor owners, SKB also receives radioactive waste from medicine, research and industry. This is regulated by agreements between SKB and the companies that produce the waste or which handle it on behalf of another company, regardless of who has legal responsibility for the waste.

1.1.2 Fundamental principles

The management of radioactive substances is regulated by laws and ordinances. Radioactive waste is material which emits ionising radiation that is harmful and constitutes waste as in Chapter 15, Section 1 of the Swedish Environmental Code (MB, SFS 1998:808) or has no planned and acceptable use. Nuclear waste is radioactive waste that is produced in nuclear power plants or other nuclear facilities. Legally, spent fuel is not defined as waste until it is finally disposed of in a repository. This has, however, no practical implications for the management of the spent fuel.

Important fundamental principles that exist in legislation are:

- A party that has generated spent nuclear fuel and radioactive waste shall also bear the costs of its disposal.
- The licensee for a nuclear activity, as well as a party that otherwise conducts activities involving radiation, is obliged to safely manage and dispose of spent nuclear fuel and radioactive waste arising from the activity.
- The state has a secondary responsibility for the spent nuclear fuel and radioactive waste generated in Sweden. This responsibility means that if there is no one who can be held responsible for safety, the state is responsible until a licensee can fulfil the obligations.
- Sweden takes responsibility for spent nuclear fuel and radioactive waste generated in the country.
- The municipalities concerned, have the right of veto against the establishment of new nuclear facilities.

Another central principle is the political decision that the spent nuclear fuel shall not be reprocessed.

SKB implements geological final disposal of short-lived radioactive waste and has long planned to also implement this for the spent nuclear fuel and the long-lived radioactive waste. Most countries and organisations, including the International Atomic Energy Agency (IAEA) and the Organisation for Economic Cooperation and Development/Nuclear Energy Agency (OECD/NEA) now agree that geological disposal is a solution that has the potential to meet the requirements for safe final disposal and feasibility. Geological disposal is also supported by the EU's community framework for the responsible and safe management of spent fuel and radioactive waste.[1]

The following principles serve as the basis for the design of SKB's final repositories:

- 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.
- Repository safety shall be based on multiple barriers (the principle of multiple barriers).
- Engineered barriers shall primarily consist of naturally occurring materials that are stable in the repository environment in the long term.
- The repositories shall be designed in such a way that safety is not dependent on active measures such as maintenance and repairs after closure.

The principle of multiple barriers is a fundamental and internationally accepted safety principle for final disposal. It entails that post-closure safety of a final repository shall be based on multiple barriers, the purpose of which is to contain, prevent or delay the dispersion of the radioactive elements in the waste. The barriers and other components that are needed in a final repository are largely depend on the content of radioactive elements, their half-lives and other properties of the waste. This means that the requirements on the barriers and their durability in SFR are different from those on the repositories for spent nuclear fuel and for long-lived radioactive waste.

The above principles, along with a number of other considerations, for example that construction of a repository must be technically feasible, have led SKB to develop and choose the KBS-3 method[2] for final disposal of spent nuclear fuel. The Swedish Government, through its decision to grant SKB a licence to build, possess and operate the Spent Fuel Repository, has determined that the activities covered by the application meet the requirements for choice of method and best available techniques (BAT).

The KBS-3 method, development of which began in the late 1970s, is based on a system of passive interacting protective barriers that together will contain the fuel and prevent radioactive elements from spreading to humans and the environment for a period of 100 000 years. The method can be summarised as follows:

- The spent nuclear fuel is placed in copper canisters with high resistance to corrosion in a repository environment. The approximately five-metre-long copper canisters have a load-bearing insert that provides the mechanical stability.
- The canisters are surrounded by a buffer of bentonite clay, a naturally occurring mineral that swells in water. The buffer protects the canister from minor rock movements and shields it from groundwater flow. It limits the amount of corroding agents in the groundwater that may reach the canister. The clay also absorbs the radioactive elements that may be released if the canisters were to fail.
- The canisters surrounded by bentonite clay are disposed of at a depth of about 500 metres in bedrock with long-term stable conditions.
- If a canister was to fail, the nuclear fuel and the chemical properties of the radioactive elements, for example their insolubility in water, constitute major limitations for the transport of radioactive elements from the repository to the surface.

Internationally, the KBS-3 method is one of the methods for final disposal of spent nuclear fuel where development has progressed the furthest. SKB in engaged in broad international cooperation and for has many years been working in close cooperation with our sister organisation Posiva in Finland (Section 5.6.3). Like SKB, Posiva has chosen to build its final repository for spent nuclear

[1] Council Directive 2011/70/Euratom of 19 July 2011 establishing a Community framework for the responsible and safe management of spent nuclear fuel and radioactive waste.

[2] The KBS-3 method is so named because it is based on the third report in the project KärnBränsleSäkerhet (Nuclear Fuel Safety) (The final stage of the Nuclear fuel cycle, Spent nuclear fuel (SKBF/KBS 1983)).

fuel according to the KBS-3 method. A major step was taken in Finland at the end of 2021 when Posiva applied for an operational permit for the final repository and the encapsulation plant to the regulatory authorities. The method, or variants of it, is also being considered for final repositories in Canada, South Korea, the United Kingdom, Taiwan and the Czech Republic.

1.1.3 Planned operating times of the reactors

The operating times of the reactors are an important factor in the planning of the nuclear waste programme. Based on the reactor owners' current planning, forecasts are made for the amounts of nuclear waste and spent nuclear fuel that will be managed in the waste system and for the outlining of when there will be a need for interim storage and final disposal.

The two reactors at the Barsebäck nuclear power plant were shut down in 1999 and 2005, the Oskarshamn nuclear power plant closed two of three reactors in 2015 and 2017, and two of four reactors at the Ringhals nuclear power plant were shut down in 2019 and 2020. All the reactors that have now been shut down were commissioned in the 1970s. The planned operating time for the six reactors in operation is 60 years. This applies to the reactors Forsmark 1, Forsmark 2 and Forsmark 3, Oskarshamn 3, Ringhals 3 and Ringhals 4. The youngest reactors, Forsmark 3 and Oskarshamn 3, will be in operation until 2045, according to the reactor owners´ current planning.

The consequences of changes to operating times for the waste system are discussed in Section 3.7.1.

1.1.4 Radioactive waste and spent nuclear fuel

Spent nuclear fuel and radioactive waste, but also conventional waste, is generated during the operation and decommissioning of the nuclear power reactors. The spent nuclear fuel and radioactive waste must be kept separate from humans and the environment in order to avoid harmful effects from ionising radiation.

The management of the radioactive waste and 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 (very low-level, low-level, intermediate-level or high-level) and the half-life of the radioactive elements (short-lived or long-lived). The level of radioactivity determines how the waste is handled before disposal. 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 radioactive elements in the waste are short-lived or long-lived, as this is of importance for the time period during which barrier performance needs to be maintained.

How much and when waste is produced are also important prerequisites in the planning of the waste system. The amounts of waste are dependent on the reactors' operating times, availability and other operating conditions. The waste system is designed to manage and dispose of nuclear waste and spent nuclear fuel from the operation and decommissioning of the Swedish nuclear reactors, and is based on the forecasts of each reactor owner. During decommissioning of the nuclear reactors, large volumes of material will be released from regulatory control.

Very low-level waste

Waste that contains a small amount of short-lived radionuclides with a half-life shorter than around 30 years[3], a limited amount of long-lived radionuclides and a dose rate for packages below 0.5 mSv/h is classified as very low-level waste. This waste is produced during both operation and decommissioning of the nuclear power plants. During operation, it is mainly generated during maintenance outages, and maintenance and service tasks. The fraction of decommissioning waste that is classified as very low-level primarily consists of dismantled systems and structural parts, as well as protective and decontamination equipment.

[3] According to IAEA Safety Standards, Classification of Radioactive Waste, General Safety Guide, No GSG-1, a short-lived radionuclide is defined as a radionuclide with a half-life shorter than about 30 years.

The management of very low-level waste is determined by the type of material and activity content and takes place mainly on site at the nuclear power plants. The very low-level waste is sorted into two fractions, compactable and non-compactable waste. For the very low-level waste produced during operation, the weight distribution is about 45 percent metal, 40 percent soft waste and 15 percent inert waste. However, soft waste is the largest fraction in terms of volume.

Final disposal of very low-level waste takes place mainly in near-surface repositories. Alternative, more complex treatment, such as combustion and melting, takes place in special plants. During a normal year of operation, 50–100 tonnes of very low-level waste is produced per reactor. Decommissioning of a reactor generates an estimated 250–500 tonnes of very low-level waste per year for a period of around 10 years. The forecasts for very low-level decommissioning waste contain a large degree of uncertainty.

Low-level and intermediate-level waste

The low- and intermediate-level waste can be either short- or long-lived. Short-lived waste primarily contains radionuclides with a half-life shorter than about 30 years and only a limited amount of radionuclides with a longer half-life. Waste is classified as long-lived if it has a significant content of long-lived radionuclides. This is completely independent of the amount of short-lived nuclides in the waste.

Low- and intermediate-level waste is generated during both operation and decommission of nuclear facilities. The operational waste consists of, for example, spent filters, replaced components and used protective clothing. The decommissioning waste includes scrap metal and building materials.

Most of the short-lived waste originates from the nuclear power plants. Other short-lived waste currently comes from Clab and later from Clink[4], as well as from medicine, research and industry. Short-lived waste is disposed of in SFR. According to current forecasts, around 180 000 cubic metres of waste will be disposed of in SFR.

Long-lived waste from the nuclear power plants consists of core components, reactor pressure vessels from pressurised water reactors (PWRs) and control rods from boiling water reactors (BWRs). The long-lived radionuclides in this waste are formed from stable elements, in for example steel, when these are exposed to a high level of neutron radiation from the reactor core. The amount of long-lived low- and intermediate-level waste from the nuclear power plants is estimated to be about 6 000 cubic metres.

Furthermore, long-lived waste contains waste from research and development within the Swedish nuclear research programmes, as well as from medicine, research and industry. The amount of long-lived low- and intermediate-level waste from these sources is estimated to be around 11 000 cubic metres, and it is managed by AB SVAFO.

Spent nuclear fuel

The spent nuclear fuel comprises a small fraction of the total volume of waste to be disposed of. However, the spent fuel contains by far most of the radioactivity, both short-lived and long-lived. Spent nuclear fuel is highly radioactivity and requires radiation shielding for all handling, storage and final disposal. Final disposal will take place in the Spent Fuel Repository.

The spent fuel generates heat (decay heat) even after it has been removed from the reactor, which means that it must be cooled to avoid overheating. The decay heat depends above all on the burnup (the amount of energy that has been extracted from the nuclear fuel) and how much time has passed since it was removed from the reactor. Due to technical advances and modifications in the operation of the reactors in order to achieve efficient use of the fuel, burnup of the fuel has gradually increased since the reactors were commissioned. A consequence of increased burnup is increased decay heat, which is of importance for interim storage and final disposal.

[4] A plant will be built adjacent to Clab for encapsulation of the spent nuclear fuel. Following completion, it will be operated as an integrated facility, the central interim storage facility and encapsulation of spent nuclear fuel (Clink).

Almost all spent nuclear fuel that will be disposed of in the Spent Fuel Repository comes from the Swedish nuclear power plants. There are, however, also small quantities of spent nuclear fuel from completed reprocessing agreements, other types of reactors and Studsvik AB 's fuel operations. The total amount of spent nuclear fuel to be disposed of is not expected to exceed 12 000 tonnes, expressed as the quantity of uranium that was originally present in the fuel.

1.1.5 Licensing of nuclear facilities

The Government is responsible for decisions on licences and permissibility in accordance with KTL and MB in respect of the construction of a nuclear facility and the implementation of major rebuilding or modifications of an existing facility. SSM is the preparatory authority for applications under KTL, while the Land and Environment Court (MMD) is the authority reviewing applications under MB, and applicants submit applications to both bodies in parallel. The applications include a preparatory preliminary safety analysis report (F-PSAR) or equivalent summary information and an environmental impact statement (MKB).

The purpose of the applications is to show that the applicant, in this case SKB, has the knowledge and ability to design the facility so that it meets regulatory, and other, requirements. Even after a licence under KTL has been obtained, research and technology development will continue, and will be presented and assessed in the continued stepwise licensing process. As part of the applications for a licence to construct the Spent Fuel Repository and encapsulation facility, as well as the extension of SFR, SKB has presented a report on the state of knowledge and the status of technology development. The licence applications also include evaluation of the importance of uncertainties for the post-closure safety of the repositories, SR-Site (SKB TR-11-01) for the Spent Fuel Repository and SR-PSU (SKB TR-14-01) for SFR.

In the application documents, SKB presents a reference design of the new facilities with a conceptual description of how requirements can be met. Continued research, technology development and design work lead to developed technical solutions or designs that are better or more efficient.

When the Government has decided on a licence in accordance with KTL and on permissibility in accordance with MB, MMD will issue an environmental permit and specify conditions in accordance with MB, while SSM will continue the stepwise licensing process – see Figure 1-1.

In the stepwise licensing process, approval from SSM of several iterations of the safety analysis reports are needed. This is governed by the regulations of SSM, which, based on international recommendations. e.g. from IAEA and OECD/NEA, state that development and licensing of nuclear facilities shall take place through a process in which the requirements for the facility, its design and technical solutions are established successively. This stepwise process, common to all nuclear facilities, includes:

- **Approval of safety analysis report prior to construction –** based on the presentation of a preliminary safety analysis report (PSAR), which describes the planned design of the facility, how the activities are organised and how the requirements are fulfilled.
- **Approval of safety analysis report prior to trial operation and prior to regular operation** – based on successive presentations of an updated and a supplemented safety analysis report (SAR). The SAR must comprehensively demonstrate how the safety of the facility will be achieved, describe the facility as built, analysed and verified, and demonstrate how the requirements for its structure, function, organisation and operation are fulfilled.

In addition, notification is made to the EU Commission pursuant to Article 41 of the Euratom Treaty by SKB prior to construction, and pursuant to Article 37 of the Euratom Treaty by the Swedish state via the SSM, based on data from SKB, prior to commissioning. Requirements relating to nuclear safeguards from both Swedish authorities and international inspection bodies must always be met for facilities where nuclear material will be handled. This means that for these facilities, SKB will submit, prior to construction, a preliminary basic technical description, Basic Technical Characteristics (BTC) and a plant description to the SSM and the EU Commission, and will then, before commissioning, submit to the SSM and the EU Commission an officially established BTC and an updated plant description.

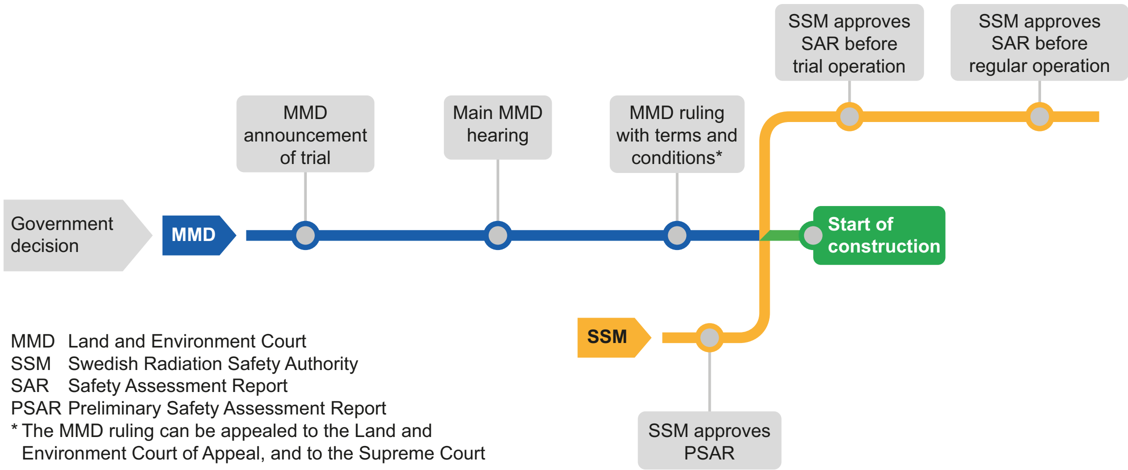


Figure 1-1. *A Government decision on a licence under KTL and permissibility under MB is followed by a process reviewing terms and conditions and a judgment by MMD, as well as a stepwise* **licensing process by SSM.**

The operating period of a final repository begins with trial operation, during which spent nuclear fuel or radioactive waste is disposed of. After trial operation, the activities shift into a management phase during regular operation. The holder of a licence to possess or operate a nuclear facility must submit a systematic, overall assessment of safety and radiation protection at least every ten years. In conjunction with these assessments, a review and compilation of the state of knowledge in the areas that are essential for radiation safety are also carried out.

Under the terms of KTL, the Government shall decide on the final closure of a final repository. The revised safety assessment, which must be approved by SSM, serves as a basis for the Government's processing of the application to close the repository. The Government can also decide on the conditions that must be fulfilled before the final closure of the repository. The Government is responsible for a sealed and closed geological final repository and the waste disposed there.

1.2 Programme for research, development and demonstration

The focus of the RD&D programmes has varied through the years, depending on the emphasis of SKB's and the reactor owners' activities. The RD&D Programme 2010 (SKB TR-10-63) includes a brief summary of the RD&D programmes published by SKB up to and including 2007. Since then, a summary has been provided of the immediately preceding RD&D Programme.

1.2.1 RD&D Programme 2019

The RD&D Programme 2019 (SKB TR-19-24) was in many respects a development and update of the programme submitted in 2016 (SKB TR-16-15). SSM's overall assessment of the RD&D Programme 2019 was that the conditions set out in the Government's decision on the RD&D Programme 2016 had been heeded and that the report was appropriate in relation to the legislator's purpose in respect of the programme and the requirements imposed on reactor owners.

According to SSM, the report in the RD&D Programme 2019 gave a sufficient overview of the reactor owners' and SKB's programmes and sufficient insight into the plan of action during the RD&D period. In December 2020, the Government decided that the RD&D Programme 2019 met the requirements of KTL and specified as a condition for the continued research and development work that future RD&D programmes should provide an overview of planned activities, corresponding to the plan of action for activities that SKB is responsible for. The report shall include a fundamental description of the management of different waste streams and the necessary and available capacity for the different parts of the system in relation to the quantity or volume of spent nuclear fuel or radioactive waste that is expected to be handled and managed. RD&D programmes will in future include all waste categories that are expected to be managed in each facility. Furthermore, it was stated that the reactor owners and SKB must consider how the RD&D programme can better contribute to openness and transparency in how the work on research, development and demonstration of methods for handling and final disposal of nuclear waste is conducted. Future RD&D programmes must also include activities regarding information preservation in accordance with the recommendations of SSM and the Swedish National Council for Nuclear Waste.

1.2.2 Milestones and development since RD&D Programme 2019

System for low-level and intermediate-level waste

When the RD&D Programme 2019 was presented, the licensing process regarding the extension of SFR, the repository for operational waste and decommissioning waste that SKB applied for in 2014, was under way in MMD and SSM. The main hearing in MMD took place in autumn 2019. In November 2019, the Court and SSM submitted their respective statements to the Government. In April 2021, Östhammar Municipal Council approved an extension of SFR. The Government decided on permissibility under MB and on a licence under KTL for the extension of SFR in December 2021. In terms of the extension of SFR. The Government's decision meant that the process will progress to negotiations on conditions at MMD at the end of 2022, followed by submission of an application prior to construction to SSM. The construction of the extension of SFR can begin after the judgment of MMD and approval of the PSAR.

In parallel with the licensing process, research, development and other activities providing supporting material, e.g. for the PSAR for the extension of SFR, have continued.

In September 2019, SKB presented an evaluation of post-closure safety for a proposed final repository concept for SFL (SKB TR-19-01), which was also summarised in the RD&D Programme 2019. The evaluation shows that in the appropriate circumstances, the repository concept for SFL has the potential to comply with regulatory regulations, given that further efforts be made in a future safety assessment to support this. As a part of the authority's review of the progress of SKB's RD&D Programme, SSM has reviewed the safety evaluation. SSM finds it appropriate that SKB bases the work of developing the concept for SFL on experience from equivalent processes that have been used for the development and siting of the other repositories, and takes a positive view of the feedback on the characterisation of the waste provided by the safety evaluation.

KBS-3 system for spent nuclear fuel

When the RD&D Programme 2019 was presented, SKB had applied to MMD and SSM in 2011 for a licence to construct and operate a final repository system for spent nuclear fuel (the KBS-3 system). In 2015, SKB had also applied for a licence to increase the quantity of spent nuclear fuel that can be placed in interim storage in Clab from 8 000 to 11 000 tonnes.

After the main hearing in MMD for the KBS-3 system in 2017, both the Court and SSM submitted their statements to the Government. SSM approved SKB's application in accordance with KTL. The Court took a positive view of several important points but requested additional supporting materials, including on the integrity of the copper canisters. SKB submitted the requested supporting materials to the Ministry of the Environment in April 2019. In addition, in June 2018, Oskarshamn Municipal Council decided to stand behind the establishment of an encapsulation plant for spent nuclear fuel adjacent to the interim storage facility in Simpevarp.

Subsequent to the RD&D Programme 2019 being submitted, in October 2020 Östhammar Municipality Council decided to permit the establishment of a final repository for spent nuclear fuel in Forsmark. The Government detached the application to increase the quantity of spent nuclear fuel for interim storage in Clab from the application submitted for a system for management and final disposal of spent nuclear fuel. In August 2021, SKB obtained a licence under KTL and permissibility under MB to increase the quantity of spent nuclear fuel in Clab. During the spring of 2022, negotiations on conditions were held in MMD and a judgment and environmental permit were issued on increasing the quantity of spent nuclear fuel in Clab from 8 000 to 11 000 tonnes. Only after approval of the SAR can the quantity of spent nuclear fuel in Clab be increased.

The Government decided on permissibility pursuant to MB and a licence pursuant to KTL for the remaining parts of SKB's application for the KBS-3 system in January 2022. The licensing process, with negotiations on conditions and submission of an application prior to construction (including PSAR), is expected to continue in 2023–2024 – see Figure 1-1. Construction of the final repository system can only begin after the judgment of the MD and SSM's approval of the PSAR for the Spent Fuel Repository and Clink, respectively.

In parallel with the licensing process, research and development work and other measures needed as a basis for e.g. the PSAR for the Spent Fuel Repository and Clink has continued.

Clarification of state responsibility for radioactive waste

A legislative amendment that came into force on 1 November 2020 clarified the state's responsibility for a geological final repository that has undergone sealing and final closure and for the waste disposed of in the repository. At that time, a further step was also introduced in the licensing process, which means that a licence from the Government is required prior to final closure of a geological repository (Section 1.1.5).

1.2.3 RD&D Programme 2022

The RD&D Programme 2022 follows the structure used for the RD&D Programme 2019, consisting of three parts and corresponding chapters, except that the chapter on site selection and safety assessment of SFL is not repeated. The parts of the programme are:

- Part I Activities and plan of action.
- Part II Waste and final disposal.
- Part III Decommissioning of nuclear facilities.

The purpose of the report in the RD&D Programme 2022 is to meet the requirement on the comprehensive research and development work that are needed to develop and implement the remaining measures specified in Section 12 of KTL (Section 1.1.1). This means that the programme presents the overall plans for implementing the remaining parts of the waste system and for decommissioning of the nuclear power reactors and SKB's nuclear facilities. It describes concrete measures in research, technology development and demonstration planned for the remaining parts of the waste system during the next six years, as well as planned measures in other areas of interest to SKB. Activities that are conducted within the framework of a given licence are reported from an overall system perspective.

This RD&D Programme takes the strategic direction one step further compared with the previous RD&D programme. The purpose is to set out the reasoning behind the presented plans and activities. For the nuclear waste programme, it is important to take a flexible approach and ensure that activities are carried out in a logical sequence which takes needs and dependencies into account. In order to allow the nuclear power reactors to operate, space must be available for interim storage of the spent nuclear fuel. For facilities that are dismantled and demolished, interim storage facilities for the waste need to be in place before the final repositories have been built. These and other dependencies are reflected in the activity and milestone schedule, the focus of which is to provide a system-wide perspective in which relations between different activities and milestones for the different facilities are more important than fixed target dates.

Research and technology development have been, and remain, the basis for building safe and fit-for-purpose final repositories. However, the stepwise licensing process means that SKB's research and technology development work for a licensed facility will be presented in more detail in this context than in the RD&D programmes. In this RD&D Programme, the programme activities in Part II are set out in bullet points in order to provide a clear overall summary of the research and technology development activities that are planned to be carried out during the RD&D period.

RD&D Programme 2022 is mainly intended for experts and decision-makers at regulatory authorities, but also for other stakeholders with knowledge of nuclear waste issues. Experts' need for information on specific issues are meet by references.

1.3 Financing

The costs of managing short-lived low- and intermediate-level waste from the operation of the nuclear power reactors are paid on an ongoing basis by the reactor owners. The financing of the decommissioning of the reactors and the nuclear waste programme is based on the reactor owners paying a nuclear waste fee per kilowatt hour of electricity produced for the reactors that are in operation and as an annual sum for the reactors that are permanently shut down. These payments are provided for in the Financing Act and an associated ordinance, and are invested in a special fund, the Nuclear Waste Fund.

Besides paying fees, the reactor owners ‘ parent companies provide collateral to cover the fees that have not yet been paid (the financing amount). Guarantees are also provided for the possibility that the Fund will prove to be insufficient due to unplanned events (the supplementary amount). The financing amount and supplementary amount, together with the reactor owners' share of the Nuclear Waste Fund, are intended to ensure that a reactor owner can fulfil its obligations even if no further nuclear waste fees are paid and no additional guarantees are provided.

Every three years, SKB prepares a calculation of costs, a plan report, on behalf of the reactor owners. The report is submitted to the Swedish National Debt Office, which reviews SKB's calculation, calculation methods and supporting data. The Swedish National Debt Office submits proposals to the Government for nuclear waste fees, financing amounts and supplementary amounts for the coming three-year period. The fees and amounts are decided by the Government. The reactor owners pay the fees to the Nuclear Waste Fund, a state authority under the Ministry of the Environment. According to Government regulations, these funds may be invested in interest-bearing accounts at the Swedish National Debt Office, in debt instruments issued by the Government and in covered mortgage bonds. A part of the fund may also be invested in corporate bonds and shares.

At the turn of the year 2021/2022, the reactor owners' shares in the Nuclear Waste Fund amounted to around SEK 81 billion (market value). In total, the two guarantees amount to approximately SEK 58 billion. In addition, approximately SEK 55 billion (current price level) has been spent on, among other things, siting process, site investigations, construction and operation of the current system and research and development work. For the period 2021 to 2023, the average nuclear waste fee is around SEK 0.042 per kWh of electricity produced by the nuclear power plants that are in operation. Barsebäck Kraft AB will pay no fee during this period, as the value of the fund is sufficient to cover the remaining costs.

2 Description of the waste system

The waste system for management of radioactive waste consists of two main parts: the system for low- and intermediate-level waste and the system for spent nuclear fuel (KBS-3 system). The transport system covers both the low- and intermediate-level waste and the KBS-3 system. Figure 2-1 shows current and future facilities within the systems with alternative waste management paths.

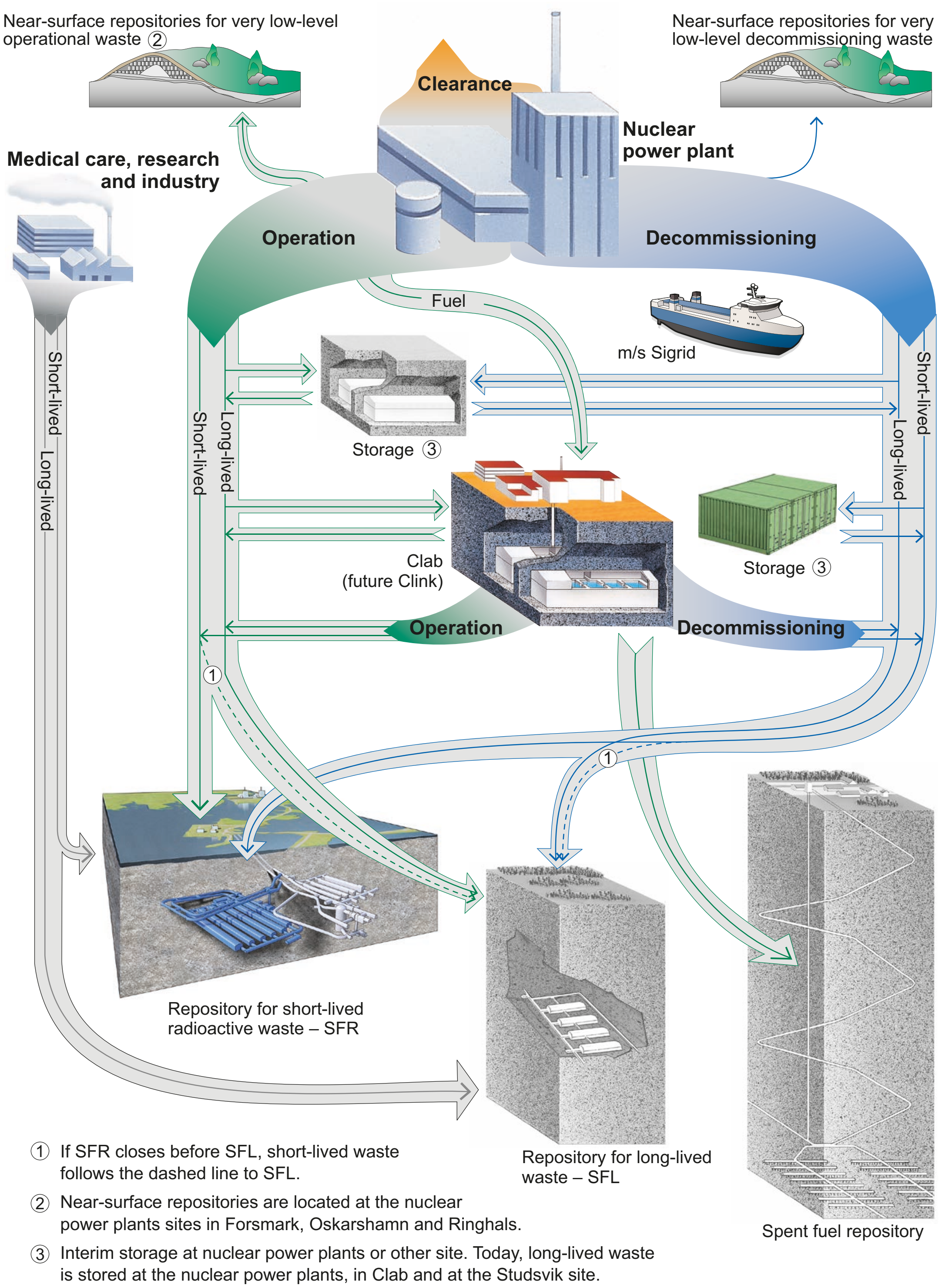


***Figure 2-1.** The system for management and disposal of radioactive waste and spent nuclear fuel generated in Sweden. Solid lines represent transport flows to existing or planned facilities. Dotted lines represent alternative waste management routes.*

This chapter presents the current situation of the systems and the design of the completed systems. It provides an overview of the management of the radioactive waste and the spent nuclear fuel, while Chapter 3 presents the plans for implementation of construction of the systems.

The system for low- and intermediate-level waste includes facilities for handling, interim storage and final disposal. Currently there is a Final Repository for Short-lived Radioactive Waste (SFR) as well as facilities for treatment, interim storage and final disposal in near-surface repositories adjacent to the nuclear facilities. Management and final disposal of all radioactive waste that is generated during operation and decommissioning of the nuclear power reactors and other nuclear facilities requires existing facilities to be extended and new facilities to be added.

The KBS-3 system includes facilities for management and disposal of the spent nuclear fuel. The system currently comprises the Clab interim storage facility. Facilities to be added to the system are Clink, which consists of the current Clab and an additional section for encapsulation of the spent nuclear fuel, and the Spent Fuel Repository for final disposal of the spent nuclear fuel.

The transport system includes ports, a ship, transport casks and port terminal vehicles for short-distance land transport. This system will be extended to accommodate transportation for the additional facilities.

When they have been completed, the three systems are designed for management and disposal of all spent nuclear fuel and radioactive waste from the reactor owners and SKB's nuclear facilities. In the event of changing circumstances, for example in the operating times of the nuclear power reactors or a forecast regarding activity inventories, this will be managed as described in Section 3.7.

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

The system for low- and intermediate-level waste consists of facilities for handling, interim storage and final disposal. The facilities and management of the waste are adapted to different categories of waste – see Section 1.1.4. The system includes facilities operated by SKB and facilities operated by reactor owners. In addition, there are facilities at the Studsvik site, at which waste from the nuclear power plants and Clab is managed or kept in interim storage as necessary. The current facilities will be extended and supplemented with new ones to manage and dispose of all the radioactive waste that arises during operation and decommissioning of the Swedish nuclear power reactors and SKB's nuclear facilities.

2.1.1 Facilities for short-lived waste

The site where short-lived waste is disposed of depends on the level of radioactivity. Waste that contains radionuclides with a short half-life and which has a low surface dose rate can be disposed of in near-surface repositories. Other short-lived waste is placed in the Final Repository for Short-lived Radioactive Waste (SFR).

Treatment of waste

There are treatment plants for short-lived waste at the nuclear power plants, at the Studsvik site and at Clab. Here the waste is treated so that it meets the requirements for clearance or for disposal in SFR or in near-surface repositories. The purpose of the treatment may be to reduce the volume of the material, concentrate activity, solidify or condition the material. Furthermore, the waste is placed in waste packages that meet the requirements for each waste type and the receiving final repository. Decommissioning of nuclear power reactors generates large quantities of waste during a relatively short time. The capacity for management of waste will therefore be increased in conjunction with decommissioning as necessary – see Chapters 15 and 16.

Interim storage

At the nuclear power plants, there are facilities for interim storage of short-lived waste. There are buffer storage facilities for operational waste prior to further handling, such as treatment and packaging, and buffer storage facilities for finished waste packages prior to transport for disposal in the Final Repository for Short-lived Radioactive Waste (SFR).

Dismantling and decommissioning of seven reactors, including Ågesta, began before completion of the extension of SFR. This means that the capacity for interim storage of short-lived waste will be increased to accommodate the waste from decommissioning at the nuclear power plants Barsebäck, Oskarshamn, Ringhals and Ågesta. A new interim storage facility for low-level waste has been built at Barsebäck. At Simpevarp, the existing interim storage facility, a storage building for low-level waste, has been extended. Interim storage at Ringhals takes place in rebuilt existing storage facilities for intermediate-level waste and in a new storage facility for low-level waste, which will be built.

Near-surface repository

Some of the low-level waste contains very low levels of activity. Waste that has a surface dose rate of less than 0.5 mSv/h and that contains mainly short-lived radionuclides, with a half-life shorter than about 30 years, may be deposited in near-surface repositories. This waste is currently disposed of in the existing near-surface repositories that are licensed for operational waste. The near-surface repositories are located on the industrial sites at the nuclear power plants in Forsmark, Oskarshamn and Ringhals. According to current practice, the area must be under institutional control for about 30 years after the last disposal of waste. The near-surface repositories that currently exist at the power plants are only licensed for operational waste. The existing near-surface repository adjacent to the Oskarshamn nuclear power plant will be extended to have capacity and a licence for disposal of the waste from decommissioning of Oskarshamn 1 and Oskarshamn 2, as well as for the remaining operational waste from Oskarshamn 3. The extension is also planned so that it will be able to receive waste from the decommissioning of Barsebäck. There are also plans to increase the capacity of near-surface repositories at Forsmark and Ringhals.

Final Repository for Short-lived Radioactive Waste (SFR)

SFR, which is located adjacent to the Forsmark power plant, has been in operation since 1988. SFR consists of an above-ground section and an underground section with two access tunnels connecting the two – see Figure 2-2. Final disposal of the radioactive waste takes place in waste vaults in the underground section, which is located in rock 60–140 metres beneath the seabed. Post-closure safety SFR is based on the fact that the amount of long-lived radionuclides is limited in the repository and that the engineered and natural barriers slow down the dispersion of radionuclides. The design of each waste vault is adapted on the basis of the activity level of the waste that is disposed of there.

The facility has a licence to receive and dispose of 63 000 cubic metres of short-lived waste. At the turn of the year 2021/2022, approximately 40 500 cubic metres of waste had been disposed of in SFR. In order to create space for final disposal of the remaining waste from operation and decommissioning of the Swedish nuclear power plants, SFR will be extended. The extension of the underground section will be located 120–150 metres below the seabed – see Figure 2-2.

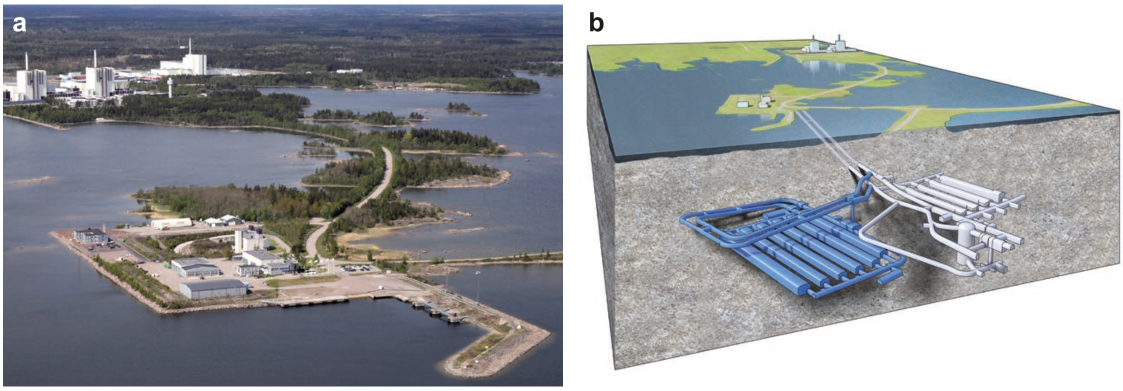


Figure 2-2. *Final Repository for Short-lived Radioactive Waste (SFR). a) View of the surface facility b) Current facility (gray) and planned extension of the underground facility (blue).*

Following extension, the capacity for final disposal will be increased by about 117 000 cubic metres. There will be a total of two waste vaults for intermediate-level waste, one in the existing part (1BMA) and one in the extension section (2BMA). For low-level waste, there will be a total of five waste vaults, one in the existing part (1BLA) and four in the extension section (2–5BLA). In the existing part, there are two waste vaults for concrete tanks (1–2BTF) and a silo for the most active waste. The extension will have a waste vault for reactor pressure vessels (1BRT) from boiling water reactors (BWR). The locations of the waste vaults in the extended SFR are shown in Figure 2-3.

2.1.2 Facilities for long-lived waste

Treatment of waste

It is currently possible to segment certain spent core components at the nuclear power plants in order to place these in steel tanks for interim storage at the nuclear sites. This has been done previously when upgrading the reactors, but presently it is mainly carried out as a part of the decommissioning projects.

AB SVAFO is currently studying the prospects for handling legacy waste, which stems from research and development within the Swedish nuclear research programmes. The study will analyse how the different waste fractions are to be managed and what the possibilities are for management and final disposal.

Interim storage

The long-lived waste is placed in interim storage until the Final Repository for Long-lived Waste (SFL) has been commissioned. At present, long-lived waste is placed in interim storage adjacent to the nuclear power plants, in Clab and at the Studsvik site. Clab is intended primarily for interim storage of spent nuclear fuel, but long-lived operational waste (control rods from BWRs and other core components) are also placed in interim storage in the storage pools. Long-lived waste generated from decommissioning of the reactors is placed in interim storage at the nuclear power plant or at another site.

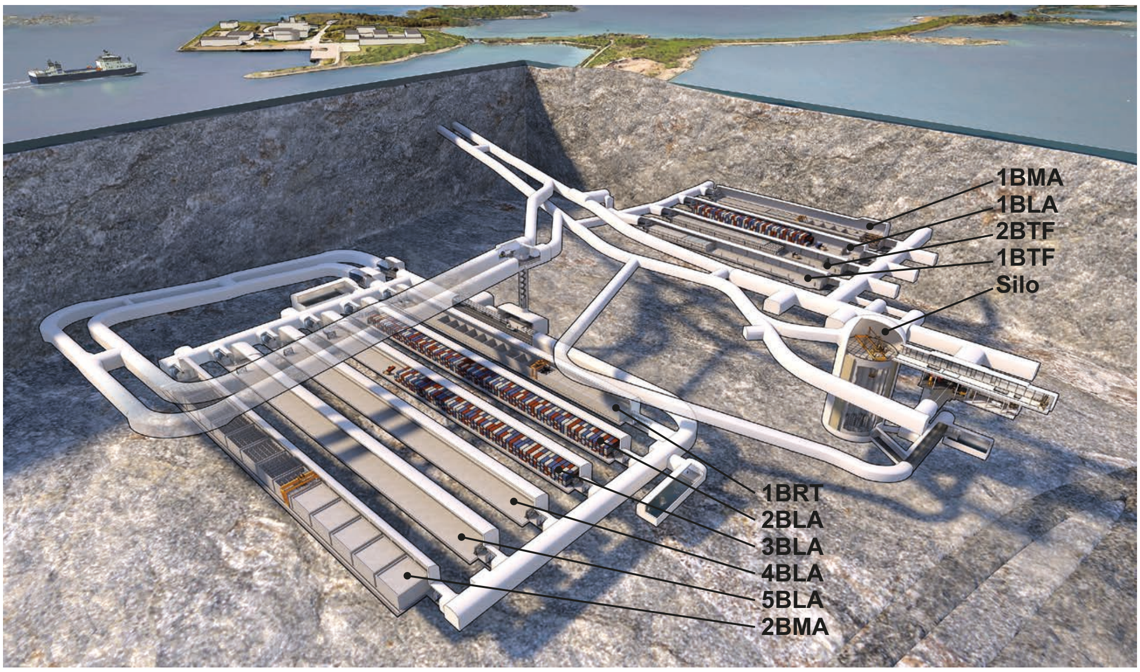


Figure 2-3. *Waste vaults in the Final Repository for Short-lived Radioactive Waste (SFR) following extension of the facility.*

Forsmarks Kraftgrupp AB currently operates an interim storage facility in a building at the site, where both short-lived and long-lived waste from maintenance stoppages and power uprates is stored. OKG Aktiebolag has an interim storage facility for low- and intermediate-level waste in a rock cavern on the Simpevarp Peninsula (BFA). The operating permit is held by OKG Aktiebolag, but BFA is licensed for interim storage of core components from all Swedish nuclear power plants. At present, waste from the Oskarshamn nuclear power plant and Clab is stored in BFA, which is estimated to have sufficient capacity for the long-lived waste that will be generated during decommissioning of Oskarshamn 1 and Oskarshamn 2.

Ringhals AB operates an interim storage facility in a building that is deemed to have sufficient capacity for the long-lived waste that will be generated during decommissioning of Ringhals 1 and Ringhals 2.

Barsebäck Kraft AB operates an interim storage facility in a building at the site, where long-lived waste from Barsebäck 1 and Barsebäck 2 is stored. The waste consists of segmented reactor internals placed in steel tanks. In order to be able to release Barsebäck Kraft AB's site from regulatory control before the Final Repository for Long-lived Waste (SFL) is commissioned, there are plans to transport the steel tanks to another interim storage facility.

At the Studsvik site, AB SVAFO has an interim storage facility for low- and intermediate-level waste (AM). The capacity for interim storage has been extended with a new building for this purpose (AUA).

Final Repository for Long-lived Waste (SFL)

SKB plans to dispose of the long-lived waste at a relatively great depth in order to avoid negative effects of permafrost on the engineered barriers. SFL will be the last final repository in the nuclear waste system to be commissioned. According to current plans, construction will begin in the mid-2040s and the facility will be commissioned just under ten years later, after which it will have an operating time of ten years. The site for the repository has not yet been decided. The storage capacity of SFL will be relatively small in comparison with SKB's other final repositories. The required storage capacity is estimated to be about 16 000 cubic metres, of which about 5 000 cubic metres is expected to come from the reactor owners.

Development of the final repository is at an early stage. SKB has developed a repository concept that includes two repository parts, one for metallic waste, mainly core components (BHK), and one for legacy waste (BHA). Post-closure safety is based on the retardation of radionuclides in the engineered and natural barriers. The core components, which consist of metallic waste, comprise about one-third of the volume, but contain (initially) the majority of the radioactivity. For the repository part for core components, SKB is planning a design based on a concrete barrier. The legacy waste is stored and managed by AB SVAFO, Studsvik Nuclear AB and Cyclife Sweden AB i Studsvik. Additional waste comes from medicine, research and industry. In the repository part for this waste, the proposed design of the engineered barrier is based on a combination of bentonite and concrete. The repository concept is illustrated in Figure 2-4.

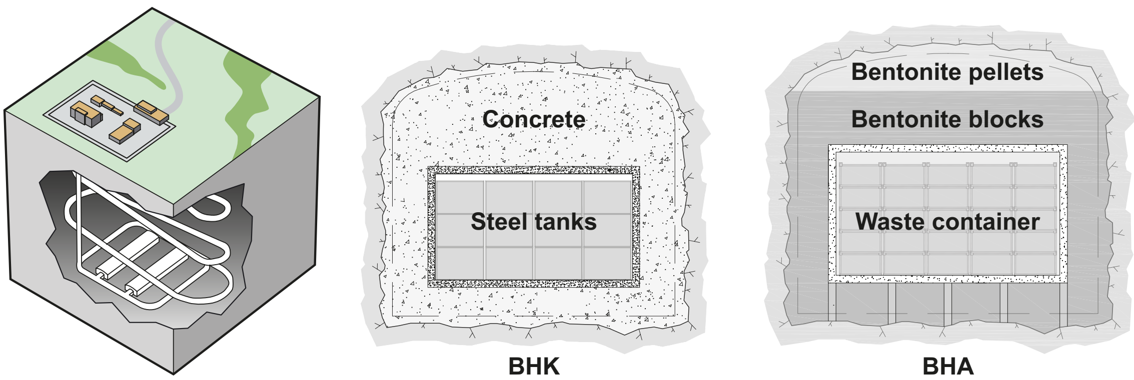


Figure 2-4. *Preliminary facility design (left) and proposed repository concept for the Final Repository for Long-lived Waste (SFL) with a waste vault for core components (BHK) and a waste vault for legacy waste (BHA).*

2.2 Facilities in the KBS-3 system for spent nuclear fuel

The KBS-3 system consists of the facilities required for implementation of the KBS-3 method. The spent nuclear fuel from the Swedish nuclear power plants is placed in interim storage in Clab, located in Oskarshamn Municipality. A new encapsulation facility will be built adjacent to Clab for encapsulation of the spent nuclear fuel. Following completion, it will be operated as an integrated facility, called Clink, for interim storage and encapsulation of spent nuclear fuel. The Spent Fuel Repository for final disposal of the encapsulated spent nuclear fuel will be built adjacent to the Forsmark nuclear power plant in Östhammar Municipality. Here, all spent nuclear fuel from Swedish nuclear power will be disposed of in copper canisters surrounded by bentonite at a depth of about 500 metres in the rock. Canister transport casks (KTB) will be developed for sea transport of the encapsulated fuel from Clink to the Spent Fuel Repository.

Central Interim Storage Facility for Spent Nuclear Fuel – Clab

The spent nuclear fuel is transported from the nuclear power plants to Clab for interim storage – see Figure 2-5. The facility, which has been in operation since 1985, consists of a receiving section at ground level and a storage section in the rock, just over 30 metres below ground level. A fuel elevator connects the receiving section to the storage section. The storage section consists of two rock caverns, both of which contain four storage pools and a reserve pool. The receiving section at ground level contains pools of water and handling equipment for transport casks and spent fuel.

Interim storage of the spent nuclear fuel takes place with about eight metres of water coverage in storage canisters, which are placed in fixed positions in the storage pools. There are two types of storage canisters for spent nuclear fuel: normal storage canisters and compact storage canisters. The two canister types have the same outer dimensions, but a compact storage canister holds more fuel assemblies.

The Government has granted SKB a licence under the Nuclear Activities Act and permissibility under the Swedish Environmental Code to increase the maximum permissible quantity of nuclear fuel for interim storage in Clab from 8 000 tonnes to 11 000 tonnes, calculated as the original quantity of uranium. There is room to receive the increased quantity of fuel in the existing facility. At the turn of the year 2021/2022, approximately 7 500 tonnes of spent nuclear fuel were stored in interim storage in Clab. According to the current forecast, the amount will exceed 8 000 tonnes in 2024.

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

Before the spent nuclear fuel is disposed of in the final repository, it will be encapsulated in copper canisters. This will be carried out in a new plant adjacent to Clab. The two plants will be operated as an integrated facility, Clink. During encapsulation, fuel is transported from its position in the interim storage facility to the encapsulation section, where it is dried and placed in the canister insert. For each canister, fuel is selected so that the total decay heat in each canister is not too great. The canister is sealed and inspections are carried out of the surface of the canister and its closure. The canister containing fuel is placed in a canister transport cask (KTB) for transport to the Spent Fuel Repository.

The canister consists of a copper shell and an insert. The copper shell – see Figure 2-6 – protects against corrosion in the repository environment, and the insert provides protection against the mechanical loads in the repository. The inserts, of which there are two types, are adapted for fuel from boiling water reactors (BWRs) and pressure water reactors (PWRs). There are also other fuel types to be disposed of. It will be possible to place these in one of the insert types.

The different components of the canister and insert will be produced by different subcontractors. After delivery to SKB, they will be inspected, assembled and processed before they are used for encapsulation of fuel.

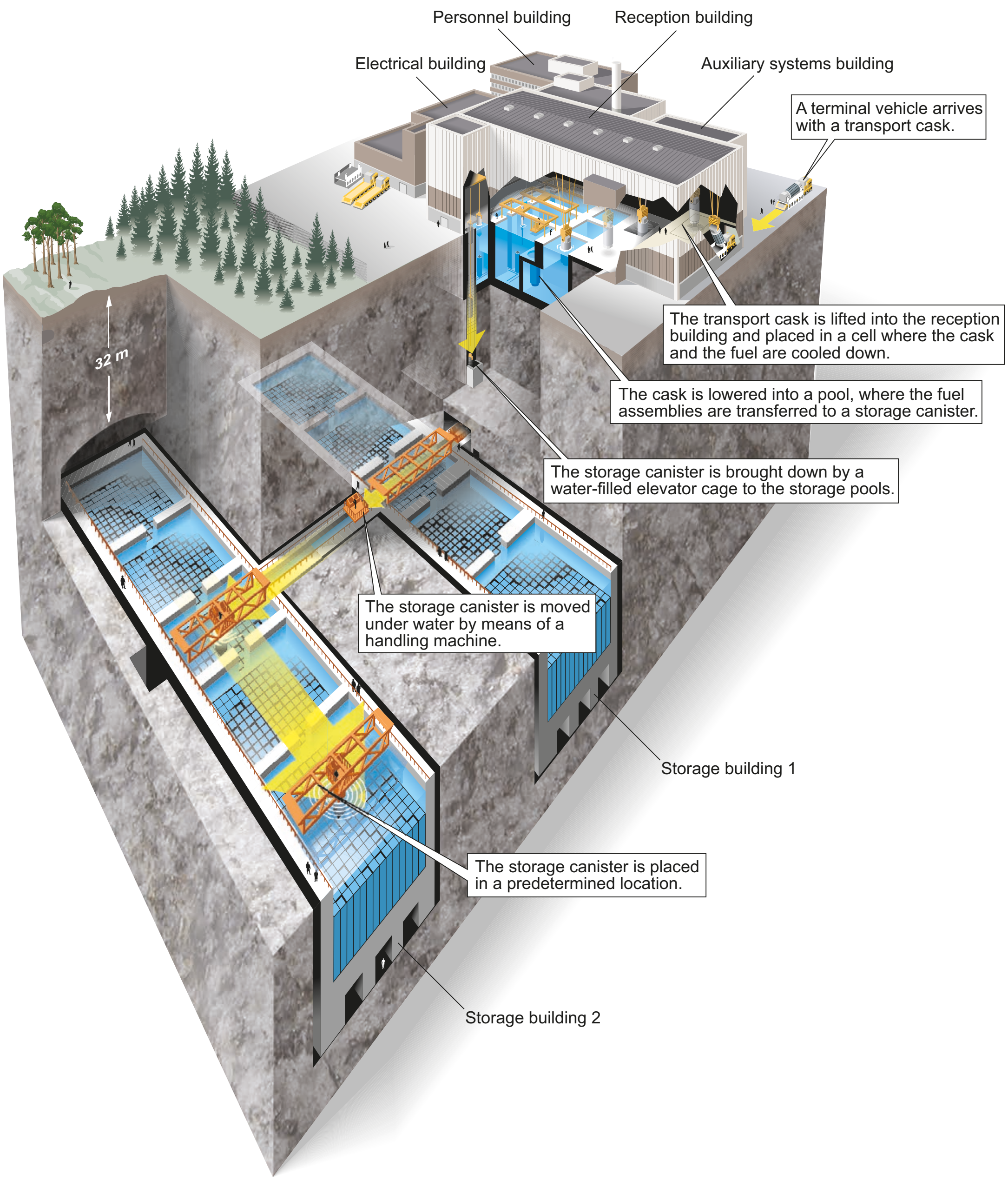


Figure 2-5. *Clab main building with receiving section and storage section in two rock caverns for spent nuclear fuel.*

Figure 2-6. *Copper canister. Length about 5 metres, diameter about 1 metre, copper thickness about 5 centimetres.*

Spent Fuel Repository

The Spent Fuel Repository will be built adjacent to the Forsmark nuclear power plant in Östhammar Municipality. All spent nuclear fuel from Swedish nuclear power programme will be disposed of there. The final repository will consist of an above-ground facility and an underground facility – see Figure 2-7. The above-ground facility consists of an operations area, rock heaps and storage buildings. The connection between the underground part and the above-ground part consists of a ramp for vehicle transport and a shaft for a lift and ventilation.

The underground facility consists of a central area and a number of deposition areas. The deposition areas together constitute the repository area. Each deposition area consists of a number of deposition tunnels with deposition holes bored into the tunnel floors. The location of the deposition tunnels, as well as the spacing between the deposition holes, is determined on the basis of the properties of the rock. Important properties include the location of large deformation zones, the presence of large or highly water-conducting fractures and the thermal conductivity of the rock. The repository depth will be 450–500 metres below ground level.

A transport vehicle is used to transport the canisters via the ramp down to the deposition level. They are then transloaded to a deposition machine that transports the canisters to a canister depot awaiting final disposal. In connection with final disposal, the canister is lifted using a deposition machine and moved from the canister depot to the deposition area for final disposal. Alternatively, the canister is loaded from the transport vehicle on to the deposition machine and transported directly to the deposition area for final disposal.

The canisters are placed in the deposition holes, surrounded by bentonite clay. When all canisters in the tunnel have been disposed of, the tunnel is backfilled with clay that will swell in contact with water. Finally, the deposition tunnel is sealed with a concrete plug. When all fuel has been disposed of, other openings are also backfilled and the above-ground facilities are decommissioned.

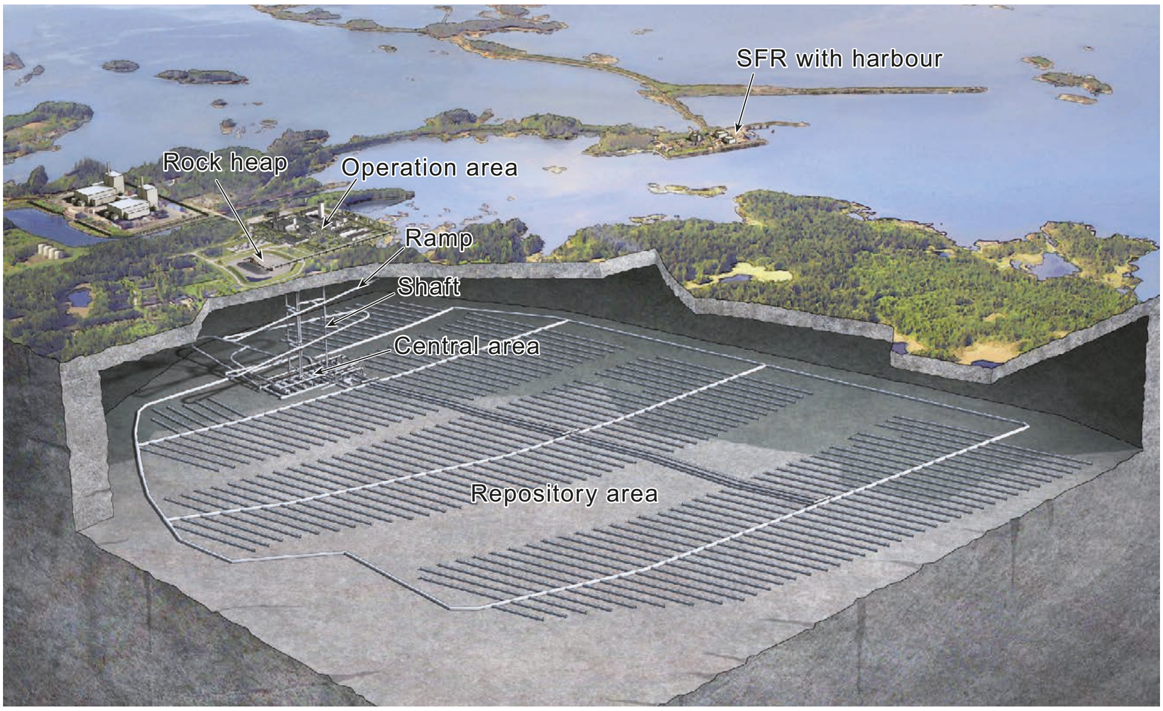


Figure 2-7. *Illustration of potential design of the Spent Fuel Repository in Forsmark.*

2.3 The transport system

The transport system consists of a ship, port terminal vehicles and different types of transport casks for fuel and radioactive waste – see Figure 2-8.

The ship, the M/S Sigrid, which was commissioned in 2014, is used for transportation of radioactive waste and spent nuclear fuel. The ship has a double hull. This structure protects the cargo in the event of grounding or collision. Typically, the ship makes between 30 and 40 trips per year between the nuclear power plants, the Studsvik site, SFR and Clab.

Short-lived low- and intermediate-level waste is transported from the nuclear power plants, Clab and the Studsvik site 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 the majority is embedded in concrete or bitumen at the nuclear power plants. The waste is transported in transport casks (ATB) with walls of steel 7–20 cm thick, depending on how radioactive the waste is.

Currently, part of the long-lived waste, including control rods from BWRs, is transported from the nuclear power plants to Clab. The waste is transported in steel transport casks whose walls are around 30 cm thick to shield against the gamma radiation from the contents. The spent nuclear fuel is also transported from the nuclear power plants to Clab in similar casks with steel walls approximately 30 cm thick. Because the fuel also emits neutron radiation and heat, these casks are also equipped with a layer of plastic for neutron shielding and cooling fins for cooling.

Work is under way to renovate and upgrade ATBs to enable the transportation of more types of radioactive waste. Transport casks are also being developed for larger core components in steel tanks.

For transportation of spent nuclear fuel, a new transport cask is being developed to meet the increased safety requirements. The design of the new transport cask differs from the current one, which is why adjustments are being made to the management of these by the nuclear power plants and Clab.

Transportation of encapsulated nuclear fuel from Clink to the Spent Fuel Repository will also take place in the future. The transport system will therefore be supplemented with transport casks for encapsulated spent nuclear fuel.

Figure 2-8. *SKB's transport system currently consists of a ship, the M/S Sigrid, port terminal vehicles and waste transport casks (ATB) for short-lived radioactive waste, core components (TK), and spent fuel assemblies (TB).*

2.4 Nuclear safeguards

Nuclear safeguards aim to ensure that nuclear material and nuclear facilities are not used for the production of, for example, nuclear weapons. As Sweden has acceded to the Non-Proliferation Treaty (NPT), all nuclear facilities must meet the requirements relating to nuclear safeguards. This means that there must be administrative systems for recording and reporting the nuclear inventory in the facilities, and technical systems for inspection and supervision so that nuclear material is not removed from the facility. All handling and processing of nuclear material must be reported, and it must not be possible to conduct any unauthorised activities in the facilities.

Aspects of nuclear safeguards are already considered in the design phase for SKB's new nuclear facilities in order to facilitate supervision and inspection during the operational phase. The purpose of the work is to ensure that interested parties are aware of the requirements in respect of nuclear safeguards so that the activities, structure and construction of facilities can be planned efficiently. An important part of supervision of the Spent Fuel Repository is being able to verify that the facility has been constructed in accordance with the drawings submitted. This is necessary so that the inspection bodies can ascertain that there are no areas or roads out of the facility that have not been reported.

When the entire KBS-3 system is commissioned, all constituent facilities and the transport system will be covered by an integrated system of nuclear safeguards to ensure that no nuclear material leaves the system. Inspections will be needed to verify the nuclear material that is to be disposed of and to ensure that it is encapsulated and ultimately placed in a final repository.

In addition to the spent nuclear fuel, some waste from legacy nuclear facilities in Sweden contain small quantities of nuclear material. This nuclear material is also covered by nuclear safeguards. At present, the waste is stored at the Studsvik site, where AB SVAFO is the licensee. Methods for verification, systems for labelling, packaging, seal handling, transportation and final disposal must be established considering the specific nature of the waste.

The regulatory authorities and inspection bodies have set up regulatory frameworks for nuclear safeguards, but these regulatory frameworks are not fully developed for the encapsulation plants and geological final repositories. SKB has participated for several years in international collaborations with the IAEA, the European Commission and the Swedish Radiation Safety Authority (SSM), among others, in order to contribute to the development of principles and methods for nuclear safeguards in these new types of facilities. SKB has prepared two draft fundamental technical facility descriptions for the Spent Fuel Repository and Clink that serve as a basis for the work of the IAEA and the European Commission on control measures. These documents will be updated as the plants are optimised and following dialogue and comments from the IAEA and the European Commission.

In order for control measures to be implemented where appropriate, work is in progress together with the control bodies on so-called Equipment Infrastructure Requirements (EIR). As part of this work, SKB submits presentations to the IAEA's regulatory body on requirements and appropriateness from an operational and safety perspective. It is important that the technical design of the facilities and the transport system is done in parallel with the development of principles and methods for nuclear safeguards, so that the facilities and transport system adhere to safe and efficient nuclear safeguards during the operational phase. After closure of the final repositories, the control measures may include possible handling above ground. SKB does, however, estimate that the design of the facilities does not need to specially take this into account.

3 Plan of action

This chapter covers planning for construction and commissioning of new and extended facilities. The reactor owners' and SKB's plans of action regarding decommissioning of nuclear facilities are also described. The chapter begins with a general account of the plans for the implementation of the nuclear waste programme. It concludes with alternative management methods and measures to deal with any major changes in the requirements relating to planning.

SKB takes a long-term approach to the planning of activities based on the plan of action in the RD&D programme, which includes all facilities until they are decommissioned, and on the basis of operational five-year plans, which are updated every year. These plans are based on planning directives based on known conditions and strategic objectives. Safe operation of SKB's facilities has top priority, and in the event of resource conflicts, priority is always given to facilities in operation. SKB needs to have flexibility in its planning, since it must be possible to adapt the activities to events in the external environment, such as decisions from the reactor owners or regulatory authorities, which may entail new planning requirements for SKB. The early shutdown of a number of reactors in Ringhals and Oskarshamn has led to a need for earlier transportation of spent nuclear fuel from these reactors to Clab, and in the long term also for transport of low- and intermediate-level radioactive waste from decommissioning. There is a growing need to commission the extended Final Repository for Short-lived Radioactive Waste (SFR) as soon as possible, to avoid having to keep large amounts of decommissioning waste in interim storage for a long time.

SKB received important Government decisions on a number of matters during the last RD&D period – see Section 1.2.2. The decisions mean that individual milestones have been reached, but the review of licensing cases under the Swedish Environmental Code and the Nuclear Activities Act continues and will continue for several years. How long each case will take before an enforcement decision is made or a judgment is issued cannot be specified in advance.

The RD&D Programme 2022 is based on Government decisions on permissibility under the Swedish Environmental Code and a licence under the Nuclear Activities Act for increased interim storage capacity in Clab, the extension of SFR and the KBS-3 system and the prerequisites for the continued licensing process. In the planning of the activities, SKB has also taken a cautious approach to the different appeals that are expected. Based on these premises, SKB has adopted the following overall strategic direction for planning of the three licensing processes:

- The application for a licence for increasing interim storage capacity in Clab to 11 000 tonnes is prioritised ahead of the KBS-3 system and the extension of the Final Repository for Short-lived Radioactive Waste (SFR) in the continued environmental licensing process and licensing process pursuant to the Nuclear Activities Act. The goal is for SKB to obtain a new licence before the licence for 8 000 tonnes is fully utilised.
- The extension of SFR is prioritised ahead of the KBS-3 system in the licensing process pursuant to both the Swedish Environmental Code and the Nuclear Activities Act. The extension of the Final Repository for Short-lived Radioactive Waste (SFR), which is needed to be able to manage waste from decommissioning of nuclear power reactors, is smaller in scope and complexity and is judged to provide valuable experience for the construction of the Spent Fuel Repository.

The continued work on the Final Repository for Long-lived Waste (SFL) has been given lower priority. The focus during the RD&D period is on work on the inventory, preliminary acceptance criteria and studies of waste casks. When necessary data become available on the inventory, which also includes the legacy waste, they will serve as the basis for an assessment of post-closure safety and pave the way for being able to apply for a licence and permissibility for the repository. This means that the start of construction of SFL will be postponed for several years.

3.1 Plan of action for the nuclear waste programme

SKB's planning for extension and construction of new facilities is based on the different licences and consents that are required according to the stepwise process, in which the steps comprise milestones – see Section 1.1.5.

The operating period for the final repositories and Clink begins with trial operation, which entails management and disposal of radioactive waste. After trial operation, the operations move on to a management phase during regular operation. Decommissioning and closure of the facilities will then take place during the decommissioning phase.

The holder of a licence to possess or operate a nuclear facility must make a new systematic, overall assessment of safety and radiation protection at least every ten years. In conjunction with these assessments, a review and summary of the state of knowledge in the areas that are essential for radiation safety are also carried out.

Figure 3-1 shows the overall activity and milestone schedule for the nuclear waste programme, including milestones for applications in the continued licensing process. The figure also shows waste streams and their extent over time.

3.2 Planning for low-level and intermediate-level waste

The final repositories that SKB plans to establish for low- and intermediate-level waste include an extension of SFR and construction of SFL. The facilities are described in Chapter 2.

The owners of the reactors that will be decommissioned before the extended SFR is commissioned intend to arrange for temporary interim storage facilities for short-lived decommissioning waste. The long-lived decommissioning waste will be placed in interim storage either at the power plants or at another site.

Decommissioning planning for the seven reactors that will be decommissioned during the 2020s is managed by each reactor owner. SKB collaborates with the different decommissioning projects on central issues related to waste management, for example acceptance criteria and waste type descriptions for decommissioning waste and transportation.

3.2.1 Overall planning

The activities planned for short- and long-lived low- and intermediate-level waste are summarised in Sections 3.2.2 and 3.2.3 respectively. Figure 3-2 shows a schematic diagram of activities and milestones for the management of low- and intermediate-level waste. To clarify the connection between decommissioning of the reactors and Clink, SFL and SFR, these are also included in the figure.

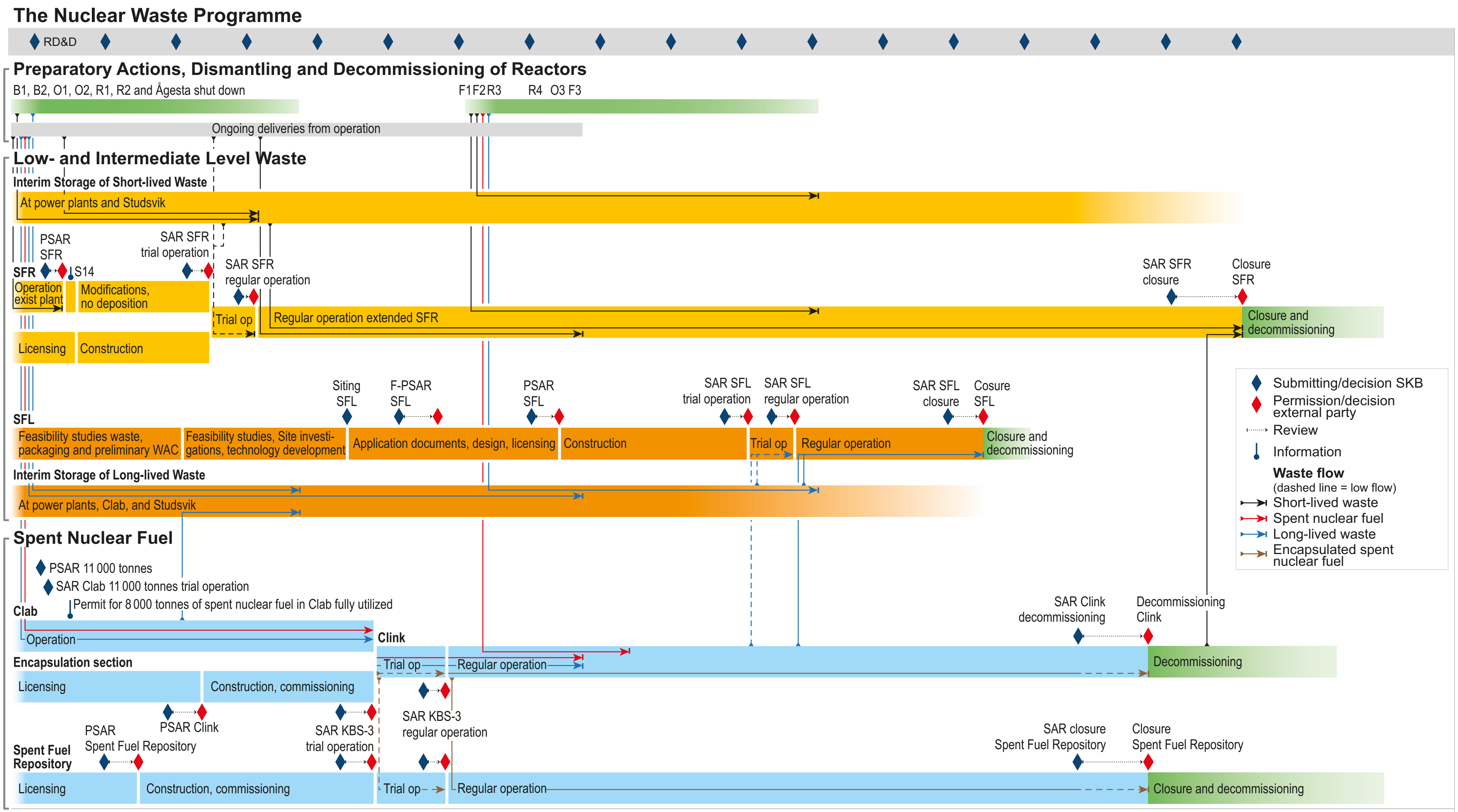
3.2.2 Short-lived waste

Interim storage of short-lived waste

The work on dismantling and demolishing the first seven reactors will begin before the extended SFR is commissioned. Barsebäck Kraft AB, OKG Aktiebolag and Ringhals AB plan therefore to place short-lived decommissioning waste in interim storage, primarily at the power plant sites, but other sites may also be considered.

Interim storage of operational waste will also be necessary during the period when construction of the extended SFR facility is in progress, since no disposal will take place during this period. Furthermore, a repository part in the existing SFR, the waste vault for low-level waste (BLA), is almost full and there are no plans to dispose of any more waste in this part until after SFR has been extended.

The strategy for the management of the reactor pressure vessels from BWRs prior to final disposal is segmentation and placement in moulds. Segmented reactor pressure vessels will be placed in interim storage with other intermediate-level short-lived decommissioning waste.



Clab	Central Interim Storage Facility for Spent Nuclear Fuel	S14	Removal S14 waste from SFR to Studsvik	Ågesta*		O1	Oskarshamn 1*
Clink	Central Interim Storage Facility and Encapsulation of Spent Nuclear Fuel	SAR	Safety Analysis Report	B1	Barsebäck 1*	O2	Oskarshamn 2*
RD&D	Programme for research, development and demonstration	SFL	Final Repository for Long-lived Waste	B2	Barsebäck 2*	O3	Oskarshamn 3
F-PSAR	Preparatory Preliminary Safety Analysis Report	SFR	Final Repository for Short-lived Radioactive Waste	F1	Forsmark 1	R1	Ringhals 1*
PSAR	Preliminary Safety Analysis Report	SSM	Swedish Radiation Safety Authority	F2	Forsmark 2	R2	Ringhals 2*
		WAC	Waste Acceptance Criteria	F3	Forsmark 3	R3	Ringhals 3
				* **Shut down**		R4	Ringhals 4

Figure 3-1. *Activity and milestone schedule for SKB's nuclear waste programme and plans for decommissioning of reactors. The figure also presents waste streams and their extent over time.*

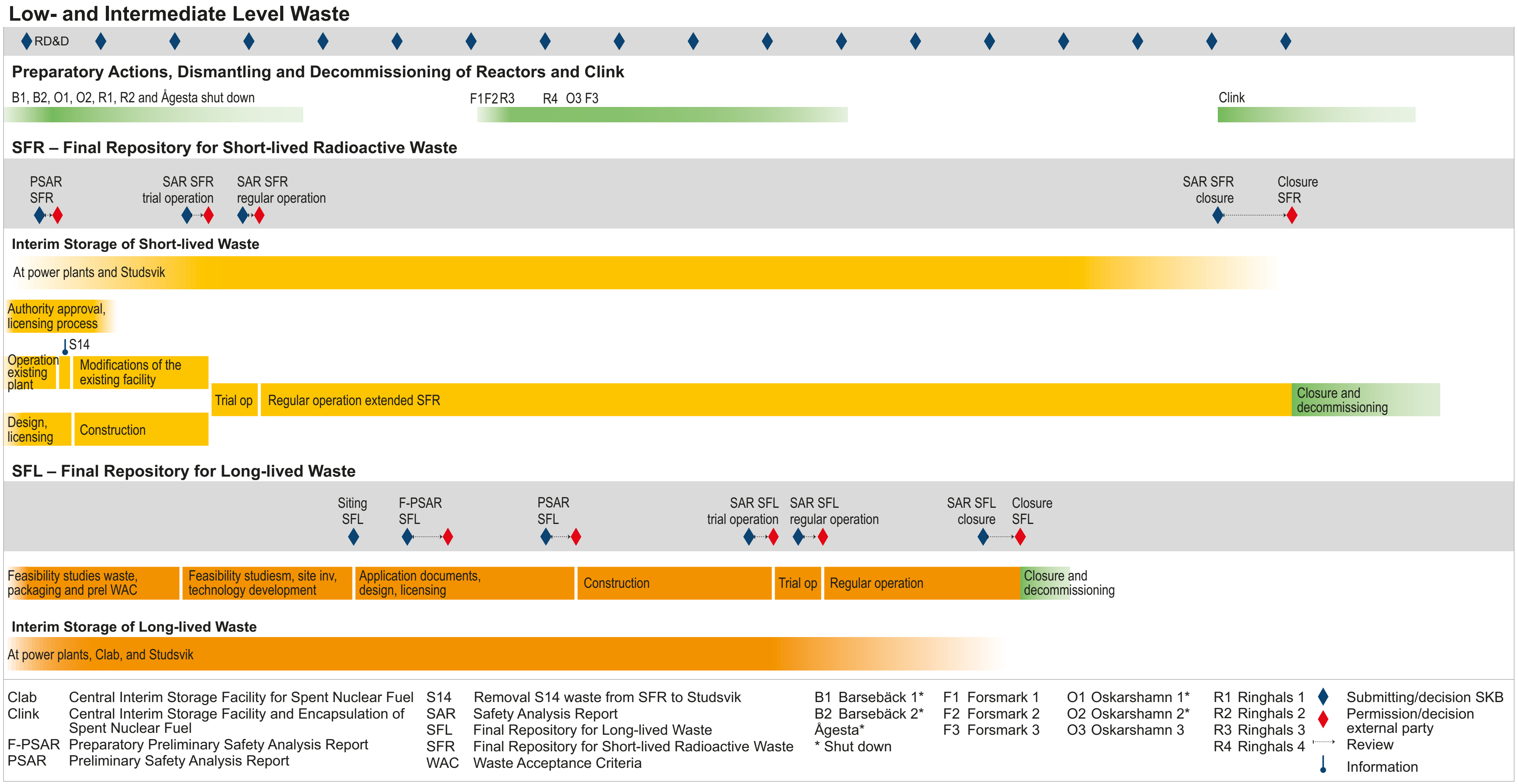


Figure 3-2. *Activity and milestone schedule for low-level and intermediate-level waste and plans for decommissioning of the nuclear power plants.*

Extension of SFR

The Government has granted permissibility under the Swedish Environmental Code and a licence under the Nuclear Activities Act for the extension of SFR. According to current planning, construction of the extension is expected to start in the mid-2020s and trial operation will begin in the early 2030s. An overall activity schedule for the extension of SFR is presented in Figure 3-3.

The Government's decision means that the licensing process will proceed to negotiations on conditions in the Land and Environment Court at the end of 2022. The next step is the submission of an application for construction with a PSAR to the Swedish Radiation Safety Authority (SSM). SKB's task is to answer questions from the regulatory authorities and to prepare and participate in the negotiations on conditions.

At the same time, SKB is commencing procurement of cooperation partners for continued design of waste vaults in rock and installations of systems in the facility. Preparatory work prior to the start of construction and detailed site investigations is also being carried out. After receiving a conditional judgment, procurement of contracts will commence and the extension of SFR will begin.

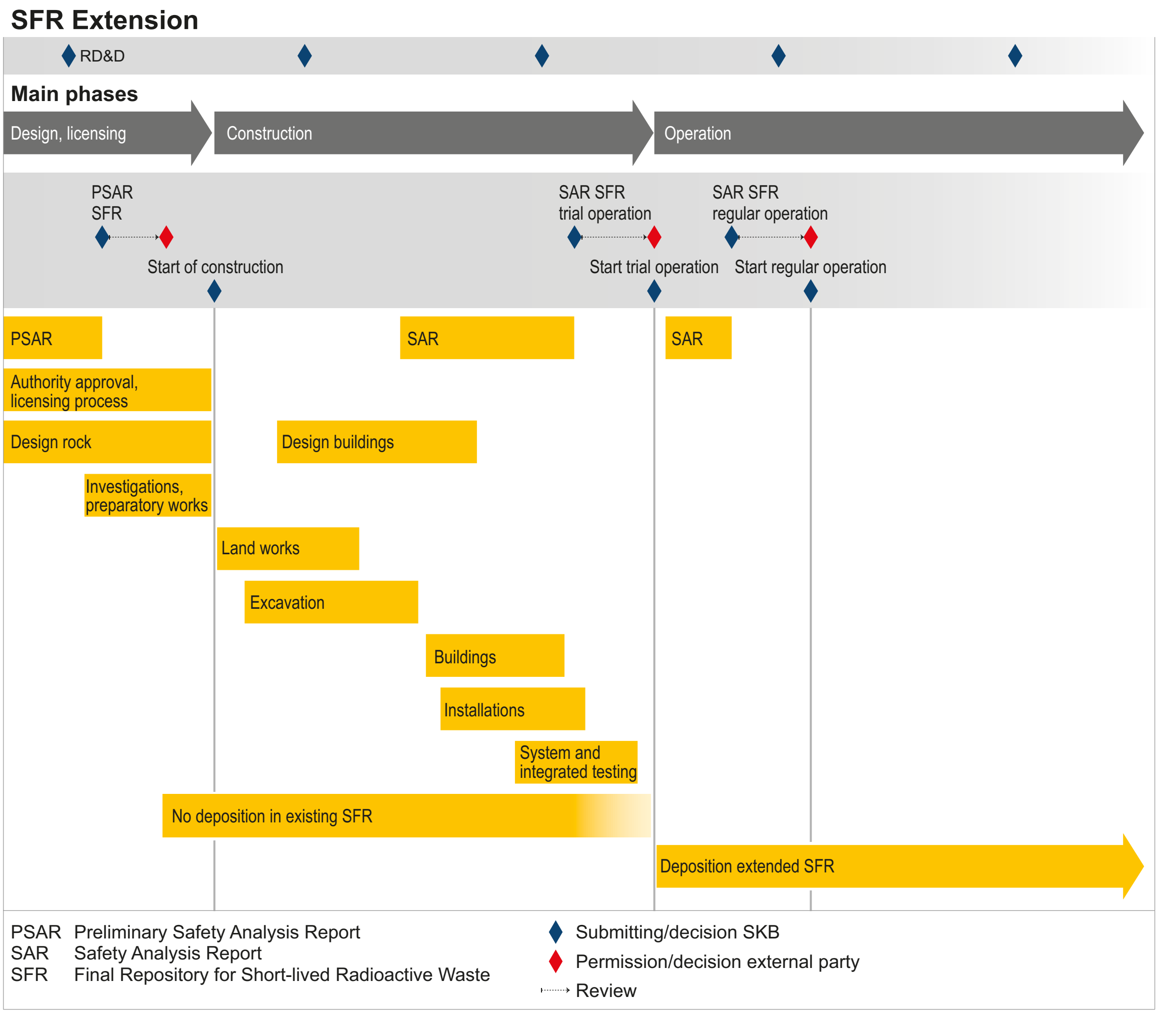


***Figure 3-3.** Activity and milestone schedule for the extension of the Final Repository for Short-lived Radioactive Waste (SFR).*

Construction and commissioning of the extended SFR

This phase includes the activities of construction, trial operation and handover to regular operation. During some of the building preparations and during construction, SKB will suspend all disposal.

At the same time as SFR is extended, the existing facility will be upgraded, among other things because the operating time has been extended in relation to the original plan. SKB plans to submit an updated SAR prior to trial operation at the end of the 2020s. It is expected that trial operation will be able to start one year later. SKB will then submit a supplemented SAR prior to regular operation.

The consequences of earlier and later commissioning SFR are presented in Section 3.7.2.

3.2.3 Long-lived waste

Interim storage of long-lived waste

SKB plans to commission SFL in the 2050s. Since several reactors will be decommissioned before SFL is commissioned, long-lived waste from decommissioning will be placed in interim storage. The reactor owners currently estimate that it is possible to create sufficient interim storage capacity at the nuclear power plants.

Existing and planned interim storage facilities for long-lived waste will be used until it is possible to transport the waste to SFL. In addition to a commissioned SFL, this will also require a new type of licensed waste transport cask.

Treatment of long-lived waste

Conditioning of the long-lived waste prior to disposal in SFL may be necessary. Before final conditioning, acceptance criteria for the long-lived waste must be established. The formulation of preliminary acceptance criteria for the long-lived waste will continue during the RD&D period on the basis of, among other things, the results of the completed safety evaluation. The current plan covers two main aspects of the long-lived waste. One is stabilisation of metallic waste from the nuclear power plants in steel tanks and the other is transloading and conditioning of waste from AB SVAFO and Studsvik Nuclear AB in containers adapted for SFL. Conditioning cannot be carried out until the acceptance criteria have been established. According to current plans, final conditioning will, if necessary, be carried out in conjunction with disposal in SFL.

Final Repository for Long-lived Waste (SFL)

SFL is the last repository that SKB is planning to commission. Several important milestones still need to be reached, such as site investigations and site selection, assessment of post-closure safety, completed licensing process and finally construction. An overall activity schedule for the work on SFL is presented in Figure 3-4. The schedule is based on a scenario where SFL is located at one of the sites of which SKB has previous knowledge. If more extensive site investigations are required, SKB believes that commissioning of SFL will be postponed. Such a scenario is further described in Section 3.7.4. During the 2030s, SKB plans to submit applications under the Nuclear Activities Act and the Swedish Environmental Code for licences to build, possess and operate SFL. According to the plans, it will be possible to commission the final repository during the 2050s and it will need to be in operation for about ten years.

Evaluation of post-closure safety of the Final Repository for Long-lived Waste (SFL)

The results of the completed safety evaluation show which waste categories that contribute the most to the calculated long-term safety consequences, which provides a basis for the work of formulating acceptance criteria. The analysed sensitivity cases for different barrier designs in the evaluation also provide supporting data for continued development of engineered barriers.

The safety evaluation forms the basis for identifying areas for further research prior to future comprehensive safety assessments. Furthermore, it provides a basis for the future siting process.

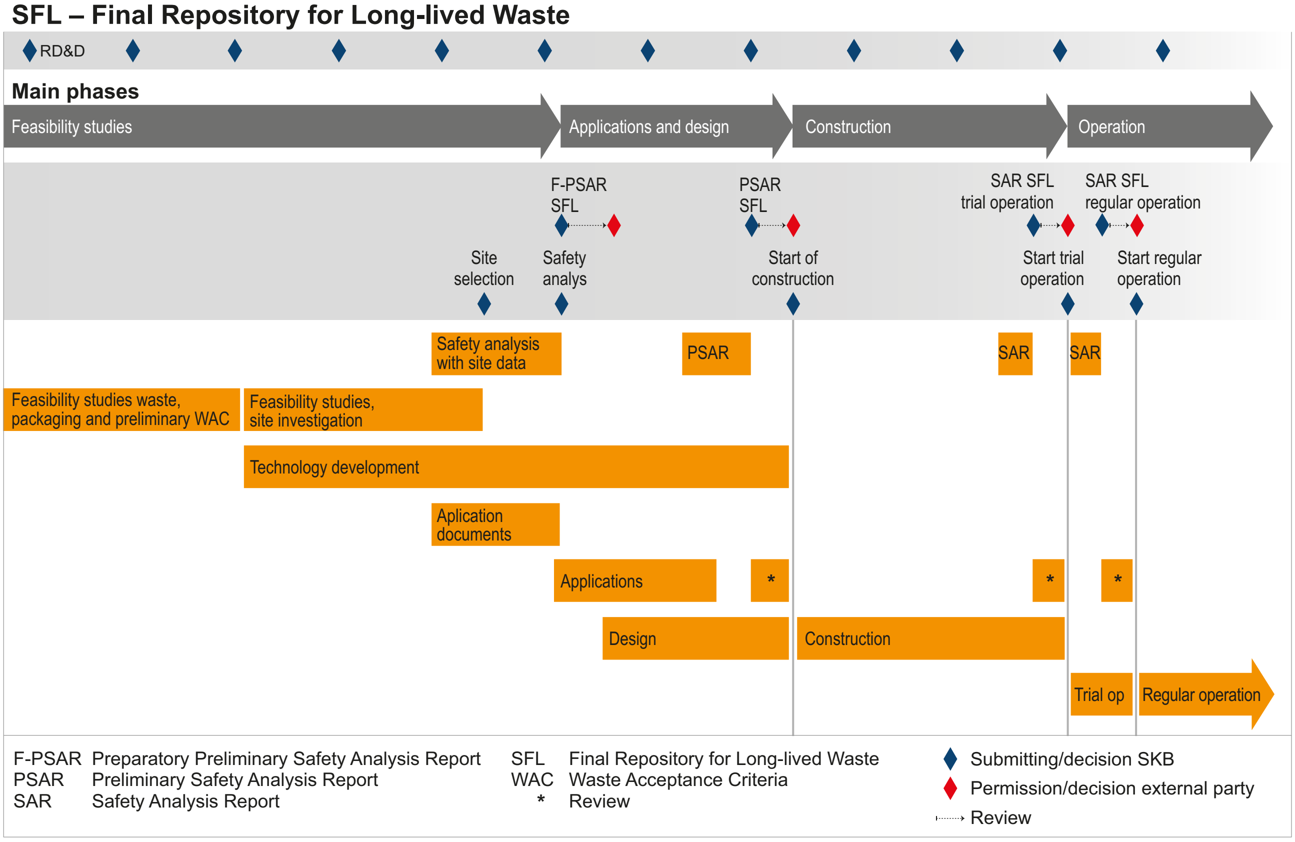


Figure 3-4. Activity and milestone schedule for work prior to commissioning of the Final Repository for Long-lived Waste (SFL).

Inventory and acceptance criteria for SFL

Better information on the properties of the waste is needed prior to future safety assessments for SFL. The waste producers are planning to characterise the waste (radiological, chemical and physical properties), especially in respect of the legacy waste, with the overall objective of obtaining a better specified inventory. The characterisation is necessary for formulating acceptance criteria, adapting treatment methods and developing waste containers and facility design.

The workby SKB in collaboration with the waste producers, also aims to provide supporting data and guidance for the waste producers' continued management of the waste.

Siting of the Final Repository for Long-lived Waste (SFL)

SKB has previously established fundamental prerequisites for the sites for final disposal of radioactive waste:

- Post-closure and operational safety as well as impact on the environment, must meet the requirements of the Act on Nuclear Activities and the Swedish Environmental Code.
- The local political and public opinion support needs to be broad and stable.

SKB plans to pursue a stepwise siting process with the objective of selecting a site for SFL in the mid-2030s. The goal is to conduct an open and transparent process in consultation with the Swedish Radiation Safety Authority (SSM), the municipalities concerned and other interested parties, where the premises for different actors are clarified early on and where the different steps in the process have been agreed upon and communicated.

The knowledge of Sweden's geology that has been gained through SKB 's previous siting processes forms the basis for the work. The supporting material consists of all areas in Sweden with data required for evaluation of safety at a depth of 400–700 metres, e.g. Forsmark, Simpevarp/Laxemar/Äspö and all so-called study sites from the siting process for the Spent Fuel Repository. A systematic approach based on grouping into siting factors has previously been developed and used by SKB for the Spent Fuel Repository. Experience from this will be applied to the siting process for SFL. Among other things, the same main groups of factors for evaluation and comparison of siting options will be used:

- Post-closure safety.
- Technology for implementation.
- Human health and the environment.
- Social aspects.

In addition to the results of the safety evaluation, the statements by SSM and the Land and Environment Court to the Government in respect of the KBS-3 case comprise important data for planning of the continued work.

Applications, construction, operation and closure of the Final Repository for Long-lived Waste (SFL)

SKB plans to submit licence applications under the Nuclear Activities Act and the Swedish Environmental Code for SFL during the 2030s. After the submission, work will continue on, for example, planning of the system and detailed design. Construction and trial operation will be followed by regular operation. Closure of SFL will take place when all long-lived waste held in interim storage and the long-lived waste from decommissioning of the last nuclear power plant have been disposed of. Before closure, SKB needs to ensure that the waste from decommissioning of Clink is suitable for SFR and does not need to be disposed of in SFL.

3.3 Planning for spent nuclear fuel

Clab is the only facility in operation in the KBS-3 system. The facilities that remain to be built are an encapsulation plant adjacent to Clab (operated as an integral part of Clab, when it will be known as Clink) and the Spent Fuel Repository, where the spent nuclear fuel will be disposed of. The facilities are described in Chapter 2.

This section describes the overall planning for future activities for the facilities in the KBS-3 system:

Planning and preparations for extension of interim storage in Clab from 8 000 tonnes to 11 000 tonnes of spent nuclear fuel.

- Completion of the licensing processes.
- SARs for the facilities.
- Planning, design, construction and commissioning of
  - the final repository in Forsmark and the production systems for buffer and backfilling material and vault seal,
  - the Central facility for interim storage and encapsulation of spent nuclear fuel (Clink) in Oskarshamn and the production system for canisters.
- Continued research and technology development.

The primary endpoint of the research and technology development being carried out for the KBS-3 system are the two facility programmes for Clink and the Spent Fuel Repository, the work on SARs and the production of the engineered barriers.

3.3.1 Overall planning

The establishment of the facilities in the KBS-3 system is divided into the following main phases: licensing (and design), construction, testing and commissioning, operation and decommissioning and closure. The activities planned during different phases are, for each facility, summarised in Sections 3.3.2 to 3.3.4. Figure 3-5 illustrates certain main activities and the different delivery phases of technology development, i.e. main periods for the development of technology components and solutions. Other milestones relate to the development of the licensing process.

During the ongoing licensing process, the progress of the projects is adapted to consider the uncertainties that need to be considered in project planning and any new information from regulatory authorities. The next milestones are:

- the main hearings on permits and conditions in the Land and Environment Court,
- submission to SSM of the PSAR prior to the construction of the Spent Fuel Repository and the encapsulation section of Clink.

As the permit-related milestones are reached, the work increases in intensity. Extensive planning is under way regarding design of facility parts and technical systems.

According to the plans, construction of the Spent Fuel Repository and the encapsulation plant in Clink will begin during the second half of the 2020s, to be commissioned during the second half of the 2030s.

Before construction of the parts of the facility that are of importance for the safety of the Spent Fuel Repository can commence, an application for construction must be submitted to and approved by SSM. The application is planned to contain the PSAR, a description of safety during construction (Suus) and plans for continued research and development for future steps in the work on SARs.

The consequences of changing the timing of commissioning of the Spent Fuel Repository and Clink are discussed in Section 3.7.5.

3.3.2 Interim storage

According to current plans, trial operation of the Spent Fuel Repository and Clink will commence during the second half of the 2030s. In connection with this, unloading of spent nuclear fuel held in interim storage can also begin. In order to be able to receive the spent nuclear fuel that is generated up until then, in addition to increasing the licensed quantity for interim storage, measures are required to free up storage space for the spent nuclear fuel. Storage capacity is increased by reloading the spent nuclear fuel stored in normal storage canisters into compact storage canisters. The compact canisters have the same dimensions as the normal canisters, but hold more fuel assemblies. An additional measure required is the unloading of core components for interim storage at another site.

Alternative methods for increasing storage capacity in the event of delayed commissioning of the Spent Fuel Repository and Clink are described in Section 3.7.5.

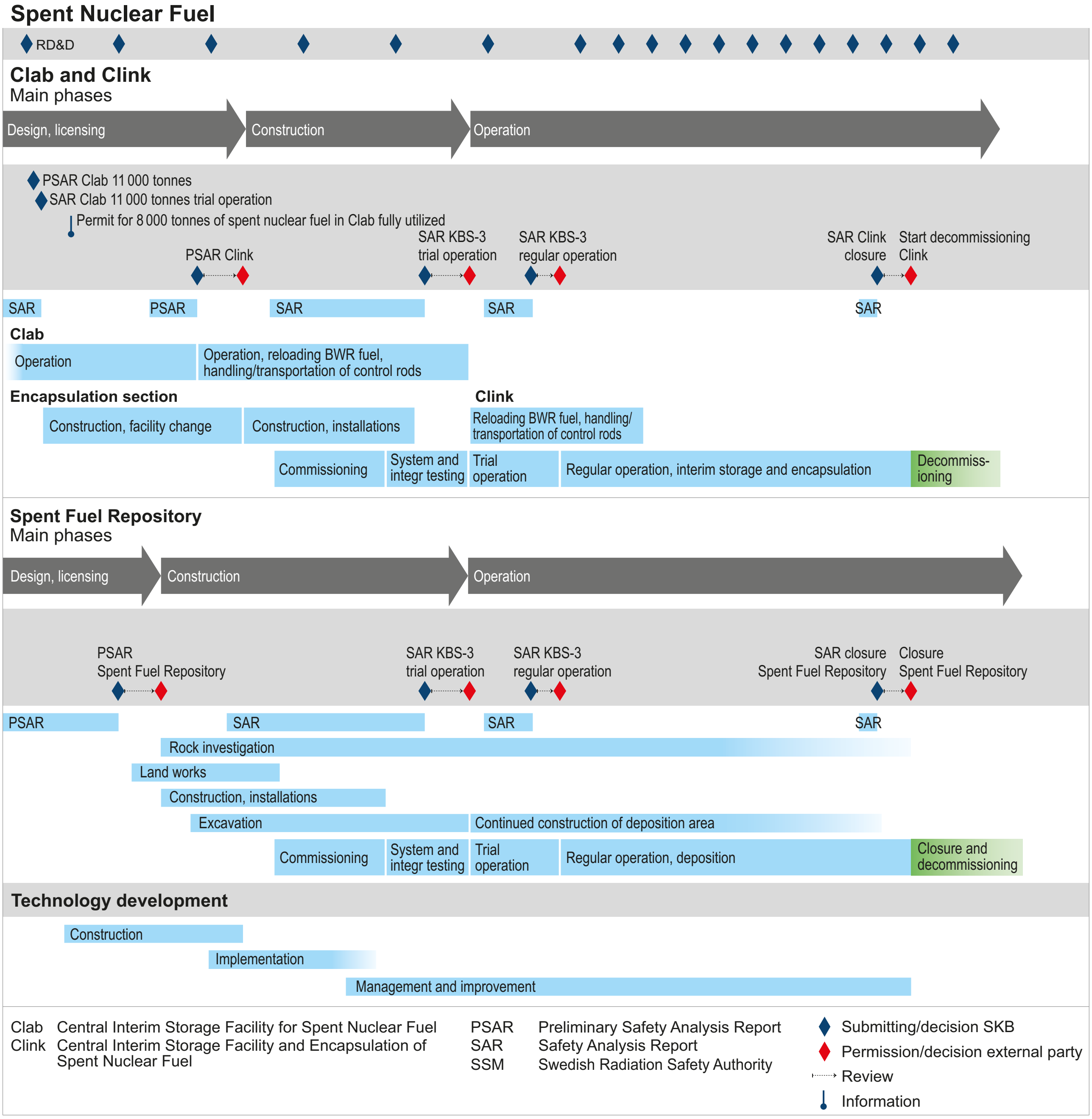


Figure 3-5. *Activity and milestone schedule for establishment of the Spent Fuel Repository and Clink based on the current situation in the KBS-3 licensing process. The technology development needed for specified milestones is presented in Part II.*

3.3.3 Encapsulation

Systems design will continue alongside the licensing process, and the results of this will serve as a basis for work during the construction phase. Planning is under way for the documents that will serve as supporting material for the application for construction that is sent to SSM.

Construction

Construction of the encapsulation plant may begin at the earliest after SSM has approved the PSAR for Clink. Besides the actual construction of the encapsulation section, construction also includes verification and validation of installed systems and the necessary qualification of the capability of certain production and measurement systems for encapsulation. Construction of the encapsulation plant will also include plant modifications in Clab. The main safety issue during the construction of the encapsulation plant is ensuring that the safety of Clab with respect to the facility and the activities can be maintained. During the construction phase, operating personnel will be trained and governing documents will be produced prior to trial operation.

Operation of Clab will continue throughout the construction of the encapsulation plant, but the reception of spent nuclear fuel may need to be limited during certain times.

Commissioning and operation

Commissioning of the subsystems of the facility will take place successively, as the systems are built and installed. Prior to trial operation, safety-related technical specifications (STF) and other documents on instructions and management of operations will be prepared. After the different functions and organisation of the facility have been tested and approved in inactive integrated testing and SSM has approved an updated SAR, trial operation can commence.

When trial operation has been completed and evaluated, an application to SSM for a permit for regular operation will be prepared. A supplemented SAR will be appended to the application, based on the experience gained and measures taken during the trial operation. After commissioning, the SAR must be continuously managed and kept up-to-date in accordance with regulatory requirements.

Operation of Clink includes both current activities in Clab and the encapsulation plant's different processes such as selection of spent nuclear fuel for encapsulation, inspections and encapsulation of spent nuclear fuel and inspection of canisters.

Production system for canisters

Renewed analyses and planning of the production system for copper canisters will be carried out during the RD&D period in order to determine the scope and time plan for the establishment of the production system.

Besides optimisation and design of canister components, current and planned technology development mainly concerns the processes for management and quality control of the production of canister components, welding of canister bottoms and sealing of the canister lid and development of inspection and testing of components and welds. This is described in Chapter 8.

3.3.4 Final disposal

Licensing and design

In line with the progress of the licensing processes, preparations required to begin construction of the access facilities to the Spent Fuel Repository will be started as soon as possible after all conditions and licences have been obtained. The construction phase will entail new requirements for SKB's organisation and activities. This applies, for example, to management of the project based on information gathered from the detailed site investigation programme and any additional surface investigations. The project organisation will be staffed successively, according to the tasks that will need to be introduced in the project.

The design of the facility will continue in parallel with the licensing process. Among other things, technical construction preparations and geological investigations will be carried out. Furthermore, surface investigations will be carried out to obtain supporting data for the location of buildings and the dimensions of foundations. The rock will be investigated, mainly at the planned locations of the access facilities. The local infrastructure at Forsmark will be prepared. This mainly involves working together with Forsmarks Kraftgrupp AB to adapt the infrastructure already in place in Forsmark to meet the needs of the Spent Fuel Repository and the extended SFR.

Work centred on the environmental impact of the Spent Fuel Repository will also continue during the licensing process. SKB has applied to the County Administrative Board (Länstyrelsen) of Uppsala for a species protection exemption. The exemption granted by the County Administrative Board has been appealed and the case has now been referred to the Land and Environment Court (MMD). Environmental permits and conditions pursuant to the Swedish Environmental Code will be issued after the main hearing has been held in MMD.

Detailed design will be carried out successively as the facility is extended, and will consider results from technology development. The results of the detailed design are needed as a basis for, among other things, in-depth planning, procurement and construction works. During the ongoing licensing process, detailed design is therefore carried out primarily for facility parts that will be built early on. This primarily includes site establishment areas, a site office, access facilities to the repository, i.e. ramp, shaft and central area, as well as some of the above-ground parts of the facility. When the facility has been commissioned, the repository area will be gradually extended.

Construction

The construction phase will begin after the PSAR has been submitted and approved by SSM.

In the first phase, parts of the area for operations will be filled in, handling areas will be prepared and temporary construction arrangements will be established. The above-ground and underground facilities will be extended at the same time.

Construction of the underground facilities is divided into three parts. The first phase involves building access facilities (shaft and ramp) down to repository level. The second phase is when the central area's underground openings are built and technical systems are installed, and the third is when the first deposition area is established and the facility is commissioned and tested. Construction of the access facilities is time-critical for the progress of the entire project. When the repository level has been reached, construction of the central area will begin. Excavation for access and central area are accompanied by installation works for the equipment that is needed to operate the facility. Construction of the above-ground facilities will keep in step with the underground works.

Construction of the access facilities and the central area will yield in-depth knowledge of rock conditions, which will be used, for example, for measures regarding rock support and sealing in tunnels or modifications of the repository design. The information will also be used to support an updated SAR prior to trial operation.

The important and challenging site adaptation of the repository will be done prior to the construction of the central area, which will take place in parallel with investigations for the first deposition area, for which an access tunnel will be driven. A few deposition tunnels, from which deposition holes will be drilled, will be driven from this access tunnel. The purpose of preparing a deposition area at this early stage is firstly to use a part of the area for production line testing and integrated testing, and secondly to gather the necessary geoscientific data required to substantiate an updated SAR prior to trial operation.

The technology development needed to complete the system for waste disposal covers buffers, backfilling, plugs and methodology and machines for installations, and is presented in Chapter 10.

Commissioning and operation

In the same way as for Clink, integrated testing will be carried out for the final repository, where all steps in operations are carried out, including disposal of a number of canisters without content of spent nuclear fuel, in order to test functions and organisation. Disposal will take place in the first deposition area built during the latter part of the construction phase. After this, once the SSM has approved an updated SAR, trial operation with spent fuel canisters can commence. Prior to trial operation, safety-related technical specifications (STF) and other documents on instructions and management of the operations will be prepared.

Before a facility can be begin regular operation, the SAR must be supplemented on the basis of experience gained from trial operation and it must be approved by SSM. After the facility has been commissioned, the SAR must be continuously managed and kept up-to-date in accordance with regulatory requirements.

Operation of the final repository includes gradual construction and completion of site-adapted deposition tunnels with deposition holes, as well as installation of buffers, transport and storage of canisters, backfilling and plugging of the deposition tunnels. Commissioning of the subsystems of the final repository will take place successively, as the systems are built and installed.

3.4 Planning for very low-level waste

Facilities for management of very low-level waste currently exist, but in conjunction with the decommissioning of the nuclear power reactors, the waste volumes that require management will increase substantially. The future decommissioning has brought to the fore the issue of handling these larger waste volumes.

The forecasts for very low-level decommissioning waste contain large uncertainties, and a review of both the estimated total quantities and the distribution between waste categories will be carried out. International experience shows that the quantities of very low-level waste and materials requiring clearance may be larger than expected.

In order to handle the large volumes of very low-level waste that are generated during dismantling and demolition of the nuclear power reactors, the reactor owners have identified a need for near-surface repositories.

Improvement work is constantly being carried out to reduce waste volumes, partly through better sorting at source, more efficient sorting of collected material, unpacking and repackaging of incoming goods, clearance and zoning of work areas during maintenance outages.

See Section 2.1.1 Commissioned and planned near-surface repositories for low-level waste.

3.5 Plan of action for decommissioning of nuclear facilities

The plan for decommissioning of the nuclear power plants at Barsebäck, Forsmark, Oskarshamn and Ringhals, the Ågesta reactor and for SKB's nuclear facilities is presented in Part III. It describes how the work has been divided between the reactor owners and SKB, and within the two groups Vattenfall AB (main owner of Ringhals AB and Forsmarks Kraftgrupp AB) and Uniper (legal name Sydkraft Nuclear Power, main owner of OKG Aktiebolag and Barsebäck Kraft AB). Additionally, the development work that remains to be done to facilitate decommissioning of the facilities concerned is presented.

The RD&D Programme 2022, the decommissioning plans of each licensee and the industry's joint cost calculation in the Plan report together comprise three interacting, required main documents that describe the planned decommissioning of the Swedish nuclear power plants and other existing or planned nuclear facilities, for example Clink and SKB 's final repositories. The three main documents supplement each other in terms of content, with the RD&D Programme presenting the development activities and other measures needed to safely decommission the nuclear power plants. The decommissioning plans present the planned execution with a focus on radiation safety and strategic aspects. The Plan report presents the estimated cost of decommissioning as described in the RD&D Programme and the decommissioning plans.

3.5.1 Overview of decommissioning

Decommissioning to release a nuclear facility from regulatory control includes a number of activities. Prior to decommissioning, the necessary licences must be in place. When a reactor is decommissioned, shutdown operation begins, when all spent nuclear fuel is transported from the reactor to interim storage. If necessary, this is followed by service operation, until dismantling and demolition begin.

The reactor owners plan to start dismantling and demolition as soon as possible after final shutdown. When the facility/parts of the facility have been released from regulatory control, conventional demolition and restoration of land can commence.

3.5.2 Current situation and overall planning

Figure 3-6 shows the overall activity and milestone plan for decommissioning of all nuclear power plants and SKB's facilities.

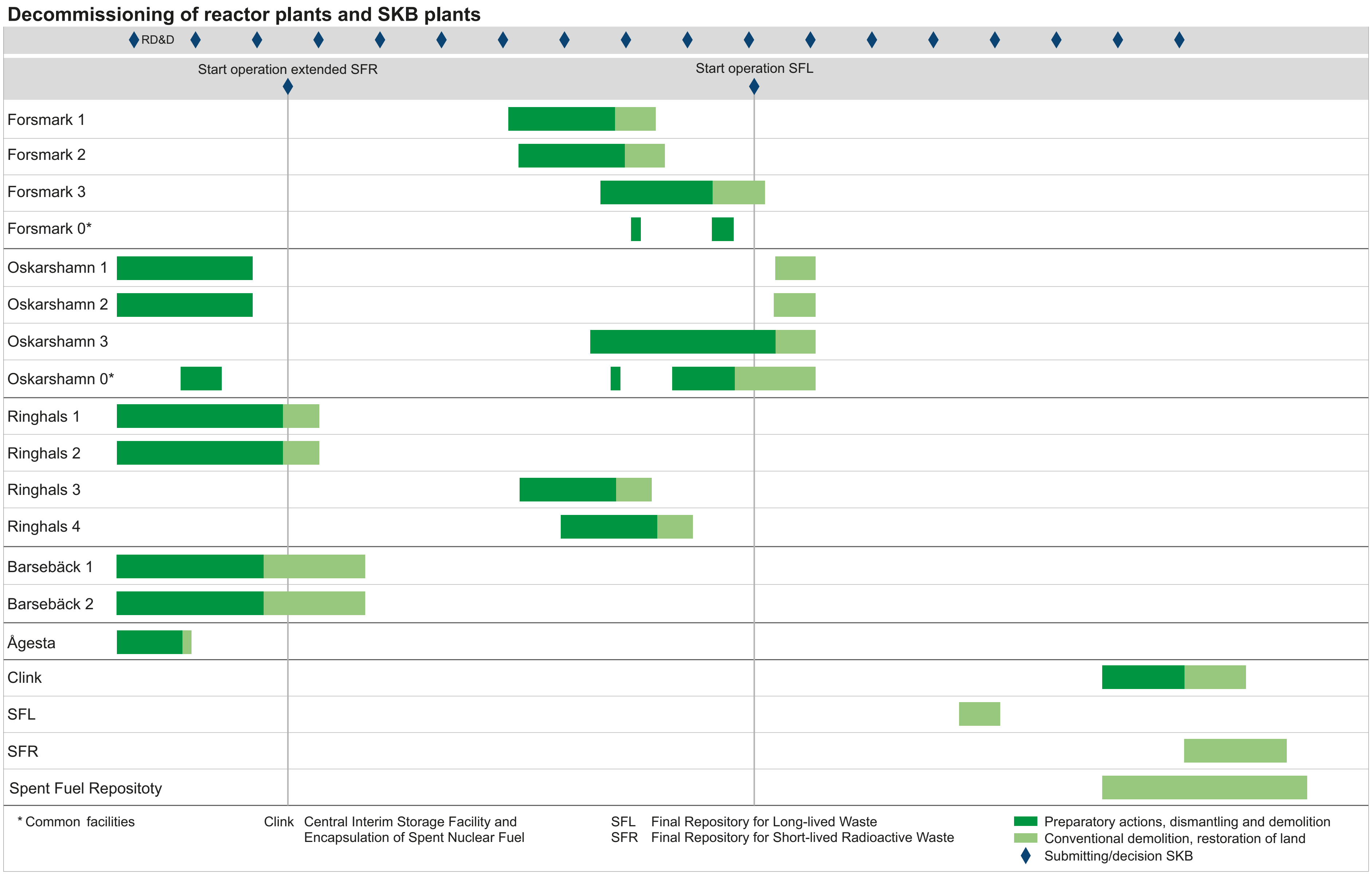


***Figure 3-6.** Activity and milestone schedule for decommissioning of all nuclear power plants and SKB's facilities.*

Barsebäck Kraft AB

The necessary permits have been obtained for dismantling and demolition, and preparatory measures required to start the work have been completed. Dismantling and demolition is under way at the nuclear power plant. Clearance of the entire site is planned to begin at the end of the 2020s and to be finished in the early 2030s.

OKG Aktiebolag

Oskarshamn 1 and Oskarshamn 2 are undergoing dismantling and demolition. Joint facilities that are not required for operation or decommissioning of Oskarshamn 3 will be demolished in parallel with the demolition of Oskarshamn 1 and Oskarshamn 2. Other shared facilities will be dismantled and demolished after the completed decommissioning of Oskarshamn 3, or left to other actors. Oskarshamn 3 is planned to be in operation until the mid-2040s.

Ringhals AB

Ringhals 1 and Ringhals 2 are fuel-free and planning and preparations are under way to begin dismantling and demolition, which is expected to be completed in the early 2030s. Ringhals 3 and Ringhals 4 are planned to be in operation until 2041 and 2043 respectively.

Forsmarks Kraftgrupp AB

All of Forsmarks Kraftgrupp AB ‘s reactors are planned to remain in operation for a total of 60 years, which means that Forsmark 1, Forsmark 2 and Forsmark 3 will be operated until 2040, 2041 and 2045 respectively.

Ågesta plant

The nuclear power plant in Ågesta has been in service operation since 1974. Decommissioning began in the summer of 2020 and is planned to be completed in the mid-2020s.

SKB’s facilities

Decommissioning of Clink and the Spent Fuel Repository can begin at the earliest when all spent nuclear fuel has been disposed of, while decommissioning of SFR can begin at the earliest when the waste from decommissioning of Clink has been disposed of. However, SFL can be decommissioned once the long-lived waste from the last reactor has been disposed of. The closure of SFL assumes that the decommissioning waste from Clink does not contain any long-lived waste and that all long-lived waste managed by AB SVAFO has been disposed of.

3.6 Plan of action for transportation

All transport takes place by sea, except from Forsmark to SFR and from Oskarshamn to Clab, which takes place overland. Since no site has been designated for SFL, it is unclear which shipments to the facility, if any, will take place overland.

3.6.1 Overall planning

Capacity utilisation in the transport system will increase from the 2030s and in subsequent decades. The number of shipments will be evenly distributed between radioactive waste and encapsulated spent nuclear fuel. The additional shipments, compared with the current situation, will mainly consist of encapsulated spent nuclear fuel that will be regularly transported from Clink to the Spent Fuel Repository and decommissioning waste from decommissioned nuclear power plants to SFR and, starting in the 2050s, also to SFL. There will also be an increased need to transport operational waste to SFR) due to the lifting of the suspension of final disposal that will exist during the period of construction of the extended facility. When the last reactors have been decommissioned, capacity utilisation will decrease and the number of shipments will then be dominated by encapsulated spent nuclear fuel to the Spent Fuel Repository.

The ship's capacity, including other components in the transportation system, is not estimated to present a barrier for the implementation of the nuclear waste programme – see Section 2.3. There is currently overcapacity in the transportation system, and the system is expected to be able to manage the increased transport volume.

3.6.2 Transportation of low-level and intermediate-level waste

Control rods for interim storage

Transportation of long-lived waste in the form of control rods from BWRs to interim storage facilities will continue as long as BWRs are in operation. A transport cask intended for core components will be used during transport.

Short-lived operational waste and decommissioning waste to the Final Repository for Short-lived Radioactive Waste (SFR)

Short-lived low-level and intermediate-level waste will be transported from the nuclear power plants, Clab and the Studsvik site to SFR for disposal. Low-level waste is transported in ISO containers while intermediate-level waste is transported in waste transport casks. Transport will take place as consolidated shipments.

Long-lived operational waste and decommissioning waste to the Final Repository for Long-lived Waste (SFL)

Transport of long-lived waste from the Barsebäck nuclear power plant to interim storage at another site is expected before commissioning of SFL. During the operating period of SFL, both the waste in interim storage facilities and the waste produced during the operating period will be transported there.

The transportation system will be supplemented with a new type of transport cask for long-lived intermediate-level waste placed in steel tanks.

3.6.3 Transportation of spent nuclear fuel

Spent nuclear fuel from the nuclear power plants to Clab/Clink

As long as there are reactors in operation, the current transport of spent nuclear fuel from the nuclear power plants to Clab will continue in a similar manner to Clink.

Encapsulated spent nuclear fuel from Clink to the Spent Fuel Repository

When the KBS-3 system is commissioned, copper canisters with spent nuclear fuel will be transported from Clink to the Spent Fuel Repository.

The transportation system will be supplemented with a new type of transport cask for transport of copper canisters containing spent nuclear fuel from Clink to the Spent Fuel Repository. The plan is that transport will take place in a steady stream, filled canister transport casks (KTB) going to the Spent Fuel Repository and empty canister transport casks to Clink.

The design of the canister transport casks will be developed in an iterative process with the supplier to satisfy regulatory requirements and SKB's own prerequisites, specific requirements and preferences. The structure and safety-related properties of the cask are described in a safety report that serves as the basis for obtaining a licence from the competent authorities in the country where it is manufactured. 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 at least seven years.

The first canister transport cask will be delivered to Clink and the Spent Fuel Repository prior to the testing of individual systems. The initial system-specific tests are planned to be carried out one year before the integrated testing of each facility. The remaining casks that are needed to achieve full disposal capacity are planned to be manufactured and delivered gradually thereafter.

3.6.4 Special transport

Experience from the transportation of spent nuclear fuel and radioactive waste using SKB's ship creates the opportunity for skilled crew to carry out transport of other radioactive goods. The experience gained from this will in turn increase competence to carry out special transport that may be needed in conjunction with decommissioning of the nuclear facilities.

The need for special transport of odd components that may occur in connection with the decommissioning of the nuclear power plants has not yet been studied. The decision to segment the reactor pressure vessels for the BWR reactors, instead of disposing of them as whole units, has simplified transport logistics. It is therefore most likely that only a few large decommissioning components may require special transport.

3.7 Alternative management methods for changed conditions

SKB 's and the reactor owners' planning for management and disposal of the waste is based on the conditions and assumptions that currently apply for the nuclear power and nuclear waste programmes. The conditions underpinning planning contain various types of uncertainties, but the activities permit relatively high flexibility. This section presents a number of possible changes in conditions and possible consequences.

In addition to the impact on the waste system of changes in the planned operating times for the remaining reactors, the timing of management and disposal of waste may be affected by the progress of the ongoing licensing processes.

3.7.1 Operating times of nuclear reactors

Since the RD&D Programme 2019, the reactors Ringhals 1 and Ringhals 2 have been shut down. For the remaining six reactors,[5] the planned operating time is 60 years, in line with the RD&D Programme 2019. This means that these reactors will be shut down during the period 2040 to 2045.

Extension of planned operating times

SKB 's facilities in the KBS-3 system will be dimensioned to handle and dispose of 6 000 canisters of spent nuclear fuel. The nuclear power companies' current forecasts, considering completed early shutdowns, estimate about 5 600 canisters. This provides a margin for possible extended operating times for the remaining reactors. For example, the dimensioned canister quantity for the Spent Fuel Repository of 6 000 canisters will be reached if the reactors' operating time is extended by around six years, i.e. to a total operating time of around 66 years. Should the operating time be extended further, it is estimated that the capacity of the Spent Fuel Repository could increase further and the operating time extended, after a proper licensing process, through the use of unutilised areas at the selected repository depth.

The need for interim storage capacity in Clab for spent nuclear fuel will not be affected by an extension of the reactors' operating time, as the additional spent nuclear fuel will be generated during the 2040s. According to the plans, disposal will then be under way in the Spent Fuel Repository, thereby releasing capacity in the storage pools. In the event of a delay of more than ten years in the commissioning of Clink and the Spent Fuel Repository, it will be necessary to increase the interim storage capacity for spent nuclear fuel. This also applies after space has been freed up in Clab – see Section 3.3.2.

[5] Forsmark 1, Forsmark 2 and Forsmark 3, Oskarshamn 3, Ringhals 3 and Ringhals 4.

The dimensioning of the extension of SFR is estimated to provide a sufficient margin for additional operational waste in the event of extended operating times. It is based on the previously planned operating times for the reactors, including an uncertainty allowance. The amount of operational waste that is disposed of in near-surface repositories will probably increase in the event of extended operating times. However, this only concerns a small fraction of the total volume of waste. The amount of decommissioning waste is not assessed to be affected by extended operating times.

In the case of extended operating times, the long-lived waste in the form of BWR control rods and other core components will also increase. If necessary, it is possible to adjust the final disposal volume in SFL until the start of construction, i.e. until around 2040 according to current planning.

Shorter operating times

Conversely, shortening of the planned operating times would entail a reduced quantity of spent nuclear fuel and operational waste and therefore lead to reduced need for storage space in the final repositories. All existing and planned facilities for the management and disposal of nuclear waste and spent nuclear fuel will nevertheless be needed. Since the Spent Fuel Repository will be built successively during operation, the size of the deposition areas can be adapted to the actual need. In this case, the number of deposition positions will decrease. If SFR has already been extended to its full size in accordance with current estimated volumes, shortened operating times for the nuclear power reactors will probably mean that the facility will not be fully utilised.

In the event that additional reactors are shut down prematurely, the total quantity of spent nuclear fuel will probably be less than the licensed quantity for interim storage in Clab of 11 000 tonnes.

According to current forecasts, the current licensed quantity of spent nuclear fuel, 8 000 tonnes, will be reached in 2024. If a license for increased interim storage is not obtained by this time, the spent nuclear fuel will need to be placed in interim storage in the storage pools at the nuclear power plants until a licence is obtained. If additional reactors of those in operation were to be shut down prematurely, their final cores would in the same way need to be placed in interim storage in the storage pools at the power plants while awaiting interim storage in Clab. Interim storage of spent nuclear fuel at the nuclear power plants after final shutdown is not a desirable scenario, since the same level of safety as in Clab cannot be achieved and it entails increased costs.

Shortened operating times of the reactors, especially if the Forsmark 3 and Oskarshamn 3 reactors (which are due to be in operation the longest) are shut down earlier than planned, would probably entail bringing forward decommissioning of the reactors concerned. It could also result in the possibility of concluding the entire nuclear waste programme earlier than planned. How much earlier would of course depend on how many reactors would be affected and by how much the operating times would be shortened. If the last reactor is demolished before SFL has been commissioned, the long-lived decommissioning waste will have to be placed in interim storage, like the waste from other reactors, SFL is commissioned.

3.7.2 Commissioning of the extended SFR

SKB plans to begin trial operation of the extended SFR in the early 2030s. According to planning, decommissioning of the seven first reactors (including Ågesta) will begin before the extension of SFR is finished. This means that other interim storage facilities for the short- and long-lived decommissioning waste are needed – see Sections 3.2.2 and 3.2.3.

Postponement

Barsebäck Kraft AB plans for the facility to be released from regulatory control as soon as possible after all radioactive waste has been transported to SFR for disposal (short-lived waste) or interim storage at another site (long-lived waste). A delay in construction of the extension of SFR would therefore also lead to a delay in clearance of the Barsebäck site, unless the short-lived waste is also transported to another site for interim storage.

For OKG Aktiebolag and Ringhals AB, the consequences of a delay in commissioning the extension of SFR are not as serious, since reactors remain in operation at these sites. It is estimated that interim storage capacity at the nuclear power plant sites could be increased to meet requirements in the case of a delay in the extension project of a few years.

Acceleration

Earlier commissioning of the extension of SFR would entail a shortened and decreased need for interim storage of short-lived decommissioning waste. For Barsebäck Kraft AB, this means that transport could take place earlier, and there is a possibility that the entire facility could be released from regulatory control before the planned date.

3.7.3 Final disposal of very low-level decommissioning waste

Since the activity level in the very low-level waste exceeds clearance levels, the options for management and disposal that exist today are disposal in near-surface repositories or in the waste vault for low-level waste (BLA) in SFR. The waste vault for low-level waste (BLA) is a more advanced facility than a near-surface repository with respect to radiation safety, and waste that is disposed of in BLA should preferably have a higher activity level than the very low-level waste to justify both costs and environmental impact (in the form of rock excavation and transportation), for example.

The dimensions of the extended SFR are designed for final disposal of all low-level waste from dismantling and demolition of the reactors, including small quantities of the very low-level waste. The waste forecasts are currently of a general nature, but the waste producers intend to update and specify them. Current thinking is in favour of final disposal of very low-level waste is near-surface repositories. This is based on studies conducted by the waste producers, in which several management route options have been evaluated.

From a technical point of view, the very low-level waste can be disposed of in BLA in the same form as in a near-surface repository. To maximise utilisation of the waste vault, however, some kind of additional treatment is preferable. Treatment could involve either moving the activity to another carrier material (through decontamination), or changing the form of the original waste (for example by means of incineration). It is mainly metals in the very low-level waste that can be decontaminated and then released from regulatory control. Remnants from decontamination are conditioned for final disposal. Soft and inert fractions, such as insulation materials, are usually technically very difficult to decontaminate, or difficult to release from regulatory control after decontamination.

For certain subsets of the waste that are combustible, controlled incineration may be an alternative method for concentrating the radioactivity content.

3.7.4 Siting and commissioning of the Final Repository for Long-lived Waste (SFL)

SKB plans to commission SFL during the 2050s, which is five to ten years after the last reactors have been shut down – see Section 3.2.3. The schedule is based on a scenario where SFL is constructed at a site of which SKB has knowledge from previous site investigation programmes.

The site for SFL has not been decided. Should SFL be built on a site where SKB has limited knowledge of the geology from previous investigation programmes, commissioning may be delayed. Such siting would require more extensive work, both to identify a site and because it leads to more extensive site investigations. A delay in commissioning would entail prolonged interim storage of the long-lived waste, both at the power plants and at the Studsvik site. This could, in turn, affect the nuclear power companies' and AB SVAFO's ability to decommission the nuclear activities on their sites.

There are also uncertainties in the current planning since the development of SFL is at a relatively early stage. A delay that results in one or more years' delay in the process up to commissioning of the repository cannot be excluded.

A possible earlier start-up of SFL is not estimated to have any negative consequences for the waste system. However, it would entail shorter times for interim storage of the long-lived waste.

Another possible alternative is that the location of SFL is divided between two different sites. According to the plans, SFL consists of two different repository parts with different barrier concepts, which, in principle, could be located at different sites. The consequences of such a route for the commissioning times are largely dependent, as is the case with a consolidated SFL, on the choice of site and the scope of the associated site investigations.

3.7.5 Commissioning of the Spent Fuel Repository and Clink

According to the plans, trial operation of the Spent Fuel Repository and Clink will commence in the second half of the 2030s, which means that SKB will then begin unloading of the spent nuclear fuel from Clab's storage pools. In order for Clab to be able to receive the spent nuclear fuel produced up to this point in time, SKB plans to increase the licensed quantity of spent nuclear fuel for interim storage to 11 000 tonnes and release storage positions in the storage pools – see Section 3.3.2. The storage space is then expected to be sufficient until the Spent Fuel Repository and Clink have been commissioned.

Postponement

A postponement of the KBS-3 system to the mid-2040s would mean that the final cores from Forsmark 3 and Oskarshamn 3 could not be accommodated in Clab, but would need to be stored in the pools at the power plants. A requirement for being able to store other produced spent nuclear fuel in Clab is that core components have been unloaded to interim storage facilities – see Section 3.3.2.

If necessary, it is also possible to increase the interim storage capacity for the spent nuclear fuel. There are two storage methods, wet and dry storage. Wet storage is the method used in Clab. Before a decision on a possible extension, the option of dry interim storage of spent nuclear fuel will be explored. This entails, among other things, analysis of aspects related to fuel properties after dry interim storage and the possible impact of this on safety after closure. Dry interim storage is currently used by a number of countries, including Spain, Germany and the USA.

Acceleration

SKB considers the likelihood of significantly/commissioning of Clink and the Spent Fuel Repository to be small. Early commissioning is not expected to have any negative consequences for the waste programme.

4 Continued research and development

This chapter presents an overview of the research and development needs that have been identified to implement the remaining parts of the nuclear waste programme. The same stepwise process for applications and licensing under the Nuclear Activities Act applies to all final repositories and Clink, even if there are differences in scope and complexity. The Government decisions regarding the KBS-3 system and the extension of the Final Repository for Short-lived Radioactive Waste (SFR) are important milestones, but do not mean the end of SKB's research and technology development.

For the KBS-3 system, research is continuing to develop and strengthen the assessments prior to future steps in the stepwise licensing process in accordance with the Nuclear Activities Act. Research and technology development are also being carried out to reduce uncertainties in the assessments, which provides margins for conducting optimisation of the final repository and its engineered barriers.

The experience gained from the operation of the existing SFR will be applied to the work of identifying processes in the waste, changing forecasts for the inventory, as well as to identify future needs for new or further developed waste containers and waste transport casks. Method development is carried out continuously to refine the determination of the inventory of low- and intermediate-level waste.

For the Final Repository for Long-lived Waste (SFL), questions remain that need to be addressed before the applications for permissibility and licences can be submitted in accordance with the Swedish Environmental Code and the Act on Nuclear Activities. A large part of the research and technology development that has been carried out and that is planned, especially in the case of SFR, will be used for SFL as well.

The specific efforts required to implement the plan for management and disposal of the very low-level nuclear waste are judged to be limited.

The basis for the planning of future research and technology development work is described in Section 4.1. The section provides an overview of how far the research and technology development needs to have progressed by the milestones that are relevant for Clink and the respective final repositories.

Sections 4.2 to 4.9 provide reasons for and summarise the research and technology development that SKB will prioritise during the RD&D period. The current situation and the programme for specific activities during the RD&D period are presented in Part II.

The concluding work that is planned to be carried out at the Äspö HRL before the hard rock laboratory is closed is presented in Section 4.10. Planning for monitoring of the Spent Fuel Repository during construction and operation is described in Section 4.11.

An overview of the technology development requirements based on decommissioning of the nuclear facilities is provided in Section 4.12, while Section 4.13 contains a description of other areas that are relevant for SKB 's mission.

4.1 Planned activities for each of the final repositories and Clink

SKB 's and the reactor owners' planning of future research and development activities for the final repositories and Clink is based on the plan of action, in which the stepwise decision process presented in Section 1.1.5 provides the basis for important milestones. Milestones related to decision steps in the form of applications and SARs dictate when knowledge and development of the technology need to have reached a certain level and when SKB will be able to commence construction and operation of the facilities – see Section 3.1.

The practical preparations prior to the start of construction include planning for the coming procurement process and building an organisational structure that will facilitate experience feedback between the various final repositories during construction. Drawing up a programme of investigations with associated method descriptions for investigations in respect of each final repository are also important preparations.

For SFL, a safety evaluation was published in 2019 (SKB TR-19-01) for the proposed repository concept. The outcome of the safety evaluation constitutes the basis for continued research and technology development. It also provides a basis for specifying site selection criteria and for designing and implementing the work of locating a site for the final repository – see Section 4.1.2.

The reports and studies described above, together with comments by the Swedish Radiation Safety Authority (SSM) in connection with the review of applications and previous RD&D programmes, as well as Government decisions that point to a need for reporting within the stepwise licensing process and future RD&D programmes, serve as the basis for this programme of research and development. It is divided into three groups:

- Increased **process understanding**, i.e. scientific understanding of the processes that affect the final repositories and their barriers, and thus the basis for assessing their importance for post-closure safety. The work is being carried out in accordance with the management of and strategy for research described in Section 5.2.
- Knowledge and competence concerning **design, structure, manufacture, and installation** of the barriers and components to be used in the facilities. The work is carried out in accordance with the technology development process described in Section 5.3.
- Knowledge and competence concerning **inspection and testing** to verify that the system barriers and components are produced and installed in accordance with approved specifications, and thereby satisfy the requirements. This also includes development of methods and instruments for inspection and testing and for monitoring of the final repositories and storage sites.

One part of the development work is to demonstrate how the developed solutions work in practice. After the ongoing experiments at the Äspö HRL have been completed – see Section 4.10 – demonstration tests will be carried out in connection with the construction of the planned final repositories, as part of the testing and verification of proposed solutions.

For the Spent Fuel Repository, monitored long-term experiments during the repository's operating time are being considered to provide in-depth information for the final safety assessment prior to closure – see Section 4.11. As an integral part of the research and development work, studies are being conducted of how the technical solutions can be optimised and made more efficient, with a maintained or increased level of safety. The development work so far has focused on finding suitable solutions for individual systems. The opportunities for technology optimisation are therefore deemed to be particularly good when it comes to interaction between the production lines for the different barriers and for the development of the technical systems within each production line.

In the Government's review prior to closure of a final repository, it is possible for the Government to address other issues, such as information preservation. This issue will therefore be relevant for SKB to follow up and work with throughout the entire operating time of the final repositories – see Section 4.13.1.

4.1.1 Final Repository for Short-lived Radioactive Waste (SFR)

Construction

During the period up to the start of the extension of SFR, some technological development of the barriers is planned in order to verify requirements and technical design requirements. This provides opportunities for optimisation of the design of the extension of SFR and the excavation. It also applies to the development of closure components, for example plugs. Updating of the properties of the installed barriers and other repository components will serve as a basis for the post-closure safety assessment prior to the SAR. A number of investigation boreholes must be sealed before excavation can begin. The sealing technique will be adapted to the conditions at SFR and a programme for quality control will be established.

At the start of construction, requirements and technical design requirements for the extension must be established. Construction will begin with the establishment of necessary infrastructure. This will be followed by construction, manufacture and installation of the barriers and preparations for commissioning. In the final phase of construction, verification and validation of systems and functions are carried out, and the construction phase concludes with integrated testing.

Operation and closure

Before trial operation begins, a renewed SAR will be prepared and the closure plan will be updated.

Prior to regular operation of the extended SFR, a supplemented SAR will be produced, incorporating experience from trial operation. The site description model will be updated on the basis of information from detailed site investigations during the construction phase. Trial operation is not expected to lead to any specific technological development needs or research prior to regular operation.

Prior to decommissioning and closure of the facility, the technology for closure, as well as closure components, will be developed with regard to materials and installation. The decommissioning plan will be supplemented with additional data, and an updated post-closure safety assessment will be included in the revised SAR that will be prepared prior to closure.

4.1.2 Final Repository for Long-lived Waste (SFL)

In the continued work on the SFL, conclusions and experience from the completed safety evaluation of the proposed final repository concept will be the starting point for the development of the engineered barriers, the acceptance criteria for the waste and the work of locating a site for the final repository. The RD&D Programme 2019 stated that a programme for planning and execution of the final disposal of long-lived waste would be started in 2020. SKB has chosen to postpone the implementation of this programme in order to first develop waste categorisation of the legacy waste. Postponement also makes it possible to make use of experience from the licensing processes for the extension of SFR and the KBS-3 system. The planned work described in the RD&D Programme 2019 (SKB TR-19-24, Section 4.1.2) remains and will be resumed and carried out at a later date. The focus of the development work during the coming RD&D period is on the inventory, packaging and acceptance criteria.

The safety evaluation identified, among other things, the following needs for research and development prior to the coming comprehensive safety assessment:

- In-depth knowledge of the inventory of the waste to be disposed of. This applies to both projected waste quantities and the material composition and radionuclide content of the waste (Section 4.2).
- Enhanced knowledge of the evolution over time of the concrete barrier in the waste vault for core components (BHK). Continued studies of the interaction between groundwater and concrete under repository conditions (Section 4.5).
- Enhanced knowledge of the evolution over time of the bentonite barrier in the waste vault for legacy waste (BHA) (Section 4.6).

A need to develop tools and methods used in the assessment of post-closure safety has also been identified. The planning to meet this need is partly dependent on the results of already planned research and development linked to SFR and the Spent Fuel Repository. Tools developed within the safety evaluation have resulted in faster feedback on technology development, allowing for greater flexibility in the work on the safety assessment. In the continued development of SFL, and specifically in the choice of best available technology and optimisation of the repository design, SKB will therefore also review the possibilities for increased functionality of the tools for modelling of groundwater flow and radionuclide transport in waste vaults and the surrounding bedrock. In this context, methods that allow flexibility in geometries and material properties are of particular interest.

Continued technology development is based on previously completed studies concerning the technology development needs, in which the set of requirements has been supplemented with conclusions from the completed safety evaluation. The main areas deemed to require development efforts specific to SFL are:

- Technical solutions for the design and construction of waste vaults.
- Technical solutions for backfilling the waste vaults with concrete and bentonite.
- Management and final disposal of different components and development and approval of the final disposal packaging for these.

To support the applications pursuant to the Nuclear Activities Act and the Swedish Environmental Code for constructing and operating SFL, technology development, information about the barriers of concrete and bentonite and the natural system, i.e. the surrounding rock and the ecosystem, need to be so advanced that it is possible for the safety assessment to show that the repository meets the requirements of the Swedish Environmental Code (general rules of consideration) and the requirements of the Nuclear Activities Act and the regulations of the Swedish Radiation Safety Authority (SSM). Requirements and technical design requirements must be presented, and it has to be shown as likely that the technical solution can be developed and installed in such a way that it is possible to verify that the requirements can be met. Because SFL differs from the other repositories, above all with regard to structures in the waste vaults and technological solutions for backfilling, these areas are expected to require development.

Preliminary acceptance criteria for the waste need to be in place, as do technical solutions for any treatment and conditioning of the waste that may be needed in order for it to meet the acceptance criteria.

Construction, operation and closure

In the coming licensing process pursuant to the Nuclear Activities Act, updated reports of the activities, the design of the facility and how requirements will be met will be described in the SARs that will be prepared in the steps prior to construction, operation and decommissioning.

4.1.3 Spent Fuel Repository and Clink

Construction

Prior to the start of construction, necessary systems, structures and components must be specified, with function and performance having been established. The submission of the PSAR is an important milestone and reconciliation point in respect of the Swedish Radiation Safety Authority (SSM). The acquisition of knowledge relating to questions concerning post-closure safety is primarily focused on providing a basis for the SAR. Enhanced knowledge will provide a basis for an assessment of whether less cautious assumptions can be used. This would provide a basis for addressing remaining uncertainties in the assessment of post-closure safety and for optimising requirements for repository components and layout. In the design of the encapsulation plant in Clink and of the final repository, the prerequisites for nuclear safeguards must be regarded – see Section 2.4.

The goal of technology development is to develop products that meet both external and internal requirements in a cost-effective manner and to ensure that the technology required to begin detailed design and procurement of the Spent Fuel Repository and the encapsulation part of Clink is available.

Already during the construction of access facilities, the central area and the first deposition area of the Spent Fuel Repository, supporting data that is required to address, issues of importance for radiation safety during the entire life cycle will be prepared. The supporting material is planned to be presented partly in a document on safety during construction (Suus) and partly in a detailed site investigation programme for the construction phase. The purpose is to identify activities and measures that are required to verify and further develop the site descriptive model of the Forsmark site prior to future assessments of post-closure safety. In addition, the technical systems (for example for disposal and backfilling) that will be present in the repository area need to be developed prior to the PSAR and SAR. When the PSAR is submitted, plans for continued development up to the SAR will be presented as a part of the application to construct the Spent Fuel Repository.

Technology and methods for encapsulation must be developed and described prior to detailed design of the encapsulation plant. The necessary technical systems must be specified, which means that the nuclear fuel measurement and the chosen method for drying of fuel assemblies need to be developed. In the same way, methods for welding and inspection of the canisters during encapsulation need to be developed and adapted to the nuclear environment in Clink.

Prior to the start of construction of the first deposition area in the Spent Fuel Repository, the technical design requirements for this area will be revised. This means that design and installation methods for buffers, backfilling and plugs must be in place, as well as methodology for excavation of deposition tunnels and deposition holes. Furthermore, the inspection methods to be applied in detailed site investigations must be verified. Also, there must be methods for investigation and modelling of the rock in the deposition area, since the deposition area must be adapted to conditions at the site on an ongoing basis from results from completed detailed site investigations.

In Forsmark, monitoring of the rock, the groundwater and the ecosystem has continued virtually unchanged after the site investigation. Monitoring is planned to continue until the start of construction of the Spent Fuel Repository. Some adjustments have, however, been made or are planned as a result of evaluations of collected measurement data. Monitoring provides a basis for establishing a reference level that can be used to assess possible environmental impact during construction and operation of the repository.

The ongoing monitoring of geosphere and biosphere parameters is planned to continue during construction and operation of the Spent Fuel Repository. What is new, compared with the site investigation, is primarily the investigations that will be performed underground. Prior to the construction of the Spent Fuel Repository, a monitoring programme for the construction and operational phases will be presented – see Section 4.11.

Operation

Prior to the operational phase of the Spent Fuel Repository and Clink, production line tests and integrated testing are planned, to verify and validate that extension and final disposal can be carried out so that the requirements for safety during operation and after closure are met. These tests will be carried out at a late stage, with the equipment and personnel that will operate the facility. This will be a final check to ensure that operation can take place as intended.

Prior to integrated testing, all systems for handling and transport of canisters, buffers and backfilling must have been manufactured, installed and tested. Process qualification with associated equipment, personnel and suppliers must be completed and documented. By that time, several systems for quality control and inspection of the barriers must have been implemented. This applies to production, handling and installation of canisters, buffers and backfilling components and to the rock construction process with associated detailed site investigations.

The site descriptive model will be updated, based on additional data from repository level, prior to the preparation of the SAR prior to trial operation. When the repository system is commissioned and in regular operation, after the Swedish Radiation Safety Authority (SSM) has approved the supplemented SAR, the activities will enter a management phase, during which the SAR must be kept up-to-date. The development work will also enter a new phase with a focus on optimisation based on experience from commissioned final repositories.

Retrieval of disposed canisters

There is no direct requirement in Sweden that it should be possible to retrieve spent nuclear fuel disposed of in a final repository. If special measures are taken to facilitate retrieval of disposed canisters, these must not jeopardise the post-closure safety of the repository. There may be situations where pre-closure retrieval becomes necessary. In tests at the Äspö HRL, SKB has assessed and demonstrated that it is possible to retrieve disposed canisters during operation of the Spent Fuel Repository. In principle, it would also be possible to retrieve canisters from a closed repository, but execution of this would require significantly more work and resources. The encapsulation part of Clink is designed so that it is possible to retrieve canisters containing fuel for re-encapsulation. It is planned for retrieval to be possible as a potential measure for dealing with any defects or quality issues that may arise or be detected during the disposal sequence. The planning for possible retrieval will be presented in the PSAR submitted prior to the construction of the Spent Fuel Repository and Clink.

In order to facilitate retrieval, some technology development is needed to ensure that the entire disposal sequence can be reversed, which includes adaptation of machines and equipment to retrieve canisters. Practical methods for handling bentonite blocks that have become partially water-saturated will also need to be developed.

Closure

A revised SAR will be prepared prior to closure of the Spent Fuel Repository. It will contain an updated post-closure safety assessment, plus a plan for closure and decommissioning. The updated post-closure safety assessment will be based on the as-built facility and the planned closure measures, and on information obtained during the operating period. It will present the technology and the procedures to be used for closure of remaining underground openings and boreholes (closure of deposition tunnels is carried out during the operating period), and the measures that are planned for monitoring and control of the repository and the operations during closure.

4.2 Planned activities for low- and intermediate-level waste

In the case of the low- and intermediate-level waste to be disposed of in SFR and SFL, in-depth understanding of the processes affecting the repositories is needed. Furthermore, information about the radionuclide content of the waste needs to be updated and improved. The current situation and the programme for research and technology development with regard to the low- and intermediate-level waste are described in Chapter 6.

4.2.1 Processes related to material properties

For both SFR and SFL, the migration of radionuclides from the repository is a key safety function.

Sorption of radionuclides to cement minerals is one of the most important processes that delays the release of radionuclides from the repositories. Complexing agents can reduce sorption and thereby accelerate the migration of radionuclides.

Gas production in repository environment can lead to pressure build-up and a subsequent impact on the function of the barriers. If the gas production rate is so high that the gas formed cannot be discharged in a controlled manner, the gas pressure may expel radionuclide-containing water, and in the worst-case damage barriers in the repository. Gas production is mainly due to degradation of material through e.g. corrosion, microbial or radiation effects.

Swelling waste may also affect the integrity of concrete barriers and thus affect the migration of radionuclides.

During the RD&D period, SKB will continue to broaden its understanding of these processes; sorption of important elements and degradation of selected organic materials to potential complexing agents will be studied further. SKB also intends to continue investigating swelling of bitumen-solidified waste under repository-like conditions.

4.2.2 Radionuclide inventory

In order to ensure a sufficiently well-determined radionuclide inventory in the low- and intermediate-level waste for future post-closure safety assessments for SFR and SFL, the state of knowledge needs to be enhanced. The radionuclide inventory is updated on a continuous basis and special measures will be taken for te so-called difficult-to-measure nuclides, which in many cases are significant for post-closure safety. Since the measured activity for difficult-to-measure nuclides only exists as indirect measurements or as analyses of individual samples, it is necessary to apply computational models for estimating the radionuclide inventory of these nuclides.

During the RD&D period, SKB will continue to develop models for activity determination of difficult-to-measure nuclides, partly by utilising data from measurements that has recently been compiled, and partly by conducting additional verifying measurements of these nuclides.

4.2.3 Acceptance criteria for waste in SFL and the extended SFR

At present there is a considerable amount of long-lived waste in the waste producers' interim storage facilities, and additional long-lived waste will be generated during continued operation and demolition of the nuclear facilities. In order to determine whether the waste from the waste producers is characterised to a sufficient extent to meet the safety requirements of future repositories, there is a need for acceptance criteria for final disposal in both SFL and the extended SFR. As the details of the repository design take shape, it will be possible to specify a set of requirements, where the results of the safety evaluation for SFL and the safety assessments for the extended SFR serve as a basis for updated acceptance criteria.

4.2.4 Waste containers and waste transport casks

In order to be able to carry out the decommissioning of the nuclear facilities in an optimal manner, the development work regarding waste containers and waste transport casks for the low- and intermediate-level waste needs to be followed up on a continuous basis. The strategy is initially to investigate the suitability of existing waste containers for disposal in future and existing repositories. The need for waste transport casks is determined by the additional waste containers that will be developed. The possibility of modifying existing waste transport casks to accommodate additional types of waste containers is reviewed on a regular basis.

4.3 Planned activities for spent nuclear fuel

Spent nuclear fuel is long-lived and highly radioactive, and requires radiation shielding in all handling, storage and final disposal. It accounts for a small percentage of the total volume of nuclear waste destined for final disposal, but contains the majority of the total radioactivity. Spent nuclear fuel can become critical and must be managed in such a way that it does not reach criticality.

If a canister in the Spent Fuel Repository were to be breached and water were to enter it, the fuel properties are crucial for determining how quickly radioactive elements might be released. The results of previous safety assessments show that the rate at which radionuclides are released from the different parts of the fuel significantly affects the post-closure safety assessment of the Spent Fuel Repository. Dissolution of spent fuel in the repository environment is therefore a key part of the safety assessment.

Research for future post-closure safety assessments and technological development regarding management of the spent nuclear fuel in the different parts of the KBS-3 system will both be needed in the coming years.

Technology for handling fuel will be developed as required as the basis for systems design and detailed design of Clink, and will be completed during construction and commissioning of the facility. The current situation and the programme for research and technology development with regard to the spent nuclear fuel during this RD&D period are described in Chapter 7.

4.3.1 Fuel integrity, fuel characterisation and fuel information

In order to ensure that it will be possible to manage all spent nuclear fuel in both Clab and the encapsulation plant, a programme for monitoring of the integrity of the fuel is being conducted (the so-called ageing programme), where changes in the properties of the fuel during storage in a pool environment are studied. SKB also follows international research and experience regarding changes in the integrity of the fuel. It is important to ensure that the fuel is not adversely affected by the planned handling in conjunction with storage and encapsulation. Non-regular fuels consist of fuels whose properties require special analysis or measures in one or more handling steps. Two important examples of non-regular fuel are failed fuel and fuel residues from analyses of various kinds, above all from the Studsvik site. The results of the activities in this area will be used in future SARs.

The work of gathering information on the spent nuclear fuel is continuing. Future studies will show how the information will best be handled and stored prior to commissioning of the complete KBS-3 system.

The decay heat from the spent nuclear fuel is a key property that influences many aspects of the entire management chain and the final repository, for instance during encapsulation, when the choice of fuel assemblies for encapsulation is largely governed by the decay heat from individual fuel assemblies. Calorimetric measurements and nuclear fuel measurements (mainly gamma and neutrons) to determine decay heat are continuing. In addition to providing information on decay heat for selection of fuel assemblies prior to encapsulation, an important purpose of the work is to provide a basis for the design of the parts of the encapsulation plant where the measurement equipment will be located. Furthermore, the fuel measurements aim to ensure that sufficient knowledge of the decay heat of the fuels is available for the coming steps in the licensing process for the KBS-3 system.

4.3.2 Criticality, radiation and nuclear safeguards

Method development is under way for analysis of criticality in different scenarios during operation and after closure of the repository. SKB's methodology for criticality analysis will be supplemented with a strategy for management of fuel bundles that do not meet the burnup requirements. The computations made using numerical methods also provide a basis for assessing the impact of the radiation on the environment.

In the area of nuclear safeguards, the ongoing development of methods for marking and identification of copper canisters containing spent nuclear fuel continues, in cooperation with the IAEA, Euratom and the SSM – see Section 2.4.

4.3.3 Fuel dissolution, radionuclide speciation and solubilities

Dissolution of the fuel in the final repository is a fundamental process in the post-closure safety assessment. An in-depth understanding of the processes that take place in connection with the radiolytic oxidation of the fuel is important for interpretation of experimental data. Knowledge of how radiation, hydrogen, oxidants and other ions in aqueous solution affect the oxidation and dissolution of uranium needs to be incorporated in a model that can calculate the fuel's dissolution rate. Further research is needed that can contribute to a better model, for example reaction rates for relevant surface reactions and how these rates change with time and other parameters.

The distribution of radionuclides in the different parts of the fuel, especially how large a fraction is situated between pellet and cladding, is also significant for assessment of post-closure safety. This varies depending on the fuel type, power history and burnup. Further research is required in this area to reflect the variation of fuels that will be disposed of in the Spent Fuel Repository. There are remaining uncertainties concerning speciation and solubility of released radionuclides, including uranium. As understanding of uranium chemistry in the repository environment is a cornerstone in the description of the chemical evolution in a leaking, water-filled canister, continued research efforts in this area are also required.

4.4 Planned activities for canister for spent nuclear fuel

The copper canister is the most important barrier in the KBS-3 system, as it must contain the spent nuclear fuel for a long enough period to give the radionuclides time to decay to an extent that they no longer pose a risk to human health and the environment.

This section describes the need for supplementary research into the properties of the canister (copper shell and insert) prior to future SARs for the Spent Fuel Repository. It also describes the technology development that is needed for the canister to be manufactured, inspected and verified in accordance with specified requirements, and used in the KBS-3 system. Chapter 8 presents the current situation and the programme of planned development activities during the RD&D period.

4.4.1 Process understanding

SSM has identified areas with a remaining need for assessment, in preparation for the coming steps in the licensing process pursuant to the Act on Nuclear Activities in respect of the Spent Fuel Repository. The majority of these areas concern the canister and its function in the repository. In order to develop the assessment of post-closure safety, both further knowledge of the processes that affect the canister in the repository and refined use of this knowledge in the assessment are needed, and these will be achieved by means of an iterative process. SKB therefore expects to continue to study processes linked to the canister in order to strengthen the supporting data for the assessment of post-closure safety of the Spent Fuel Repository.

Corrosion

If the clay buffer surrounding the canister is eroded such that advective conditions occur, sulphide corrosion is the most significant corrosion process affecting the copper shell. By means of electrochemical experiments in particular, studies are being conducted to investigate the detailed mechanisms of corrosion, the morphology of the copper surface after corrosion, and the conditions that make the corrosion more concentrated (local), which results in deeper attacks for a specific amount of corrosion than more even distribution. SKB will also continue to study the mechanisms for stress corrosion cracking and to develop the modelling of localised corrosion in the assessment of post-closure safety. The latter of which will especially focus on the period with an unsaturated bentonite buffer. A better understanding of the sulphide concentrations that can occur in the repository environment, both in the clay materials and in the rock, is important in order to avoid, as far as possible, overestimates of the transport of sulphide to the canister (and thereby of corrosion).

Material properties of canister material

As the material for the canister, SKB will use an oxygen-free copper, with low concentrations of impurities, to which small quantities of phosphorus are added to obtain favourable creep properties (sufficiently high ductility). The effect of phosphorus is attested by extensive creep testing, but for the assessment of post-closure safety, a better understanding has to be developed to show that the material properties do not change over long periods of time. SKB will therefore continue its studies, both experimental and theoretical, of the effect of phosphorus on the creep properties of copper. There are also plans for further investigations to study possible penetration of hydrogen into copper, and whether this affects the material properties in any significant way. Ageing of the insert materials, both static and dynamic, will also be studied further.

4.4.2 Technical design

SKB will continue to develop the requirements for the canister and its constituent components. The work includes requirements for both manufacture and handling of each canister component and the canister as a whole.

In order to ensure reliable future industrial production of the canister and its components, SKB will continue to develop and optimise both manufacturing methods and design. One example is the detailed design of the insert, where alternative structures and manufacturing methods are evaluated. In order to be able to produce a robust insert, it is important that both manufacturing and control aspects of the canister and the loads to which the canister may be subjected, both during handling and after final disposal, are considered in the detailed design. This involves, for example, optimising the design of the steel lid with associated gasket in order to ensure safe and reliable handling in Clink and to ensure that the requirement for the environment in the insert is met.

4.4.3 Manufacturing, inspection and testing

In order to ensure that the canister and its constituent components can be manufactured in a reliable manner in accordance with established requirements, it is of great importance to have a sufficient understanding of the manufacturing process for each component. This understanding is obtained through process mapping of the manufacturing chain for each canister component. On the basis of process mapping, process windows and detailed specifications for each process in the manufacturing chain will be tested and established. This constitutes the conditions for future qualification of the processes. On the basis of process mapping, production and measurement systems that are key to post-closure safety must undergo testing and qualification prior to trial operation.

4.5 Planned activities for cementitious materials

Cementitious materials occur in the waste matrices, barriers, and structures in SFR and SFL. In all repositories, cementitious materials are used in the plugs, rock support and for grouting. This section provides an overview of the research that is needed to improve the process understanding of the properties of the cementitious materials used in repository structures and other parts of importance for post-closure safety of the repositories. Furthermore, the technology development needed for the design of concrete structures, materials and production methods for the various repositories is described.

The current situation and the programme for cementitious materials for the RD&D period are presented in Chapter 9.

4.5.1 Process understanding

Over the time period covered by the assessment of post-closure safety, the chemical and mechanical properties of the cementitious materials will slowly change. These changes can be caused by both chemical and mechanical processes or combinations of these. Cementitious materials have a key function in maintaining post-closure safety in several respects, and SKB is conducting ongoing work to maintain and strengthen understanding of the processes that can affect the evolution of the materials' properties over time.

During the RD&D period, SKB intends to continue to pursue and develop the programme for studies of the evolution of the barriers' chemical and mechanical properties over time, under the influence of the chemical processes and mechanical loads to which they will be subjected during the lifetime of the repositories. The work will continue to be conducted in the form of both experimental studies and computer-based modelling.

4.5.2 Design of concrete structures and materials

In preparation for the extension of SFR and the construction of the Spent Fuel Repository, SKB is carrying out extensive development work linked to the design of concrete structures and materials for the various repositories.

Final Repository for Short-lived Radioactive Waste (SFR)

During the RD&D period, work will continue on the design of repository structures and development of materials for these. The programme builds on the experience and results gained from the work already carried out, with a focus on:

- design of concrete structures,
- further development of manufacturing methods for materials,
- production method for caissons,
- system for gas transport.

The focus will shift from material development and development of technology for construction to more production-related aspects.

Spent Fuel Repository

Current technical design requirements and requirements on the Spent Fuel Repository assume use of materials with a low pH level to ensure that leaching products from the materials do not adversely affect the bentonite in the buffers and backfill.

During the RD&D period, the ongoing analysis of the dispersion of a pH plume in the rock mass, from cementitious materials used for grouting and rock support, will be completed. The data received from this will form the basis for formulation of the programme for cementitious materials in the Spent Fuel Repository.

The task of specifying requirements and technical design requirements for the plug for the deposition tunnels will continue, based on the latest tunnel cross-sections.

4.6 Planned activities for clay barriers and closure

The main purpose of the clay barriers in the Spent Fuel Repository (buffer and backfill), SFR (between the rock and the concrete silo) and SFL (backfill in the waste vault for legacy waste (BHA) is to restrict the water flow around canisters and around waste containers for low- and intermediate-level waste. This is achieved by means of a material with low hydraulic conductivity and a swelling capacity that makes the installed barrier homogenise, fills cavities and seals against the rock and other repository components. For the assessment of post-closure safety, several questions remain concerning the clay barriers. For the Spent Fuel Repository and the extended SFR, this applies primarily until SAR, with the state of knowledge in the PSAR and prior to a decision on start of construction as important reconciliation points. Knowledge of the properties and function of bentonite also needs to be strengthened in respect of the continued work on the safety assessments SFL.

Most of the processes in the bentonite barriers are the same for all the different facilities. The results from the research being conducted in respect of the Spent Fuel Repository can, for the most part, also be used for the barriers in SF) and for the silo in SFR. The current situation and the programme for clay barriers and closure during the RD&D period are presented in Chapter 10.

4.6.1 Process understanding

Development after installation until saturation

In both the Spent Fuel Repository and the waste vault for legacy waste (BHA) in SFL, the clay barriers will be installed as a combination of compacted blocks and pellets or granulate made of bentonite. The bentonite barrier therefore initially has neither swelling pressure nor low hydraulic conductivity. These properties will evolve as the bentonite absorbs water from the surrounding rock.

In the rock in Forsmark, a large fraction of the deposition holes for final disposal in the Spent Fuel Repository are expected to be partially unsaturated for 1 000 years or longer. This is of no major importance for the function of the bentonite, but the chemical environment in contact with the canister will be different from that prevailing under saturated conditions. Transport of elements in gaseous form in an unsaturated buffer can take place much more rapidly than the transport of elements in a saturated buffer. A better understanding of gas composition during the unsaturated period is therefore necessary to be able to evaluate with greater certainty the significance of any corrosion on the copper surface during this period. In order to draw conclusions regarding the long-term function of the bentonite in SFL, its evolution in the waste vault for legacy waste (BHA) during the unsaturated period has to be evaluated when the site for the repository has been chosen.

During the period from the installation of the bentonite barriers up to the point at which a swelling pressure has been established, there may be very high water pressure gradients in the barriers. Together with inflows of water, this may cause channel formation and erosion of material, particularly in pellet and granulate backfill. In order to be able to evaluate the consequences of this, it is important to understand how water is absorbed in pellet backfills under different conditions.

In order to be able to better understand and describe the evolution of the bentonite material up to saturation, further research and study efforts are needed, above all concerning the following issues:

- Gas composition and its evolution in the unsaturated bentonite (primarily with regard to oxygen and hydrogen sulphide content).
- Channel formation/erosion.
- Water uptake and swelling, and homogenisation of blocks and pellets.
- Microbial sulphate reduction/sulphide formation.

Properties in a saturated state

As mentioned above, the main function of the clay barriers is to restrict water flow around the waste in the various repositories. This is achieved by means of low hydraulic conductivity, resulting in diffusion becoming the dominant transport mechanism, and by means of a swelling pressure that makes the buffer self-sealing. In the Spent Fuel Repository, the buffer will also keep the canister in place in the deposition holes, limit rock shear movements, restrict microbial activity on the canister surface, and filter colloidal particles. The most important properties of the bentonite, i.e. low hydraulic conductivity, high swelling pressure and high shear strength, are all related to its density. The relation is unique for each bentonite type and may vary for each type of bentonite depending on its composition (for example montmorillonite content). A fundamental understanding of the connection between the composition of the bentonite material and its properties is important. Therefore, SKB will continue to conduct detailed studies of the connection between the composition of the material and its properties for different bentonite materials.

For the Spent Fuel Repository and the Final Repository for Long-lived Waste (SFL), continued efforts concerning bentonite loss due to colloid release/erosion are also needed. The results may directly affect the outcome of the assessment of post-closure safety by allowing less conservative assumptions regarding buffer losses.

Work is also needed concerning the long-term stability of the bentonite with regard to temperature, iron content and the interaction with cement. The interaction between bentonite and cement is one of the most important processes for post-closure safety in SFL. Knowledge about this interaction can also be applied to the assessment of post-closure safety for the silo in SFR.

4.6.2 Barrier design, manufacturing, inspection and testing

Advances have been made in technology development for mechanical excavation of tunnels, and SKB is investigating the possibility of using the technology for the production of deposition tunnels in the Spent Fuel Repository. Backfill design is also being studied from the starting point of facilitating industrialisation of the installation. Further development of backfill design is needed regarding verification of the ability of the backfill to provide a constraint for upward swelling buffer. Furthermore, the requirements need to be updated and clarified.

Requirements relating to the installation sequence and installation of buffers and backfill components, which are dependent on water inflow into the deposition holes, will be updated and clarified. An important question is how thermal, hydraulic and mechanical processes in the buffer affect density from the time the buffer is placed in the deposition holes until the backfill is installed on top of the deposition hole. Continued full-scale tests will be carried out to investigate the maximum water inflows the buffer is able to withstand during installation. In the work on ensuring a reliable production process for buffer blocks, further development is focused on segmented buffer rings, and specifically on developing tolerances to ensure function and suitability for industrial production and installation.

Measurements of different bentonite materials that are needed to ensure compliance with requirements, such as hydraulic conductivity and swelling pressure, will continue, both to gain a better understanding of different materials and to develop competence and measurement methods.

4.6.3 Installation of buffer and backfill

Technical systems for disposal include deposition machines, backfill robots and transport systems for buffers and backfilling components. Prototypes and schematic designs for the disposal work are available. The disposal process is intended to be automated and an overriding control system is being developed in order to control and monitor such a system. Further development of machines will continue and the equipment will need to be adapted for segmented buffer blocks.

4.6.4 Closure of boreholes and repositories

The closure plan for the Spent Fuel Repository will be updated, as will the requirements for efficient closure.

A method for borehole sealing has been developed. Further development will take place for the purpose of simplifying installation and making it possible to carry out installation even in subhorizontal and horizontal boreholes. As part of the review, an inventory will be made of the need for verifying tests, and testing will be carried out.

4.7 Planned activities for rock

The most important function of the bedrock in the final repositories is to ensure stable mechanical and chemical conditions and to limit the water flow, in and adjacent to, the repositories over the time that the waste needs to be isolated from humans and the environment. In order to achieve the intended function, there is a need for sufficient knowledge of bedrock conditions and of the processes that alter the mechanical and chemical conditions in and around the repository. The underground openings also need to be designed in such a way that the long-term stable conditions are not jeopardised. The research and technology development issues that relate to the rock and the final repositories' underground openings are to a large extent the same for all three repositories.

The technological development in the rock area that is needed for the Spent Fuel Repository as a basis for the PSAR has been completed and the report on safety during construction of the final repository (Suus) and prior to the start of construction is being prepared. Further development of methods prior to the SAR is expected to lead to more efficient deposition holes selection criteria, which in turn means that fewer holes need to be rejected. Sections 4.7.1 to 4.7.4 summarise the areas with remaining rock issues with respect to research and post-closure safety assessment. The current situation and programme relating to rock for the RD&D period is presented in Chapter 11.

4.7.1 Characterisation and modelling of rock properties

Modelling of the mechanical properties of the rock mass with coupled models requires that the properties of the rock can be specified independently of the modelling tool and numerical resolution. There are knowledge gaps in the fundamental understanding of the mechanical properties of individual fractures and fracture systems and of how they interact with thermal and hydraulic properties, which in turn affect the hydrogeological, geochemical and transport properties of the rock.

Fracture propagation in crystalline, hard rock is dependent on the mechanical, thermal and hydraulic properties and on the prevailing rock stress conditions. In a long-term perspective, fracture growth is also affected by the chemical conditions of groundwater. Integrated modelling is therefore needed to assess the implications of induced movements, both in the immediate and distant area, as a result of thermally, seismically or glacially induced loads.

The computations that have been made so far of rock stresses within the repository volume of the Spent Fuel Repository in Forsmark are associated with uncertainties, mainly due to scarcity of data. In order to gather more data, both in situ measurements and modelling are needed, which will also provide a basis for improving the description of the spatial variability of the stress field with respect to size and orientation.

4.7.2 Modelling of discrete fracture networks

Discrete fracture networks (DFN), are used in the modelling of rock mechanics, groundwater flow and solute transport. The description and parameterisation of individual fractures are important input data in the assessment of many issues that play a role in the safety of the repositories after closure, as well as questions that need to be answered during the design and operational phases. A DFN methodology has been developed and will continue to be tested and developed in the coming years.

4.7.3 Seismic impact on repository safety

The research on earthquakes carried out by SKB can roughly be divided into the partially overlapping disciplines of paleoseismology, instrumentation and modelling. The main purpose of the research is to ensure that the seismic risk is not underestimated, that the negative impact on the repository system calculated in the models has not been underestimated, and to study the potential for more efficient use of the repository volume.

Earthquakes in the vicinity of the Spent Fuel Repository could induce shear movements along fractures intersecting canister positions. If the earthquakes are large enough and occur close enough to the repository, secondary shear movements are induced. These may, if they occur along inappropriately located and oriented fractures, exceed the canister's strength. For SFL, the possible consequences of earthquakes must be assessed when the site and concept for the repository have been chosen.

A remaining uncertainty concerns the relationship between earthquake frequency and magnitude and its variability during a glacial cycle. This uncertainty is partly addressed by studies of paleoseismic events and in-depth studies of measured data from earthquakes. The modelling of secondary movements in fractures also needs to be developed to permit less cautious assumptions prior to calculations in future safety assessments.

4.7.4 Groundwater flow, groundwater chemistry and solute transport

In recent years, hydrogeological modelling has developed so that geochemical processes and transport processes can now be integrated with the flow modelling. Work is required to further develop and test these new tools and extend their application areas (for example by including microbial processes) for use in site modelling and safety assessments. Furthermore, development of the hydrogeological simulation tools is planned, so that groundwater flow and radionuclide transport computations can be done in the same tool without export/import of data files with intermediate results.

In respect of solute transport, work is also needed, above all regarding matrix diffusion and sorption, in terms of conceptual understanding, reduced uncertainty in transport parameters and further development of modelling tools. This mainly affects the Spent Fuel Repository and the Final Repository for Short-lived Radioactive Waste (SFR). The work may above all lead to the possibility of using less cautious assumptions in the assessment of post-closure safety by being able to use site-specific data with a reduced uncertainty interval compared with the data used today.

4.8 Planned activities for surface ecosystems

SKB 's research programme for surface ecosystems aims primarily to produce supporting data for calculations of potential radiation doses to humans and the environment in the assessments of post-closure safety for the various repositories. The programme also provides supporting data for environmental monitoring, assessments of any environmental changes and for assessment of safety in the facilities in operation. The current situation and programme for the RD&D period is presented in Chapter 12.

Research questions related to the surface ecosystems primarily concern conditions and processes that affect radionuclide cycling and how existing and new knowledge will be applied in dose calculations in post-closure safety assessments. The research questions for surface ecosystems are in principle the same for all three final repositories. Prior to the SAR for the Spent Fuel Repository and for the extended SFR, and prior to the preparatory PSAR (F-PSAR) for SFL, there are several issues where further work is required, mainly in order to reduce the use of over-cautious assumptions in the assessment of post-closure safety. The most important remaining issues are in four different areas:

- Uptake pathways and uptake mechanisms for different organisms.
- Temporal and spatial heterogeneity in the landscape.
- Transport and accumulation processes.
- Radiological, biological and chemical properties of certain substances (e.g. carbon, chlorine and the uranium decay chain).

4.9 Planned activities for climate and climate-related processes

The overall purpose of the work on climate issues is to provide SKB's safety assessments with scientifically substantiated and updated scenarios for future climate change, as a foundation for the assessment of post-closure safety of the repositories. Depending on possible future climate scenarios, there are specific issues relating to the different repository concepts that must be covered by the safety assessments.

In order to handle the uncertainties that exist around future climate change over the time scales covered by the post-closure safety assessments, and to be able to answer the specific questions posed for the different repository concepts, a set of tailored climate scenarios is required for each repository concept and safety assessment. The climate scenarios are based partly on knowledge of historical climates and partly on modelling of the future climate. Overall, the work on climate issues therefore entails developing process understanding, compiling climate history, updating climate scenarios and validating the models and the methodology used to describe the span of climates and climate-related processes that the repositories may be subjected to during the coming 100 000 to one million years. Furthermore, other disciplines included in SKB's assessments of post-closure safety will be provided with input data, assumptions and boundary conditions for climate and climate-related processes, such as the evolution of climate parameters, permafrost, ice sheets, erosion of the bedrock surface and changes in sea level.

There are questions with a bearing on all repositories that need to be studied further in order to reduce uncertainties and to improve the credibility of the post-closure safety assessment. In addition, research is required to evaluate the methodology used to handle climate issues in the safety assessments, and work needs to be carried out to ensure that the climate scenarios are in step with the prevailing state of knowledge. In addition to providing the safety assessments with information and data, some of the issues also relate to the design of the surface facilities of the Spent Fuel Repository and SFR.

One of the main tasks of the climate research programme is to update and, if necessary, evaluate and supplement the climate scenarios used in SKB's safety assessments. In order to achieve this, the following need to be studied further:

- Analysis of extremes in climate and climate-related processes.
- History of climate change during the last glacial cycle.
- Sea-level variations and shoreline displacement in the short and long term.
- Denudation processes affecting the rock surface in Forsmark, including quantification of historical and future glacial erosion.
- Ice sheet dynamics and behaviour.

The current situation and programme in respect of climate for the RD&D period is presented in Chapter 13.

4.10 Closure of the Äspö Hard Rock Laboratory (HRL)

In the RD&D Programme 86, SKB presented the plans for an underground rock laboratory. The laboratory was, among other things, intended to provide opportunities for development and testing of investigation methods, increase understanding of groundwater flows in a large rock domain and provide a site for geoscientific investigations and experiments, for example concerning nuclide transport in groundwater. Various in situ tests, large-scale demonstration tests and testing of engineering and execution techniques for final repositories were to be carried out in the rock laboratory. The plans resulted in the Äspö Hard Rock Laboratory (HRL), which was put into regular operation in 1995 and which has been an important part of SKB's work on design, construction and operation of final repositories. After three decades of research and development at Äspö, the original goals of the rock laboratory have been fulfilled. The Äspö HRL as a facility is described in greater detail in Section 5.4.1.

4.10.1 Ongoing long-term experiments

It will be possible to conclude the field work involving long-term experiments still being conducted under active surveillance in the RD&D period.

Prototype repository

The single largest remaining task in the rock laboratory concerns the excavation of the prototype repository, a full-scale experiment installed at a depth of 450 metres. It contains canisters and buffers with a similar reference design to the planned final repository, together with obsolete versions of backfill and plugs. The experimental tunnel originally consisted of two sections, an outer section with two deposition holes that were retrieved in the early 2010s, and an inner section with four deposition holes that are planned to be excavated during the RD&D period. Plans for excavation of the outer section and a summary of results (Svemar et al. 2016) have been presented in previous RD&D programmes.

The canisters in the prototype repository are equipped with heaters to simulate decay heat from the spent nuclear fuel. The experiment has been designed to be able to study coupled processes in a way that is not possible in more specialised trials, and the large scale means that boundary and upscaling effects are relevant. When the experiment was installed, during the period 2001–2003, final disposal and installation techniques could be tested under conditions approximating real-world conditions. This meant that SKB was also able to show that it is possible to build the final repository. According to the original plans, the prototype repository was intended to be in operation for up to 20 years so that the results of the experiment could serve as a basis for the Spent Fuel Repository's operating permit. Figure 4-1 shows photos from the installation, illustrating the size of the components and installation equipment.

Figure 4-1. *Photos from the installation of the prototype repository in the early 2000s. On the left, a canister placed in a deposition hole. On the right, a buffer block to be placed in a deposition hole.*

The remaining inner section consists of four deposition holes with canisters and a total of about 80 tonnes of buffer. The deposition tunnel is 40 metres long and backfilled with more than 1 000 tonnes of backfill of type 30/70 (proportion of bentonite/crushed rock). Excavation and sampling will continue for almost two years and will be followed by an analysis and modelling phase. Since the current reference design is based on a different material being used as backfill in the deposition tunnels, a pure bentonite product, less importance will be placed on evaluating the status of the backfill. Since the backfill is part of the integrated system, some efforts will nevertheless be made to understand how it has developed and become integrated with other parts of the system (e.g. as a mechanical constraint to the swelling buffer).

One of the overall goals of the experiment is to confirm that the system works as intended and that the development can be understood with the models and tools used in modelling of post-closure safety. This work also includes understanding the differences between the installed system (which is based on e.g. 30/70 backfill) and the current reference design.

In order to best support the reasoning in a coming SAR, the execution and reporting of the analysis work will to a large extent be structured in a similar way to relevant parts of the reference development in the assessment of post-closure safety in the PSAR.

MiniCan

The MiniCan experiment was installed in 2007 and contains five miniature canisters with a length of 31 centimetres and a diameter of 14.5 centimetres. They have been installed in virtually horizontal boreholes at a depth of 450 metres in the rock. The copper quality used for the canisters is the same as for the canisters for the Spent Fuel Repository. Four of the five canisters have been assembled in packages together with bentonite clay. In three of the packages, the clay has a low density, while the fourth contains high-density bentonite. The variation in the occurrence and density of the clay makes it possible to study specifically the function of the clay for the canister. With the aid of measuring instruments and regular water sampling, a number of parameters have been measured during the course of the experiment, for example groundwater chemistry, microbiology, redox potential, pH, corrosion rates and tensile stresses in the canister material.

Two experimental canisters from the MiniCan experiment still remain. These will be retrieved during the RD&D period, after more than 15 years. The canisters that remain were surrounded with low compacted bentonite and are most similar to the canister that was retrieved in 2010.

Long-term testing of buffer materials

The radioactive decay of the fuel will increase the temperature in the repository and this, together with uptake of groundwater, is expected to cause minor mineralogical changes in the bentonite. The purpose of the long-term test of buffer materials (LOT) is to identify and quantify mineralogical changes in the bentonite that may occur in a under repository-like environment. Furthermore, related processes in the bentonite with respect to copper corrosion, diffusion of cations, and survival and activity of bacteria are being investigated. A total of seven test packages were installed, containing a central copper tube around four metres in length, surrounded by cylinder rings of compacted bentonite. An electric heater inside the copper tube is used to simulate the decay heat from the spent nuclear fuel. Temperature, total pressure, water pressure and humidity are measured via sensors placed in the bentonite.

Three of the test packages (S1 to S3) have been exposed to typical KBS-3 conditions with a maximum temperature of less than 100 °C, and four test packages (A0 to A3) have been exposed to particularly unfavourable conditions, above all elevated temperature to a maximum of about 140 °C. Six of the packages have been retrieved and the bentonite clay has been analysed thoroughly, as described in a number of reports. Most recently, two packages, S2 and A3, were retrieved in 2019 and assessed, e.g. by Johansson et al. (2020). Package S3 remains and will be excavated during the RD&D period, after the experiment has been in progress for about 25 years.

Alternative buffer materials

The Alternative Buffer Materials (ABM) experiments are, like the LOT tests, designed to study the long-term stability of the bentonite buffer. The biggest differences are that in ABM, a large number of different bentonite clays are included (instead of only MX80) and the heater is made of steel instead of copper. Otherwise, the design is very similar to LOT, and the target temperature in ABM has been around 130 °C, with the exception of ABM5, which was intentionally much hotter. The ABM experiments will be completed during the RD&D period and are described in more detail in Section 10.3.5.

Concrete and Clay

The first experimental packages in the Concrete and Clay project were installed in the Äspö HRL in 2010 (Mårtensson 2015), and since then additional test specimens have been installed. Several retrievals and analyses have been carried out. The main purpose of the project has been to study interactions between the different types of materials that can occur in final repositories for low- and intermediate-level waste under realistic repository conditions. This includes studies of interactions between both different barrier materials and between barriers and material samples representative of low- and intermediate-level waste.

Implementation of this project is justified by the need for studies under relevant repository conditions to complement the often accelerated laboratory studies, which comprise the majority of the data used in the safety assessment work. Results from analysis of retrieved samples have also served as a basis for the design of supplementary laboratory experiments for increased process understanding. Retrieval of the last remaining test specimens is planned during the RD&D period. Sub-projects that form part of the Concrete and Clay project are described in more detail in Section 9.1.3.

Long-term development of backfill material for the Final Repository for Short-lived Radioactive Waste (SFR)

A long-term experiment is under way with macadam backfill in conditions imitating SFR for the purpose of studying how its properties are affected by microbial growth over time. The experiment has been runningsince 2020 and will be completed during the RD&D period – for a more detailed description, see Section 9.1.7.

4.10.2 New field experiments

A small number of new and shorter duration tests will be carried out in the rock laboratory during the RD&D period. They concern the early evolution of the buffer in the deposition hole under the influence of groundwater inflow and heat from a full-scale canister, as well as further evaluations of conceptual installation techniques for buffers and backfill.

4.11 Monitoring during construction and operation

4.11.1 Monitoring programme

SKB will develop a monitoring programme prior to construction of the Spent Fuel Repository. The programme aims to provide a comprehensive picture of SKB ‘s future monitoring activities during construction and operation of the final repositories. It describes which data will be collected and what they will be used for. The information that is collected will provide a basis for:

- the assessment of post-closure safety,
- the design and construction of the final repository,
- inspections of the external environment.

In terms of the supporting data for the second and third points, SKB has previously conducted studies in this area. In the monitoring programme, this work will be summarised and the relationship with other types of monitoring will be described.

The post-closure safety assessment is based on an understanding of the thermal, hydraulic, mechanical, chemical and biological processes that control the evolution of the final repositories. It serves as a basis for demonstrating the repository's ability to contain or retard the dispersion of radioactive elements and for verifying that the installed barriers, tunnels and rock caverns meet the technical design requirements.

The data that SKB has built up through decades of research will be supplemented with results from a detailed site investigation programme adapted to the specific conditions at the selected site. Methods and procedures will be developed to ensure that technical design requirements are met. This will serve as a tool for identifying manufacturing or installation errors and other deviations in materials, equipment or handling. The monitoring will help to:

- verify SKB's understanding of the evolution of the repository,
- support assumptions made in the post-closure safety assessment,
- identify any previously unknown processes and events.

The safety of the final repositories is based on passive barriers, and monitoring must not adversely affect these. The installation of monitoring equipment in a barrier may involve a risk for post-closure safety. This limits the choice of technology, location and time frames for conducting the monitoring. The risk for loss of signal or incorrect signals from sensors in the engineered barriers is also a reason why they will not be used. Incorrect signals could lead to unfounded decisions on measures associated with high costs and radiological risks.

There are other possibilities for monitoring that give relevant information on the evolution of the barriers at the repository site without jeopardising safety. One such possibility under consideration is the installation of long-term tests down in the rock in the Spent Fuel Repository at representative locations in the repository. The focus would be on the most important aspects of the engineered barriers, and experiments can be excavated and evaluated during the operating period and prior to closure in order to provide a basis for and confidence in the decision to close and seal the repository.

Monitoring programmes for the Spent Fuel Repository will be prepared and submitted to SSM as a basis for the application to begin construction. Parameters and experiments that are suitable for monitoring will be identified and their relevance for post-closure safety will be explained. In addition, qualitative descriptions of anticipated development must be prepared. A rationale for the type of measures that may be adopted to handle any situations where results deviate from expectations will be presented. Monitoring to support the post-closure safety assessment is planned in the following areas:

- Hydrology.
- Groundwater chemistry.
- Mechanical and thermal behaviour of the rock.
- Cementitious materials, clay barriers and closure.
- Copper corrosion.

4.11.2 International development

SKB also follows international monitoring efforts, and has been and remains committed to this through participation in the EU project Modern2020 and subsequent sub-project MODATS, which is being pursued within the European Joint Programme on Radioactive Waste Management (Eurad, https://www.ejp-eurad.EU/).

Modern2020, which was completed in 2019, had several main objectives. One was to refine methods and provide guidance for the development of monitoring programmes, including technical design requirement, strategies for monitoring and a process for decision-making based on monitoring data. Another purpose was to research and develop monitoring technology in terms of sensors, signal and energy transmission, as well as non-intrusive monitoring methods. Another objective was to refine various methods for effective dialogue with the public and local stakeholders at an early stage of the development of monitoring systems. These objectives were achieved to some extent and generated data for, among other things, continued work in e.g. MODATS.

The ambition of the MODATS project is to consolidate strategies for the implementation of monitoring systems by developing methods that can support confidence in the collected data and thereby benefit the implementation of the final repositories. In order to develop this ambition, research and development is carried out with respect to the acquisition, management and reporting of data, as well as its utilisation as a basis for understanding of the system. In addition, research and development is also being conducted concerning new technology for monitoring engineered barriers.

4.12 Decommissioning

Decommissioning of nuclear power plants is under way or planned at most of the Swedish nuclear power plants. Areas requiring development are largely the same for all the licensees, and relate more to being able to establish requirements in respect of waste management and management routes than to fundamental research and pure technology development, although adjustments of available technologies will be required. Challenges regarding waste management include preparing and developing acceptance criteria for future final repositories, gaining good knowledge of the waste that is generated with regard to the content of radionuclides and materials, and establishing approved waste containers for long-lived waste for the planned final repository. Issues relating to waste and waste management are described in more detail in Chapter 6. Part III describes planned activities linked to decommissioning and decommissioning planning.

4.13 Other topics

SKB is following developments in a few other areas that are not directly linked to the development of SKB's final repository, but that are of interest for the operations. Such areas include issues concerning preservation of information and knowledge through generations and development of other methods for final disposal.

4.13.1 Preservation of information and knowledge through generations

SKB has for many years been working on issues concerning archiving and preservation of knowledge and information about the final repositories, both during their operating time, covering a few generations, and in the significantly longer term after their closure. The purpose of the work has been to try to create the best possible conditions for retaining essential information.

In its decision on a licence for the KBS-3 system in January 2022, the Government writes that the issue of knowledge and information transfer is an important part of the continued stepwise licensing process and that it forms part of the RD&D programme. The issue has been relevant throughout the licensing process for the KBS-3 system, and SKB intends to continue to follow development and present its work on the issue in the RD&D Programme.

Questions regarding preservation of information and knowledge for future generations may be considered most urgent for the Spent Fuel Repository, but also need to be considered for SFR and SFL. In purely practical terms, the solutions for preservation of information concerning the existence of the repositories do not need to be in place until a final repository is sealed, which will not take place until the end of this century. Neither SKB, regulatory authorities nor other parts of society are in a position to determine definitively today how best to proceed with something that is to take place so far into the future. However, as early as during construction of the final repositories, SKB will need to devise strategies for its own work on preservation of data and information on the repositories and their contents until closure.

In its day-to-day work and until the closure of the final repositories, SKB manages and preserves documents, data and information in accordance with external requirements set by SSM and the National Archives of Sweden. Some of these requirements entail storage with no clear end date. Pursuant to SSM 's regulations on archiving at nuclear facilities, SSMFS 2008: 38, the archives of a final repository must be handed over, in an ordered and indexed form, to national or regional archives if activity ceases. SKB takes a structured approach to the management and preservation of documents, data and information as a part of its regular work involving SARs, and this is a key valuable starting point for the future assessment and selection of which information to preserve after closure. The work is already governed by existing data and information management plans in accordance with SKB's management system, and is intended to be supplemented with procedures for sorting and marking linked to preservation status (up to or after closure). Procedures for handling of documentation regarding choice of technology, strategies and working methods are planned to be further developed prior to the start of construction of the Spent Fuel Repository. The procedures for handling physical samples and image material are also planned to be developed in a similar manner to those for documents.

SKB considers it important to have a procedure that aims to keep the issue alive, develop strategies and disseminate knowledge about the need. A prerequisite for success, besides the participation of SKB, is input from regulatory authorities and the municipalities, as well as from society in general. SKB has previously participated in the international initiatives "Preservation of Records, Knowledge and Memory across Generations" and "Assembling Alternative Futures for Heritage", which were presented in the RD&D Programme 2019. The final report from the OECD-NEA initiative RK&M was published in 2019 (OECD NEA 2019) and pointed to a preservation strategy involving several different methods that operate on different time scales, consist of different media and content, use different transfer methods, involve different actors and are located at several different sites. Choosing a diversified preservation strategy is considered to provide the conditions for information and knowledge preservation across generations.

Information preservation is not only an issue for the nuclear area. The needs are also similar for other types of hazardous waste and contaminated areas, and are also a common international concern. On this basis, SKB will continue to participate in national and international forums and working groups where these issues are discussed and addressed during the operating time of the final repositories. The aim is to be aware of the current state of knowledge and to contribute to developing appropriate principles, methods and tools, as well as taking appropriate action. Lessons learned will be reported to interested parties on an ongoing basis. On the basis of this and the reporting made by SSM in the autumn of 2021 to the Government in the form of "Methods for transfer of information and knowledge about the final repository for radioactive waste" (SSM 2021d), SKB plans to continue to manage this issue. This planning will shed light on both future research needs and internal work, as well as the connection between information preservation and fuel information and the prevention of inadvertent intrusion.

In spring 2021, a two-year research project was initiated at Linköping University, partly funded by SKB, on method development and the writing of a so-called KIF (Key Information File, which was proposed in the final report from the RK&M initiative) for the Spent Fuel Repository.

Project Memory beyond generations

A feasibility study for a planned project, Memory beyond generations, was carried out in 2019. The purpose of the feasibility study was to produce a proposal for a project with the objective of proposing and implementing practical examples of solutions to the challenge of preserving knowledge and information on hazardous waste across generations through collaboration with cultural heritage sectors. Besides SKB, the participants included the Linnaeus University (the initiator and overall coordinator), Östhammar Municipality, SSM, the National Archives and others. The feasibility study was financed by Vinnova, but unfortunately no funds were provided to start the proposed project. The work on the feasibility study laid the foundations for fruitful collaboration between various national stakeholders. In spring 2022, the group gathered to look at the possibility of resuming the work and possible financing. As a result, SKB, among others, participated in an application from Linnaeus University for funding from Formas for a two-year project – Change forever: sustainable long-term memory of the final repositories for nuclear waste. The decision on any allocation of funding will be made in November 2022.

Information, Data and Knowledge Management

In 2018, three projects within the framework of NEA's Radioactive Waste Management Committee (RWMC) connected to the preservation of knowledge and information on final repositories for radioactive waste were completed. SKB actively participated in two of the projects: Radioactive Waste Repository Metadata Management (RepMet) and Records, Knowledge and Memory (RK&M). The third project was the Expert Group on Waste Inventorying and Reporting Methodology (EGIRM). NEA launched a continuation of the three projects under a common umbrella – Information, Data and Knowledge Management (IDKM), comprising four working groups: Knowledge Management, Archiving, Safety Case and Awareness Preservation.

From the start, SKB has participated in the working group Awareness Preservation, which constitutes a continuation at RK&M. The purpose of SKB's participation is to continue sharing experience of and developing ideas for practical execution of preparations to preserve knowledge and information on final repositories for spent nuclear fuel and radioactive waste and to explore proposals for solutions that were not covered before RK&M was concluded. Examples of questions suggested for in-depth exploration are ethical aspects of IDKM, the possibilities of utilising the internet across generations, and possibilities of using different types of time capsules. Since the spring of 2022, SKB has also participated in the working group on Knowledge Management.

4.13.2 Other methods for final disposal

In all the years of consultation and licensing processes relating to the encapsulation plant and the Spent Fuel Repository, questions relating to the development of other methods for final disposal, especially disposal of spent nuclear fuel in deep boreholes, have kept re-occurring. Since the RD&D Programme 2019, nothing has come to light that gives cause to re-evaluate the view that the most realistic strategy for final disposal of the spent nuclear fuel as waste is geological disposal. In the Government decision on a licence for the KBS-3 system, the Government established that alternatives to the KBS-3 method are only available on a conceptual level and that there is therefore no other method available today that could constitute Best Available Techniques (BAT). SKB is continuing to follow the development of other methods for final disposal of radioactive waste with the aim of acquiring new knowledge in order to further optimise its final repositories in the future. An example of developments that SKB intends to follow is the Swedish Scientific Drilling Program (www.ssdp.se), in order to thereby obtain data and other results that are relevant for SKB´s activities. SKB does not, however, plan to conduct any research or development in this area.

5 Procedures, competence and resources

In order to be able to manage the radioactive waste and spent nuclear fuel in a safe and cost-effective manner, SKB has developed a systematic approach to the research, development and demonstration that are needed to construct and commission new facilities. The facilities in operation are subject to the regulations of the Swedish Radiation Safety Authority (SSM), and information on the implementation of these in respect of procedures, resources and competence is not specified here. The chapter focuses on the iterative process of developing, implementing and evaluating final repositories for radioactive waste, which includes research and technology development as well as evaluation of safety during operation and after closure. This chapter also gives an overview of how SKB ensures that the necessary competence, resources and tools are available.

SKB's procedures for carrying out research, development and demonstration are being developed on an ongoing basis, but the presentation in this chapter mainly follows the corresponding chapter in the RD&D Programme 2019. Management of organisation, resources and competence in connection with the decommissioning of the nuclear power reactors is handled mainly by the reactor owners and is described in Section 14.4.

5.1 Role of the RD&D programme for openness and transparency

The periodicity and content of the RD&D programme are subject to the regulations of the Nuclear Activities Act. The RD&D Programme 2022 is the 17th of its kind, and SKB can conclude that the RD&D process is an important part of the continuous build-up of knowledge and the choices on development of final repository systems that are progressively being made. Universities and university colleges, environmental groups, municipalities and society in general follow the RD&D process and have thereby influenced the development of the nuclear waste programme over the decades. The RD&D programme, and the adjacent process, have contributed to openness and insight into research and development issues for the final repository systems that the licensees are responsible for implementing, through SKB. This applies to the state of knowledge in general as well as to plans and programmes. Regarding the RD&D Programme 2019, the Government also notes that "...SKB provides a clear overview and understanding of the plans of the company and the reactor owners, and permits the openness and insight into plans and programmes that the legislation is aiming for."

Some of the original conditions for the RD&D process have changed since the Nuclear Activities Act was introduced. In the most recent RD&D programmes, SKB has had the ambition to avoid duplicate reporting and not anticipate comments on SKB's applications relating to the ongoing licensing processes for KBS-3 and the extension of SFR, in conjunction with the review and evaluation of the programmes. It is SKB's view that the focus will shift and the scope of the RD&D programme will be reduced when planned facilities transition to licensed facilities under the supervision of SSM. At the same time, the report should include an overview of all activities and operations necessary to provide a sufficient understanding of the whole.

The Government is also providing feedback regarding the RD&D Programme 2019 with the comment that "The reactor owners and SKB should take into account how the RD&D programme can better contribute to openness and insight into how the work on research, technology development and demonstration of methods for management and final disposal of nuclear waste is being conducted."

In the RD&D Programme 2022, SKB presents a more system-wide perspective in order to clarify relationships between different activities and milestones for the different facilities and also the need for continued research and development to be related to future milestones. In this context, SKB also wants to specify the process that is being carried out in conjunction with the submission of the RD&D Programme and that is being progressively developed. In addition to reporting at a major seminar for review bodies of the RD&D Programme 2022, arranged by SSM, SKB is conducting a dialogue with the review group within SSM. Information to, and dialogue with, the municipalities about the programme is also carried out in conjunction with publication. SKB also intends to arrange an evening on the topic of RD&D for the general public and interested parties in the municipalities concerned.

SKB continuously publishes relevant results from research, technology development and demonstration in scientific and other journals and encourages its own personnel, research institutions and consultants with whom SKB collaborates to publish. SKB's results are also presented and discussed openly at scientific conferences and presented in the conference proceeding. International collaboration has been and remains of particular importance for testing the results of experiments and analyses in a constructive technical and scientific spirit. The exchange is also a recurring source of new ideas and suggestions for continued initiatives.

SKB is also working to disseminate research results outside the scientific community – see Section 5.2.3.

5.2 Research

The objective of SKB's research programme is to ensure that the knowledge needed to design, locate, obtain licences for, plan, build, operate, decommission and seal planned facilities and to maintain safe operation of SKB's existing facilities is available. This means that the research should:

- provide sufficient knowledge of post-closure safety and ensure that safety can be assessed for SKB's existing and planned facilities also in the future,
- provide sufficient data for the continued technology development and planning that is needed in order to achieve efficient and optimised solutions that at the same time ensure safety both during operation and after closure of SKB's final repository.

5.2.1 Management of research

The focus of SKB's research programme is established in the RD&D programmes which, in accordance with the Nuclear Activities Act, are submitted to the Swedish Radiation Safety Authority (SSM) every three years. SKB has a research council with representatives from different parts of the organisation to support the research and assist in prioritising the research.

In connection with the annual planning of activities, the persons responsible for the different disciplines included in the safety assessments conduct a review of new, ongoing and concluded research issues, including detailed development of the plans for the coming years. This means the established research programme is subject to continuous review and follow-up. As a part of the process of updating SKB 's RD&D programme, research seminars are conducted, where SKB 's experts in different disciplines present their suggested plans, including estimated resource requirements for the coming years to the research council and other interested parties within SKB. The presentations provide the research council and SKB's project sponsor with an overview of the research needs, enabling them to make an overall assessment of the need for activities during the coming years. This assessment then serves as a basis for planning of activities and provides key supporting material for the work on coming RD&D programmes.

5.2.2 Future research focus

The purpose of the repositories is to protect human health and the environment from radiological harmful effects of the waste in the long term. Post-closure safety is therefore essential for the design, siting, construction and operation of the repositories. Formally, the question of whether a repository has an acceptable level of safety is determined by authorities reviewing SKB 's assessments of post-closure safety. All SKB 's work in this field has, however, emerged from the fact that SKB itself must be convinced that the proposed repository concepts are safe in the long term. SKB 's research programme is therefore largely based on the need to assess post-closure safety of the repositories. An important part of each post-closure safety assessment is the evaluation of the state of knowledge with regard to both processes and input data in the assessment. Such evaluations have been included in the most recent safety assessments SR-Site for the Spent Fuel Repository and SR-PSU for the extension of the SFR, as well as the safety evaluation for the SFL. The results of these evaluations and the review of SKB 's licence applications constitute the basis for the research initiatives that are planned in this RD&D programme.

Future research initiatives concern primarily SKB ‘s existing and planned final repositories for radioactive waste and spent nuclear fuel (SFR, SFL and the Spent Fuel Repository). With the start of construction of the new facilities, the focus will shift from research and proposed solutions to technology development of quality controlled industrialised systems. However, research initiatives and safety assessments will be needed to support technology development and in particular for the development of practical technical design requirements for the final repositories, as well as for being able to verify that the developed technical solutions fulfil these requirements.

Although extensive knowledge is now available in all the areas that are of importance for post-closure safety, it can be anticipated that new questions will arise. Questions that may require research efforts may originate in many different places:

- Internally at SKB during development of repository concepts and in conjunction with the execution of safety assessments during operation and after closure. Site investigations may also generate new questions, in line with sites increasingly being investigated in detail. New or modified waste types may also lead to new questions.
- Externally, for example, within the scientific community and in SKB’s sister organisations in other countries.
- From SSM and review bodies in connection with reviews of RD&D programmes and supporting material for licensing processes.
- Changes in external requirements.

Following research in adjacent areas that may be of importance for the Swedish nuclear waste programme in the future is also part of SKB’s responsibility. This concerns mainly the development of methods for treatment and final disposal of radioactive waste. This is primarily implemented by maintaining international contacts and by following industry publications.

5.2.3 Review, openness and transparency

SKB’s research is conducted on the basis of the requirement that the results must be correct, traceable, reproducible and relevant for SKB’s mission. To achieve this, SKB has developed and applies procedures for quality assurance of the execution of research projects and tasks. There are also special procedures for quality assurance of safety assessments, which include approval of research results, data and models to be used in the assessments – see also Section 5.5.4.

The fundamental principle is that SKB ‘s research results will be published open access to facilitate external review. Since the research programme was initiated in the 1970s, research results have been published and will continue to be published in SKB ‘s report series, which are available on SKB ‘s website. Before they are published, the reports have undergone internal and/or external review in accordance with established procedures. Quality-assured data from SKB ‘s site investigations, technology development and research are saved in databases and are available for the authorities in their review.

SKB also strives to publish relevant results in scientific journals and encourages its own personnel, as well as research institutions and consultants that SKB is collaborating with, to publish. The results then undergo an independent review through the peer review that takes place prior to publication. SKB ‘s research results are also presented and discussed at scientific conferences, and are published in the conference proceedings. From 2019 to 2021, SKB’s employees and researchers funded by SKB published almost 100 scientific articles and more than 20 other publications (books and conference contributions).

In addition to publishing or funding research itself, SKB has shared data with other researchers or institutes who have requested it for their research. Examples of this are data from SKB’s research on Greenland, which are available in open databases, chemistry data from the Forsmark area, which has been shared with SMHI and with Svealand’s coastal water conservation association (Svealands kustvattenvårdsförbund), and material samples of copper and bentonite, which have been shared with other research teams.

SKB is also working to disseminate research results outside the scientific world, for example by publishing in popular scientific journals, publishing on SKB's website and in SKB's magazine Lagerbladet, conducting themed evenings in Östhammar and Oskarshamn municipalities, and providing information at schools and universities. Employees at SKB also participate in various industry days to obtain and provide information within their specialist areas (for example rock and concrete).

5.3 Technology development

The objective of technology development is to make sure that the processes, systems and equipment needed to manage and dispose of the radioactive waste and the spent nuclear fuel are available when the facilities are commissioned. Management and final disposal of nuclear waste must take place in a regulated, controlled and rational manner while the requirements for post-closure safety, low radiation dose during operation of the facilities and limited environmental and climate impact are met.

Technology development concerns both physical objects – for example, in the case of the Spent Fuel Repository, the spent nuclear fuel, engineered barriers, plugs in deposition tunnels and underground openings, and production lines with associated process steps. The process steps, in turn, generally include some form of production process and some form of measurement process to facilitate quality assurance of the outcome of the production process. Technology development also includes describing how all parts interact with the surroundings, i.e. people, technology and organisation.

5.3.1 Assessment of technology maturity

Technology readiness levels (TRL) is a procedure used to assess the degree of maturity in each area of technology. The TRL methodology is based on a scale from 1 to 9 and is used to support the planning and prioritisation of development projects. TRL1 means that basic functional needs and possible technical solutions have been identified. Basic technology development continues up and including level TRL6. The work of realising and validating the technology includes TRL7-8, and at TRL9 construction and trial operation has been carried out according to requirements and expectations.

The TRL assessment is performed by a group with overall expertise within the technical area in question. The responsible technical sponsor determines the assessment.

5.3.2 Management of technology development

Management of technology development is based on a strategic technology development plan. Plans of action are linked to this, with the purpose of clarifying and justifying which technology development is needed for each final repository system, production line and technology area right up to commissioning. The plans of action are based on what needs to be finished in time for the milestones identified for the various construction projects described in Chapter 3.

The plans of action describe the route to completion of the fundamental technology development (confirmed reference design, TRL6). The subsequent realisation of selected technology and methods (further development/optimisation, design and implementation, TRL7) and finally the adapted validation activities (TRL8) that need to be carried out during construction (of underground openings and concrete barriers) or prior to trial operation of all production and measurement processes is controlled via programme plans in accordance with the management system.

Validation (TRL9) is considered to be completed in conjunction with approval of the SAR for regular operation and the trial operation phase is thereby concluded.

Technology development is commissioned from the functions in SKB that are responsible for construction of the new facilities. The scope of the work is governed overall by how well-proven a technology, method or equipment is, i.e. how standardised it is.

Technology development and RD&D programmes

As part of the preparation of the RD&D programme, a detailed review is performed of the current status of the technology development and research portfolios. The new RD&D documentation is reconciled and calibrated against management plans, the investment plan and the main time plan for the future facilities.

RD&D programmes and management plans are presented to the company management and SKB's Board of Directors for decisions. The plans are then harmonised annually between the construction projects, the sponsors of technology development and the functions within SKB that carry out the actual development work.

5.3.3 Technology development process

SKB applies an overall systematic approach to development of production lines, including technical equipment and methods that are used and which are based on SKB's management system. The goal of the systematic approach is to create the conditions for identifying, planning and executing all the detailed activities and measures required to be able to operate the completed final repository in such a way that all applicable requirements in respect of the engineered barriers and underground openings, including those related to post-closure safety, are met.

The process for management of technology development starts with ideas of different concepts and ends with commissioning of systems and methods in the facility in question. It is divided into the following phases:

- Develop concept proposals (set of requirements and functionality).
- Develop concept description (details, simulation and validation in laboratory environment).
- Develop reference design (demonstration and evaluation).
- Realise technical solution (detailed design, optimisation and implementation).
- Validation and qualification (interoperability of processes).
- Verification of compliance with requirements.

Initially, development takes place with a strong focus on post-closure safety, since such compliance is the main goal of the development. Other requirement areas also need to be considered, however, in order to ensure a robust facility that can be operated efficiently while producing an acceptable final repository. Examples of other areas of requirement are a good safety and working environment during operation, maintenance aspects, adaptation to human ability, low impact on the external environment and reasonable investment and life cycle costs.

For each development phase for each product or process, there is overall control over what is to be achieved and thus what supporting data should be available before a decision is made to proceed with the next phase of development.

This process involves an overall approach to what should be done in each phase and what should be delivered. Implementation of technology development (independent of phase and whether the work spans one or more phases) is described in SKB's management system for management of tasks or projects.

Project scope is determined on a case by case basis. Normally, a project does not extend over several phases, but is limited to a specific technology development phase. In order to coordinate the development activities, for example for a facility, the projects can be governed and coordinated by a programme that normally extends over several phases. Technology development is not an independent process. Decisions regarding technology take the planned facilities' limitations into account in order to achieve well-functioning production lines.

5.3.4 Technology development and technical design

The technology development process is related to SKB's design management model, which is described in the management system and is applied in the design of new facilities. The design management model covers the following principal phases:

- Facility configuration.
- Facility design.
- System design.
- Detailed design.
- Installation.
- Final documentation.

An important relationship between the technology development process and the design management model is that TRL6 (confirmed reference design) coincides with the conclusion of the systems design phase, i.e. delivery of the detailed design documentation. Technology realisation (TRL7-8) is then carried out, including testing of installation, implementation of verification and validation according to the design management model, with the addition of qualification and certain validating integration tests.

5.3.5 Quality assurance, control and inspection

An important objective of technology development is that it should be possible to verify that the developed technical solutions meet the requirements that have been set. Quality assurance, control and inspection refer to measures that need to be taken in order to ensure and instil trust that the requirements set for the facilities during operation and after closure of the disposal facilities and completed final repositories are fulfilled. The results obtained must conform to acceptable values for properties that contribute to safety and radiation protection.

SKB applies a systematic requirement management governed by the management system. Statutes, licences, conditions and internal requirements are identified and broken down into requirements for facility, system and component so that they can serve as a basis for verification and validation. This applies to both individual requirements and to the coordination of requirements for a holistic approach. Requirements that are applied must be relevant, documented, communicated and traceable during all phases of technology development for the design, construction and operation of future facilities and after their decommissioning and closure. Changed or new requirements may arise due to changes to regulations or changes to internal requirements. The implementation of new or changed requirements requires an assessment to be made to ensure that the change will not have a negative impact on operation or on post-closure safety of the final repositories, even if the change is not related to radiation safety.

The technical solution that is established in the development work should result in the production of a final product that conforms to the established design. Before production can begin, the manufacturing processes and testing methods that SKB intends to employ must be proved to be stable. The ability of measurement and production systems to achieve and perform quality assurance of physical objects that comply with specifications is demonstrated in qualifications. In order to describe how quality is achieved and ensured, a systematic approach and traceability are needed in how SKB confirms that the production of final repositories, and the parts included in these, achieve sufficient quality. Reporting and approval of the results from technology development, including verification and validation activities, is synonymous with a qualification of production lines.

Qualification of each measurement and production system is adapted to the manufactured or tested component's importance for post-closure safety, available proven technology, available standards and norms, and the conditions that will prevail, both physical and organisational, in the execution of the planned production. This means that each qualification is unique; some, in principle, only refer to the standards and norms to be employed, while others require extensive analyses and demonstrations. SKB's continued efforts to establish principles for the qualification process will be presented in the preliminary safety analysis reports (PSAR) for Clink and the Spent Fuel Repository and their descriptions of the production lines.

Testing and inspections will also be carried out during construction of the facilities to confirm that construction is functioning as intended and to ensure that no errors or deviations remain with significance for post-closure safety. Inspections are incorporated in inspection programmes, the design and contents of which are dependent on the types of requirements that will be verified, for example inspection programmes for external environment, programmes for investigations of rock, programmes for inspection of rock excavations and programmes for inspection of work environment.

5.4 SKB's facilities for research, development and demonstration

SKB's facilities for research, development and demonstration include the Äspö HRL, which consists of an underground facility and three facilities above ground, and the Canister Laboratory. SKB plans to finnish the experiments in the underground facility of the Äspö HRL during the current RD&D period. The surface facilities of the Äspö HRL will continued to be needed into the 2030s.

5.4.1 Äspö HRL

The activities at the Äspö HRL, which was built between 1990 and 1995, are a continuation of the work that was previously carried out at the Stripa Mine in Bergslagen. The laboratory is situated on the island of Äspö, north of the Oskarshamn nuclear power plant. The underground facility consists of a tunnel from the Simpevarp peninsula, where the Oskarshamn nuclear power plant is located, to the southern part of Äspö. Under Äspö, the main tunnel descends in two spiral turns down 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 laboratory is shown in Figure 5-1 and current experiments are presented in the Äspö HRL annual report (SKB TR-21-10).

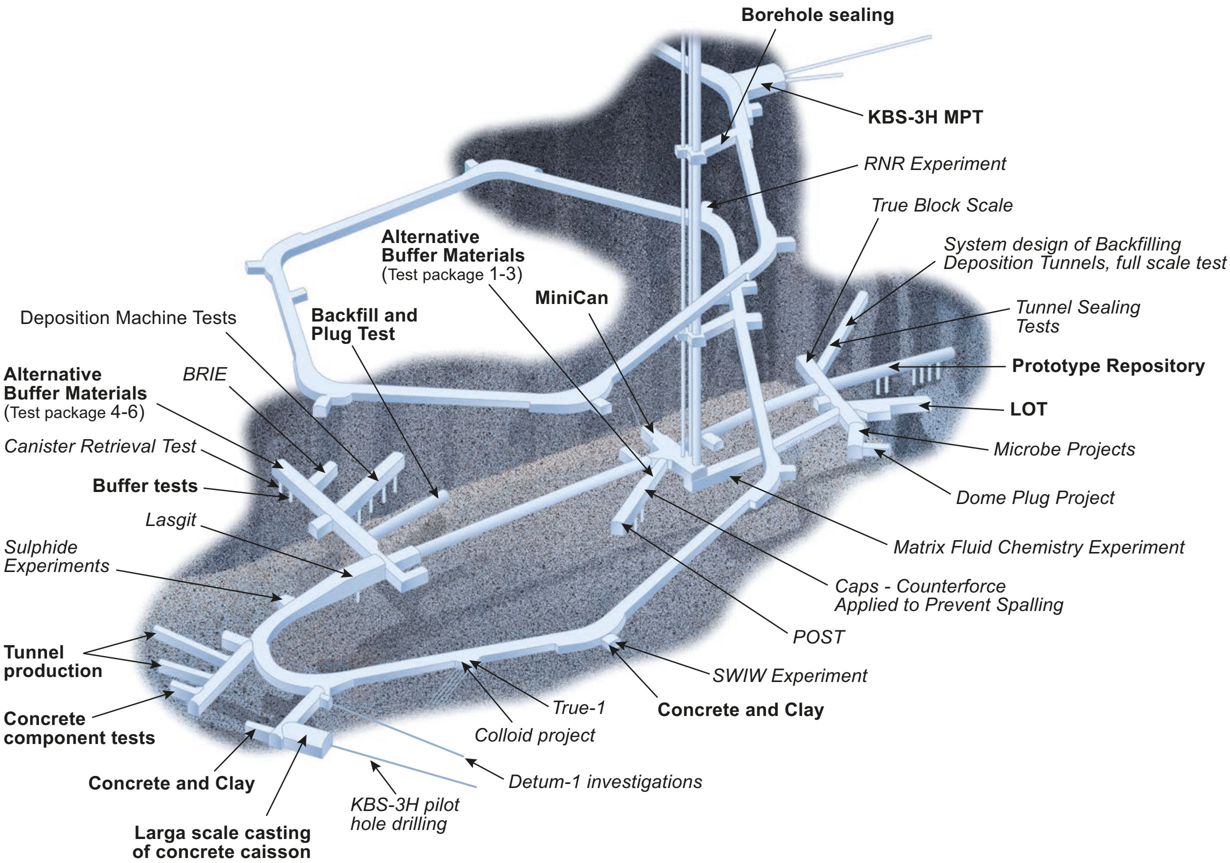


Figure 5-1. Äspö Hard Rock Laboratory (HRL) with ongoing (in bold) and completed (italics) experiments.

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 and for execution of investigations during ongoing construction. This experience will be of benefit for the coming detailed site investigations for the Spent Fuel Repository, and the extension of SFR in Forsmark, as well as for the siting, design and construction of SFL.

The properties of the rock and the hydrochemical processes that take place in the rock were studied in depth during the construction of the facility and the first decade that the laboratory was in operation. Results and knowledge from these activities have served as a basis for defining the (safety-related) function of the rock in relation to the other barriers.

Following the start of operations in 1995, experiments gradually began, involving investigation of how the barriers and other components of the Spent Fuel Repository (canister, buffer, backfill and closure) can be designed and installed in order to provide optimal functionality. Another important purpose has been to develop and demonstrate methods for constructing and operating the Spent Fuel Repository. Tests have been carried out on almost all of the KBS-3 method's subsystems in a realistic setting, a number of them full-scale. The results from several of these experiments comprised important supporting data for SKB's application for the KBS-3 system. The Äspö HRL will continue to play an important role in the development of the KBS-3 system, for example through the ongoing long-term tests that will be completed in the coming years. Prior to the future extension of SFR and construction of SFL, experiments have been carried out connected to the evolution of structural concrete and other cementitious materials and the technology for construction of the barrier structures in SFR and SFL.

Today and in the coming years, activities at the Äspö HRL will be focused on excavation and evaluations of ongoing long-term experiments and continued optimisation of the engineered barriers, where the focus of technology development will be testing and inspections of clay barriers, prototype equipment and systems for application in the Spent Fuel Repository. See also Section 4.10.1 for a description of the long-term tests that are still in progress.

After three decades of research and development at Äspö, the original objectives of the rock laboratory (underground facility) have been achieved. The long-term tests that are still in progress will be able to be concluded in 2024. At this time, SKB estimates that remaining underground tests and experiments can wait and be carried out in conjunction with the construction of the Spent Fuel Repository. SKB has for several years actively attempted to find actors who want to carry on managing the Äspö HRL as an open test and demonstration environment. In the absence of sufficient business interest and social benefit, this initiative has now been closed. SKB plans to initiate decommissioning of the rock laboratory during 2025, but still look after interest in alternative uses of the rock volume, for example as energy storage. The surface facilities of the Äspö HRL – the Water Chemistry laboratory, the Materials Research Laboratory, the Multi-purpose test facility and offices and storage facilities – are needed by SKB for use beyond 2030 for continued research and development.

Water Chemistry Laboratory

The water chemistry laboratory on Äspö is accredited to ISO17025 for analysis of the chemical components in groundwater that are of particular importance for the performance of the final repositories after closure. During the site investigation phase, the laboratory was responsible for the handling of all analyses and result summaries for the site investigations in both Forsmark and Laxemar. The laboratory's combined competence has been used for the establishment and accreditation of the corresponding laboratory in Forsmark.

Materials Research Laboratory

SKB also operates laboratory activities that focus on research regarding the physical and chemical properties of clay materials, mainly with respect to issues of importance for ongoing and future safety assessments. The laboratory also develops standardised testing and investigation methods that will be used for inspection of bentonite deliveries during the operational phase. Methods for analysis of tracer elements in copper materials will also be established in the coming years. The laboratory is currently housed in the same building as the water chemistry laboratory on Äspö, but the activities may be relocated in the future, based on SKB 's needs.

At the beginning of 2022, two new instruments for analysis of copper were installed. One analyses the ppm levels of hydrogen and oxygen and the other the level of other impurities, such as zinc, bismuth and silver.

Multi-purpose test facility

Since 2007, SKB has been conducting research and development in what is now called the Multi-purpose test facility, which is located above ground directly adjacent to the Äspö HRL. The experiments being conducted in the Multi-purpose test facility complement the experiments being conducted underground and in the other laboratories on Äspö.

In the Spent Fuel Repository, copper canisters are surrounded by highly compacted bentonite. Bentonite also surrounds the silo in SFR and is planned to be used as a barrier in SFL. Bentonite will also be used for backfilling of the tunnels in the final repositories. In the Multi-purpose test facility, SKB conducts studies of the properties of bentonite, for example by simulating water conditions in a controlled manner. Installation methods are also being developed for backfilling of the repository's tunnels with backfill material and for building plugs to seal the deposition tunnels and boreholes.

The investigations performed in the Multi-purpose test facility are often preparatory tests of varying scale and scope, in preparation for full-scale tests at repository level in the Äspö HRL. The Multi-purpose test facility also has equipment and space for receiving bentonite deliveries and for mixing bentonite to the desired water content.

5.4.2 Canister Laboratory

The Canister Laboratory, which is situated in the harbour area at Oskarshamn, was built between 1996 and 1998. Among other things, the Canister Laboratory tests and develops the technology for welding the bottom of the copper tubes and for sealing the lids on the canisters. The different parts of the canister, before and after welding, are also inspected and tested here. Most experiments are performed at full scale.

The development of manufacturing methods for all the canister components is led by SKB, after which the manufacturing tests are conducted by external suppliers. Subsequent investigation and evaluation largely takes place at the Canister Laboratory. The goal is to develop methods for manufacturing and techniques for inspection and testing. These methods and techniques must meet established quality requirements and be sufficiently reliable for use in canister production and in Clink. Key equipment in the laboratory includes a system for friction welding with rotating tools, equipment for non-destructive testing including an X-ray system, ultrasonic testing of the friction weld, eddy-current measurement, tensile testing equipment, instruments for chemical analysis of the composition of copper components, light optical and electron microscopes and a management system for full-size canisters. Figure 5-2 shows the equipment for friction welding.

5.5 IT tools

Execution of the research, development and design needed to decommission the nuclear power plants and dispose of the nuclear waste requires access to powerful and relevant IT tools. SKB and the reactor owners have developed or acquired a set of such tools. This section provides a brief overview of the management and development of the databases, modelling and computational tools, as well as investigation methods and instruments for site modelling that are used.

Because all nuclear power companies are subject to the same overall legislative and regulatory requirements, there are good opportunities for coordinating IT solutions, as well as forums for joint coordination of planning, strategic issues, interpretation of requirements etc.

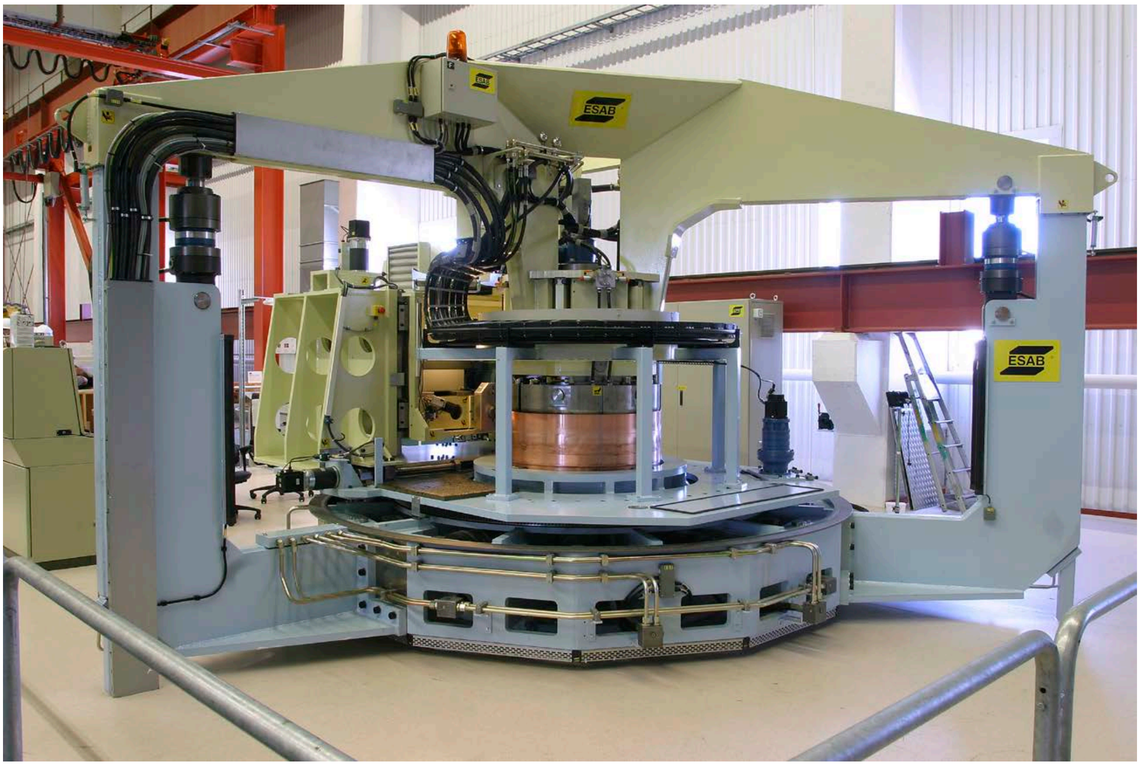


Figure 5-2. *Equipment for friction welding.*

5.5.1 Databases

Management of radioactive waste and spent nuclear fuel entails management of large quantities of data that is collected and structured in databases. The IT field is developing rapidly, which means that tools and databases are added, managed and replaced on an ongoing basis in order to keep up with development in terms of both security and functionality. Databases are managed and integrated with systems in SKB's operating activities, research and development as well as with the construction programmes and projects that are now being carried out and planned in the future.

SKB uses the databases Dark and PlutoWeb for spent nuclear fuel and long-lived waste that is kept in interim storage in Clab, and Gadd (shared waste management database) for all low- and intermediate-level waste disposed of in SFR. The Gadd database has been developed in stages and now includes nuclear waste from the nuclear facilities that deliver waste to SFR. For research, and in order to conduct safety assessments, a number of databases with data obtained from SKB's own research experiments are used, but sources of data also include public sources, e.g. with radionuclide data and thermodynamic data. Data from e.g. site investigations and from analyses conducted at SKB's various laboratories are stored in the Sicada database, and a GIS database is used for geographic information management and modelling. Requirements identified and applied according to SKB's systematic processing (Section 5.3.5) are documented and followed up in the Doors NG database.

Several applications and system solutions are applied to support different operations within the company. One example of the applications used to document the design of the facilities is Autodesk Autocad. There are also databases of a more administrative character, such as Bibas, which is SKB's library database, and SKBdoc, which is a document management system.

5.5.2 Modelling and computational tools

Models and computations are a key part of the work on design and on evaluation and assessment of safety during operation and after closure. In order to be able to carry out all the analyses and calculations that are required to manage the radioactive waste and the spent nuclear fuel, SKB uses both modelling and computational tools developed in-house and commercial tools that have been adapted for application by SKB as needed.

To perform the assessments of post-closure safety for the Spent Fuel Repository (SR-Site) and for the extended SFR (SR-PSU), a large number of modelling and computational tools (SKB TR-10-51, TR-14-11) are used. The computational tools include both commercial software with hundreds of thousands of users, and software specially developed for safety assessments with perhaps only a few dozen active users and developers. To be able to use the software as a part of the safety assessments, SKB has established quality requirements that the computational tools must meet. According to these requirements (Chapter 2 in SKB TR-10-51), there must be documentation confirming that:

- the software is suitable for its task in the safety assessment,
- the software has been developed in an appropriate manner, and the calculations give correct results,
- the software has been used in a correct manner and there is a description of how data have been transferred between different calculation tasks.

The data used in the calculations comes mainly from SKB's databases or commercially available databases (see Section 5.5.1).

Modelling and computational tools are maintained and upgraded on a continuous basis in line with the general development of computers and operating systems. SKB's computation capacity is also continuously being improved to meet the increasing calculation and data storage needs.

SKB strives to have its own competence in respect of all software used in safety assessments, and is conducting development work to optimise some of the commercial tools for the needs of the safety assessments. Our own computational tools also undergo continuous development.

Models relevant to SKB's safety assessment activities are stored in a proprietary model database. The database enables version management and preservation of used models used in previously performed safety assessments.

5.5.3 Investigation methods and instruments for site modelling

In order to gather the data that are needed for modelling and evaluation of the safety of a final repository, SKB has, in many cases, together with sister organisations and cooperation partners, developed special investigation methods and instruments, which are being further developed as necessary. SKB manages a set of measuring instruments for site investigations. These instruments are kept in an instrument storage building located adjacent to the Canister Laboratory in Oskarshamn. From the storage building, a one-kilometre deep borehole has been drilled, which is used for testing and calibration of borehole instruments.

A site descriptive model is an integrated description of multiple scientific fields. The models constitute a compilation of measurement data, conceptual models, structural geological models, surface ecological models and quantified descriptions of the site's hydrogeological and hydrogeochemical evolution up to the present. A site descriptive model is one of the cornerstones for repository design, environmental impact assessments and for post-closure safety assessment for a repository. SKB has developed site descriptive models for Forsmark (SKB TR-08-05, TR-11-04) and Laxemar (SKB TR-09-01).

By selecting Forsmark as the site for the final repositories, both for spent nuclear fuel and for short-lived, low- and intermediate-level radioactive waste, it is today only deemed justified to maintain discipline-specific models for Forsmark. An update of the site descriptive model for Laxemar may be needed as part of the work of selecting a site for SFL.

Since the site investigations were concluded, a monitoring programme has been conducted in Forsmark which involves collecting data on groundwater pressure, groundwater composition, seismic events, precipitation, temperature, development of ecosystems etc. This means that new data that can be used for updating the site-descriptive model is gradually added. Among other things, the model is used as a basis for design and will be updated in conjunction with the gradual construction of the repositories. Larger updates will be carried out if there is substantial new information prior to the updated SARs in accordance with the activity schedule described in Chapter 3.

Prior to the start of construction, SKB is preparing for the establishment of methodology for discipline-specific and integrated geoscientific modelling on different scales and for different purposes, as well as investigation programmes with associated method descriptions for investigations for each repository.

5.5.4 Quality assurance

In order to ensure that results from research and technology development are correct and maintain high quality, SKB has procedures for quality assurance of the results. SKB 's management system includes procedures for procurement, approval of suppliers with the correct skills and who are able to live up to SKB 's requirements, approval of content in databases, approval of modelling and computational tools and approval of input data for models and computation results.

Results from SKB's research and development are usually presented in SKB's report series or in scientific publications. Reports and other documents of importance for safety which are prepared by SKB undergo a documented review process, in accordance with SKB's management system – see Section 5.2.3. The review is carried out to ensure that the documents meet the requirements that have been set regarding scope and content, and that the information provided is factually correct and based on approved sources and computational tools.

5.6 Competence and resources

SKB's broad and multifaceted activities mean that the company needs access to competence in many different areas. A large part of the knowledge and technology that SKB needs is included in the general scientific body of knowledge and is developed within the scientific community, while other parts are specifically linked to the management and final disposal of nuclear waste. This applies, for example, to large parts of the detailed knowledge of the function of the repository barriers in a geological environment. It is so specific to the nuclear waste field that the knowledge needs to be generated by SKB itself or in cooperation with other actors.

SKB needs to ensure that the company has sufficient competence to be able to utilise the knowledge that exists in the research community and that is of importance for management and final disposal of nuclear waste. There also needs to be sufficient competence to identify new research needs and to be a competent procurer of research. It is therefore essential for there to be a coherent group of persons with knowledge of the methodology for assessment of post-closure safety and with a broad and interdisciplinary insight into how the different processes that affect repository safety interact. The group also needs people with in-depth knowledge of the disciplines that affect safety. By conducting its own research, SKB ensures that this competence is maintained.

The need for competence to manage and carry out technology development is based on the plans that are prepared (Section 5.3.2). In order to be able to evaluate the development that is needed and be able to manage it, SKB needs expert knowledge within each production line. Development can in many cases be carried out by different research institutions or consulting companies, but in certain areas, where there are few other actors, SKB needs in-house competence for development. This particularly applies to the areas of radioactive waste, spent nuclear fuel, construction of canisters, development of cementitious materials and clay barriers and methodology for investigation of the rock. SKB's process for skills supply seeks to ensure that competence in such areas exists and is developed, both in the short and long term.

In order to be able to carry out some tasks of importance for SKB 's activities, there is a need for access to special laboratories and special instruments or tools, such as SKB 's materials research laboratory on Äspö for conducting specific analyses of bentonite and the Canister Laboratory (Section 5.4).

The personnel involved in carrying out transport are specially trained in the areas of radiation protection, physical protection, measures in the event of an accident and such training as is required by the Swedish Civil Contingencies Agency's (MSB) regulations on the transport of dangerous goods. Furthermore, it is ensured that personnel participating in transport are familiar with SKB's transport system, SKB's transport manuals, SKB's transport documents and locally prepared instructions linked to transportation.

5.6.1 Building, developing and maintaining competence

As the operator, SKB is obliged to ensure that tasks are performed by suitably qualified persons. Whether the competence requirement is to be met through the use of in-house personnel or through external suppliers or consultants is to some extent a strategic question. The decision made is based on an assessment of which tasks are of such strategic importance or essential for safety that they must be performed by SKB's own personnel, the risks of being dependent on external suppliers, and financial considerations. The outcome of the assessments relating to competency requirements and the deliberation on having in-house personnel or using external suppliers may vary over time.

SKB's starting point is that SKB should have its own personnel with the competence to be able to manage and lead the work on research, development and operation of the system for management of radioactive waste and spent nuclear fuel. This includes SKB having the necessary competence to procure and evaluate the services and goods that SKB orders from external suppliers. SKB's strategy is that its own resources should have the necessary competence and carry out the tasks that are of strategic importance for radiation safety. Examples include safety assessment methodology and the development of canisters for spent nuclear fuel, where SKB has chosen to rely largely on its own resources in order to build and maintain competence in the company in the long term. If other products or services are available through SKB 's owners or commercially, the general strategy is to use these.

SKB has a defined process for skills supply in the management system. It is structured in accordance with the Systematic Approach to Training (SAT) model developed by the IAEA, which is a methodology for working with competency areas, qualifications based on levels, competency assessments and area-specific training and management of identified gaps. The process means that SKB has a systematic approach to compliance with internal and external requirements to ensure that adequate competence is available for maintaining a high level of safety and achieving the goals of the organisation in the short and long term. Competence assurance also includes preparing for future changes in activities and external requirements.

Competence requirements and competence development are analysed on an ongoing basis and planning is carried out at both individual and group level and over a time frame of four to five years. Given the time it takes to develop competence, a future perspective is always essential. Strategic competence analyses with a time horizon of around ten years are also carried out regularly. Training programmes are established for individuals and groups as necessary. It may be mentioned that a programme has been initiated to ensure competence transfer in the areas of hydrogeochemistry and safety assessment methodology for the Spent Fuel Repository. Further description of this can be found in Section 11.4.1.

In order to store and preserve the information that is identified to ensure competence and staffing in the short and long term, SKB uses the Competence Tool (CT) support system. Documents from SKB's document support system are linked to CT regarding competence analyses, qualifications (competence descriptions), job and role descriptions, training, support documents, performance appraisals and development reviews, as well as competency reviews for both individuals and groups within the organisation.

SKB collaborates with Vattenfall and participates in an industry-wide network with representatives from Kärnkraftsäkerhet och Utbildning AB (KSU), Ringhals AB, Forsmarks Kraftgrupp AB and OKG Aktiebolag, with the purpose of improving and developing the approach of competence analyses and joint training activities.

5.6.2 Development areas within competence

The routines and procedures that have been established and the strategic competence and staffing analyses that are carried out form a good basis for managing competence development and skills supply in the long term. Development of the process for skills supply takes place on an ongoing basis, considering future conditions and requirements. Strategic competence analyses, qualifications and competence development are fundamental parts of competence assurance of SKB's personnel. Examples of development elements that are planned in the area of competence assurance include:

- **Support system for competence analyses** – A new support system is about to be implemented at SKB, Vision from the company Focus Learning. For many years, Vision has been used in the nuclear power industry in the USA to safely store and assess the importance and levels of difficulty of competencies identified through competence analyses. Vision also assists in the design of training for positions and roles.
- **Filming of infrequent work** – SKB has started a project to film work that is seldom carried out in SFR in order to maintain knowledge in the long term. A further development of this is to identify and film this type of work for all of SKB's commissioned facilities.

5.6.3 Competence networks and partnerships

The fundamental needs for competence within research, safety assessment and technology development are met by SKB's own personnel. In several of these areas, there is also a need for in-depth competence and access to larger personnel resources for research and development activities. External specialists are engaged for this, often from national and international universities and university colleges, research institutes and consulting companies, but also from SKB's sister organisations in other countries. Many have to a varying extent been associated with SKB's activities for decades. External suppliers are generally used in cases involving temporary needs for competence and resources, for example for larger projects regarding planning and design of facilities. SKB's owners are also an important resource.

Collaboration with universities, university colleges and research institutes

SKB collaborates with a large number of Swedish and international universities, university colleges and research institutes in order to obtain critical knowledge in areas where this knowledge is lacking. Through this, SKB has developed a broad competence network with deep knowledge in several key areas. Access to such expert competence has in many cases been, and will continue to be, a key factor in being able to solve critical research and investigation issues. The collaboration may involve both short-term, focused research assignments and longer assignments that can take place within the framework of a post-doctoral position or a PhD project.

SKB-funded doctoral candidates constitute a prospective competence reserve for SKB and others in the industry. SKB currently finances or partially finances around ten PhD projects. During the past RD&D period, nine SKB-funded PhD students have completed their studies and PhD, and over the years SKB has financed or partially financed well over 100 PhD students. Many of these have, during the research programme and after their PhD degree, played key roles in advancing SKB 's work. Today, around ten former PhD students funded by SKB are employees of SKB.

Cooperation with trade associations and external suppliers

SKB also collaborates with other waste organisations around the world. These collaborations have been and will continue to be important for ensuring access to competence and experience from similar development work in other countries. Collaboration takes place both bilaterally and in constellations comprising multiple organisations. One example is the international cooperation that SKB is coordinating at the Äspö HRL. Another is the now concluded Greenland Analogue Project (GAP), a collaboration between SKB, Posiva of Finland and NWMO of Canada, which included studies of how a future ice sheet can affect a final repository for spent nuclear fuel. In the copper field, SKB is collaborating with NWMO of Canada, both in specific research projects and in international collaborations where NWMO has a driving role, for example in the MICA project (Michigan International Copper Analogue) and the research consortium at Canmet MATERIALS.

Specialists and modelling groups from several countries are collaborating on selected issues of importance for final disposal of nuclear waste in special forums. Two forums are established in groundwater and transport modelling and engineered barriers: SKB Task Force on Modelling of Groundwater Flow and Transport of Solutes (Task Force GWFTS) and SKB Task Force on Engineered Barrier Systems (Task Force EBS). These collaborations are aimed at evaluating different concepts and modelling methods and facilitating collaboration between experimentalists and modellers.

Within surface ecosystems, several organisations collaborate in Bioprota on developing methodology, compiling data and comparing models to calculate consequences for human beings and the environment.

Edram is an international association for organisations responsible for the management of radioactive waste. It is a forum where strategic questions of a shared character are discussed in order for organisations to exchange experience and support each other. An example of areas addressed is "knowledge management" viewed from a long-term perspective.

In order to ensure access to the necessary competence and sufficient resources during periods of high workload, especially during ongoing assessments of post-closure safety for the different repositories, SKB is collaborating with several consulting companies from Sweden, Europe and North America. Several of these collaborations have been going on for a long time, and experts within these consulting companies constitute an important part of SKB's competence network.

Collaboration with Posiva

For a number of years, SKB has been engaged in in-depth collaboration with its sister organisation Posiva in Finland. Like SKB, Posiva has chosen to build its final repository for spent nuclear fuel according to SKB's KBS-3 method. In 2001, the Finnish parliament ratified the Finnish Government's decision in principle regarding the method and site for the Finnish final repository. The plant will be built in Olkiluoto in Eurajoki. In 2004, Posiva began building an underground rock facility (Onkalo) in Olkiluoto and reached the planned repository level in 2010. Onkalo has been used and is still used for research and development, but will also constitute the access facilities to the actual final repository.

At the end of 2012, Posiva applied for a licence to build an encapsulation plant and a final repository for spent nuclear fuel according to the KBS-3 method. In 2015, the Finnish Radiation and Nuclear Safety Authority (STUK) announced its statement on this application and recommended that the Finnish Government should grant a licence. In its statement, STUK identified a number of issues that Posiva needs to solve and report before it is possible to provide the operating licence. In the same year, the Finnish Government granted permission for construction of the final repository and the encapsulation plant. Posiva is now building the facilities, the first new shafts, the first deposition tunnels and the encapsulation plant, and intends to commence operation, if a licence for this is granted, around 2024. A major step was taken at the end of 2021 when an application for a licence for operation of the final repository and the encapsulation plant was submitted to the regulatory authorities. In connection with this, the questions asked by STUK in 2015 were also answered.

Since 2013, the cooperation between SKB and Posiva has sought to develop common technical solutions for the final repository system prior to commissioning. The agreement is valid for five-year periods and is renewed on a rolling basis. The collaboration includes canister, bentonite and rock issues, research questions, machine design and issues linked to finding financially optimal solutions without compromising safety.

In addition to the efficiency gains that this collaboration brings, SKB's ability to carry out research and development is also enhanced. The preparation of joint plans and joint projects means that these will be subject to broader and more comprehensive preparation and review than if SKB were to conduct this work on its own. Through the cooperation, SKB gains access to Posiva's facilities, especially Onkalo, and also gains access to the research institutions, institutes and other experts Posiva collaborates with. The collaboration is also important for future skills supply.

International collaboration within the EU, IAEA and OECD/NEA

Active participation in international collaboration has always been and continues to be an important part of SKB's work. This is part of business intelligence and a basis for the exchange of expertise and external review of research. For many years, SKB has been actively participating in international collaborations via projects and working groups in the EU, the OECD-NEA and the IAEA.

The EU's work in the field of nuclear power is regulated by the Euratom Treaty. Among other things, the European Commission strives to harmonise nuclear waste management in Europe and has issued directives on both nuclear safety and waste management. Research in the field of nuclear waste has been a part of the EU 's research programme for many years. Since 2019, research has been conducted within the framework of Eurad. The programme has been designed by a number of actors in the waste area, such as waste organisations (e.g. SKB), technical support organisations and research institutes. The program started in 2019 and will run for five years. The first phase consists of seven different research projects, two strategic projects and initiatives concerning knowledge transfer. SKB is participating in projects within the programme and has experts who are participating in reference groups for projects. Planning for a continuation of Eurad was initiated in 2021 and SKB intends to participate in the future programme (Eurad 2).

SKB plays an active role in the Implementing the Geological Disposal of Radioactive Waste Technology Platform (IGD-TP), which has the vision that a final repository for high-level waste and/or spent fuel should be in operation by 2025. IGD-TP is the waste organisations' part within the Eurad programme. The platform has been important for the focus of the EU's research programme. Members of the platform steering committee are organisations with responsibility for waste programmes in eleven countries, ONDRAF/NIRAS (Belgium), Posiva (Finland), Andra (France), BMWi (Germany), Enresa (Spain), SKB (Sweden), Nagra (Switzerland), RWM (UK)[6], Puram (Hungary), Covra (Netherlands) and Surao (the Czech Republic).

SKB is participating in the OECD's joint organisation for nuclear energy issues NEA. NEA includes RWMC and the Committee on Decommissioning of Nuclear Installations and Legacy Management (CDLM), which consists of representatives from regulatory authorities and waste organisations. Every four years, the OCED/NEA organises the International Conference on Geological Repositories, the sixth of which was held in Helsinki in April 2022. Participation in these is an important opportunity for exchange of experience for SKB.

The International Atomic Energy Agency (IAEA) also offers a number of projects and groups, and SKB and its owners participate in several projects on both final disposal and decommissioning issues. In the autumn of 2021, the IAEA organised a workshop in the area of knowledge management, in which SKB participated as a rapporteur.

Sweden has ratified the IAEA's waste convention (Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management) and SKB is contributing to the national report that the SSM, in accordance with the convention, produces every three years. The report is reviewed by the convention parties in special review meetings.

5.6.4 Current challenges and maintaining competence in the long term

In the autumn of 2021, SSM circulated a report for comment regarding the National strategy for skills supply in radiation safety, where SKB had the opportunity to respond. In its statement, SKB highlighted several different areas suitable for development. It is very important to have coordination at national level and for the entire industry to work together in order for long-term skills supply to be successful. SKB, together with the owners of the reactors and other nuclear facilities, has taken a leading role in this work.

General and fundamental prerequisites

SKB requires competence in many areas, especially persons with training in the natural sciences and technology. The areas in which SKB will require competence are expected to remain relatively unchanged over time, but apart from the continued need for competence in research and technology development there is a great future need for competence in construction, operation and decommissioning of facilities.

In general, the national trend of a declining interest in degrees in technology and the natural sciences is a problem for SKB and for Swedish industry as a whole. Besides leading to a shortage of graduates within the relevant areas, this also has negative effects on the opportunities for higher educational institutions to conduct teaching programmes in relatively narrow subject areas such as nuclear technology and to maintain important research environments, which, among other things, is highlighted in the SSM study (Hällgren 2018). Promoting interest in science and technology in order to ensure a sufficient number of graduates is thus an important issue for several different actors.

To secure competence in areas important for the final disposal of nuclear waste and decommissioning of nuclear facilities is, for SKB and its owners, a strategically important issue in both the short and long term. Final disposal and decommissioning entails a broad need for competence in the fields of nuclear technology and radiation protection, but also in geoscience, geotechnology, materials science, construction technology, instrument and measurement techniques and competence concerning climate evolution. The starting point is that society is responsible for providing basic education, e.g. for civil engineers, and for maintaining basic competence in relevant areas, but the industry will also need to make specific efforts. In order to secure competence within the country, collaboration with universities and university colleges will be a part of long-term skills supply, as will the continued encouragement of collaboration within the industry in order to both retain existing personnel and attract new personnel in these areas.

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[6] Since January 2022, RWM has been part of Nuclear Waste Services (NWS).

SKB is working together with the owners to increase the attractiveness of the industry, and intends to continue the work of strengthening confidence in SKB's task by highlighting the many interesting and challenging tasks and roles that are needed to carry out the work. SKB collaborates with companies and schools, primarily at the locations where SKB is active, and participates in labour market days, fairs and various industry events.

In general and in a long-term perspective, SKB expects most of the need for personnel to be satisfied by personnel with relevant basic education (upper secondary school, university college or university) in the subject field, who are then further trained by SKB for company-specific applications. In addition, there is a need for a small number of persons with in-depth competence, for example postgraduates, in combination with long experience of areas important for SKB.

Ongoing and planned activities

SKB is continuing to contribute to different professorships and other positions at university colleges and universities, and to disseminate information on its task and the kind of competencies that will be sought after at universities and colleges that have relevant teaching programmes in order to safeguard long-term skills supply. Where appropriate and possible, SKB supports PhD projects that can contribute to the creation and preservation of good research environments. In addition to developing knowledge in issues of importance for final disposal of waste, the PhD students funded by SKB constitute a significant injection of skills, particularly expert skills, for SKB as well as for other actors in the area.

Other parts of the industry also promote training in the field of technology. For example, Vattenfall is funding an engineering programme focused on nuclear power at Uppsala University, and at the upper secondary school Vattenfallgymnasiet in Forsmark, in order to secure access to qualified competence well into the 2040s. Furthermore, Forsmarks Kraftgrupp AB, Ringhals AB, OKG Aktiebolag and Westinghouse Electric Sweden AB are funding a national research centre, the Swedish Centre for Nuclear Technology (SKC). SKC supports education, research and development within nuclear applications at universities and university colleges in Sweden.

On a local level, at the sites where SKB has operations, SKB contributes competence and equipment to Teknikcollege, which offers additional courses. These courses can be taken at upper secondary school, especially through the technical programmes.

For narrow competence areas in which there may be only one or two experts at SKB or even in Sweden, the international collaboration SKB has with other waste organisations is essential (Section 5.6.3). Through this collaboration, SKB has the opportunity to engage experts from these organisations either to carry out assignments for SKB or to train personnel within SKB. The purpose of the collaboration with Posiva mentioned in Section 5.6.3 is, among other things, to maintain competence and to support each organisation in narrow fields of expertise within the nuclear waste field.

Questions relating to the maintenance of competence are discussed in depth internationally, for example in the IAEA, the NEA and the EU, and among other nuclear waste organisations. SKB considers the international collaboration to be important in order to develop and secure competence in the long term.

Related challenges

A related challenge for SKB is the construction of the new facilities that will then be operated for a long time. SKB is building up a purchasing organisation to procure, manage and follow up the contracts that the construction of each facility will be divided into. Since the construction of the facilities mainly includes using well established technology, SKB is planning to engage established suppliers (contractors) with the necessary competence. SKB already has experience, for example from the construction of Clab and the site investigations, of managing suppliers and ensuring that they comply with SKB's often specific requirements.

One of the difficulties in finding suitable suppliers for the facility programmes is the competition for personnel from other major infrastructure projects in the coming decades. Another could be access to the senior competence that is needed, given the generational change that is taking place now that many of the people who have built up the Swedish nuclear waste programme will soon retire or have already retired.

Long-term challenges

When it comes to skills supply in the very long term, i.e. a 50–100 year perspective, there are two important prerequisites that need to be taken into account:

- SKB's activities are planned to continue until around 2090, and it is already necessary to plan for which competences will be important in the long-term perspective.
- SKB is the dominant player in Sweden in the management of radioactive waste.

The long-term perspective can be an advantage, as competence development and skills supply can be planned for the long term in a way possible for only few other companies. SKB has well developed processes for competence development and skills supply, as presented in Sections 5.6.1 and 5.6.2, which provide a good basis for SKB to secure the competence needed to carry out the planned activities according to specified internal and external requirements.

One aspect which especially needs to be considered during the entire life cycle of the facilities is to have personnel with facility knowledge and, for the final repositories, to also possess knowledge of the surrounding rock. Special initiatives will be needed when it comes to the preservation of knowledge and competence in these areas, such as good documentation, training of personnel and redundant staffing. Work to develop processes and procedures for the transfer of knowledge and experience between employees and to new recruits (knowledge management), where the focus is well in the future, is under way at SKB – see Section 4.13.1.

SKB's role as the main employer in management of radioactive waste makes the company sensitive to the general trend of increased mobility in the labour market that exists in today's society. If an employee leaves the company, it may be difficult to find a replacement with the appropriate skills. SKB is, however, not the only employer in the area, and it is assumed that equivalent competencies will continue to exist with owner companies, suppliers, consulting firms and regulatory authorities.

A risk identified by SKB is the risk that some part of the activities might be paused, and the problems that might then arise in retaining competence within various areas. Operational pauses may be planned or unplanned. Planned pauses in operations could for example be the site investigations that have already been carried out for the Spent Fuel Repository and the extension of SFR, and which will be carried out for SFL in a few years. A pause in part of SKB's activities normally leads to a need for relocation of personnel, but could also lead to redundancies. The possibilities of relocating personnel are naturally dependent on the area of competence. SKB considers the prospects for retaining personnel and competence in respect of the experts who work with safety assessments to be relatively good. Safety assessments are to be performed at regular intervals as part of the stepwise licensing process for the three final repositories. Those working with research and technology development may also work to an increased extent within international projects or be sent on loan to waste organisations in other countries via SKB International. For other personnel categories, such as operational operating personnel, there may be opportunities to lend personnel to SKB's owners, where similar tasks occur.

SKB considers potential problems to be manageable and believes that, by applying and further developing the procedures for competence development and skills supply, it is possible to accomplish SKB's task of disposing of the radioactive waste from the nuclear power plants with retained competence.

Part II

Waste and final disposal

Part II of the RD&D Programme 2022 describes planned research and technology development activities during the RD&D period. The focus is on issues identified by SKB as priority issues for the continued management and final disposal of radioactive waste and spent nuclear fuel. The current situation and programme for the low-level and intermediate-level waste, the spent nuclear fuel and the different parts of the repository system are presented, which means that some research and technology development is described in an integrated manner for the three final repositories. The description of the current situation is an overview and refers to more detailed reporting of results in background reports etc. The presentation of the prioritised research and technology development activities during the RD&D period is presented as bullet points in the programme sections.

For licensing activities, the results of ongoing research and technology development will be reported in more detail in the stepwise licensing process pursuant to KTL for each facility.

6 Low- and intermediate-level waste

Low- and intermediate-level waste consists of operational waste and decommissioning waste from the Swedish nuclear facilities and radioactive waste from research, medicine and industry. Waste with a limited amount of long-lived radionuclides is regarded as short-lived and is disposed of in the Final Repository for Short-lived Radioactive Waste (SFR), while long-lived waste will be disposed of in the Final Repository for Long-lived Waste (SFL). This chapter presents the research and development that SKB intends to conduct during the RD&D period in order to learn more about the processes and properties related to low- and intermediate-level waste in SFR and SFL.

6.1 Impact on sorption

In the SFR, sorption of radionuclides on cement minerals is one of the most important processes that delay the release of radionuclides from the repository. Complexing agents can reduce the sorption of radionuclides and thereby accelerate their migration from the repository. Through requirements in waste acceptance criteria the amount of complexing agents in the waste is minimised. There are also organic materials that under certain conditions may degrade into complexing agents. A better understanding of the degradation processes and the products that are formed by degradation is therefore of great importance for assessments of which materials require restrictions in SFR and how the sorption of radionuclides is to be managed in SKB's post-closure safety assessment.

The sorption in SFR is quantified in the form of sorption coefficients, K_d. For the radionuclides where specific K_d-data are lacking, SKB uses chemical analogy reasoning, i.e. the assumption that the same K_d-value can be applied as for a chemically similar substance for which data are available.

Different types of organic matter occur in the engineering material and the waste to be deposited in the Final Repository for Short-lived Radioactive Waste (SFR). Of the organic material categories that can give rise to complexing agents, SKB has conducted studies of cellulose, cement additives and filter aids, among others.

Under alkaline conditions, cellulose can be broken down into isosaccharinate (ISA) (Glaus et al. 1999). The degradation mechanism is well studied (Glaus and Van Loon 2008) and SKB has used the rate derived in the study to estimate the concentration of ISA in SFR (Keith-Roach et al. 2014, 2021). ISA can act as a ligand and form soluble organometallic complexes with the majority of radionuclides. ISA can also sorb on cement minerals (Ochs et al. 2014), which reduces the availability of ISA for complexation with radionuclides.

In both the extended SFR and the future SFL, large amounts of concrete will be used as an engineering and grouting material. In order to obtain a high-quality concrete with suitable properties, concrete additives such as superplasticisers will need to be used. Superplasticisers consist of organic polymers that may degrade over time into easily soluble organic compounds. These degradation products can affect the speciation and transport of radionuclides when they are in solution. However, this effect has not been observed for superplasticisers in solid concrete or cement, which is explained by the fact that they are strongly bound in the hardened material (NDA 2015, 2017, Keith-Roach and Höglund 2018, Bamforth et al. 2012, Young et al. 2013, Glaus and van Loon 2004). For this reason, experiments with radionuclides and superplasticisers embedded in concrete show no impact on radionuclide sorption (NDA 2015). Additional leaching tests with varying ratios between the amount of solid and liquid phases confirmed the results that superplasticisers do not increase the transport or mobility of radionuclides in concrete (NDA 2015). These experiments are conducted under more repository-like conditions than traditional experimental sets and therefore provide information that is more relevant for SFR.

Another type of organic material that often occurs in the operational waste from the nuclear power plants is filter aids such as UP2. This filter aid consists mainly of polyacrylonitrile (PAN). Under alkaline to hyperalkaline conditions, the polymer structure breaks down and forms highly soluble compounds that can affect the sorption and/or the solubility of radionuclides (Duro et al. 2012).

Current situation

Two studies of the impact on sorption of cellulose degradation products, funded by SKB, were concluded in 2018 in the form of PhD theses (González-Siso 2018, Tasi 2018). The first study involves nickel sorption in the presence of ISA. Nickel forms complexes with ISA under reducing alkaline conditions such as in the Final Repository for Short-lived Radioactive Waste (SFR). Complexation was shown to lead to an increase in the solubility of nickel, and equilibrium constants for a number of different Ni-ISA complexes have been developed. A summary of the results from González-Siso (2018) is available in Bruno et al. (2018) and the results of this study regarding sorption coefficients K_d for nickel on cement and speciation of uranium under Final Repository for Short-lived Radioactive Waste (SFR) conditions have been applied directly in the PSAR for the Final Repository for Short-lived Radioactive Waste (SFR).

In the second thesis, sorption experiments served as a basis for a model that describes sorption of plutonium on cement in the presence and absence of ISA. In the same study, sorption reduction factors for plutonium could be derived for different concentrations of ISA. The updated sorption reduction factors for plutonium (Tasi 2018, Tasi et al. 2021), as well as sorption coefficients K_d for plutonium on cement in the absence of ISA, have been used in the PSAR for SFR.

A systematic literature review and thermodynamic calculations of how ISA and other complexing agents in waste affect the sorption of different substances have been carried out by Keith-Roach and Shahkarami (2021) within the framework of waste acceptance criteria (WAC). The study confirms that the previous ISA WAC limit of 10^{-4} M is still appropriate for use. On the other hand, they propose that the limit values for certain other complexing agents be raised, which for di- and tricarboxylic acids means that the concentration of these no longer needs to be limited with regard to complexation. The reason for this is above all competition from stable Ca^{2+}- and OH^--ions from cement that occur in high concentrations in SFR. This reduces the degree of complexation between radionuclides and organic complexing agents, which has not been fully taken into account in previous acceptance criteria.

The studies of the impact on sorption of degradation products from cement additives used in structural concrete that SKB started in 2016 have been concluded (Chernyshev et al. 2018, 2021). The structure of the superplasticiser was characterised in the study and it was shown that the polymer includes organophosphate groups. The rate constant for degradation of the superplasticizer and the release of phosphate was determined and shown to be fast. The studied superplasticiser increases the solubility of both nickel(II)and europium(III) by complexation at pH 12.4. At an even higher pH, degradation of the superplasticiser and release of the phosphate are accelerated, which in the study led to precipitation of Ni- and Eu-phosphate respectively. It is important to note that both studies by Chernyshev et al. were performed in solution phase. In SFR, superplasticisers will be used in structural concrete where they are considered to be attached and can thereby be expected to degrade more slowly and have less impact on the speciation of radionuclides, as described at the beginning of this Section 6.1. The two studies are therefore of limited relevance for SFR.

SKB has developed limit values for superplasticisers in embedment concrete and solidification cement in packages, since radionuclides there come into contact with superplasticisers in dissolved form (Hedström 2019a, b). The limit values have been adjusted slightly in WAC 5.0 due to results from Keith-Roach and Shakarami (2021).

The content of concrete additives is often not publicly available, as the companies that manufacture these do not usually openly report the composition of the additives. This can limit the possibility for SKB to study in advance what an additive contains. On the other hand, more information about the composition can generally be obtained when purchasing a product. In good time, before the start of construction of the extension of SFR and construction of SFL, it is important that SKB gets the opportunity to make a decision on the suitability of the superplasticiser that the concrete contractor intends to use so that its suitability can be assessed.

In a study conducted at the Royal Institute of Technology (KTH), the impact on sorption of degradation products from the filter aid UP2, sorption of nickel(II) and europium(III) on cement in the presence of degradation products from UP2 under repository-like conditions was investigated (Tasdigh 2015). UP2 mainly consists of PAN. The study showed high formed concentrations of degradation products from the filter aid and that such high concentrations reduced sorption of Eu(III) and that the solubility of Ni(II) increased. In contrast to this study of UP2, previous studies of degradation products of PAN have shown that they do not significantly affect the sorption of Eu(III) (Dario et al. 2004, Duro et al. 2012).

Further studies of the filter aid UP2 began in 2020 within the framework of Cement-Organics-Radionuclide interactions (CORI), a work package within Eurad. Preliminary results show that degradation of UP2 under reducing conditions at pH 12.5 is initially slow, but becomes relevant after about 800 days (Gaona et al. 2021). Furthermore, it was shown that the high concentrations of degradation products from UP2 in Tashdigh (2015) were greatly overestimated due to the use of acetate filters, which give a large indication of the measured content of dissolved organic material. This could be reproduced in terms of magnitude by Gaona et al. (2021) and they therefore use consistently inorganic filters that produce measured concentrations that are consistent with Duro et al. (2012).

Based on the presumed degradation mechanism of PAN, three proxy ligands have been identified by Gaona et al. (2021): glutarate, α-hydroxy-isobutyrate and 3-hydroxy-butyrate. A proxy ligand is a known ligand that can be measured directly, while most of the degradation products can be considered unknown and therefore cannot be measured directly. The identified proxy ligands can be assumed to have similar reactivity to most of the degradation products and their impact on the solubility of Ca(II), Nd(III), Ni(II)and Pu(III/IV) is currently being studied. Preliminary results show that the impact is negligible for Ca(II), Nd(III) and Pu(III/IV), while the solubility of Ni(II) increases slightly. In general, the solubility impact of the three proxy ligands is considerably less than that of the ISA. Preliminarily, the presence of the proxy ligands also shows a very weak effect on Pu sorption. Similar experiments with Ni and Eu are ongoing.

Programme

- Molybdenum, selenium and technetium are important for the safety assessment, but lack robust sorption coefficients. A research assignment aimed at supporting national competence concerning radionuclide sorption on cement is planned to be started during the RD&D period. K_d-values for cement in the near-field will be experimentally determined for Mo, Se and Tc, and the sorption processes will be modelled for better process understanding.
- The study of degradation of the filter aid UP2 has recently been concluded. The decomposition products actually formed are planned to be used in future sorption studies.
- Further studies of the three proxy ligands are planned to determine their impact on radionuclide sorption.
- SKB is co-financing and actively participating in the PAN research project within CORI. SKB also follows CORI in general through participation in annual meetings and review of annual reports.
- SKB will, in good time before the start of construction of the SFR extension and SFL, make a decision on the superplasticiser that the concrete contractor intends to use so that its suitability can be assessed.
- SKB is not presently planning to continue studying the impact of degradation products from cellulose and superplasticisers on the sorption of radionuclides. SKB will, however, continue to follow the state of knowledge.

6.2 Gas formation

Gas production in the final repository environment can lead to pressure build-up and a subsequent impact on the function of the barriers. If gas production is so extensive that the produced gas cannot be discharged in a controlled manner, the gas pressure may expel radionuclide-containing water and in the worst case damage the barriers in the final repository. It is therefore important to understand the degradation processes leading to gas production in order to assign realistic parameter values in the post-closure safety assessment and to be able to specify requirements for the content of the waste. Gas production is mainly due to degradation of materials by e.g. corrosion, microbial processes or radiation effects. Anaerobic corrosion of aluminium and zinc in waste can form large quantities of hydrogen gas in a short time, and it is therefore important to quantify the corrosion rate for zinc and aluminium in a final repository environment.

Current situation

SKB has, together with researchers at KTH, studied the corrosion of aluminium and zinc embedded in concrete and exposed to anoxic artificial groundwater. In the now completed study, a series of trials were conducted with different exposure times (2, 4, 12, 52 and 104 weeks) (Herting and Odnevall 2021).

For both zinc and aluminium, corrosion was initially shown to be rapid, but when the systems reach an equilibrium state, the corrosion rate declines. As expected, aluminium corrodes faster than zinc. Corrosion rates for concrete-embedded aluminium and zinc exposed to artificial groundwater are around 100 µm/year and 1 µm/year respectively after 104 weeks. When it comes to aluminium, a void is formed, above all on top of the metal, but also around it, which is attributed to hydrogen gas evolution. Cracks may also form in the concrete due to the increase in volume caused by the formation of corrosion products. Fracturing could not be observed for zinc. Corrosion rates for aluminium and zinc in artificial groundwater without concrete embedment are of the same order of magnitude as results from experiments performed on embedded samples. This means that corrosion products on metal surfaces are rate-limiting, and not their diffusion through the concrete barrier.

Programme

- Corrosion of aluminium alloys, for example in the reactor pressure vessel from the Studsvik site, and corrosion of other metallic materials may be needed for further studies during the RD&D period.
- SKB is not currently planning further studies of anaerobic corrosion of aluminium and zinc, since already published studies (for example Fujiwara et al. 2017) provide sufficient information. SKB will, however, continue to follow the state of knowledge.

6.3 Swelling of ion exchange resins

Like gas production, swelling waste may also affect the integrity of concrete barriers and thereby affect the outward migration of radionuclides from SFR. It is therefore important to quantify the swelling pressures that swelling waste can give rise to, and to impose requirements on waste to avoid waste forms that risk creating excessive swelling pressures.

Current situation

It is known that bitumen-conditioned hygroscopic materials, such as dried ion exchange resins and evaporator concentrates, can absorb water and thereby swell. The rate at which this occurs and how the swelling pressure evolves over time depends on factors such as the ratio between constituent material, bitumen type, type of ion exchange resin, crosslinking ratio of the ion exchange resin, solubility of the constituent salts, particle size and the extent to which the initial swelling can take place freely.

Supplementary investigations have been carried out, both by SKB and on behalf of SKB, in order to reduce the uncertainties associated with swelling and swelling pressure. In order to achieve the purpose, however, further investigations need to be carried out. Experiments with conditioned ion exchange resins are impeded by the fact that resaturation of bitumen-conditioned ion exchange resins takes a very long time and is also dependent on counter pressure.

Programme

- SKB intends to carry out new experiments and calculations in order to better understand and describe swelling of bitumen-conditioned waste under repository-like conditions, in order to reduce the uncertainties associated with swelling waste and swelling pressure in SFR.

6.4 Radionuclide inventory

SKB is continuously working on improving methods for estimation and prognosis of the radionuclide inventory in low- and intermediate-level waste. As the state of knowledge improves, the reference inventory calculation is updated and supplemented.

SKB 's efforts to improve the estimation of radioactivity in low- and intermediate-level waste are primarily focused on the nuclides that in safety assessments have proved to have the greatest impact on the radiological risk after closure. Most of these nuclides are categorised as difficult-to-measure nuclides. Difficult-to-measure nuclides are nuclides whose activity cannot be routinely measured directly on waste packages.

Certain difficult-to-measure nuclides can be measured routinely in samples from, for example, process water at the waste producers. For measurement of other difficult-to-measure nuclides, however, extensive concentration and sample preparation is required, as well as advanced analyses in external laboratories. Because of this, the availability of measured values for these nuclides is limited.

Since the measured activity for difficult-to-measure nuclides in the low- and intermediate-level waste only exists as indirect measurements or as analyses of single samples, it is necessary to apply calculation models for estimating the radionuclide inventory. For the calculated inventory to be reliable, the models must, as far as possible, be based on established physical relationships and the assumptions and parameters in the models must be verified with measured values. Physically plausible models are also a prerequisite for extrapolating data for difficult-to-measure nuclides for older waste produced when the availability of measured values was more limited. The models will be adapted and verified against measured activity for difficult-to-measure nuclides and against more extensive sets of measurement data for related easy-to-measure nuclides.

SKB’s continued work with difficult-to-measure nuclides includes increased measurement of these, as well as further development of the calculation models that are used to estimate the radionuclide inventory. Furthermore, SKB is working to improve the estimation of uncertainties in both measured and calculated activity by means of further measurements and collection of process data from the waste producers.

6.4.1 Reference inventory

Current situation

SKB and the reactor owners, with ongoing dismantling and demolition of nuclear power reactors, carry out regular cross-checking in order to inform each other about ongoing work and to exchange experience. Since the RD&D Programme 2019, SKB has written a memorandum (Ahlford 2021) to clarify expectations regarding activity determination in waste from dismantling and demolition. In addition, a template has been prepared for regular reporting of waste forecasts for decommissioning waste. This means SKB is able to continuously update the inventory as the reactor owners carry out studies, calculations or measurement programmes for the purpose of characterising the decommissioning waste. SKB has also begun the development of systems so that information from radiological characterisation can be received prior to dismantling and demolition.

The work of characterising the legacy waste that is planned to be disposed of in the coming Final Repository for Long-lived Waste (SFL) is under way and has been developed by AB SVAFO. Characterisation is a necessary step in updating the reference inventory for SFL prior to the coming application for a licence to build this final repository, since there is currently uncertainty regarding the content of both materials and radionuclides in the waste. Since the end of 2019, SKB and AB SVAFO have carried out regular cross-checking to plan and structure the characterisation work and develop a plan of action regarding what information needs to be collected in the first phase for the different waste fractions.

Programme

· Request continuous updates of waste forecasts for the decommissioning waste and regularly follow these up with radionuclide transport calculations (calculation model from the PSAR for SFR). These calculations form an important basis for SKB's assessment of whether changed forecasts for the inventory are accommodated in the ongoing licensing process for the extension of SFR.

- Continue to monitor the results of sample analyses of segmented reactor internals and reactor pressure vessels in order to be able to update the estimate of induced activity in low- and intermediate-level waste.
- Uniform inventory calculations for the decommissioning waste with the calculations made for the operational waste.
- Start the work of reviewing the SFL inventory, which entails, among other things, obtaining data and updating the inventory with the latest information. More concrete initiatives, such as characterisation of specific waste fractions (e.g. control rods), will also be carried out within the framework of the work.
- Support AB SVAFO continuously with advice and guidance, including follow-up radionuclide transport calculations for the SFL inventory, with the aid of the calculation model developed within the safety evaluation (SE-SFL). Calculations are made when updated information on the radionuclide content has been produced for a specific waste fraction in order to be able to answer whether characterisation and management strategy can be deemed to be sufficient for the fraction in question.
- Continue to follow the planning of the coming nuclear facilities in Sweden in order to be able to continuously consider the possibility of disposing of waste from operation and decommissioning of the facility.

6.4.2 Method development for difficult-to-measure nuclides

Current situation

To facilitate further development of calculation models for both establishment of source terms and activity determination of difficult-to-measure nuclides, SKB has collected available operational and measurement data for all Swedish reactors since the start of operation, and in 2021 a database for this data was established.

Since the RD&D Programme 2019, SKB has developed a calculation model for determination of molybdenum-93 in operational waste. The model currently includes BWR reactors and describes the corrosion product release from molybdenum-alloyed fuel spacers and corrosion product release from other molybdenum-containing core components that are exposed to sufficiently high neutron fluxes to be activated. The methodology has also been updated for operational waste from Clab with respect to molybdenum-93, where results from verifying measurements serve as a basis for the updated estimate.

SKB has, together with the reactor owners, carried out verifying measurements of the difficult-to-measure nuclides Mo-93, Tc-99, I-129 and Cs-135. Integrated samples from reactor water and pool water (Clab) were collected in 2019 and 2020 and sent to the Risø laboratory for analysis. Between the two sampling campaigns, the method for sample preparation of Mo-93 was further developed by the contracted laboratory, which resulted in enabling detection of this nuclide too in the samples from the second campaign. Gamma measurement and element analysis were done in connection with sampling, so that the quantity of difficult-to-measure nuclides could be related to relevant easy-to-measure nuclides and measured metal content in the samples.

SKB has also carried out single, supplementary measurements of difficult-to-measure nuclides or element composition when material has been available in a laboratory, for example when samples have been sent for analysis within the framework of the different radiological characterisation projects. These measurements are aimed at increasing understanding of the mechanism for distribution of different substances in the reactor systems, and provide an important basis for the activities carried out in the model development of difficult-to-measure nuclides.

Programme

- Update and supplement the database established by SKB for the reactors' operational and chemistry data on an annual basis and use the data as a basis for parameterising and verifying calculation models for difficult-to-measure nuclides. The work also includes improving the uncertainty analysis of these nuclides with the aid of the data.
- Continue development of calculation models to quantify the quantity of difficult-to-measure nuclides in operational waste. This primarily includes further development of the model for molybdenum-93 so that it also includes PWR reactors, but also development of source term models for crud and fuel damage respectively.
- Supplement the source term models so that all difficult-to-measure nuclides in operational waste from the Swedish nuclear power reactors are included, using measurement values from already established activity determination methodology (such as regular measurement of a number of transuranic elements) for verification and estimation of uncertainties.
- Continue the work of developing an improved model for calculation of induced activity in control rods held in interim storage using boron burnup data and other operational history in order to provide more detailed information of the irradiation history. The calculations will also be used as support for radiological assessments regarding the continued handling of the control rods up to disposal in SFL when it comes to e.g. dimensioning of the radiation shield for handling, but also in order to make decisions on final disposal.
- Collect neutron transport models for each reactor to facilitate future updates of activity calculations.
- Improve the reliability of how measured and calculated activity quantities in systems at the waste producers are distributed per waste package by obtaining and applying to a greater extent information that increases the possibility of tracing waste packages to the systems where the waste has been generated.
- Together with the reactor owners, carry out further measurements of molybdenum-93.
- Design a programme for sampling of the control rods that are kept in interim storage in Clab in connection with their handling, in order to verify and adjust the calculation model that is being developed for the purpose of determining the radioactivity in the control rods.
- Perform supplementary analyses for increased mechanism understanding.
- Improve the uncertainty analysis by performing further verifying measurements of difficult-to-measure nuclides.

6.5 Waste acceptance criteria in the Final Repository for Long-lived Waste (SFL) and the extended Final Repository for Short-lived Radioactive Waste (SFR)

At present, a considerable amount of long-lived waste is kept in the waste producers' interim storage facilities, and additional long-lived waste will be generated during the continued operation and dismantling of the nuclear facilities. In order to avoid impeding future transportation and final disposal, it is important to clarify how the long-lived waste is to be managed and characterised, and which requirements can already be imposed on the waste today.

With respect to the decommissioning projects, there is also a need for clear prerequisites for how the short-lived decommissioning waste is to be managed in order to, as far as possible, comply with future acceptance criteria for the extended SFR.

In order to determine whether the waste produced by the waste producers is characterised to a sufficient extent to meet the safety requirements in a future final repository, there is thus a need to present acceptance criteria for final disposal in both SFL and the extended SFR. As the details of the repository design are finalised, it will be possible to further define the set of requirements.

Current situation

During the past RD&D period, SKB has begun formulating waste acceptance criteria for the waste vault for core components (BHK) in SFL. The results of the safety evaluation of SFL serve as a basis for deriving requirements related to post-closure safety.

Acceptance criteria for waste to be disposed of in the extended SFR were submitted in conjunction with the application for the construction of the extension to SFR. During the past RD&D period, SKB has further developed the waste acceptance criteria for the extended SFR and developed and derived waste acceptance criteria for all new repository parts. The criteria are based on the latest knowledge of the design of the repository and the transport system and the post-closure safety assessment that formed part of the preparatory PSAR (F-PSAR). These acceptance criteria are included in the PSAR and will then be further developed on the basis of the results of the assessment of post-closure safety in the PSAR.

Programme

- Continued development of acceptance criteria for final disposal of waste in SFL, both for the waste vault for core components (BHK) and for the waste vault for legacy waste (BHA). A first version of preliminary acceptance criteria for the two repository parts will be developed during the RD&D period. Requirements linked to post closure safety of SFL are derived, in the first version, from the completed safety evaluation, and will be developed as SFL evolves. Requirements linked to the operating activities in SFL are developed in conjunction with the preparatory PSAR (F-PSAR) and further developed as the design of the facility evolves. Examples of investigations that will be carried out in connection with the development of the acceptance criteria are requirements for packaging – see Section 6.6.
- Continued development of the acceptance criteria for final disposal of waste in the extended SFR, based on the results of the safety assessment in the PSAR, will be planned during the RD&D period. Harmonisation will be done of the acceptance criteria for the extended SFR so that they follow the same structure as the acceptance criteria for the existing SFR. Prior to the SAR for trial operation, the waste acceptance criteria for the extended SFR will be incorporated into the waste acceptance criteria for the existing SFR.

6.6 Waste containers and waste transport casks

In order to be able to carry out the decommissioning of the nuclear facilities in an optimal manner, the development work on waste containers and waste transport casks for the low- and intermediate-level waste needs to be followed up on a continuous basis.

6.6.1 Waste containers

Current situation

Work on waste containers during the past RD&D period has focused on existing types of waste containers and on evaluation of whether these can be used in different waste vaults and final repositories. For example, the strategy for application of waste containers in the waste vault for reactor pressure vessels (1BRT) in the extended SFR has been refined to mainly handle waste containers of mould format, which permits efficient handling.

Work has also been initiated for SFL, where the suitability of existing types of waste containers is being evaluated. This is carried out on the basis of the waste that is planned to be disposed of and the set of requirements that are expected to apply for long-lived waste.

A study has been conducted to analyse the suitability of different waste containers linked to the handling of BWR control rods from a BAT and ALARA perspective. The study included an early concept for the waste container long mould. However, the proposed design of the long mould was judged to be unable to meet applicable requirements with respect to radiation safety, transportability and interim storage.

Programme

- SKB will continuously monitor the need for new or further development of suitable waste containers for existing and future final repositories.
- During the RD&D period, a project is planned to investigate the possibility of designing and using a sealed waste container for SFL.

6.6.2 Waste transport casks

Waste transport casks aim to permit safe and efficient transport of waste from the waste producers to new interim storage facilities or final repositories. The need for waste transport casks is determined by the choice of waste containers and the need for capacity.

Current situation

Existing waste transport casks ATB12K (intended for twelve single moulds) and ATB3T (intended for three tanks) have been updated to packaging classes Type A and IP-2, which increases the transport capacity of these containers. A conversion of ATB16K (previously intended for 16 single moulds) to packaging classes Type A and IP-2 is under way, which will enable both a steel or concrete tank, or two double moulds, to be transported.

Programme

- To develop, in cooperation with American Holtec International Power Division Inc, a new waste transport cask, ATB1T, for transport of steel tanks with higher activity. The intention is for the new waste transport cask to be certified and commissioned during the RD&D period.
- The need for modification of existing containers to accommodate a different waste type than they were originally intended for, and whether a different packaging class may be needed, will be reviewed prior to future decommissioning waste.

7 Spent nuclear fuel

According to the planning premises (Section 1.1), the total quantity of spent nuclear fuel will correspond to about 6 000 canisters in the Spent Fuel Repository. One canister contains about two tonnes of fuel. The amount of spent nuclear fuel is given as the quantity of uranium that was originally present in the fuel. In addition to all spent fuel from the current Swedish nuclear power plants, the amount of spent nuclear fuel to be disposed of in the Spent Fuel Repository also includes fuel from the Ågesta reactor, spent fuel from testing programmes at the Studsvik site and a number of fuel assemblies with MOX fuel (mixed oxide fuel). These fuel types comprise a very small fraction of the total amount of spent nuclear fuel. Approximately 20 tonnes of spent nuclear fuel from the Ågesta reactor and approximately three tonnes of spent nuclear fuel from Studsvik Nuclear AB's research activities are currently being kept in interim storage in Clab. Furthermore, a number of leaking fuel rods, corresponding to a total of 1.4 tonnes of fuel, are included; these are placed in special sealed containers for storage in Clab and later disposal in the Spent Fuel Repository. Also stored in Clab are 23 tonnes of MOX fuel that has been obtained from Germany in exchange for the fuel that at an early stage of the Swedish nuclear power programme was sent to La Hague in France for reprocessing.

SKB's programme for management of fuel comprises several parts, from requirements for information on the properties of the fuel before it is used in the fuel cycle, to formulating an internationally approved programme for nuclear safeguards. The development work that is being pursued and planned in these areas is described in this chapter. Development of nuclear safeguards is an area where SKB works in close cooperation with international bodies.

If a canister in the Spent Fuel Repository is breached and water enters, the fuel properties are crucial for determining how quickly radioactive elements might be released. The results of previous safety assessments show that the rate at which radionuclides are released from the different parts of the fuel significantly affects the post-closure safety assessment of the Spent Fuel Repository. An in-depth understanding of the mechanism for dissolution of the fuel matrix is needed to support the interpretation of the experimental results.

7.1 Non-regular fuels and fuel integrity

Non-regular fuels are e.g. fuel from the Ågesta reactor, spent fuel from testing programmes at the Studsvik site, MOX fuel and failed fuel.

The integrity of the fuel is significant for the encapsulation plant, as handling of the fuel assemblies in Clink must be done in a predetermined way. SKB therefore follows ageing of nuclear fuel in Clab, from nuclear power plants and other industry, as well as international research findings. A number of individual fuel assemblies are continuously inspected in Clab to monitor changes in their properties during interim storage. As handling in Clab takes place in water-filled pools, inspection of the fuel assemblies is relatively uncomplicated, unlike dry storage in canisters, where it is not possible to perform such inspection in a simple and continuous manner. For experience feedback, contact is also maintained with fuel suppliers and other facilities with experience of interim storage.

Current situation

A campaign to empty the Swedish nuclear power plants of failed fuel was carried out during the period 2015–2020. The fuel consists of uranium dioxide pellets enclosed in cladding material. The cladding material may be damaged, and in order to restore the barrier function that it is intended to perform during operation of the facilities where fuel is handled, the failed fuel has been treated so that it does not need further treatment before final disposal. Failed fuel in breached cladding is also called leaking fuel.

SKB has used two different methods for management of leaking fuel rods. Both methods entail that the failed fuel is placed in containers with the dimensions for PWR or BWR fuel. These containers are water-tight and thereby replace the failed cladding. One method is developed by Westinghouse and is called Quiver. It has been developed for use in the storage pools of the nuclear power plants. The second method has been developed by Studsvik and entails that the leaking fuel rods are sent to the Studsvik site where they are placed in sealed containers inside transport boxes for further transport to Clab.

Existing documentation shows that there are a small number of fuel assemblies with leaking fuel rods in Clab. How this will be managed, and how it affects the chemical environment in the canister during the first years after encapsulation, has been studied (Spahiu 2021). The models used to calculate how oxidants are formed and consumed in a sealed canister in the presence of water, air and argon (Jonsson 2021, Henshaw and Spahiu 2021) show that iron corrosion in the canister rapidly consumes remaining water and that anoxic iron corrosion produces hydrogen gas and ammonia. The modelling activities serve as a basis for the strategy for management of the leaking fuel in Clab's storage pools and are based on an optimisation of the greatest possible benefit with regard to reducing the risk contribution to safety after closure and reducing the risk for personnel in connection with fuel repairs.

The long-term inspection programme, which is part of the ageing programme, includes for example MOX fuel from Germany and fuel from the Ågesta reactor. The results of the programme have so far not shown any ongoing ageing phenomena in these fuels. In addition, test coupons of borated stainless steel and of stainless steel, of the same type that is used in load-bearing parts of fuel assemblies, have been studied. Gaps and welds in these test coupons have been examined by means of metallographic testing and optical microscopy and no corrosion has been found.

Through Vattenfall's cooperation with the Electric Power Research Institute, Inc (EPRI), SKB now has access to experience concerning fuel degradation from EPRI's database of fuel-related issues.

Studies of failed fuel have been initiated at Studsvik during the past RD&D period. The purpose is to learn more about this type of non-regular fuel so that it can be managed less conservatively in the assessment of post-closure safety. Although it is a relatively small quantity, it has a noticeable negative effect on the safety assessment, since parts of the failed fuel dissolve much faster than intact fuel. Some interim results from these studies have been reported, but since the investigations continue, open access publication of the results is not expected until the RD&D period covered by this programme. Another study aimed at better describing how fuel with damaged cladding is affected by prolonged water contact, for example during storage in Clab, has been carried out on old fuel included in the "Series 3" experiment at the Studsvik site. The results are described in greater detail in Section 7.7.

In order to be able to assess the risk contribution of failed fuel to the assessment of post-closure safety, an estimate has been made of how quickly the oxidised fraction of the fuel dissolves in a repository environment. The estimate is based on already published scientific studies (Evins and Hedin 2020).

Both failed fuel and MOX fuel were studied in the EU project DisCo. Optical microscopy of fuel from an old gas reactor, which had been kept in a storage pool in Sellafield for about 40 years, showed surprisingly little impact (O'Neill et al. 2019). More investigations were planned to confirm this, but were delayed due to the Covid-19 pandemic. Used MOX fuel was studied at two laboratories in Germany (KIT-INE and JRC Karlsruhe), while unirradiated MOX fuel with a high concentration of plutonium was studied in France. The results show that spent MOX fuel dissolves in a similar manner and at similar rates as regular uranium oxide fuel (Herm et al. 2020) and that an anoxic repository environment with corroding iron also keeps unirradiated MOX fuel with high alpha radiation reduced (Kerleguer et al. 2020).

Programme

- A systematic compilation of the status of fuel types with structure weakness will be carried out, since these types may create problems during management.
- A project has started that will provide greater knowledge of different degradation mechanisms in fuel and structural materials as a consequence of its handling during the fuel's entire life cycle. In particular, the project will look at the Ågesta fuel and MOX fuel.
- Update of the ageing programme for nuclear fuel as a result of ongoing work to investigate degradation effects.

- Follow-up of experience from the operation of the nuclear power plants in Sweden and internationally via participation in international conferences, such as the IAEA groups for collaboration and the new OECD-NEA project Scip VI (Studsvik cladding integrity project). Experience of handling nuclear fuel is also gained through contact with representatives of the Swedish nuclear power plants and the nuclear power industry in Sweden and internationally.
- Development of a model for radiolysis of remaining liquids and gases in a sealed canister is planned to support the work on requirements for drying process and fuel handling.
- Studies of fuel samples which have undergone leaching in oxygenated environments are planned with Studsvik AB to investigate how oxidation has affected the uranium dioxide matrix. The samples include doped fuel and the fuel samples that have leached for about 37 years in oxygenated water ("Series 3").
- Further studies of failed fuel at Studsvik are planned to gain a better understanding of how this fuel has been affected by water contact in the reactor and in connection with the subsequent pool storage.
- Investigations of different minor fractions of non-regular fuel are planned for the purpose of determining properties, requirements and contributions to risk in the assessment of post-closure safety.

7.2 Fuel characterisation, decay heat and radiation

The fuel 's decay heat is a key parameter that must be known in the different steps of the nuclear fuel cycle, such as transportation, interim storage, encapsulation and final disposal, because it gives rise to increases in temperature. The decay heat is often the limiting factor for how much fuel can be placed in transport casks, storage pools and copper canisters, which means that it is of significant importance both in terms of safety and in terms of operational and economic importance in both the short and long term (Sjöland 2020).

During fuel characterisation, all relevant available information in the form of fuel properties and operational history, computer codes and measurements is used. The radionuclide inventory of the fuel assembly is calculated, and on the basis of this the multiplicity (linked to criticality), radiation field and decay heat of the fuel can be obtained. The fissile substance content of the fuel is also determined, which means that the fuel can be verified with respect to nuclear safeguards. There are requirements stating that all fuel assemblies must be verified prior to encapsulation, which is the prerequisite for planning for developing understanding, methods, computer codes and measuring instruments. The Spent Fuel Repository is special in the sense that the fuel can no longer be inspected. The last time this will be possible is in connection with encapsulation.

Current situation

The second of three phases of a fuel characterisation project was concluded in 2021. Methods and instruments, including detectors, have been developed here for use in the encapsulation plant's measurement station for the fuel (Perrey et al. 2021, Bengtsson et al. 2022, Favalli et al. 2016, Jansson et al. 2020a, b).

A large number of measurements have been made in the project for the set of fuels called SKB-50. The set includes 25 PWR and 25 BWR fuels that have been characterised in a number of ways, for example calorimetrically (with the calorimeter in Clab) and by gamma and neutron measurements. SKB-50 constitutes an international benchmark, which is also used in the EU programme Eurad in cooperation with American research institutes, in activities within NEA and EPRI cooperation.

The methods developed by the Los Alamos National Laboratory, the Differential Die-Away Self-Interrogation (DDSI) and Differential Die-Away (DDA) methods (Thompson et al. 2022, Trellue et al. 2021, Trahan et al. 2020), have been tested with good results on a full scale in Clab. The methods constitute an important basis for the characterisation system in Clink, above all within validation linked to nuclear safeguards (Lindgren et al. 2019).

Blind testing with the purpose of comparing codes for and measurements of decay heat has been carried out in cooperation with the OECD-NEA and around 25 international organisations with virtually all codes for calculation of fuel properties in the world. Results from the different codes have been compared with each other and with calorimetric decay heat data, where the participants have only received basic operational information for five different spent nuclear fuels (Jansson et al. 2022a). The conclusions from the blind test show that more and better experimental data would facilitate better determination of decay heat. It was also clear that the degree of detail in the operational data for the fuel, as well as the simplifications that are made, affect the results of the decay heat calculations, something that needs to be studied further in the future.

Important questions linked to decay heat are the temperatures that are reached in different parts of the KBS-3 system. The temperature often limits e.g. how much fuel can be placed in a canister or transport cask and has a great impact on the possibility of optimising the KBS-3 system. The capacity to carry out thermal modelling within SKB has been strengthened.

Within the radiation field, the radiation of the fuel and its effect on the material properties of the barriers in the final repository have been investigated by means of Monte Carlo simulations of neutron and gamma radiation from different common types of BWR and PWR fuels, conducted by Uppsala University (Jansson et al. 2022b). See also Section 8.1.4 concerning the effects of radiation on the canister.

Since 2019, SKB has led the Spent Fuel Characterisation (SFC) work package within Eurad. SFC is divided into four areas: 1) Administration and coordination, 2) Fuel properties and related uncertainty analyses, 3) Fuel behaviour after removal from the reactor and 4) Accident scenarios and consequence analyses. The websites https://www.ejp-eurad.eu/ and https://cordis.europa.eu/project/id/847593 describe the project in detail (Hernandez-Solis et al. 2021, Papaioannou et al. 2020, Ma et al. 2020, Shama et al. 2021, Schillebeeckx et al. 2021, Király et al. 2021, Rochman et al. 2021).

Residual heat calculations for Clab and for the transport system are done with the SNF code (Studsvik Nuclear Fuel) that considers the real history of the fuel. This tool has shown good conformity to measurements. SKB has an ongoing collaborative project with EPRI, which aims to increase and improve the validation data that exists regarding decay heat.

Programme

- A new phase (the third) is started in the development of fuel characterisation methods and instruments. The purpose of the work is to develop industrially applicable methods and build a full-scale prototype for characterisation of fuel prior to encapsulation. Future testing and verification of the methodology is planned to be carried out in Clab. The project will also develop verification methods that are suitable for non-regular nuclear fuels, such as fuel from the Ågesta reactor and MOX fuel, as well as the new casks for management of failed fuel from the nuclear power plants. Guidelines will be prepared for which uncertainties need to be associated with fuel where there is no information.
- Coordination and harmonisation of issues relating to nuclear safeguards is continuing on the basis of the Swedish policy, which, among other things, is based on SKB carrying out extensive measurements of the fuel assemblies before encapsulation, as well as on robust monitoring systems after encapsulation up to insertion into the Spent Fuel Repository. The policy is described in more detail in af Ekestam et al. (2018).
- In the area of fuel characterisation, the well-established collaborations with various international groups and organisations continue, not least the collaboration with the U.S. Department of Energy (DOE) through Los Alamos National Laboratory and Oak Ridge NL. In Sweden, the collaboration with Uppsala University and Lund University (where SKB holds an adjunct professorship) continues.
- Continued participation in SFC, which contributes to the development and evaluation of methods for fuel measurement and determination of fuel parameters. The project will run until 2024. Discussions on a continuation through Eurad 2 take place at European level.

- SKB will actively participate in the newly formed working group concerning determination and evaluation of decay heat. Within the IAEA Coordinated Research Project (CRP), the SFC project continues under the chairmanship of SKB.
- Thermal analyses will be carried out in the vicinity of the fuel and the canister, including transport casks. This will also include development of the EPRI benchmark for measurement of temperatures on the fuel surface and analysis of prudence in calculations.
- Capacity to carry out its own thermal modelling has been established within SKB, and modelling of different cases relevant for Sweden's back-end and the KBS-3 system will continue during the period. The temperature evolution of the canister is described in Section 8.2.4.
- Continued work with EPRI concerning validation of decay heat and reconstruction of existing calorimeter in Clab.

7.3 Fuel information and fuel selection optimisation for encapsulation

The composition of the fuel is the starting point for all calculations of radioactivity, criticality, decay heat and radionuclide inventory that are required for fuel that is to be transported, kept in interim storage and disposed of. In order to be able to make correct calculations, detailed information is needed on the fuel's geometry, material, initial enrichment and operational history.

The degree of detail of the fuel information needed for verification of fuel properties depends on the measurement methods and precision requirements required in the different steps of the transport of the fuel from the nuclear power plants to interim storage and final disposal in canisters. The more detailed the fuel data that can be obtained, the more exact calculations can be carried out, which means that the allowance for different types of uncertainties can be minimised.

Current situation

During the past RD&D period, SKB has worked to identify and gather the fuel information that is needed for management and final disposal in the KBS-3 system. The work has also included quality assurance of the fuel information that already exists in the databases. This is carried out partly through automated comparisons with data from the nuclear power plants, and partly by going through fuel documentation from the fuel suppliers and verifying it against the information in the database.

In the case of the older fuels, the documentation often contains less detailed data. In some cases, there is also no detailed operational data saved at the nuclear power plants. Guidelines need to be developed for which uncertainties should be associated with such fuels.

The information on the spent fuel needed for safe management and final disposal includes detailed information on the operational history, geometry, material and nuclear design (including axial distribution) of each fuel assembly. With this information, detailed calculations can be carried out in respect of e.g. nuclide inventory, decay heat and reactivity. For all fuel, it is also important to save information on reconstructions, damage and observed weaknesses in materials, since this information is needed both for nuclear safeguards and for the practical handling in Clink. This collection and quality assurance of fuel data is also a part of the work being done to ensure the preservation of important information for future generations – see Section 4.13.1.

The decay heat in individual fuel assemblies affects how the selection of fuel assemblies to be placed in a particular canister is carried out in order to minimise the number of canisters (encapsulation optimisation). Other parameters that must be included in analyses to optimise the number of canisters are the encapsulation rate, the start date of encapsulation and the requirement to minimise the number of transfers of fuel in the interim storage facility. In the selection and placement of fuel assemblies, criticality and radiation levels on the outside of the canister need to be taken into consideration. An algorithm and programme for encapsulation optimisation, based on available fuel information, will be developed by ongoing projects and the result will be available at the time of encapsulation.

Programme

- Continue the work of collecting and performing quality assurance of fuel information from nuclear power plants and fuel manufacturers.
- Different computational programmes will be linked to the fuel database, so that calculations can be based on quality-assured data from the power plants instead of estimated data. This applies to calculations of radionuclide content, radiation, decay heat and criticality.
- Develop and evaluate algorithm for fuel assembly selection in encapsulation.

7.4 Acceptance criteria for fuel

SKB currently applies acceptance criteria for reception and interim storage of fuel in Clab in the form of two requirements and criteria documents. Compliance with requirements is verified on a step-by-step basis and the process for this is described in SKB's management system. The acceptance criteria include all the steps involved in SKB's handling of fuel, i.e. criteria regarding transport, interim storage, encapsulation and final disposal. The criteria are based on the safety assessments that have been carried out for each step. The acceptance criteria ensure that the fuel fulfils the properties required for it to be managed and disposed of in a safe manner in all steps up to final disposal and after closure of the Spent Fuel Repository.

A number of acceptance criteria for spent nuclear fuel and copper canisters have been identified for both encapsulation and the Spent Fuel Repository. As the details of facility layout are concretised and their safety assessments are updated, it will also be possible to specify the requirements.

Regulation SSMFS 2021:7 of the Swedish Radiation Safety Authority (SSM) concerning management and disposal of nuclear waste includes requirements on documentation of nuclear waste. Some of the requirements relating to spent nuclear fuel introduced in SSMFS 2021:7 are new. This means that there is a need to study what the overall documentation concerning the fuel and copper canister will look like in order to meet applicable regulatory requirements and to meet other needs regarding documentation in the organisation. This also includes design of acceptance criteria for transfer from interim storage to encapsulation and for final disposal in the Spent Fuel Repository.

Regulation SSMFS 2022:1 (amending regulation SSMFS 2008:1) states that acceptance criteria must be included in the SAR in accordance with Chapter 4, Section 2, Annex 2 SSMFS 2008:1. Since the requirement and criteria documents which today constitute acceptance criteria for receiving fuel in Clab are not covered by the SAR for Clab, SKB has applied for a time-limited exemption for this until 1 January 2024.

Current situation

During the past RD&D period, SKB began work on compiling acceptance criteria for reception and interim storage of fuel in Clab according to the same structure that is used for acceptance criteria in the Final Repository for Short-lived Radioactive Waste (SFR), and for introducing them in the SAR. The work also includes reviewing procedures linked to acceptance criteria, for example with regard to the documentation of derivation, safety review of acceptance criteria and receiving inspection.

Industry-wide work was initiated during the past RD&D period within the nuclear power industry's safety coordination group (KSKG) to study how the overall documentation on the fuel and copper canister should be designed to meet applicable regulatory requirements according to SSMFS 2021:7 and to meet the other needs regarding documentation that exist in the organisations. The work also includes looking at the link between acceptance criteria and other documentation.

Programme

- Continued work on formulating acceptance criteria for receiving and interim storage of fuel in Clab according to the same structure as is used for acceptance criteria in the Final Repository for Short-lived Radioactive Waste (SFR) and on introducing these in the SAR for Clab.
- Continued industry-wide study of how the overall documentation concerning the fuel and copper canister, including connection to acceptance criteria, should be designed to meet applicable regulatory requirements pursuant to SSMFS 2021:7 and meet the other needs regarding documentation that exists in the organisation.
- Compile acceptance criteria for the activities planned at the coming facilities Clink and the Spent Fuel Repository.

7.5 Criticality

For criticality calculations, SKB applies burnup credit for PWR and BA credit for BWR. Burnup credit entails crediting the decrease in reactivity that occurs when the fuel is irradiated in the reactor and burnup increases in the criticality analyses. BA credit refers to consideration of the fact that the fuel contains a so-called burnable absorber (BA), mainly gadolinium-155, which is a strong neutron absorber and thereby reduces the reactivity of the fuel.

SKB ‘s programme for criticality safety continuously develops the methodology used and validates new versions of computational programmes.

Current situation

If the integrity of the canister is lost and water enters, a number of corrosion processes will start. These will affect the internal materials and their geometries. SKB is currently working on developing previous work (Agrenius and Spahiu 2016) on how this occurs and what impact it will have on reactivity. The previous work has now been supplemented with a study of the long-term stability of different fission products and other nuclides that are credited in the criticality analysis. It is particularly important that gadolinium has been shown to remain in uranium dioxide for millions of years (Hidaka and Holliger 1998), and that other elements such as molybdenum, technetium, ruthium and rhodium are released congruently with uranium dioxide when the spent fuel leaches under conditions similar to the repository environment (Ekeroth et al. 2020).

The international standards and guidelines that SKB has used in its development of the methodology for criticality analysis do not fully cover all aspects that must be considered in the very long-term process that applies to the Spent Fuel Repository. SKB is therefore participating in an international collaboration on Post Closure Critical Safety (PCCS) as a part of the collaboration within Implementing Geological Disposal of Radioactive Waste Technology Platform (IGD-TP).

For PWR fuels, the criticality analysis for the Spent Fuel Repository states a minimum burnup as a function of enrichment which must be achieved in order to place the fuel in a canister. Not all fuel assemblies will have achieved this burnup, e.g. one-year fuel from the Ringhals 2 final core will not meet the current requirements. SKB’s methodology for criticality analysis must be developed in order for these fuels to be managed and disposed of. This can preferably be done by performing canister-specific calculations. This means that instead of using a conservatively assumed burnup and operational history for all constituent fuel assemblies, the calculations are based on the actual operational history taken from the core follow from the nuclear power plants.

If detailed fuel information is used to make canister-specific calculations of criticality, high demands are made on the accuracy of the fuel information used. The work on this is described in Section 7.3. It will also be of further importance to be able to show that the correct fuel assemblies are selected and that each assembly has the characteristics that are expected. The development work with verification of this is described in Section 7.2.

Programme

- Further development of the methodology for criticality analysis regarding the aspects that are unique to the Spent Fuel Repository. The work includes:
  - Selection of experiments for validation of computer codes. At present, few experiments are available in the OECD-NEA database that satisfactorily cover the specific conditions in the Spent Fuel Repository.
  - International cooperation, for example to evaluate which scenarios should be considered in the criticality analysis for extremely long time periods.
  - Effects of possible geometry and material changes in connection with the development of the canister insert.
- Complete the work of developing methodology for criticality analysis of fuel that does not meet today's burnup criteria.
- Continued method development for canister-specific criticality calculations.

7.6 Nuclear safeguards

The application of nuclear safeguards must ensure continuous knowledge of the nuclear fuel. This entails inspections and verification to ensure that the spent nuclear fuel does not go astray. This means that the development of nuclear safeguards for the KBS-3 system mainly concerns the following areas:

- Method for verification of declared nuclear material.
- Logistics.
- Identification of copper canisters.
- Seal.
- Management of abnormal events.

The development work will be carried out cohesively so that the bigger picture concerning nuclear safeguards is considered, in order to ensure continuous knowledge of nuclear material in the KBS-3 system.

Current situation

SKB has contributed to the European Commission's research and development of methods for marking and identification of copper canisters. The assessment is that engraving of identity marking on the canister lid is the most reliable method and is acceptable in the production environment. Continued development will focus on technology for secure reading of the identity marking. Marking of the canister with engraving constitutes the reference design.

Reversal of canisters from the final disposal facility to Clink will be possible with the same equipment and systems that are used in normal operation. It must be possible to handle the bookkeeping of the transfers in a reversible manner in the system for nuclear safeguards.

SKB has submitted supporting data to the European Commission, which has then submitted proposals for seals for use on the canister transport cask. The details will be developed and specified in conjunction with the development of the transport cask.

Through ongoing reconciliation in the design work with respect to nuclear safeguards, the facilitation of nuclear safeguards in the facilities is considered. Examples of areas covered are the transport system, logistics and transfer routes for the fuel and the canister.

Programme

- Methodology for verification of nuclear material will be developed, linked to the development of fuel measurement (Section 7.2) and fuel information (Section 7.3).

- The development of the methodology for combining fuel data with characterisation measurements for verification against declared nuclear material is under way and will continue. Development of the monitoring station layout and the methodology and technology for nuclear material verification is also in progress in this area.
- A logistics study with respect to nuclear safeguards for finished copper canisters in Clink will be prepared in conjunction with plant design and will be discussed in cooperation forums with the control bodies.
- Development of the identification method for copper canisters will focus on secure reading of engraved identity markings and more precisely on how the marking should be done and where in the process it is most appropriate to do it.

7.7 Fuel dissolution

Fuel dissolution leads to release of radionuclides, and the process is essential for the consequence analysis included in the post-closure safety assessment. Research in this area is being conducted to gain a better process understanding of fuel dissolution, mainly in reducing environments, but also in oxygenated environments. The fuel consists of uranium dioxide pellets enclosed in cladding and construction materials. All parts may release radionuclides if water comes into contact with the fuel. This takes place at different rates depending on where in the fuel the radionuclides are located. Some radionuclides are located in the gap between pellet and cladding and in fractures and grain boundaries in the fuel pellets. This part of the radionuclide inventory is called the gap inventory (or the instant release fraction, IRF). The gap inventory is expected to be released rapidly upon water contact. The majority of all radionuclides in the fuel are released at the rate of dissolution of the fuel matrix, i.e. the uranium dioxide that constitutes the fuel pellet itself. Research on fuel dissolution focuses on matrix dissolution and release of the gap inventory, but also includes corrosion of cladding and other metal alloys in the fuel.

Current situation

During the past RD&D period, a number of research projects in Studsvik have continued and have been concluded in full or in part. The task of long-term leaching in ampoules is under way, but further work on analysis and interpretation of data is required before the results so far are ready for publication.

One Studsvik assignment concerns closure and analysis of the long-standing fuel leaching project "Series 3", which started in 1982. In this project, pieces of fuel were leached in flasks with air contact, and the experiments thereby yielded valuable data on oxidative fuel dissolution and radionuclide release in an oxygenated environment. The results from the final sampling of "Series 3" are ready, but it remains to compile all data from 1982 onwards before a summary article or report is ready for publication. However, certain additional studies have been conducted on these fuel samples for the purpose of investigating how much of the samples has been transformed to secondary phases during the approximately 37 years that the two samples were leached. These results are provided by Roth et al. (2021) and show that secondary phases were formed mainly on the sample leached in deionised water, while the sample leached in synthetic groundwater (containing carbonate) had minimal secondary phases, which was expected. The secondary phases that were found were mainly studtite and metaschoepite, and they contained, in addition to uranium, a certain amount of radionuclides such as iodine and caesium (Roth et al. 2021).

Data from fuel leaching experiments in an oxygenated environment, which does not correspond to the expected repository environment, can provide relevant information for determination of the gap inventory as well as an understanding of what can affect oxidative fuel dissolution and release of radionuclides. During the past RD&D period, data from a number of such experiments have been compiled and analysed (Roth et al. 2021), with the conclusion that the selected sample preparation can affect how quickly different radionuclides are released from the samples. Specifically, it seems that the results may vary depending on whether it is a segment of a fuel rod or fragments of fuel that is leached.

An experiment with chromium-doped fuel has also been conducted to investigate the gap inventory in particular. The experiment was carried out in an oxygenated environment with fuel fragments, which proved to have a very small grain size (Barreiro Fidalgo et al. 2021). The results show that even if the gap inventory is at similar levels as normal, non-doped fuel, it appears that the fraction of iodine, molybdenum and technetium is a little higher than for standard fuel. However, this may be a result of the fine grain size of the sample. Another experiment, aimed at determining the fraction of radionuclides in the grain boundaries of the fuel (Fidalgo et al. 2019), was carried out via grinding and simultaneous leaching. The idea was to grind to grain size, so that the grain boundaries were exposed. This proved to be difficult, however, and the results probably reflect an increased release rate from the pulverised fuel matrix.

For fuel dissolution under repository-like conditions, experimental results have shown that hydrogen gas counteracts oxidative fuel dissolution (Ekeroth et al. 2020). An important observation from the experiment is that the surface of a fuel sample leached under hydrogen gas consists of uranium dioxide and not higher oxides. Previously obtained data from leaching of an American high-burnup fuel under hydrogen have been analysed and published (Puranen et al. 2022). The results show that hydrogen gas can effectively counteract oxidative fuel dissolution even for fuels with a burnup of 70–75 MWd/kgU.

Studies included in two now completed PhD projects at KTH (Maier 2019, El Jamal 2021) have been reported in open access scientific literature. A licentiate dissertation from Chalmers (Hansson 2020) with associated articles has also been published. The PhD projects are aimed at gaining greater knowledge of the processes and mechanisms that regulate oxidative fuel dissolution, such as catalytic decomposition of hydrogen peroxide and the hydrogen effect. During the period, new SKB-funded PhD projects have also started at KTH (one fully financed and one partially financed) and at Chalmers (fully financed). For details of the research conducted by the PhD students, see published scientific articles (Maier et al. 2020a, b, Li et al. 2019, 2021, Hansson et al. 2020, 2021, El Jamal et al. 2021a, b, c, d). In summary, the results have improved the fundamental understanding of how the redox reactions that take place on the surface of uranium dioxide affect the dissolution of uranium in different environments. This understanding permits further work on developing a model for fuel dissolution in the repository environment.

Within the framework of the now completed EU project DisCo, matrix dissolution of spent fuel and different fuel dioxide variants was studied. The purpose of the project was mainly to investigate whether doped fuel, i.e. fuel with additives of, for example, chromium or aluminium and MOX fuel, dissolves in a similar way and at a similar rate as standard fuel. The experiments included were mainly carried out in a reducing environment, but other types of experiments were also carried out to gain a better understanding of oxidative fuel dissolution and how different additives affect uranium dioxide. Project results have been reported on a continuous basis via reports (https://cordis.europa.eu/project/id/755443/results), but also via scientific articles (Evins et al. 2020, Fidalgo et al. 2020, Riba et al. 2020, Cordara et al. 2020, Curti and Kulik 2020, Miléna-Perez et al. 2021, Cachoir et al. 2021, Kegler et al. 2021). An important conclusion from the project is that the slow dissolution of the fuel matrix in the modern fuels studied (Adopt and MOX) does not differ significantly from standard fuel. This is supported by both fuel experiments and experiments with analogous materials. Data from different experiments in the project have also been used for development of models of fuel dissolution, with successful results. However, there are some remaining questions regarding the effects of the repository environment (for example, the presence of hydrogen and iron). Some fuel experiments were delayed due to the Covid-19 pandemic, which means that certain results are not expected to be available until during the RD&D period.

During the past RD&D period, efforts have also been made to study how the environment in a fuel-filled and sealed copper canister can affect the properties of the fuel cladding. This has been investigated above all to estimate whether hydrides can be expected to cause fractures in the fuel cladding, so that the fuel matrix risks coming into contact with the gases and liquids in a sealed canister. It is estimated that approximately three per cent of the fuel may have failed fuel cladding after the first warm period in the copper canister (Evins 2020).

Programme

- Ongoing PhD projects at Chalmers on how hydrogen and alpha radiation at a uranium dioxide surface affect radiolysis and oxidation of uranium will be concluded at the end of 2022. A new PhD project has been started, with a different focus – see Section 7.8.
- An ongoing PhD project at KTH on the effects of water radiolysis on dissolution of uranium dioxide will be concluded in 2023. A new PhD project concerning mechanisms and kinetics for chemical processes relevant for radiation-induced dissolution of spent nuclear fuel started in autumn 2021 and will continue until 2025.
- Development of a computational model for fuel dissolution is being carried out in cooperation with KTH and in parallel with the PhD project. This assignment is expected to continue until 2025.
- An ongoing assignment with Studsvik aimed at investigating how fuel in closed glass ampoules dissolves will continue until 2026. The last ampoules will be stored, pending opening in 2026.
- Closure of ongoing assignments and start-up of new assignments regarding doped fuels will be carried out with Studsvik. Leaching of chromium-doped fuel in an autoclave will be started and data from completed experiments with two different chromium-doped fuels will be compiled and published in scientific articles.
- Further studies concerning the Instant Release Fraction will be started. The focus of this is the grain boundary inventory, which in a first phase is planned to be investigated with Studsvik's laser ablation instrument.
- Work together with Studsvik on data compilation concerning the now completed experiment "Series 3". This data collection will be carried out for documentation and modelling purposes.
- Leaching tests with fuel in the presence of sulphide are planned to investigate whether sulphide can affect the redox reactions at the fuel surface.

7.8 Radionuclide speciation and solubilities

This section describes the research activities being carried out to gain a better understanding of the processes that affect the radionuclide chemistry in a broken copper canister. When radionuclides are released from the fuel, their concentration in the water increases, and the expected concentration depends on the solubility and speciation of the element. The concentration in solution is limited by the solid phase with the lowest solubility. Knowledge in this area is based on thermodynamics and the use of thermodynamic data, which are collected in databases for use in modelling tools.

Current situation

Uranium chemistry is a central issue for the Spent Fuel Repository. In the reducing repository environment, two minerals are expected to be stable: uraninite (uranium dioxide) and coffinite (uranium silicate). During the past RD&D period, an international project regarding coffinite formation was concluded and results from the project are available in open access scientific literature (Szenknect et al. 2020). Data from oxygen-free, room-temperature experiments with uranium dioxide and silicate in aqueous solution show that coffinite can form on the surface of a uranium dioxide pellet in a system that is kept at room temperature and a negative redox potential (Eh around −50 mV, Szenknect et al. 2020). At this redox potential, oxidised uranium species in aqueous solution are stable, and coffinite formation in the experiment probably occurred via oxidation followed by reduction. Since the expected redox potential in the repository environment is lower than that achieved in the experiments, the implications for the final repository needs some further studies.

An experiment to investigate the effects of possible calcium-uranyl-carbonate complexes in groundwater of the Forsmark type has recently been carried out within the framework of a PhD project at Chalmers. The question is whether formation of this complex can prevent the reduction of uranium in a reducing system. When the results are compiled, they will be published in open access scientific literature.

The EU project DisCo has also involved questions concerning speciation and solubilities. In the modelling part of DisCo, thermodynamic data have been used to model the experimental systems. Results from Riba et al. (2021) show that the model, which includes thermodynamic calculations, can simulate measured radionuclide concentrations and provide indications of which redox pair controls the redox potential in the system. Another modelling project within DisCo involved unirradiated MOX fuel in a clay-based final repository, and the results showed that the concentrations of uranium and plutonium were controlled by their tetravalent, amorphous phases, and that iron reacted with the oxidants and precipitated as the minerals magnetite and chukanovite (De Windt et al. 2021).

Quality-assured thermodynamic data for repository relevant elements are required in order to be able to carry out solubility calculations for the Spent Fuel Repository and use the chemical models where these calculations are included. SKB has therefore participated in the NEA-TDB project for many years. The importance of this project for final repository programmes around the world is highlighted by Ragoussi and Costa (2019). In the current phase, NEA-TDB-6, an electronic database has become available (Martinez et al. 2019) and two volumes of reviewed thermodynamic data have been published: an update for actinides (Grenthe et al. 2020) and one on iron (Lemire et al. 2020).

Programme

- A PhD project at Chalmers with a focus on actinide chemistry and possible co-precipitations in the repository environment was started in the autumn of 2021 and is expected to continue until 2026.
- Parts of an ongoing PhD project at Chalmers, which is expected to be concluded in 2022, are investigating how complexes with UO_2^{2+}, Ca and CO_3^{2-} can affect the reduction of UO_2^{2+} in the expected final repository environment.
- SKB will also continue to participate in the ongoing NEA-TDB project, and a number of new publications are expected in the coming years.
- A summary review of the knowledge on coffinitization that was obtained within the completed coffinite project will be carried out.
- Research activities concerning solubilities, speciation and precipitation take place within Eurad, specifically within the work package Fundamental understanding of radionuclide retention (Future). SKB is not actively participating in the work package, but relevant issues, for example Ra-Ba co-precipitation, will be monitored via Eurad.

8 Canister for spent nuclear fuel

In the Spent Fuel Repository, the spent nuclear fuel will be stored in canisters consisting of a cylindrical copper shell with a load-bearing insert. The canister is 4.84 m long and has a diameter of 1.05 m, and the thickness of the copper shell is 5 cm. The copper shell protects against corrosion in the repository environment and the insert provides protection against the mechanical loads in the repository. The canister with its insert constitutes the most important barrier in the KBS-3 system in that it must contain the spent nuclear fuel for a sufficiently long time for the radioactivity of the radionuclides to decay until they no longer pose a risk to humans or the environment.

This chapter describes the current situation and the supplementary research that is planned regarding the properties of the canister for future SARs for the Spent Fuel Repository. Furthermore, it describes the technology development that is needed for the canister to be manufactured, inspected and verified against specified requirements, and used in the KBS-3 system.

8.1 Corrosion

There are only a few situations in the Spent Fuel Repository that can cause corrosion of the canister. Some initial corrosion is to be expected from the amount of oxygen trapped in the bentonite pores together with oxidative species formed by the ionising radiation of the nuclear fuel during the first hundred years of the repository. Sulphide corrosion, which is expected to be the most important corrosion process in the long term, is limited by a low supply of sulphide and is characterised mechanistically by mainly general corrosion instead of localised corrosion (pitting). Sulphide corrosion will therefore continue to be studied, together with different mechanisms for stress corrosion cracking. In order to analyse the durability of the copper material as a barrier, modelling to characterise the corrosion process over time and space will also need to be developed.

In its statement to the Government on permissibility pursuant to the Swedish Environmental Code for a final repository in Forsmark, the Land and Environment Court requested additional supporting data on a number of research questions regarding the integrity of the canister. In order to comply with this request, SKB presented a report (SKB TR-19-15) in 2019 that describes in detail the results of the research conducted in the requested areas after SR-Site (SKB TR-11-01) was published. This report will in future be referred to as the Canister supplementary report.

All activities presented under “Programme” in this section are aimed at strengthening the supporting data for future post-closure safety assessments for the Spent Fuel Repository and at reducing the conservatism in a number of cautious assumptions made in the assessment.

8.1.1 Sulphide corrosion

In the long-term perspective, sulphide is the most significant corrosive agent for copper in the repository environment. In the SR-Site safety assessment, canister failure due to corrosion by sulphide in cases where the buffer had eroded away, constituted the dominant risk contribution. The scientific basis for sulphide corrosion in the final repository environment is extensive, as can be seen from e.g. the Canister supplementary report. In order to be able to reduce conservatism in a number of cautious assumptions in future safety assessments, a better understanding is needed of the detailed mechanisms in the corrosion process.

Current situation

The formation of a copper sulphide film on copper in sulphide solution, and the mechanisms for this have continued to be studied by electrochemical methods, different types of microscopy, and in corrosion experiments. Previously, the focus of the research has been the question of what limits film growth, which has been shown to be sulphide fluxes to the copper surface. Presently, the focus has shifted towards issues concerning passivity and properties of the sulphide film, as well as conditions for localised corrosion. A passive film can be a prerequisite for localised corrosion, while a corrosion film without passivating properties normally gives rise to more even, general corrosion limited by the supply of corrosive agents. SKB's work in this field is mainly being carried out at the University of Western Ontario (UWO) in Canada.

The studies with electrochemical methods of mechanisms for film growth have been expanded with experiments with other groundwater ions present (Martino et al. 2020). SKB is also monitoring the studies being conducted within the framework of the Canadian programme to study sulphide corrosion under conditions of relevance for the Canadian repository, for example at a high chloride ion concentration (Senior et al. 2021).

In order to gain a better understanding of whether some form of localised corrosion can occur in connection with corrosion of copper in sulphide solution, SKB has commissioned studies under varied experimental conditions. These have been summarised in the Canister supplementary report, where detailed results and appurtenant analyses are also found. The studies reported therein includes studies of bacterial formation of a biofilm, which can act like a passive film and cause localised corrosion. The general conclusion was that in all experiments where biofilm and/or localised corrosion from bacterial activity was observed, the concentration of sulphide in the solution was higher than the highest concentrations measured in the groundwater in Forsmark. The sulphide flux in the experiments was also higher than that expected in the final repository (since the experiments were done without bentonite that limits the transport of sulphide to the copper surface), and therefore such observations cannot be directly applied to the canister in the final repository. In order to enable characterisation by microscopy of the copper surface under the formed sulphide film, a method is required to remove the film with as little impact on the underlying copper surface as possible. From the initial studies of how the sulphide film can be removed (Chen et al. 2019), an improved method has been developed, which is intended to be reported as a publication in a peer-reviewed journal.

Whether the formed film is passive or not, has been the subject of previous studies and discussions, for example in the Canister supplementary report. Studies that SKB has subsequently carried out of the different layers that arise in connection with film formation under different conditions indicate that a passive film cannot arise at the corrosion potentials and sulphide concentrations (and therefore not the sulphide fluxes) that can be expected under repository conditions (Guo et al. 2019, 2020, 2021). This confirms the previously described transport-limited mechanisms for sulphide corrosion.

Quantum chemical calculations have been carried out to gain a better understanding of the electrochemical interface between copper, copper sulphide and groundwater (Halldin Stenlid et al. 2020a, b), and for the reaction mechanism in sulphide corrosion of copper (Halldin Stenlid et al. 2019). Furthermore, a study has been conducted to characterise differences in the growth mechanism and morphology of sulphide and oxide films in the early stages of the corrosion process (Halldin Stenlid et al. 2021).

A summary of the state of knowledge regarding sulphide corrosion, under both saturated and unsaturated conditions in the bentonite, is presented in the Canister supplementary report. The most important conclusion is that as long as the bentonite is intact, and regardless of its degree of water saturation, the sulphide fluxes through the clay system are well below the sulphide fluxes in experiments where local corrosion phenomena, such as micro-galvanic corrosion and superficial fracturing under applied tensile stress, have been observed (Section 8.1.5).

Programme

- Laboratory experiments will continue to be conducted at UWO, primarily electrochemical experiments to study sulphidation of oxidised copper, and corrosion exposures in order to systematically investigate the occurrence of localised corrosion, including micro-galvanic corrosion.
- SKB is part of a consortium (together with NWMO, Nagra, and ONDRAF/NIRAS) that conducts research at CanmetMATERIALS, Canada. The copper studies include exposures in different sulphide environments where corrosion is measured as hydrogen gas evolution. Within the consortium, SKB has also initiated studies of hydrogen penetration from sulphide corrosion.

- The long-term evolution of copper exposed to an environment that changes over time is being studied in the Michigan International Copper Analogue project (MICA), where the ongoing phase I includes an inventory of available copper samples from natural analogues and applicable methods for specimen analysis. The analyses are then planned to be carried out in a phase II. Within the framework of studies of natural analogues, SKB also follows the work carried out by the Canadian programme, where the present knowledge has been compiled in King (2021a).
- SKB plans to review the need to also study high chloride ion concentrations in combination with sulphide in the solution, the effects of gaseous sulphide, for example under unsaturated conditions, as well as the effects of a biofilm on the canister surface.

8.1.2 Corrosion under oxidising conditions

Corrosion under oxidising conditions primarily includes the general corrosion to which the canister is exposed before closure and in the repository until residual oxygen has been consumed, where the latter is expected to occur within decades after closure. In SR-Site, localised corrosion (pitting) under oxidising conditions is treated as uneven general corrosion (surface roughening) with a maximum additional corrosion depth, instead of the previously used pitting factors.

Current situation

The pore water compositions that are relevant in the repository favour general corrosion and generally do not give rise to a passive film (King and Lilja 2013). Such a film is a prerequisite for localised corrosion under oxidising conditions to occur. In order to assess the uncertainty and variability in the data to an even greater extent, SKB has begun the development of probabilistic models for localised corrosion. Modelling has so far been done for the systemically less complex case with oxidising conditions in saturated bentonite (Briggs et al. 2020, 2021). However these conditions are unlikely to occur in the repository since the time for resaturation in most deposition holes is expected to be longer than the initial oxidising period. The model shows the importance of available oxygen for passivity, but also for pit growth, which stops when oxygen has run out. A new source of supporting data is access to the database that NWMO in Canada has commissioned UWO to produce (summarised in Qin et al. 2017) and which contains a large number of measurements of electrochemical potential under different conditions (temperature, pH, concentrations of chloride, sulphate and carbonate). For further planned modelling studies, see Section 8.1.6.

Data from previous field experiments with copper in repository-like environments have been compiled and analysed with respect to general corrosion (mass loss) and to which environmental parameters co-vary with the corrosion depth (Johansson et al. 2019, Johansson 2019). Analysis of data from the MiniCan, LOT, ABM and FEBEX experiments shows that corrosion depth correlates with the estimated total amount of initial oxygen in bentonite and air-filled gaps in the near field.

Retrieval and analysis of copper components from the test packages LOT S2 and LOT A3 have shown corrosion which, in nature and scope, is in line with previous field experiments (Johansson et al. 2020). The observed corrosion products are mainly oxides, but small quantities of sulphur were detected on the surface, which in some cases could be assumed to originate from the formation of copper sulphide. The analysed copper surfaces exhibit a rough topography with pits and defects. It is uncertain how much of this topography originates from manufacturing and how much may have been caused by corrosion. Even if the pits were to be caused by corrosion, they are measured to be within the depths expected from previous field experiments and modelling of localised corrosion under oxidising conditions.

Analysis of general corrosion of copper within the framework of the Lasgit experiment (Cuss et al. 2010; see also Section 10.3.1) provides results consistent with previous field experiments. No signs of pitting were observed. Only isolated points with limited corrosion were found on the analysed copper surfaces, which could possibly be attributed to sulphide corrosion. (Wendel et al. 2022).

Experiments have been carried out at UWO in Canada to measure the electrochemical potential of copper in bentonite (i.e. in contact with bentonite pore water), and have been reported in a PhD thesis (Martino 2018) and in Martino et al. (2021).

Programme

- Ongoing laboratory studies of oxygen consumption rate and copper corrosion in unsaturated bentonite will be presented during the RD&D period. (For experiments with oxygen consumption in bentonite without presence of copper, see also Section 10.1.1.)
- As a complement to field experiments, SKB has initiated laboratory experiments for studies of copper–bentonite interactions in anoxic environment, to investigate how adsorption of copper affects the properties of the bentonite (Section 10.3.5).
- SKB is planning for excavation of the inner section within the Prototype repository at the Äspö HRL. Analysis and reporting of canister components from the Prototype tests are included in the plan for excavation.
- Excavation and analysis are planned for copper components from remaining test packages within the LOT, ABM and MiniCan experiment series (Sections 4.10.1 and 10.3.5).

8.1.3 Copper in pure water

The issue of copper corrosion in pure, oxygen-free water has attracted a great deal of attention for several years, since a group of researchers at KTH presented the view that the extent of this form of corrosion is considerably greater than established science predicts.

Current situation

As reported in the previous RD&D programme, SKB has studied the process in depth, without finding any support for the standpoint of the KTH researchers. In October 2021, he Swedish Radiation Safety Authority (SSM) stated (SSM 2021b) "that there is no evidence for the importance of this form of corrosion and no reasonable explanation as to why traditional thermodynamics are not applicable" (quoted text translated from Swedish by SKB). The Swedish National Council for Nuclear Waste stated in its state-of-the-art report in February 2022 (Swedish National Council for Nuclear Waste 2022): "Hultquist and co-authors have not shown that the hydrogen gas formed in their experiments correlates to the mass that the formed copper corrosion products should have." (quoted text translated from Swedish by SKB.)

SKB's work on developing a kinetic model for the reactivity of copper in anoxic aqueous solution has been completed with data from experiments in perchlorate solution. The model has then been used to extrapolate to an aqueous solution without added ions, with the conclusion that the measured hydrogen gas comes from a surface reaction limited by the number of available adsorption sites on the copper surface (Betova et al. 2021). Also Senior et al. (2021) comes to the conclusion that the very small hydrogen quantities measured in experiments with copper in water come from a reaction with a very limited amount of reactive surface atoms.

After the initial study to determine the possibility of formulating a "mixed-potential model" (King and Orazem 2017), the work is not being pursued further due to the scarce availability of kinetic data.

Programme

It is SKB's clear conclusion that there is no scientific support for the claim that copper corrodes in pure, anoxic water in any other way than what established science states, and that the issue is covered in the safety assessment for the Spent Fuel Repository. SKB therefore plans no further studies in the area.

8.1.4 Radiation-induced corrosion

Radiation-induced corrosion of the copper canister occurs due to the oxidising radiolysis products formed when water on the outside of the canister absorbs gamma radiation from the fuel in the canister. It is predominantly during the first few hundreds of years in the Spent Fuel Repository, that the dose rate on the canister surface will generate radiolysis of water to any significant extent. In the SR-Site safety assessment, it was estimated by theoretical means that this corrosion process could give an average corrosion depth of at most 14 µm, which is negligible, both in relation to the thickness of the copper shell and in comparison with the scope of other corrosion processes.

Current situation

After the study of the SR-Site, the continued effects of gamma radiation on corrosion of copper were studied in a PhD project at KTH and in the Canadian programme, described in depth in the Canister supplementary report. NWMO has published a summary of the state of knowledge regarding the effects of gamma radiation on copper corrosion, and concludes that the effect is minimal (a few ten μm) and therefore not a decisive factor for the life expectancy of the canister (King and Behazin 2021).

New experiments and analyses that SKB has commissioned from KTH have shown that, the relative importance of the different oxidants formed during radiolysis of water at the surface differs drastically between for instance uranium dioxide and elemental copper, and that different models therefore need to be used to describe the surface reactions for the two materials. The revised model for radiolytic copper corrosion can now predict the experimentally observed corrosion of copper under gamma irradiation, with relatively good precision (Soroka et al. 2021). The model has also shown that relevant groundwater ions in repository-like concentrations only have a minor effect on radiation-induced corrosion at neutral and alkaline pH (Jonsson 2021b).

Programme

The model that has been developed to describe radiolytic corrosion of copper and the influence of a number of groundwater components is being compiled and is intended to be published in a scientific peer-reviewed journal – see Section 8.1.6.

SKB is considering the prospects of proceeding with the studies of the effects of irradiation on the microstructure of the material that have been done in vacuum (Padovani et al. 2019) and also on Cu-OFP (oxygen-free, phosphorus doped copper). By performing irradiation in solution, the simultaneous effects of irradiation and corrosion on the material could be studied.

8.1.5 Stress corrosion cracking

For stress corrosion cracking to occur, a sensitive material is required, in combination with tensile stresses and specific ions. Stress corrosion cracking has been covered in previous safety assessments and has then primarily concerned corrosion under oxidising conditions in the presence of nitrite, ammonium and acetate (SKB TR-11-01). Since the necessary ions are not present in sufficient concentrations during the initial oxidising period in the repository, stress corrosion cracking has been deemed not to have an effect on the integrity of the canister.

The question of whether stress corrosion cracking may also occur in the presence of sulphide or not, has been discussed, particularly since a Japanese research group (Taniguchi and Kawasaki 2008) presented results that indicated such a process.

Current situation

Since the study by Taniguchi and Kawasaki (2008), four further groups have conducted experiments to investigate whether stress corrosion cracking can occur in a sulphide environment, as described in SKB's summary of the state of knowledge in the Canister supplementary report. From the studies that have been done, SKB concludes that stress corrosion cracking in sulphide does not threaten the integrity of the canisters in a KBS-3 repository in Forsmark, even though the mechanistic understanding of the process is not entirely elucidated.

Several tests have been conducted to, through fatigue (repeated mechanical loading), reproduce the pre-cracked samples installed in the field tests MiniCan 4 and 5 (Gordon et al. 2017, Johansson et al. 2017). The purpose was to be able to characterise the initial state of these samples better than what was done prior to the installation of the field experiments. Preliminary results have shown difficulties in achieving these fractures without obtaining a large plastic deformation. The bent samples that were included in MiniCan to study stress corrosion cracking, have also been newly manufactured in order to investigate the microstructure after manufacture in the most exposed parts.

SKB has continued the studies of slow strain rate testing of copper rods in sulphide solution at Research Institutes of Sweden (RISE), with an expanded experimental matrix, in order to test the hypothesis that intergranular corrosion is enhanced by strain. Lower strain rate and lower temperatures have been used, as well as variations in the buffering environment (lower ionic strength and borate buffer instead of phosphate buffer). Preliminary results show that all these changes produce fewer and/or rounder pits, indicating that mechanisms that cause fracturing are not active. As before, the pits are localised to grain boundaries. Figure 8-1 a shows an example of a more fracture-like pit, while Figure 8-1b indicates that switching to a lower ionic strength leads to the formation of a rounder pit. Updating of the mechanistic description is under way, but the new results overall confirm that stress corrosion cracking in sulphide does not take place in a repository environment.

The Swedish Radiation Safety Authority (SSM) has continued the commissioned testing, but at a lower strain rate (Becker et al. 2020, Forsström et al. 2021) and noted corrosion in grain boundaries rather than the occurrence of fractures. At the higher sulphide concentration studied (10^{-3} mol/L), grain boundaries had opened and were often filled with corrosion products. At lower sulphide concentrations (10^{-5} mol/L), no opening of grain boundaries was observed, but some corrosion was observed in grain boundaries and along slip-lines. Sulphide exposure to unloaded samples did not show any such surface defects.

In order to study the so-called Aaltonen mechanism, copper has been creep-tested during anodic and cathodic polarisation in anoxic sulphide solution (Ikäläinen et al. 2022). The results show that the creep rate is briefly affected by both polarisation and sulphide exposure. However, these changes were clearly transient and further studies would be needed to provide an in-depth mechanistic understanding of the observed processes. From the study, no signs of stress corrosion cracking could be detected in sulphide solution with a concentration of 25 mg/L (7.8×10^{-4} mol/L).

Another way of studying stress corrosion cracking in sulphide solution is being investigated by VTT within the Finnish programme KYT2022 (nationally funded research on nuclear waste). This entails measuring the rate of repassivation and comparing it to the corresponding rate in nitrite solution (Huotilainen et al. 2020). An initial experiment has been conducted, and the results, which the authors state as preliminary, show that in sulphide solution with a concentration of 100 mg/L (3×10^{-3} mol/L), the rate of repassivation is one order of magnitude lower than in nitrite solutions, in which stress corrosion cracking is well known to occur. The lower rate of repassivation means that dissolution in the fracture tip continues for a longer time and that the fracture becomes less pointed, thereby reducing the stress concentration that could lead to fracture growth.

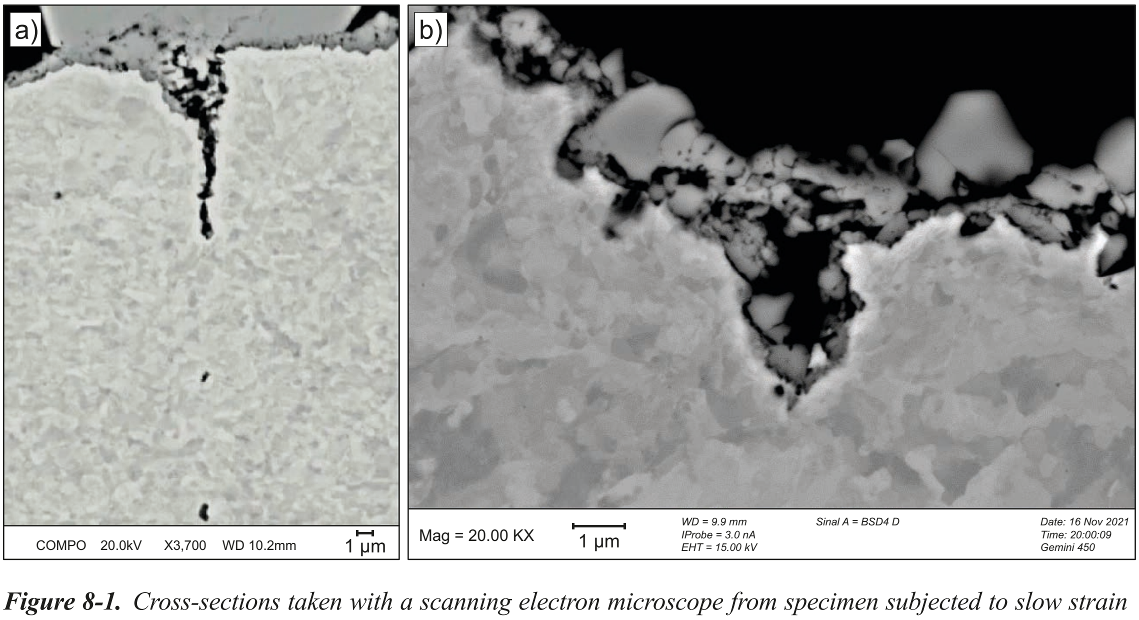

Figure 8-1. *Cross-sections taken with a scanning electron microscope from specimen subjected to slow strain rate testing in a solution of 10^{-3} mol/L sulphide at 60 °C, a) 10^{-2} mol/L phosphate buffer and 10^{-2} mol/L chloride (Taxén et al. 2019), b) 10^{-3} mol/L phosphate buffer and 10^{-3} mol/L chloride. The rounder pits on the right indicate that there is no active fracture mechanism. Note that the two images have a different scale.*

Posiva has evaluated stress corrosion cracking with a fault tree analysis, for both oxidising and sulphide-containing environments, and concludes that the risk of canisters failing due to stress corrosion cracking is small (King 2021b). As can be seen from both this and previous safety assessments (e.g. SKB TR-11-01, TR-19-15), the conclusions are based on a number of factors, which are of different importance in different environments. Posiva has gathered additional experimental data through strain rate testing of copper in acetate solution under oxidising conditions, but even at acetate concentrations as high as 0.4 mol/L, no stress corrosion cracking is detected (Ikäläinen et al. 2021). For sulphide environments, SKB notes in the Canister supplementary report that it is not sufficient to compare the sulphide concentration in experiments (like the ones described above) with the measured sulphide concentrations in the groundwater in Forsmark (a distribution where max is 1.2×10^{-4} mol/L, but most values are $< 10^{-5}$ mol/L sulphide). Instead, it is the sulphide flux nearest the canister surface that is crucial and this in turn is determined by the presence of bentonite.

Programme

- Continued experiments are foreseen at RISE, to gain a better understanding of the mechanisms that give the stress corrosion cracking-like attacks on copper in sulphide solution.
- The effect of injection of vacancies in the copper during sulphide corrosion will continue to be investigated, preferably at VTT, the Technical Research Centre of Finland. Studies of this type can also provide information on the role of hydrogen for possible stress corrosion cracking in sulphide solution.
- Quantum chemical models of the copper–copper sulphide interface may be used to study vacancy dynamics and the possible impact of defects on fracture initiation.
- SKB is reviewing the possibilities of conducting further experiments to produce samples and investigate the microstructure in such pre-cracked samples of the type used in MiniCan, for the purpose of studying crack initiation and crack propagation in copper. However, it has proved difficult to induce crack initiation and propagation in copper, as shown for example in studies within creep research (Wu et al. 2013).

8.1.6 Development of corrosion models and integration of corrosion analyses

Models for corrosion processes are used at different levels and with different degrees of quantification (a conceptual description of a process can also be said to be a model) in the assessment of post-closure safety. However, when modelling is mentioned, this usually refers to the models that calculate the corrosion that occurs under different conditions. The calculations can be done as (mathematically) simple mass balances or, for example, one-dimensional transport calculations. Alternatively, the calculations can be carried out with software that uses numerical calculations to link together corrosion processes, the surrounding chemical environment (for example hydrogeological data) and processes therein (for example radiolytic processes and solute transport).

In the description of research for the concerned corrosion modes (Sections 8.1.1 to 8.1.5), computational models are often developed to describe and analyse the mechanisms in more detail. These models are also suitable for sensitivity analyses, where the influence of different parameters, uncertainty in data and the use of conservative assumptions can be studied. The sensitivity analyses can then serve as a basis for formulating calculation cases for the more integrated corrosion analysis in the assessment of post-closure safety. The procedure of first describing corrosion processes in a final repository and then making more or less complex quantitative estimates of the extent of corrosion is also used in other nuclear waste programmes.

Current situation

Assessment of the reference evolution constitutes a central part of the post-closure safety study, and at several steps in this evaluation, coupled models are used, and results from one model study are used as input data for another study. The main example from corrosion modelling is the Excel model that calculates buffer loss in specific deposition holes due to erosion and sedimentation (SKB TR-11-01). The same model is also used to calculate corrosion at the elevated sulphide fluxes that occur during advective conditions, for the deposition holes where the loss of buffer was extensive enough to lead to advective conditions. Output data in the form of positions for deposition holes, where canister failures due to corrosion have occurred, are then used in the modelling of radionuclide transport from these deposition holes with damaged canisters. This model takes into account the variation in flow conditions and sulphide concentrations between the approximately 6000 deposition holes in the repository and thus handles corrosion from a repository perspective.

In Canada, an overview has been presented (Hall et al. 2021), where four corrosion processes are evaluated quantitatively (corrosion from initial oxygen, radiolytic corrosion, anoxic corrosion, and microbially influenced corrosion) and additional processes are evaluated qualitatively. A compilation has been made of processes that may produce and consume sulphide in the repository environment (Behazin et al. 2021). Furthermore, a 3D model in the software COMSOL has been used to study the influence of including the water saturation process and the initial temperature evolution, with the conclusion that these are of little importance for total sulphide corrosion (Rashwan et al. 2022). Within the Finnish programme, the evolution of the canister in the repository has been described in an extensive report (Posiva 2021a) that includes both corrosion and mechanical analyses, from which conclusions are subsequently included in the safety assessment. Sulphide corrosion is also analysed in the safety assessment as a function of sulphide concentration and flux, in an extensive modelling study (Posiva 2021b).

Work has been under way for many years on a kinetic and more mechanistically detailed model for sulphide corrosion, the Copper Sulfide Model (CSM) – see King and Kolář (2019) and King et al. (2021). It describes the most recently developed version that includes corrosion from sulphide in the groundwater, microbial production of sulphide, the initial oxidising phase, resaturation, and the temperature evolution. The model has also been compared with other models in a cooperation between SKB and Posiva (Posiva SKB 2021) – see Section 11.4.2 for more information.

For localised corrosion, a probabilistic model has been developed, so far for oxidising and saturated conditions (Briggs et al. 2020, 2021). The model handles uncertainties in local chemistry, including the time of oxygen consumption, and uncertainties in literature data on corrosion potentials (Section 8.1.2).

The previous model for radiolytic corrosion on the outside of the canister has been further developed with respect to which oxidant dominates the corrosion (molecular oxygen instead of hydroxyl radical in the previous model), so that the modelling results now agree better with experimental data in respect of the extent of oxide formation (Soroka et al. 2021). The model has also been developed so that a more realistic decline in dose rate corresponding to decay of caesium-137 is used. Furthermore, effects of different groundwater components have also been included in the model (Jonsson 2021b). Regarding the initial formation of oxidants due to radiolysis of water, air, and argon inside the canister, and the consumption of these species, the process description has become more detailed and the models have been updated, see also Section 7.1.

Programme

- A probabilistic model for localised corrosion under oxidising unsaturated conditions is under development in cooperation between SKB, NWMO, and Integrity Corrosion Consulting in Canada. The model will consider uncertainties regarding the saturation process, relative humidity, pore water composition, and the time it takes for oxygen to be consumed.
- In connection to updating the assessment of post-closure safety, SKB will review how different models for corrosion are used and whether there are more processes that should be modelled to a greater extent. These concerns, primarily, the different parts of corrosion in sulphide and the description of an integrated model of film growth, localised corrosion (including micro-galvanic corrosion), and the stress corrosion cracking-like process that has been observed in strain rate testing in sulphide solution.
- The model for radiolytic corrosion on the outside of the copper canister, which has been developed in several iterations at KTH, today includes the most relevant groundwater components. As stated in Section 8.1.4, a compilation of the work is under way and the results are intended to be published in a scientific peer-reviewed journal.
- The model for the radio-chemical environment inside a canister will continue to be developed – see Section 7.1 – and will also be available for use in modelling of corrosion on the canister materials in the assessments of post-closure safety.

8.2 Material properties of canister material

The canister insert is load-bearing for the mechanical load, while the surrounding copper shell constitutes a corrosion barrier. An important process in the analysis of canister integrity is the impact of the expected ambient pressures in the final repository. These will deform the canister copper elastically, viscously through creep deformation and, if pressures are high, also plastically. In view of the required time frame of the repository and the fact that it is crucial that the properties of the canister materials do not change over time, extrapolation will be required from the results of testing of creep deformation. A fundamental understanding and characterisation of material properties is needed for a reliable extrapolation. The risk of hydrogen embrittlement of the canister materials can be reduced by defining requirements and using well-characterised canister materials, with low concentrations of hydrogen and oxygen at the time of deposition. Gamma and neutron radiation can affect the material properties of the canister. To minimise these risks, well-founded requirements, modelling and extrapolation from experimental data are also applied.

In the same way as described in Section 8.1, the activities described under the Programme in this section aim to strengthen the supporting data for upcoming post-closure safety assessments for the Spent Fuel Repository, and to reduce conservatism in a number of cautious assumptions made in the assessments.

8.2.1 Copper creep

SKB has chosen an anoxic copper with a low content of impurities, to be used for the canister. In order to achieve favourable creep properties (sufficiently high ductility), phosphorus is added, and the material is often called Cu-OFP. The effect of phosphorus is attested through extensive creep testing, but for the assessment of post-closure safety, a better understanding has to be developed (down to an atomic level) to show that the material properties do not change over long periods of time. For a long time, SKB has conducted research and development work to be able describe the creep processes in copper via representative models.

Current situation

The modelling of creep deformation during the final phase before failure (tertiary creep) was previously included in the modelling of canister copper, and has now been shown to be applicable up to 250 °C (Sandström and Sui 2021). In the work of extrapolation of creep rupture data, artificial neural networks have been employed, and in connection, methods for fault analysis are under development. The applicability of the approach to predicting creep properties using fundamental models, without involving a number of fitting parameters, has also been demonstrated through application to other materials and in other technical applications. One example is the prediction of creep rupture data for an austenitic steel (Sanicro 25) with high creep strength around the critical temperature 700 °C, for advanced types of thermal power plants (He and Sandström 2019).

Time to creep rupture decreases with increasing temperature and stress, which means that creep rupture is unlikely for the canister in the final repository, since temperature and stresses are low. In order to investigate underlying mechanisms and indicate margins, SKB has previously measured time to creep rupture in canister copper at higher temperatures and stresses than those expected in the final repository. The measurements have facilitated empirical modelling and mechanistic studies. Empirical creep rupture models can reasonably well describe degradation over time and are necessary when experiments are to be planned.

In the work of using quantum chemical calculations to study the effect of how impurity atoms interact with grain boundaries, additional grain boundary types have been included (Lousada and Korzhavyi 2020a) and the binding energy has been estimated for sulphur and phosphorus. The results of these calculations have then been applied in modelling of cavity formation in Cu-OFP (Sandström and Lousada 2021). The data used to include more types of CSL grain boundaries (CSL, Coincidence Site Lattice) comes from preliminary evaluations of the distribution of grain boundaries with the aid of Electron Backscatter Diffraction (EBSD) performed at Swerim of Cu-OFP from SKB (Hagström et al. 2020), and showed that sigma9 grain boundaries are more common than previously shown for copper (Mishin et al. 1997).

Further extensive quantum chemical calculations have been carried out and are being prepared for publication. Preliminary results show, for example, that the sigma9 grain boundary (which is common) and the sigma5 grain boundary (which has special properties) can act as strong depressions for vacancies, and these are in turn necessary for segregation of phosphorus and sulphur atoms to grain boundaries.

The effect of phosphorus on the creep properties of copper has continued to be studied experimentally. After an initial study of recrystallisation (Sundström et al. 2020), work has continued on more systematic testing of material from SKB copper tubes and at temperatures primarily between those previously tested, i.e. in the range 230–255 °C. Since Auger spectroscopy of creep-tested material has shown particles of phosphorus (Sundström et al. 2020), these studies have also continued, now also with transmission electron microscopy (TEM). The purpose was to further investigate the hypothesis that recently deformed material contains enrichment of phosphorus, which has been preliminarily confirmed. The method of using slow strain rate testing at different temperatures has also been used on a material with an increased content of zinc, bismuth and tellur (approx. 5 weight ppm of each), which shows a slightly reduced ductility compared with Cu-OFP (Andersson-Östling and Sundström 2021). Creep testing of the material is under way to provide a more complete picture of what the slow strain rate testing can show with respect to the material properties. SKB's results from creep testing 1985–2018 have been compiled from different background reports and published in an integrated report (Andersson-Östling 2020).

Programme

- Using the extended series of recrystallisation data, a recrystallisation model based on the Avrami equation (IUPAC 1997) will be developed and used for extrapolation to repository times. If recrystallisation does not take place, it is therefore not expected that the material properties of the copper will change.
- The study of phosphorus enrichment will be completed. Further plans include studies of whether diffusion of phosphorus can explain the enrichment during deformation. Factors that can be varied are phosphorus content in the material, temperature, and time for creep testing. Particles can be studied with TEM, while Glow Discharge Optical Emission Spectroscopy (GDOES) can probably be used to measure phosphorus content.
- Creep testing of the material with a slightly elevated content of zinc, bismuth and tellur will be completed and documented.
- Commissioned by SKB, a systematic study is under way using EBSD at Swerim, characterising samples from different positions on two copper tubes and a lid, with samples taken in different directions. The samples are relatively large (20 × 50 mm) and represent the entire thickness of tubes and lid. The measurements provide distributions of CSL grain boundaries, grain size and crystal planes, as well as images of these. The developed method can then be used in more focused studies.
- The quantum chemical calculations of the relative stability of different atomic defects in the structure of copper continue at KTH Material Science. The diffusion of phosphorus, sulphur and hydrogen atoms in copper, as well as the self-diffusion of copper, will also be investigated there.
- The compilation of the long-term work on development of creep models carried out by Rolf Sandström, KTH, will be published.
- SKB plans to measure time to creep rupture for test-manufactured canister copper at high temperatures and high stresses. The results will be used to adjust SKB's empirical models used in practical creep testing and operational situations. The effects of minor variations in the composition, structure, and grain size of the canister copper can also be evaluated in the tests.

8.2.2 Hydrogen embrittlement in copper, nodular cast iron and steel

Hydrogen can occur in metals both in atomic form (H) and as hydrogen molecules (H_2). If the concentration of hydrogen is sufficiently high, it may negatively affect the mechanical properties of the metal. In the atomic form, hydrogen can be bound to different types of defects and impurities in the material. It may also form molecular, gaseous hydrogen in microscopic pores within the material, which can have a negative effect on the mechanical properties of the material (hydrogen embrittlement).

Nodular cast iron and low alloy steels have a space-centred crystal structure (bcc) and hydrogen embrittlement may occur. The risk of hydrogen embrittlement increases with the strength of the metal alloy and the presence of tensile stresses in the structure. Copper and many other metals and alloys with a surface-centred crystal structure (fcc) are less likely to suffer from hydrogen embrittlement.

The requirements for anoxic copper used for the canisters serve to prevent hydrogen embrittlement and also the material degradation termed hydrogen sickness (where hydrogen reacts with oxygen in the form of an oxide and forms water vapour). The starting material for the copper canisters should be sufficiently clean that hydrogen embrittlement does not pose a risk. Nevertheless, it needs to be ensured that the material is not affected by welding of the canisters or in the final repository (by radiolysis or corrosion) in such a way that hydrogen embrittlement could occur and negatively affect the canister properties.

Current situation

The solubility of hydrogen in copper is very low (Magnusson and Frisk 2017) and to study whether hydrogen affects the material properties, hydrogen has to be forced into the metal. However, it has proved difficult to force in any large quantities into the material, other than superficially (Leijon et al. 2018), and also to reliably measure the content at different places in the material (Granfors 2017).

In some experiments, moderate increases of the hydrogen content have been observed when copper in water has been exposed to a radiation dose corresponding to that which the material will be exposed to in the final repository. In these experiments, superficial corrosion effects were observed, but no pore formation at depth. For a more detailed report, see the Canister supplementary report.

The extensive quantum chemical calculations that have been carried out in order to better understand how hydrogen interacts with the metallic copper surfaces, with its internal crystal structure, and with the grain boundaries, have now been published in a scientific, peer-reviewed journal (Lousada och Korzhavyi 2020b). As previously stated, the calculations show that, even if the copper surface was completely covered with hydrogen, the concentration inside the metal is very low. The work has since continued with studies of hydrogen segregation to, and diffusion in grain boundaries (sigma9 and sigma5 – see further in Section 8.2.1) compared with the matrix in a crystal grain in the copper.

The risk of hydrogen penetrating and a resulting influence on the properties of the copper material at depth, has also been analysed theoretically on the basis of diffusion calculations – see the Canister supplementary report. The results show that the sulphide flux (H_2S/HS^- as a hydrogen source) under repository conditions is too small for the released hydrogen to have any significant impact on the canisters. Despite this, there is at least one study where the results have been interpreted to mean that hydrogen has penetrated into and interacted with the copper material, but under considerably higher fluxes of sulphide than those under repository conditions (Zhang et al. 2021). This interpretation was not based on detection of hydrogen, but on measurements of small changes in the crystal structure of the copper material after sulphide exposure. Even the reference sample oxidised at elevated temperature exhibits variations in the crystal matrix at the same sample depth as the samples exposed to sulphide.

The question of hydrogen embrittlement in nodular cast iron has also been discussed and analysed, even though the experimental material is limited and mainly consists of charging hydrogen into the metal with electrochemical methods in relatively aggressive solutions (Martinsson et al. 2009, Wu et al. 2015, Forsström et al. 2019, Sahiluoma et al. 2021). The hydrogen was found to penetrate the material superficially and also exit the material again.

The material was also tensile tested or creep-tested at the same time, which increases the charging of hydrogen (Sahiluoma et al. 2021), but creep testing after hydrogen charging also increases the outflow (Martinsson et al. 2009). The material properties change in the form of reduced ductility, but also in the form of a slightly reduced yield strength and ultimate tensile strength (Martinsson et al. 2009, Forsström et al. 2019, Matsuno et al. 2012). The binding of hydrogen in the material has been studied with Thermal Desorption Spectroscopy (TDS) and can be related to two different types of "traps" in terms of energy (Sahiluoma et al. 2021), related to small cavities and defects (lower hydrogen binding energy) and storage in graphite nodules (higher hydrogen binding energy). However, only small amounts of hydrogen are anticipated to be available under repository conditions, and experiments with exposure to distilled water, which is more similar to repository conditions than the aggressive solutions used to introduce the hydrogen, show ductile failure mechanisms (Forsström et al. 2019, Sahiluoma et al. 2021).

Programme

- Hydrogen penetration and diffusion in copper is being studied with a type of Devanathan–Stachurski cell within the consortium work at CanmetMATERIALS, Canada (Section 8.1.1).
- Studies are under way at VTT, Finland, to quantify possible hydrogen charging in copper due to sulphide exposure under tensile stress – see Section 8.1.5 on stress corrosion cracking.
- Work is also under way to better ensure the accuracy of measurements of hydrogen in copper.
- The quantum chemical calculations at KTH Material Science are continuing, partly with respect to how hydrogen interacts with dislocation cores (provides knowledge of diffusion and segregation), and partly to better understand how vacancies are injected into the copper in the formation of a copper sulphide film and how it interacts with hydrogen.
- SKB is reviewing the need to set limits for, and analyse the quantity of hydrogen in the insert materials.

8.2.3 Radiation effects on copper, nodular cast iron and steel

A background report (SKB TR-10-46) for the SR-Site safety assessment concluded that the radiation doses to which the canister materials will be subjected have negligible effects on the mechanical properties of the canister. This is based primarily on previous calculations of radiation damage in the materials as a result of gamma and neutron radiation.

Gamma and neutron radiation can also cause embrittlement of the material properties of the nodular cast iron, through precipitation of copper particles or formation of Late-Blooming Phases (LBP), or by affecting the diffusion of e.g. phosphorus to grain boundaries. The extent of the precipitation of copper particles has to be studied so that well-founded requirements may be imposed on the maximum permissible copper content in the iron material.

Current situation

In the Canister supplementary report, SKB presented updated calculations of radiation damage, as well as experiments with gamma irradiation, but has not conducted any further studies since then. However, the modelling work at KTH Reactor Physics has continued (Yang et al. 2022). The model for calculating precipitation of copper particles in nodular cast iron can now be used to confirm SKB's required threshold of max. 0.05 per cent copper in the nodular cast iron to avoid embrittlement.

Programme

- SKB is not planning any further studies of radiation effects on the canister materials in the RD&D period.

8.2.4 Ageing of nodular cast iron and steel

Iron and steel dissolve carbon and nitrogen, which are mobile atoms, and atoms can form atmospheres around deformations and slow down further deformation. As a result of this ageing effect, iron and (ferritic) steel can become harder and less ductile when stored after cooling, casting or deformation.

Current situation

Dynamic ageing may occur during strain rate testing at elevated temperature; it is identified as inhomogenous deformations in the tensile test curve and is called the Portevin–Le Chatelier effect. Simply put, the inhomogenous deformations can be attributed to dissolved substances interacting with the ongoing deformation of crystal grains in the metal, and both the diffusion and the deformation of the substances are temperature-activated. This effect can lead to a slight reduction in the ductility (brittleness) of the material.

Ageing is affected not only by the concentrations of dissolved atoms, deformation, and deformation rate, but also by the temperature of the canister. Several reports on canister temperature and temperature in the near-field of the canister have recently been published (Ikonen 2020, Renström 2020, Jansson et al. 2022b), concluding that the insert temperature is mainly affected by the gaps and the conductivity of the bentonite. In addition, the temperature increase outside the canister mainly takes place through conduction and only to a certain extent through direct energy deposition of gamma radiation.

Furthermore, nodular cast iron is the form of cast iron that is most resistant to strain ageing and only becomes brittle at 300–400 °C (Yanagisawa and Lui 1983). After a survey of the available literature and the load cases in the final repository, SKB's assessment (2017) was that neither static nor dynamic strain ageing will affect the mechanical properties or functions of the canister insert in the final repository. This is partly because plasticising loads are required for the phenomena to occur, while the canister insert is designed so that plasticisation is not achieved except in unlikely cases, and partly because temperatures higher than around 100 °C are not expected in the final repository. This is considerably lower than the temperatures at which specifically dynamic ageing occurs.

In addition to the assessment from 2017, SKB has evaluated cold ageing, dynamic and static strain ageing, and possible brittleness by means of strain rate testing of nodular cast iron from test-manufactured inserts (Sarnet 2022). As can be seen in Figure 8-2 dynamic strain ageing with serrations in the stress–strain curve can only be observed at 300–400 °C for SKB's nodular cast iron (Sarnet 2022). No signs of dynamic strain ageing can be seen up to 125 °C. The ultimate strength is highest at room temperature, 384 N/mm^2. The ultimate strength at 300 °C is as high as at 125 °C as an effect of the dynamic strain ageing at 300 °C. The dotted lines show the elastic component of the material, so that the elongation can be determined from the maximum stress on the stress-strain curve. The elongation is highest for the measurements at 125 °C and lowest for the measurements at 400 °C: 18.6 % and 10.9 %, respectively.

Static strain ageing requires a strong pre-deformation, three per cent elongation, which is several times higher than the elasticity limit. It has not been possible to show that the elongation deteriorates with increased static strain ageing (Björklund 2021, Sarnet 2022).

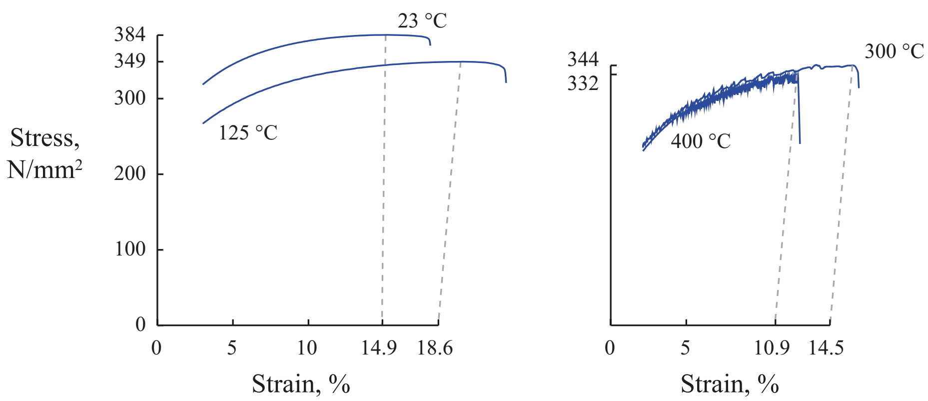

***Figure 8-2.** Stress–strain curves for SKB's nodular cast iron at four temperatures (Sarnet 2022).*

Programme

SKB plans to supplement the investigations of the influence of the traction speed and temperature on dynamic ageing and the influence of pre-deformation on static ageing for the canister insert. Furthermore, the factors of nitrogen content and metal structure can be varied in the experiments.

8.3 Technical design

Current situation

The requirements for the composition of the material for the copper shell have essentially remained the same since they were first specified in Werme (1998). The basis for the requirements is the material designation Cu-OFE (SS-EN 1976:2012), which also specifies requirements for impurities such as nickel and iron. The most recently published and currently applicable technical design requirements (Posiva SKB 2017) state an oxygen content of a maximum of 5 weight ppm, which is an update from what was stated in the application (SKB TR-10-14), but the same as in Werme (1998). Other requirements for material composition remain unchanged since the application. Furthermore, SKB adds a requirement on the minimum yield strength of the copper material to ensure that the intended handling of the canister does not affect its properties.

SKB reported in RD&D Programme 2019 that an updated design analysis of the canister had been prepared. Based on these calculations, the design requirements with respect to mechanical properties in the insert have been further specified. More specifically, the design requirements have been diversified for central parts that do not take any load, so that no requirements are made on elongation after fracture in a circular area with a radius of 100 mm in the centre of the insert's cross-section (Posiva SKB 2018). In addition to the mechanical properties, manufacturing requirements in respect of microstructure have been further specified for the parameters (graphite and perlite) where a direct link to the material's properties has been identified.

To ensure that the insert is sufficiently tight so that the requirement on gas tightness in Clink can be met, a seal is required for the insert's steel lid. In the previously presented design, this seal consists of a traditional rubber O-ring, which does not meet the overall requirements that the canister must not contain organic materials. Therefore, an alternative design of both the steel lid and its seal has been evaluated. The basic principles of the new design are a threaded joint in the periphery of the lid instead of a central bolt and a flat gasket placed in this threaded joint, with similar principles as for a cylinder head gasket.

Programme

- For the copper shell, new studies will be performed concerning the maximum residual deformation that the copper shell may be subjected to. Permissible handling loads will be investigated, as will the permissible size of handling defects in the copper shell.
- The effects of tensile stress in the copper lid will be investigated and, if necessary, the detailed canister design will be optimised.
- Preparation of requirements for the purity of the copper shell and how the requirement is to be verified in production.
- The requirements with respect to acceptable defects for the different parts of the canister as input data for the qualification of non-destructive testing will be further specified.
- A project is being pursued for the canister insert, for the purpose of developing an alternative design of the insert that permits robust fabrication and easier inspection and testing. The project includes an alternative design where the insert is fabricated without an embedded steel cassette, and alternative designs that are fabricated from conventional standard products. In conjunction with this investigation, a review is made of the manufacturing requirements for the insert.

8.4 Manufacturing, inspection and testing

8.4.1 Manufacturing of copper parts

Current situation

In the RD&D Programme 2019 it was reported that SKB had identified variations in grain size and the occurrence of cold shuts in the forging of copper lids. As a result, SKB has further developed the forging process. Manufacturing is carried out as before by means of drop forging, and by using much greater forces than before, it has been possible to simplify the process to use of a simpler forging tool and fewer process steps. Among other things, the concluding step of flattening, which in the previous process caused the cold shuts, has been excluded.

Manufacturing of copper lids using the optimised process has resulted in components that meet the requirements for geometry and material structure. Ultrasonic testing of manufactured lids indicates homogeneous low sound attenuation throughout the volume, which indicates a homogeneous fine-grained material structure, which in turn permits good testability. Figure 8-3 shows typical examples of how variations in the material structure are indicated by ultrasonic testing of lids manufactured by means of the original forging process and by means of an optimised forging process. In order to quantify the grain size, a number of samples have been taken from some of the manufactured lids, and the results confirm a homogeneous and fine-grained material structure with an average grain size of 60–120 µm – see Figure 8-4.

Additional copper tubes have been extruded for development of the process for fabrication of copper tubes. The purpose has been to gather additional information on suitable process parameters and to support previously tested temperatures for the different steps in the hot forming process. These experiments have shown that further work is required to establish the limits within which different process parameters are allowed to vary. No development is currently taking place on the alternative fabrication method of pierce and draw processing.

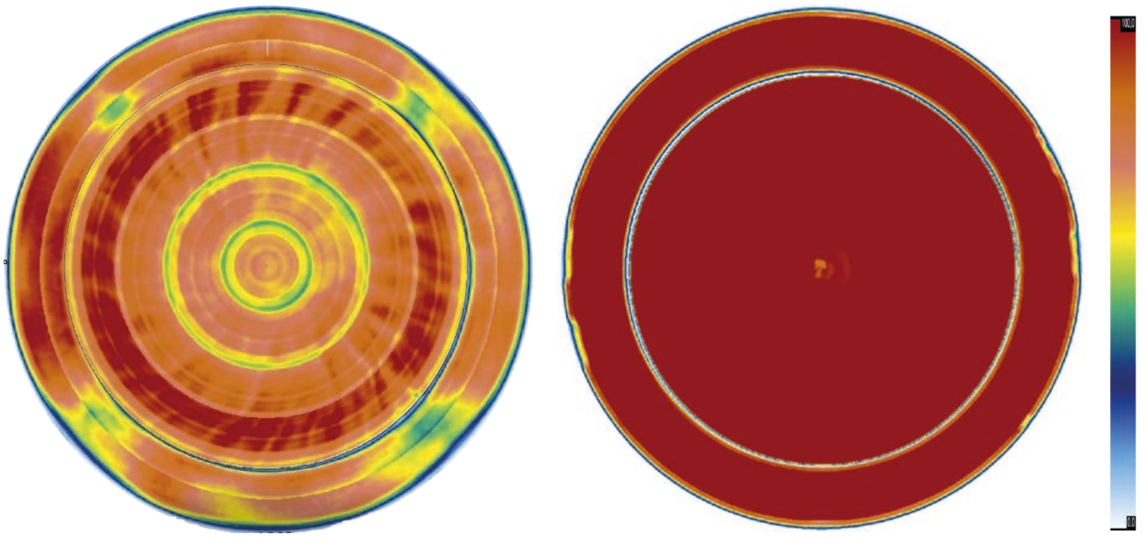

Figure 8-3. *Ultrasonic testing results with 3.5 MHz array finder of copper lids, TX250 on the left (original forging process) and TX261 on the right (optimised forging process). Data is presented in the form of C-scan images for bottom echo amplitude, where high amplitude indicates low sound attenuation and thereby a fine-grained material.*

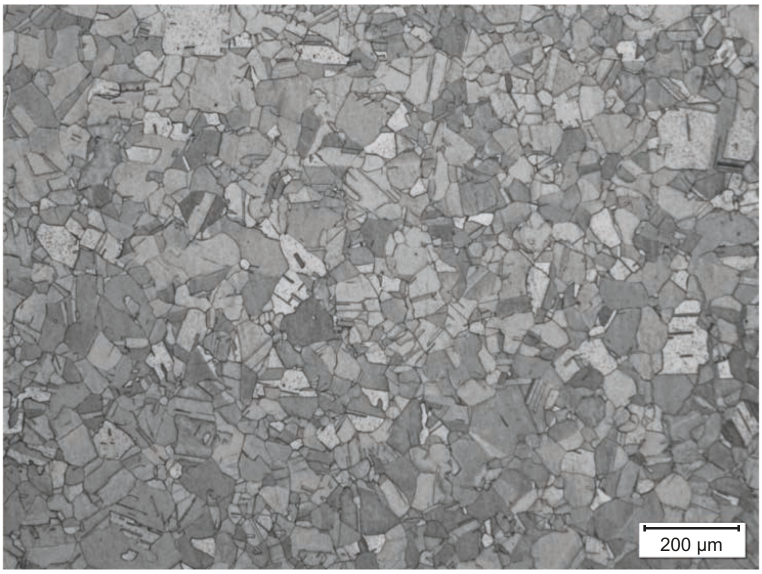

***Figure 8-4.** Results from grain size measurement of samples from copper lid TX261.*

Programme

- Carry out process mapping in order to obtain data to be able to carry out experimental planning of process parameters, which in turn will serve as a basis for future qualification of the manufacturing processes for the entire manufacturing chain for copper components.
- Implement an optimised process for extrusion of copper tubes.
- Perform manufacturing tests for the purpose of compiling data for process windows for process parameters at the different steps for extrusion of copper tubes.
- Conduct initial experiments to evaluate alternative manufacturing methods for copper tubes to ensure access to alternative suppliers during production.
- Perform manufacturing tests for the purpose of compiling data for process windows for temperature for forging copper lids.
- Draw up a testing programme for destructive testing with the purpose of identifying the need for testing in connection with qualification and production of copper components. Review of systems for handling of the copper canister.

8.4.2 Manufacturing of insert

Current situation

An extensive testing programme has been carried out for the last three manufactured BWR inserts and two most recent PWR inserts. Testing has included non-destructive testing of both inserts and samples, as well as destructive testing of samples from a samples from the overlength and the insert volume. Material testing shows that requirements for fracture toughness and yield strength are fulfilled in the tested positions. On the other hand, there are uncertainties regarding elongation after fracture, but also ultimate strength. In addition, certain defects have been indicated that are assessed to be on the limit.

Changed manufacturing requirements have been introduced for the steel cassette after casting. Tensile testing of material from the steel cassette's channel tubes after casting has shown that the properties do not meet the new requirements regarding yield strength and elongation after fracture. In order to investigate the causes of worse material properties in the nodular cast iron between channel tubes and in the steel cassette's channel tubes after casting, experimental studies of contraction and expansion of cast iron as a function of cooling rate have been carried out. Numerical modelling of the stress distribution in nodular cast iron inserts due to thermal expansion and structural investigation and fractography of nodular cast iron and steel has been included. The experiments show that small microfractures form at high temperatures and that the main reason is that the nodular cast iron and the steel in the cassette expand and contract differently during temperature changes. Analysis of the microstructure shows that carbon has diffused from the nodular cast iron into the steel cassette, due to a concentration gradient between the two materials. The carbon diffusion means that the material properties change both in the steel and in the nodular cast iron. The steel's microstructure changes and the material loses ductility. In the nodular cast iron, silicon will prevail in the decarbonized zones in the vicinity of the channel tubes, and the material is expected to become more brittle. These results indicate the need for continued technology development and studies to reduce uncertainties regarding the fabrication of inserts.

Programme

- Conduct process mapping for the manufacturing chain of the canister insert.
- Development of alternative methods for fabrication of the insert, in which the channel tubes are created without an embedded steel cassette.
- Evaluation of manufacturing of alternative insert designs based on standardised and proven materials.
- Carry out detailed design of the new design for lid design with screw joints and flat washer.
- Draft a testing programme for destructive testing with the aim of identifying the need for testing in connection with qualification and production of inserts.

8.4.3 Welding of the copper shell

Current situation

The development work for welding of lids on copper tubes has focused on production adaptation including simplified joint geometry, determination of tolerances for the lid/tube fitting, development of internal gas shield and quality assurance of the probe manufacturing. In order to investigate the tolerance for the fitting between lid and tube and to simplify the joint line geometry, welding has since 2019 been carried out using a new straight and slightly inclined joint line design (instead of the current reference design with a male-female rebate). Both non-destructive and destructive testing show equivalent results to the current reference design.

An external gas shield has been functional for a long time during the welding development and a possible need for internal gas shield to minimise the risk of oxide particles in the weld root has been identified.

The two welding probes that have been tested in respect of service life/safety factor against failure each held four full welds, but extensive work is currently being carried out to ensure that all probes have the same properties by carrying out quality assurance of the machining, heat treatment and surface treatment of the probes.

The full welds performed according to the reference procedure have been compiled in a report to describe the process windows for the different input and resultant parameters. This report is to be used as a basis for qualification of the welding procedure, as the qualification is planned to be carried out through so-called pre-production welding test (according to ISO 15613).

Programme

- A qualification cooperation project has been initiated with Posiva and an accredited qualification body to qualify a welding procedure for welding of the copper bottom on the copper tube and a welding procedure for the seal weld.
- Supplementary destructive testing of welds.
- Test and implement internal gas shield (in the form of a so-called welding shroud) in the FSW machine at the Canister Laboratory (FSW, friction stir welding).
- Determine the upper limit for the oxygen content in the internal gas shield.

8.4.4 Inspection and testing

Current situation

SKB has carried out an assessment of how the principles of the Swedish Radiation Safety Authority (SSM) regulation SSMFS 2008:13 can be applied to the set of requirements and testing of the canister. The assessment is that certain parts of the regulation can be used, but adaptations need to be made, since testing of the canister is not done with respect to operationally induced defects and only a limited number of objects should be tested over a long time period in specially adapted testing systems.

The technology for ultrasonic testing of the copper components has been further developed with the support of extensive simulations with the established commercial software CIVA. Based on the simulations and practical experiments, optimised array finders have been developed and, based on these, adapted test settings for the copper tube have been tested.

The occurrence of actual defects in the copper components has been rare. However, a limited number of surface-breaking defects in the form of creases in the surface have been identified. In order to get a clear picture of the characteristics of the defects, some have undergone extensive analysis. They have been tested in several steps using ultrasound, X-ray and eddy-current technology. Based on the results, a number of smaller samples have been cut out for further investigation with high-resolution computer tomography and conventional metallography. The results of these investigations are then used for development of eddy-current technology for the outside of the copper tube and the flat sides of the copper lid. Development has been done by studying the signal response from artificial defects in the form of sparked traces and from real defects in the form of creases. In addition, CIVA simulations have been carried out to study the effect of certain defect parameters.

For testing of the copper shell's FSW welds, new X-ray equipment has been commissioned at the Canister Laboratory. In the specification of the equipment, testing of different X-ray configurations (radiation energy and detector type) has been investigated and optimised. The new equipment has been used to carry out parameter tests, which has made it possible to develop adapted testing settings for the welding equipment.

Programme

- Detailing of requirements and prerequisites for qualification based on applicable parts of the consolidated version of regulation SSMFS 2008:13.
- Preparation of draft plan for qualification.
- Further development of technology for ultrasonic testing of copper lid and bottom, for eddy-current testing of copper lids, copper bottom and inside of the copper tube and for ultrasonic testing of the copper shell's welds.
- Assessment of the opportunities for using virtual defects for training and qualification of personnel.
- Development of adapted ultrasonic technology for testing of the canister insert.
- Development of a method for measurement of oxide layer on the copper surface by means of colour differences after welding.
- Development of methods for inspection with respect to handling damage on the canister surface.

9 Cementitious materials

The Final Repository for Short-lived Radioactive Waste (SFR) contains large amounts of cementitious materials in waste matrices, engineered barriers and other structures. This also applies to the concept for the Final Repository for Long-lived Waste (SFL) which has undergone safety evaluation and tor the Spent Fuel Repository, where cementitious materials are planned in plugs, grouting materials and materials for rock support.

This chapter presents SKB's programme concerning cementitious materials for the three final repositories. The research programme, which is aimed at a better understanding of processes, is presented jointly for the three final repositories (Section 9.1) since the results achieved within this programme are often relevant for all final repositories. The programmes for design and production methods for other materials, on the other hand, are presented separately for SFR (Section 9.2) and the Spent Fuel Repository (Section 9.3) since the relevant issues are more facility-specific, even if certain common issues exist. Since further work on SFL has been postponed, no separate programme concerning the design, material and production method for SFL is presented in this chapter.

9.1 Cementitious materials – evolution after closure

This section describes the research that SKB plans to conduct in order to gain a better understanding of how the function and properties of the cementitious materials change in the repository environment during the time periods covered by the safety assessments.

9.1.1 Impact of groundwater on cementitious materials

Cementitious materials which come into contact with groundwater are affected by the chemical composition of the water and by the magnitude and direction of the water flow. Dissolution or precipitation of minerals alters the mineral composition of the concrete, which in turn affects the hydraulic, mechanical and chemical properties of the material. This in turn leads to changes in the concrete structures' strength and sorption capacity for radionuclides, but changes in mass transport properties can also be expected.

Current situation

Over the years, SKB has commissioned a number of studies concerning the evolution of the properties of the concrete structures in SFR. The two most recent are Höglund (2018) and Höglund (2019), in which the pH evolution in 2BMA (waste vault for intermediate-level waste in extended part of SFR) and 1BRT (waste vault for reactor pressure vessels from boiling water reactor (BWR)) during the first 100 000 years after closure were studied. In agreement with previous studies in this research area, both of these studies showed that degradation of concrete by dissolution or transformation of the cement minerals under repository conditions is a very slow process, but also that the predicted pH evolution is affected by the choice of calculation parameters. The studies showed that pH values exceeding 12 (2BMA) and 11 (1BRT) can be expected during the entire analysis period, i.e. 100 000 years.

During the previous RD&D periods, SKB has commissioned the development of a computational tool for reactive transport modelling in fractured systems (Nardi et al. 2014). Further development of the computational tool could facilitate studies of the evolution of the transport properties of concrete with the presence of cracks.

Formation of cracks in the concrete structures can lead to an increased proportion of groundwater penetration into the waste domain taking place through the cracks instead of through the concrete matrix. This means that the penetrating groundwater does not interact to the same extent with the concrete's pore water before it reaches the waste domain. The consequence of this may therefore be a change in the chemical environment in the waste domain, for example an initially lower pH and lower concentrations of species originating from dissolution of cement minerals, which can affect, for example, the corrosion rate of the metallic materials present in the waste.

Programme

In view of the results obtained in the studies that were carried out within the framework of the SFL Safety Evaluation, which were presented in the RD&D Programme 2019, SKB intends to continue to develop its programme for studies of interactions between groundwater and concrete under repository conditions. The programme is not finally established at present. Of the possible development areas identified, the following are worth mentioning:

- Studies of how the mineral composition of the materials evolves as a function of precipitation and dissolution reactions and how this evolution is linked to mechanical processes.
- Further studies of the properties of concrete with different binder composition.
- Evolution of the properties of materials in the systems for gas transport over time.
- A study to gain a better understanding of the pH evolution in the waste vaults 1–2BTF.

9.1.2 Modelling of gas transport

During the period after closure, different types of gases may be formed through degradation of materials present in SFR and SFL (Section 6.2). In the event that these gases cannot be transported out through the concrete structures, an internal pressure could build up and affect the structural integrity of the concrete structures.

Current situation

Since the activities focused on modelling of gas transport were carried out, as described in the RD&D Programme 2019, SKB has commissioned an update of the model of gas-producing processes. However, the work has not yet been finally reported.

Programme

SKB's programme for studies linked to modelling of gas transport:

- Final reporting of the update of the model for gas-producing processes.
- Development of models for gas and water flow in the near-field in SFR.
- On the basis of completed modelling, carry out assessments regarding how different types of gas transport systems can be designed so that they have as little impact as possible on radionuclide transport. See also Section 9.2.2.

9.1.3 Impact of degradation of organic waste

Organic materials that degrade in a cement matrix can affect the properties of the cementitious materials. This can alter the composition of the pore water, but also affect the ability of the concrete to limit releases of radionuclides, e.g. through the formation of complexing agents.

Current situation

The project Concrete and Clay, which has been conducted by SKB at the Äspö HRL since 2010 (Mårtensson 2015) studies, among other things, what types of degradation products are formed from different types of organic material, representative for low-level and intermediate-level waste, and how they are dispersed in a concrete matrix. Implementation of this project is justified by a need for studies under relevant final repository conditions, and constitutes a complement to the often accelerated laboratory studies that comprise a majority of the supporting data used in the safety assessment work. Results from the analysis of samples from these experiments can also serve as a basis for the design of additional supplementary laboratory experiments.

In 2019 and 2020, steel containers containing groundwater from repository depth, organic material representative of low-level and intermediate-level waste and some crushed cement paste were sampled and analysed with respect to degradation products from the organic materials. In addition, concrete cylinders with organic material, which had been stored in the bedrock for about 10 years, were retrieved and analysed. The analyses showed that degradation of organic material was very limited, and the only degradation products that could be detected with any certainty consisted of plasticisers from plastic gloves (Szabó et al. 2020).

Programme

SKB's programme for studies of degradation of organic material in a cement environment during the period up to the closure of the underground facility at the Äspö HRL:

- Retrieval and analysis of the remaining two samples containing organic material in the Concrete and Clay project at the Äspö HRL.

9.1.4 Impact of corrosion of metallic waste

In corrosion of metals embedded in a cementitious material, the corrosion products can react with the cement minerals and thereby change the properties of the cementitious materials. In addition, the mechanical pressure that occurs if voluminous corrosion products congregate on or around metal surfaces can lead to fracturing in the matrix.

Current situation

During 2020, retrieval and analysis of a concrete cylinder containing different metal samples, corresponding to those previously reported by Kalinowski (2015) was carried out within the Concrete and Clay project (Kalinowski 2021).

In agreement with the results from the previous retrieval (Kalinowski 2015), these analyses showed that the corrosion rate of all metals included in the experiment was very low. Of particular interest was the fact that the long-term corrosion rates measured for aluminium and zinc were far below the value of 1 mm/year used in previous assessments of post-closure safety for SFR. Kalinowski (2021) here supports the finding that the corrosion rate for aluminium is less than 0.1 mm/year, but it is in fact probably much lower than this, since corrosion had clearly stopped between the first and second retrievals.

Support for this conclusion can also be obtained from the study conducted by SKB in cooperation with researchers at KTH (Herting and Odnevall 2021) which is discussed in Section 6.2. For both aluminium and zinc, the study showed an initially very high corrosion rate caused by reaction between the metals studied and the wet cement paste. However, the corrosion process clearly stopped after the cement paste had hardened and the average corrosion rate moved asymptotically towards very low values, as a layer of corrosion products was built up around the embedded test pellets.

The fact that similar results were obtained from long-term experiments under repository-like conditions at the Äspö HRL as under controlled conditions in a laboratory environment can here be taken as evidence that previous experiments clearly overestimated the corrosion rate of these metals, since they did not consider long-term effects of embedded material. This shows that experiments performed under repository-like conditions and time scales provide a more complete picture of how waste is degraded in SFR.

Programme

SKB's programme for studies of metal corrosion in the cement environment during the period up to the closure of the underground facility of the Äspö HRL:

- Retrieval and analysis of the remaining specimen in the Concrete and Clay project containing metallic material and final reporting of the results from this series of experiments.

9.1.5 Impact of bentonite on cementitious materials

In the existing silo in SFR and the planned waste vault for legacy waste (BHA) in SFL, direct contact between cement and bentonite will occur. When these materials become water-saturated, the chemical interactions may lead to changes in the cement material's composition, properties and structure. Degradation products from the waste may affect the evolution processes.

Current situation

In the case of the cement pellets consisting of regular cement, a lower ratio of calcium/silicon could be observed in the interface with bentonite, while slightly elevated levels of calcium and magnesium were noted in the bentonite at a distance of up to 10 mm from the interface. For samples where the pellet consisted of low-pH cement, the affected zone was smaller and the levels lower (Mårtensson and Kalinowski 2019).

The experimental packages Concrete and Clay 17 and 19 were retrieved in 2021, and analyses of these are in progress. The focus of the analysis will be, as in the case of previous retrievals, studies of ion transport and mineral transformations in the interfaces between cement and bentonite, and dispersion of degradation products from the materials mixed into the small cement pellets in the bentonite.

Programme

SKB's programme for studies of the impact of bentonite on cementitious materials covers the period up to the closure of the underground part of the Äspö HRL:

- Analysis of the samples from the Concrete and Clay experimental packages 17 and 19 and reporting of the results.
- Retrieval, analysis and reporting of the two remaining experimental packages within the Concrete and Clay project.

On the basis of the results of the Concrete and Clay project, the following is planned:

- Modelling studies for the purpose of verifying the modelling results from previously completed studies of cement/bentonite interactions.

9.1.6 Impact of changes in binder composition and additives

In the manufacturing of cementitious materials, in addition to cement, water and aggregate, different types of additives can also be used. These materials – for example silica, finely ground limestone or fly ash – can be added either during cement production or during preparation of the cementitious material. The purpose of these additives may be to reduce the environmental impact of the material or to control the properties of fresh and hardened material. The additives may, however, affect the long-term evolution of the material during the period after closure, since they lead to a change in mineral composition compared with materials without these additives.

Current situation

During the previous RD&D period, SKB concluded the experimental investigations of the properties of concrete of different compositions (Villar et al. 2019).

Modelling studies conducted by Idiart et al. (2019) showed that the long-term chemical leaching of concrete is clearly dependent on the concrete's initial porosity and transport properties. Of the different types of concrete investigated in this study, the concrete developed for 2BMA (Lagerblad et al. 2017) showed the best properties. This was explained by its very low initial porosity rather than its chemical composition.

Programme

- A sensitivity analysis is planned to study how changes in the composition of binders in cementitious materials affect the long-term evolution of the material.

9.1.7 Long-term evolution of backfill material

In conjunction with the closure of SFR, 1–2BMA, 1BRT and 1–2BTF will be backfilled with crushed rock (macadam 16/32 mm). This material has a high hydraulic conductivity, which contributes to limiting the flow of groundwater through the waste domain and reducing the risk of rock fall-out damaging the concrete structures.

Current situation

During the previous RD&D period, SKB identified a need to improve the state of knowledge in respect of long-term evolution in the macadam backfill, since there is some uncertainty about the extent to which microbial growth on the grain surfaces of the backfill and in its pores affects its hydraulic conductivity. The studies conducted by Anderson et al. (2006a, b) showed that biofilm develops within a few months on polished rock surfaces exposed to anaerobic granitic groundwater. How biofilm growth occurs under repository conditions like those in SFR has not been studied, however.

In order to gain a better understanding of the processes that can take place in a backfill material under repository conditions, a long-term test under conditions like those in SFR has been initiated in the Äspö tunnel. Four steel containers have been installed and connected to a borehole where the water composition corresponds to what can be expected in SFR during the initial period after closure and at a corresponding depth, i.e. about 100 metres below sea level, to obtain an adequate pressure of 0.5–1.2 MPa.

The rock material used corresponds to that specified as backfill material in 1–2BMA, 1–2BTF and 1BRT. This material consists of granitic macadam with a size of 16–32 mm. The experiment started in November 2020. Two of the containers were excavated at the end of 2021 and then refilled with new material and reinstalled.

Programme

SKB's programme for studies of the long-term evolution of the backfill material will be carried out until the closure of the underground facility at the Äspö HRL. It will focus on microbial growth and accumulation of fine-grained fracture-filling minerals:

- Analysis of material from the retrieval that was carried out at the end of 2021 with a focus on the occurrence and growth of bacterial material and accumulation of fracture-filling minerals.
- Continued operation of the four experimental casks.
- Retrieval and analysis of the material in all casks.

9.2 Design of concrete structures and materials for SFR

This section presents SKB's programme for design of concrete structures, material development and production method for SFR.

9.2.1 Waste vault for intermediate-level waste, 2BMA

The planned waste vault for intermediate-level waste (2BMA) in the extension of SFR will consist of a rock vault over 250 metres in length, in which a number of free standing caissons will be built of unreinforced concrete.

Current situation

During the previous RD&D period, SKB continued the technological development work that has been in progress since 2015, and focused on the development of a method to enable the use of tie rods in the construction of formwork for the external walls of the caissons. This work was based on the large-scale test casting of a caisson on a quarter-scale carried out at the Äspö HRL in 2018 and 2019 (Mårtensson and Vogt 2020) (Figure 9-2).

There was a requirement that the external walls of the caisson should be constructed without the use of tie rods to keep the formwork together during casting, in order to avoid the formation of hydraulically permeable zones. However, during the course of the work it became apparent that the use of tie rods would significantly facilitate formwork construction, and work to identify different alternatives was therefore initiated.

The completed study (Mårtensson 2021a) showed that the use of protective tubes of concrete around tie rods in conjunction with casting could both ensure that tie rods could be removed in conjunction with formwork demolition and that they could be effectively filled with a cementitious grout in such a way that a structure with very low hydraulic conductivity could be achieved.

Measurements showed that the hydraulic conductivity of the filled and embedded tubes corresponded to that of the concrete in the external wall of the caisson. Figure 9-2 shows a cross-section of a drill core from the wall that was erected as a part of this evaluation work.

During the previous RD&D period, SKB also commissioned a number of studies concerning the method of construction of the inner walls of the caissons. The work carried out so far has been of an overall evaluative nature and has included evaluation of three different methods for the construction of inner walls:

- Placement and grouting of prefabricated concrete elements.
- Slipform construction.
- Traditional casting with fixed formwork.

The studies have shown that all methods are associated with different advantages and disadvantages, and no final choice of method has yet been made.

Figure 9-1. *Cast caisson at the Äspö HRL, externally and internally (Mårtensson and Vogt 2020).*

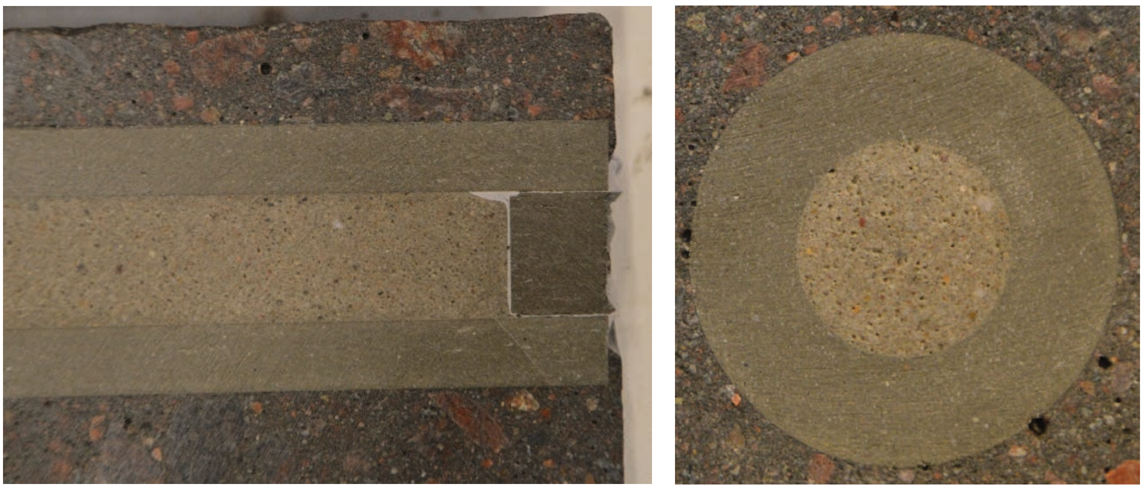

Figure 9-2. *Cross-section of a drill core from testing of different concepts for facilitating the use of tie rods in the construction of formwork for external walls for the caissons in 2BMA (Mårtensson 2021a).*

Programme

SKB's programme for development of the caissons and for determining the production method for 2BMA prior to the start of underground construction:

- Development of a method for construction of the inner walls of the caissons.
- Verifying tests of design and production technology.
- Preparation of operational and maintenance programmes.

9.2.2 Design of gas transport system

During the period after closure, waste, waste containers and any reinforcement will degrade, resulting in expected gas formation. Although some gas transport can take place through the concrete's pore system, a system for gas transport needs to be installed to facilitate passage through concrete structures with low gas permeability.

Current situation

Systems for gas transport are currently planned to be installed in the silo and 2BMA (Mårtensson et al. 2022). Studies are also being conducted into the impact of a system for gas transport in 1BMA on the post-closure safety of the repository. The studies are called for by the plans to construct new and tighter walls outside the current concrete structure and to increase the thickness of the lid that will be built in conjunction with closure, which will significantly reduce the gas permeability of the concrete structure.

The current concept for a system for gas transport entails making holes in the lid of the concrete structure that are filled with a gas-permeable material in combination with gas-conducting channels in the waste domain. These channels can either be left empty or filled with a permeable grout. The gas transport system is designed in such a way that the structural integrity of the concrete structure is not affected.

Programme

- Further studies concerning the design of the gas transport system in 2BMA. See also Section 9.1.2.
- Experimental studies concerning transport properties of different types of materials relevant for use in gas transport systems.
- Development of a gas transport system in 1BMA.

9.2.3 Mechanical properties of concrete tanks

In 1–2BTF mainly concrete tanks with dewatered ion exchange resins are disposed of. In the assessment of post-closure safety, these concrete tanks are assumed to have a certain flow resistance, which assumes that cracking does not occur during the initial period after closure.

Current situation

In 2021, SKB investigated the penetratability of water through the walls and bottom of a concrete tank. The study (Mårtensson 2021b) showed that the walls and bottom of the concrete tank were watertight and no water penetrated into the tank during the six months duration of the experiment. Given the fact that the contact surface between the concrete tank lid and the actual tank is sealed with a combination of different materials, the conclusion of the study was that the concrete tank as such will not be entirely filled with groundwater during the resaturation phase after closure. The concrete tanks will therefore be subjected to an external water pressure corresponding to the pressure at repository depth when the waste vault becomes fully resaturated.

Supplementary structural mechanical studies (Könönen and Malm 2021) showed, however, that the concrete tanks are not designed for this water pressure, which is why cracking can be expected to occur at the end of the resaturation phase, unless measures are implemented to ensure faster filling of the concrete tanks.

Programme

- Experimental investigation of the mechanical strength of concrete tanks against an external groundwater pressure.
- Extended structural mechanical calculations of concrete tanks exposed to high pressures.
- Development of a coupled model that solves the structural mechanical problem and flow through the tank wall simultaneously.

9.2.4 Repair and reinforcement of the concrete structure in 1BMA

SKB has for a number of years carried out investigations and studies linked to the current status and method for repair and reinforcement of the concrete structure in 1BMA. On the basis of these studies, SKB has decided to provide the concrete structure with new walls outside the existing walls and to increase the thickness of the concrete structure's lid. Both of these measures are intended to be implemented in conjunction with closure of the facility.

Current situation

SKB has previously shown that the planned measures will ensure that the repository's intended initial state can be achieved. In addition, SKB has also shown that new walls and the lid can be designed to withstand both the unilateral water pressure in conjunction with repository resaturation and the earth pressure from the backfill material, for at least up to 20 000 years after closure (Mårtensson 2017).

During the previous RD&D period, SKB carried out studies of the strength of the walls in the concrete structure in 1BMA against internal and external loads (Eriksson 2021). The studies showed that the walls of the existing concrete structure do not have the capacity to withstand loads from swelling waste.

In 2021, SKB commissioned studies linked to costs for repair and reinforcement of the concrete structure (Wimelius 2021) and calculation of dose load to personnel (Becker 2021). The studies showed that neither the costs nor the dose load to personnel in connection with this work would be such that they clearly disfavour a repair according to the intended methodology.

Programme

- In-depth analysis of the extent to which loads caused by swelling waste may affect existing and new concrete structures.
- Update of method for repair and reinforcement based on analysis work performed.

9.3 Design of concrete structures and materials for the Spent Fuel Repository

9.3.1 Plugs for deposition tunnels

Plugs for deposition tunnels in the Spent Fuel Repository keep the backfill material in the deposition tunnels in place and limit the transport of gas and water between deposition tunnels and adjacent parts during the period up to closure of the facility. The plugs have no function after closure.

Current situation

The final report from the full-scale plug test Domplu was presented by Enzell and Malm (2019). The test showed that it is possible to construct the plug system in an appropriate manner with an unreinforced concrete plug of low-pH concrete that meets the performance requirements (Recipe B200 developed by Vogt et al. 2009).

Since demolition and final evaluation of Domplu (Enzell and Malm 2019), no further practical work has been carried out with respect to plugs for deposition tunnels.

Programme

- Specification of requirements and technical design requirements for the deposition tunnel plug.
- Update of the plug design on the basis of the latest orientation of the tunnel cross-section of the deposition tunnels.

9.3.2 Low-pH cement materials for grouting and rock support

Current technical design requirements and requirements for the Spent Fuel Repository assume the use of low-pH cement materials for grouting and rock support between 200 metres below surface level and repository level. This is based mainly on a study of the dispersion of a plume of leaching products in the bedrock from the grouting of the ramp and shafts (Sidborn et al. 2014), in which the risk of impact on the properties of the buffer was noted.

Current situation

During the previous RD&D period, SKB did not carry out any development work on low-pH cementitious materials for grouting and rock support. For standard cementitious materials, which do not contain pH-reducing additives, analysis of the extent of the pH plume in the Spent Fuel Repository that was initiated during the previous RD&D period is in its final phase. The new analyses have, in addition to cementitious grouting materials, also included shotcrete and grout for rock bolts. The studies are based on more realistic assumptions and calculation assumptions than those used in the study carried out by Sidborn et al. (2014), and also includes analysis of some uncertainties. The results have not yet been fully analysed, but are expected to provide additional data with more realistic assumptions prior to the formulation of requirements.

The possible effects of the pH plume on radionuclide transport will also be analysed (Section 11.4.3).

Programme

- Revision of the requirements for maximum depth for the use of standard cementitious materials (i.e. without pH-reducing additives) for grouting, rock support and grout for rock bolts. This will take place when the ongoing analysis concerning the extent of the pH plume during the use of standard cementitious materials for grouting and rock support has been completed.

10 Clay barriers, plugs and closure

The main function of the clay barrier is to limit the water flow around the canister and in the deposition tunnels in the Spent Fuel Repository, around the waste transport casks and in the plugs for the silo in the Final Repository for Short-lived Radioactive Waste (SFR) and in the BHA in the Final Repository for Long-lived Waste (SFL) (Figure 10-1) and to limit advective transport in the deposition tunnels in the Spent Fuel Repository. This is achieved by means of low hydraulic conductivity in the clay, so that diffusion is the dominant transport mechanism, and by means of a swelling pressure that makes the buffer self-sealing. In the Spent Fuel Repository, the buffer will also hold the canister in place in the deposition hole, mitigate the shear movements of the rock and retain its properties during the period being analysed. In addition, the buffer should limit microbial activity on the canister surface and filter colloidal particles. An important function of the backfill in the Spent Fuel Repository's deposition tunnels is also to keep the buffer in place in the deposition hole. The clay barriers must not significantly impair the function of the other barriers.

Closure include plugs, material installed in boreholes and the material used for sealing all underground openings outside the waste vaults (the waste vaults in SFR) and SFL as well as the deposition tunnels in the Spent Fuel Repository).

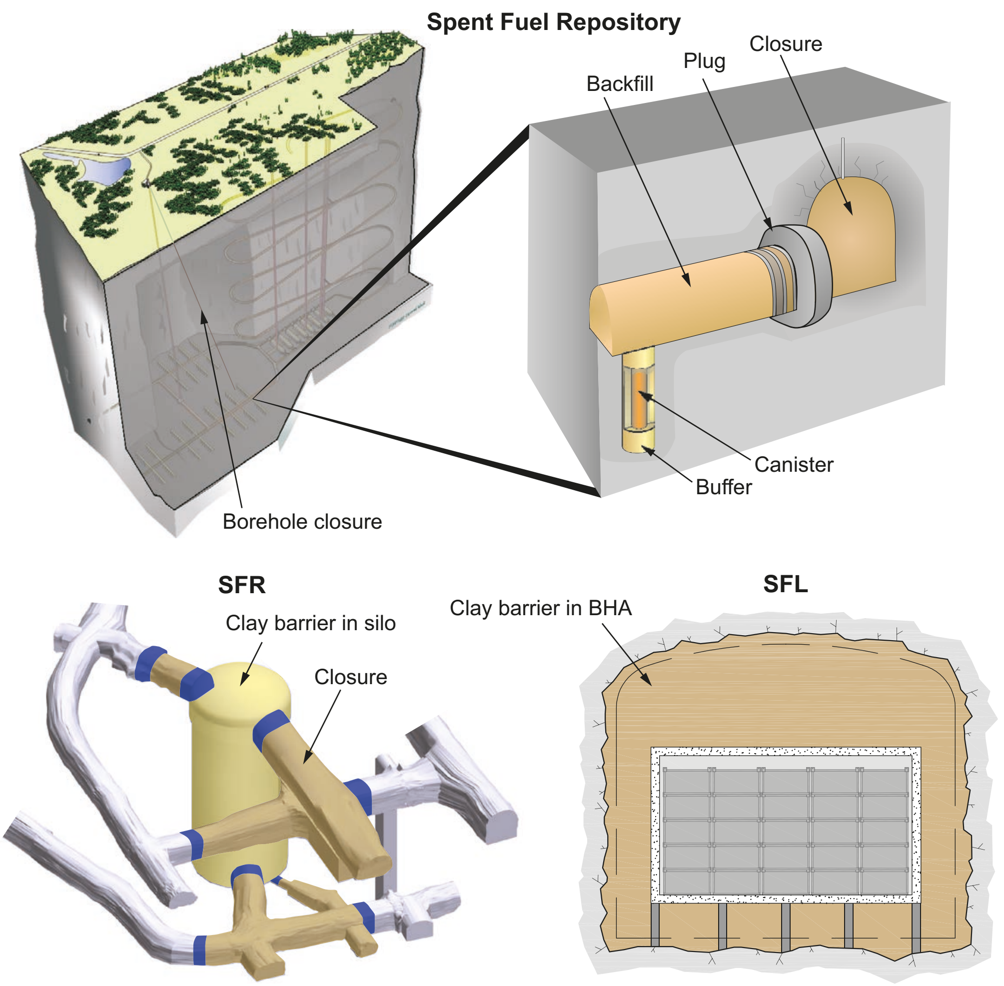

Figure 10-1. Upper: Clay barriers and closure of the Spent Fuel Repository: buffer, backfill, closure and borehole seal. Lower, left: silo clay barrier and closure in SFR. Lower, right: clay barrier in the waste vault for legacy waste (BHA) in the Final Repository for Long-lived Waste (SFL).

The closure of the repositories must maintain the principle of multiple barriers by preventing the formation of conductive water paths between the repository area and ground level and preventing the backfill from expanding out of the deposition tunnels and the waste vaults. In the upper part of the ramp and shafts, closure must considerably impede unintentional intrusion into the repository.

10.1 Evolution of the bentonite material after installation until saturation

In both the Spent Fuel Repository and the waste vault for legacy waste (BHA) in SFL, the clay barriers will be installed as a combination of compacted blocks and pellets made of bentonite. The installed barrier will therefore initially have neither swelling pressure nor low hydraulic conductivity. These properties will evolve as the bentonite absorbs water from the surrounding rock.

In order to understand in more detail and describe the bentonite material's evolution until saturation, work is required with respect to chemical evolution during the unsaturated period, channel formation/erosion, swelling, homogenisation of blocks, pellets and cavities, water vapour circulation and microbial sulphide formation under unsaturated conditions.

10.1.1 Gas phase composition during the unsaturated period

In the rock in Forsmark, the saturation time for the buffer around the canister in the Spent Fuel Repository is expected to vary from tens to thousands of years, depending on the position of the deposition hole in the rock. In most positions, the saturation time is expected to be more than 1 000 years. This means that the canister surface may be exposed to unsaturated conditions for a relatively long time. One possible problem for canister corrosion is the chemical composition of the gas in the unsaturated bentonite. Of particular interest are the oxygen content (O_2) and hydrogen sulphide (H_2S). Questions that need to be answered are:

- Will oxygen be consumed by the repository components? If so, what is the reaction rate?
- Will hydrogen sulphide be generated from minerals in the buffer?
- Can hydrogen sulphide be microbially generated in an unsaturated buffer?
- Could other gases form?

These questions are of particular interest for the Spent Fuel Repository, since the gas composition will primarily affect the copper canister. However, it is not impossible that the composition of the gas phase during the unsaturated period may also be of interest for the BHA in SFL.

Current situation

Laboratory tests that study the evolution of the gas composition in water-unsaturated bentonite have been presented by Birgersson and Goudarzi (2018), Åkesson et al. (2020) and Åkesson and Laitinen (2022). The methodology in these tests has been developed successively. Initially, a rather complex experimental set-up with a thermal gradient and a copper heater surrounded by bentonite blocks and pellets was studied, and only the oxygen content was analysed at regular intervals. In 2021, the tests focused instead on test equipment with isothermal conditions in which bentonite is placed in a glass vessel (without copper) and where five different gas components can be analysed regularly (O_2, CO_2, H_2, H_2S and SO_2) (Figure 10-2). This equipment is located in the material research laboratory on Äspö. After the initial measurements with copper heaters, the interpretation was made that oxygen was mainly consumed by aerobic copper corrosion. On the other hand, the new measurements clearly show that oxygen may very well be consumed in bentonite and that this appears to occur through oxidation of pyrite.

Programme

- Continuing investigations concerning the evolution of the gas composition. This work will mainly include investigations of how the processes are affected by different conditions, for example regarding temperature and water ratio.

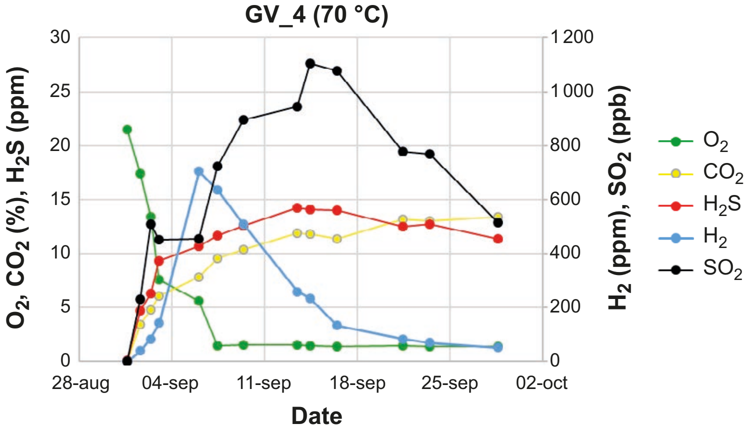

Figure 10-2. *Test equipment with glass vessel in heating cabinet (left). Evolution of gas composition during heating of Milos bentonite with a water ratio of ~17 % and a temperature of 70 °C (right). From Åkesson and Laitinen (2022).*

10.1.2 Channel formation/erosion

A hydraulic problem during the operational phase concerns formation of channel and related erosion effects in the buffer and backfill. The inflow of water to the deposition holes that is required for wetting of the buffer will take place mainly through fractures in the surrounding rock. If the inflow is localised to fractures that supply water at a more rapid rate than that the swelling buffer can absorb, the resulting water pressure in the fracture will affect 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 formation of channel in the bentonite and a continuous water flow, plus progressive erosion of bentonite particles. The continued evolution is then determined by the bentonite swelling rate, the flow rate through the buffer and the buffer's erosion rate.

The same phenomenon may occur in the backfill in the BHA in SFL. In that repository, however, the amount of bentonite is so large that a mass loss at an early stage will hardly be of any significance for the barrier's function.

Current situation

Studies of, among others, erosion, self-healing of erosion channels, channel formation and the ability to stop channel formation were carried out within the so-called EVA project (Börgesson et al. 2015). The goal of this project was to understand and develop models for critical processes that occur at an early stage after the installation of buffer and canister, and to create opportunities for setting requirements on the tightness of the end plugs in the deposition tunnels. The results of these studies led inter alia to the following observations:

- Channel formation with subsequent erosion occurs and is maintained until the water pressure gradient is taken by the plug (i.e. when hydrostatic pressure prevails behind the plug) and the flow rate out from the backfill is less than 10^{-4} L/min. This means that channel formation/erosion can continue during the entire life of the plug, i.e. as long as the downstream main tunnel is kept open.
- Erosion channels of limited radial extent (1–2 cm) will self-heal to such an extent that they do not have a significant influence on the hydraulic properties of the bentonite when stagnant water pressure conditions are achieved.

Programme

- Erosion measurements under realistic test conditions. This work is aimed at quantifying the potential redistribution of bentonite from deposition holes to the tunnel and is planned to include erosion tests (for example on a scale of ~1:10) in which both buffer and backfill are represented.
- Development of a conceptual model (Figure 10-3). The work aims to improve understanding of these complex processes, and includes both modelling and various laboratory tests.

10.1.3 Water uptake

When the bentonite blocks and pellet fill have been installed in the deposition holes and deposition tunnels in the Spent Fuel Repository, or in the backfill in the BHA in SFL, the bentonite will absorb water from the surrounding rock. During the saturation phase, the bentonite will develop a swelling pressure that mechanically affects the rock and surrounding barriers. Water transport in the unsaturated bentonite is a complicated process that is dependent on, among other things, temperature, density, montmorillonite content and water ratio in the different parts of the barrier. The most important driving force for achieving water saturation is the low chemical potential of the clay water (which can be expressed as a negative liquid pressure or suction), which leads to water uptake from the rock. The hydraulic conditions in the rock closest to the barrier determine the course of the saturation process. If the availability of water is unlimited, full water saturation will be achieved within a few years in the buffer in the Spent Fuel Repository and in tens of years in the backfill in the deposition tunnels and in the BHA. Since the water supply will be limited by the flow in the rock, water saturation will in reality take much longer.

Water uptake and the time for full water saturation are in themselves of no direct importance for the function of the barrier in any of the repositories. However, the process may be of indirect importance for the functioning of the repository, for example because of:

- the maximum temperature, where a saturated buffer has a higher thermal conductivity,
- the swelling and homogenisation process,
- the time when species (for example corroding agents) in aqueous solution can be transported between rock and canister,
- the time relating to how long a gas phase may be in contact with the canister.

The water saturation process therefore provides boundary conditions for a number of other processes.

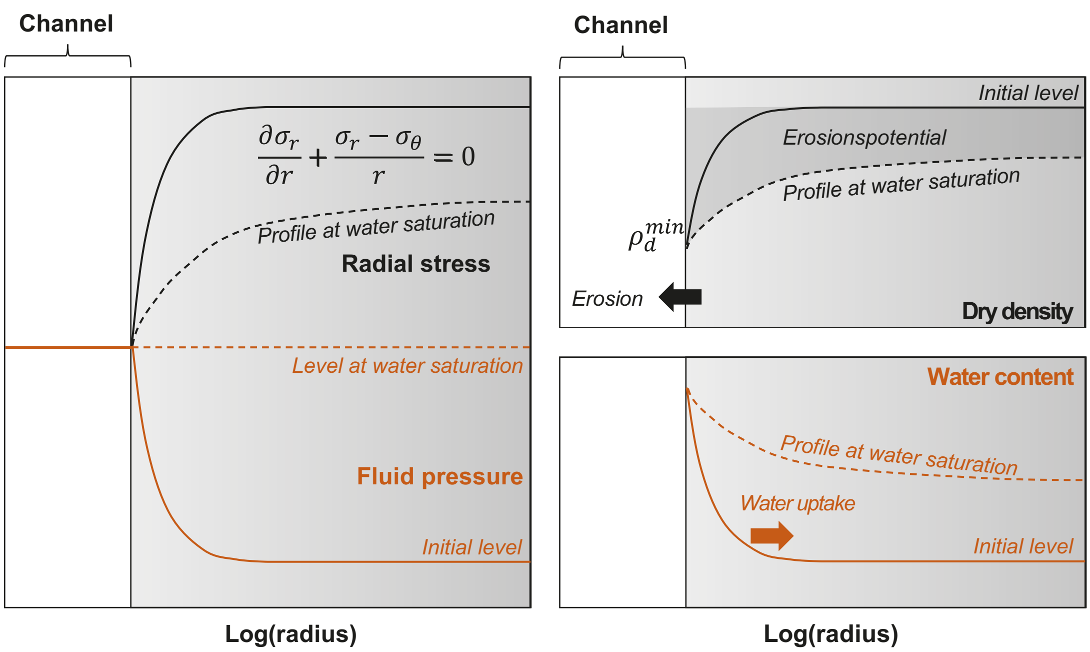

Figure 10-3. *Schematic illustration of conceptual model of channel formation/erosion. The radial stress distribution is characterised by mechanical equilibrium and by the fact that this stress at the boundary is the same as the pore pressure in the channel (upper left). The pore pressure in the channel also defines the boundary conditions for the liquid pressure distribution (lower left). Dry density distribution is strongly related to stress distribution, especially in water-saturated parts, and assumes that there is a minimum density for which the swelling pressure is zero (upper right). The water ratio distribution is strongly related to the dry density (saturated parts) and the liquid pressure (unsaturated parts). The difference between the initial density and the density profile at water saturation constitutes a potential for the maximum extent of erosion. The time scale for water uptake also determines the time it takes to achieve maximum erosion.*

Current situation

Water saturation predictions for buffer and backfill were carried out in conjunction with SR-Site (Åkesson et al. 2010). By combining these models with fracture data from a hydrogeological model of the Forsmark repository site, a description of the distribution of the water saturation time for the different deposition holes could be obtained (Sellin et al. 2017).

After this, water uptake in different experiments has been modelled, both for different field experiments (Prototype Repository (Svemar et al. 2016), BRIE (Malmberg and Åkesson 2018) and Domplu (Åkesson et al. 2019)), and for laboratory experiments where water transport in pellet-filled gaps has been investigated (Åkesson 2020, Eriksson 2019, 2020). These studies have highlighted the need to develop the conceptual description and the modelling tools that are used for water uptake predictions, both regarding the hydraulic properties of the rock and in order to be able to represent channel formation in the bentonite (Figure 10-4).

Programme

- Impact of the fracture network in the rock on water uptake. The purpose of this work is to improve water uptake predictions by including a more realistic description of the hydraulic properties of the rock, and primarily includes modelling.
- Water transport through channel formation. This work aims to improve water uptake predictions by including a representation of channel formation in the material models for buffer and backfill. This includes scale tests of a tunnel section with local inflow, in order to investigate the occurrence of channel formation and the extent to which this contributes to accelerated water uptake.
- Water uptake modelling within Task Force EBS and in conjunction with the evaluation of the Prototype Repository. These studies are aimed at developing and validating the modelling tools that are used for water uptake predictions by applying the tools to different experiments, which may vary from small-scale laboratory experiments to large full-scale experiments such as the Prototype Repository.
- Laboratory measurements of retention curves for different bentonites.
- Supplementation/revision of water uptake predictions for the Spent Fuel Repository. This work will result in an updated description of the water saturation time distribution for the different deposition holes. The work will probably not be carried out during the RD&D period.
- Supplementation/revision of water uptake predictions for SFL. This work will result in an updated description of the water saturation time for the backfill in the BHA.

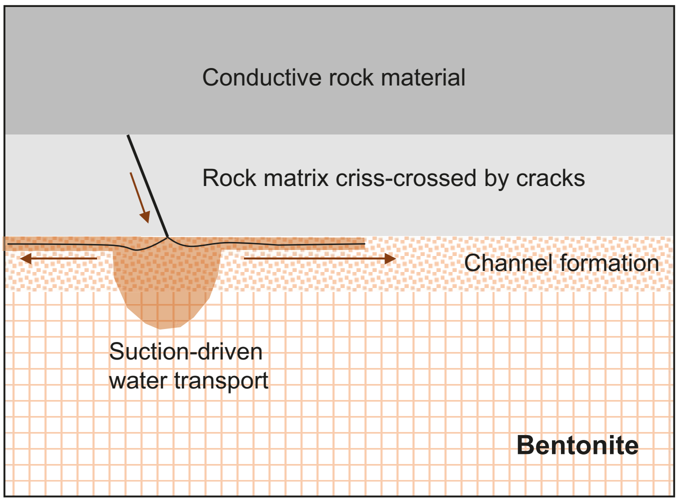

Figure 10-4. *Schematic illustration of conceptual description of water uptake in bentonite with local inflow from rock. The rock is handled as a conductive homogeneous material, except in a limited cylinder around the deposition holes, in which the rock is represented by a low-conductive rock matrix intersected by fractures. Water uptake in the bentonite is characterised by a "suction-driven" water transport and by rapid channel formation, especially along the rock wall.*

10.1.4 Swelling, homogenisation of blocks, pellets and cavities

The buffer in the Spent Fuel Repository will be installed as a combination of blocks with high initial density and a pellet fill with considerably lower density. In the current reference design, this also applies to the backfill in the deposition tunnels in the Spent Fuel Repository and in the BHA in SFL. During water uptake, the components will swell, which means that cavities are filled and that the density differences are equalised. The swelling will also mean that any loss of material, caused for example by channel formation (Section 10.1.2) or colloid release (Section 10.3.3) is able to self-heal. In the Spent Fuel Repository, the swelling of the buffer will lead to it being able to expand into the tunnel backfill, which in turn could lead to a decrease in density in the deposition hole.

A sufficient density of the water-saturated material in the clay barriers is crucial for the safety functions to be fulfilled. This also assumes that the material is sufficiently homogeneous. The swelling leads to homogenisation, but the final state will always have a residual heterogeneity. The degree of remaining heterogeneity has been studied in both laboratory experiments and field experiments, but the process is strongly linked to both initial state and boundary conditions, which means that experimental observations are only representative of the conditions under which the experiment has been conducted. It is therefore necessary to have numerical models that can simulate the mechanical evolution of the buffer with sufficient precision.

Questions that need to be clarified:

- Will the clay barriers, installed according to the reference design, be sufficiently homogenised to fulfil the safety functions? If not, can the design be improved, or does the case involving a heterogeneous barrier need to be addressed in the safety assessment?
- Can the barriers self-heal to a sufficient extent to fulfil the safety functions even after a mass loss due to erosion?
- What requirements must be set for the mechanical properties of the backfill in order for it to limit the upward swelling of the buffer from the deposition hole?

Another requirement for the buffer is that it should hold the canister in place so that it does not sink to the bottom of the deposition hole. In a water-saturated buffer, the swelling pressure together with the friction between canister and buffer will hold the canister in place. In an unsaturated buffer it is, instead, mechanical properties that prevent the canister from sinking. This is probably not a problem, but the process has not been studied.

Current situation

SKB coordinated the EU project Beacon, which was concluded in May 2022. The goal of the project was to develop, verify and validate quantitative models that can simulate the mechanical evolution of bentonite barriers. At the start of the project, there were only a few groups with experience of this type of task. After the project, there are about ten groups which can handle the issue. The conclusion from the project is that there are tools available that can predict the remaining heterogeneity in an installed barrier with relatively good precision. The results from Beacon are available at https://www.beacon-h2020.eu/, and some results are also published elsewhere (Bosch et al. 2020a, b, Narkuniene et al. 2021, Sellin et al. 2020, Gens et al. 2018, Sun et al. 2019, Villar et al. 2021, Ferrari et al. 2022).

Alongside the work in Beacon, SKB has continued with model development and experiments under its own auspices. A hydro-mechanical material model, called HBM, developed for bentonite clay, has been implemented in the general finite element solver COMSOL Multiphysics. The current HBM model has proved to have a good ability to represent the clay, as it has been used in very different situations without the need to recalibrate material parameters, and it has been successfully used in the Beacon project.

In Dueck and Börgesson (2021), three experimental studies are presented where the purpose has been to compare homogenisation of calcium and sodium bentonite, homogenisation during fast and slow wetting, and homogenisation in long tubes (Figure 10-5) after two, four and six years. In Dueck et al. (2022a) interim results from ongoing experiments are reported. The tests deal with swelling pressure and density in expansion tests in cylinders, self-healing tests where the intention is to study the consequences of mass losses, and homogenisation in long tubes where the focus is on wall friction.

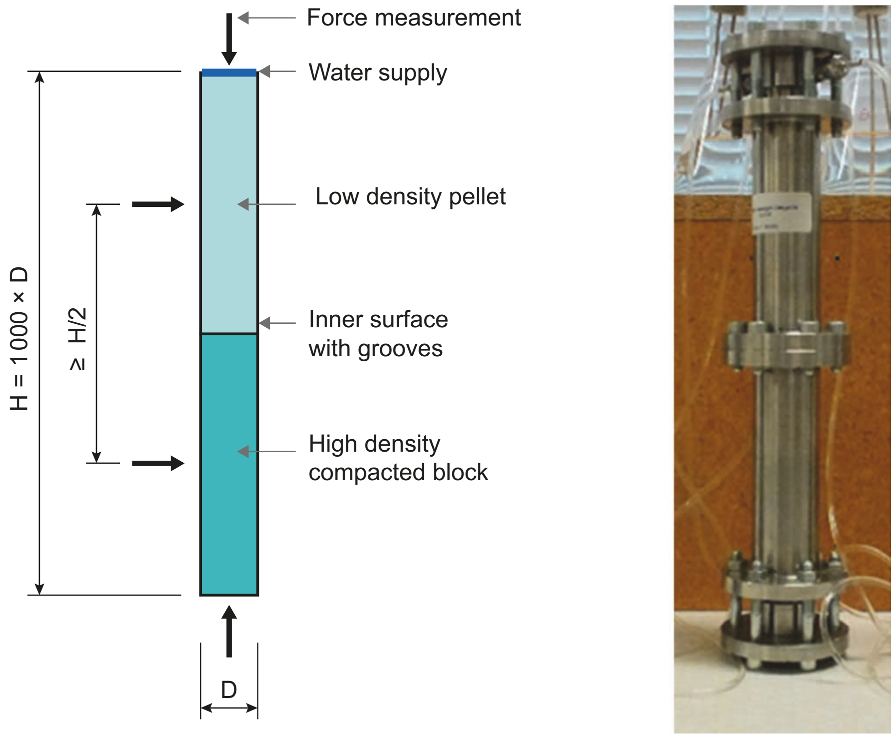

Figure 10-5. Drawing and photo of the experimental set-up with long tubes. Axial and radial pressures are measured at the arrows (Dueck and Börgesson 2021).

Swelling and homogenisation under conditions when the empty space in the experiment is large compared with the original volume of clay are presented in Harrington et al. (2020). In Daniels et al. (2021) the effect of an elevated temperature on homogenisation is evaluated.

In order to gain a better understanding of how different parameters can affect the friction between the deposition hole wall and the buffer, modelling has been initiated. It investigates how upward swelling of the buffer is affected by, for example, the position of the inflow, initial dry density and stiffness of the backfill. The calculations are done with several different models in order to estimate uncertainties in the models.

Programme

- Development and testing of the HBM model in COMSOL Multiphysics will continue. The purpose is to have access to a modern computational tool in future safety assessments and technological development.
- Laboratory and model studies of swelling and homogenisation will continue. The focus will be on operation and excavation of ongoing long-term experiments and on obtaining data for the parameters around which large uncertainties remain.
- Formulation of requirements for the ability of the backfill to counteract upward swelling of the buffer, on the basis of the knowledge obtained from modelling of upward swelling of the buffer. These requirements will be formulated as a stiffness requirement for the backfill.
- Laboratory tests to study creep in unsaturated bentonite.

In conjunction with the excavation of the Prototype Repository (Section 4.10.1)

- predictive pre-modelling of buffer and backfill homogenisation will be carried out.
- homogenisation of pellets and blocks in the buffer and in the backfill material compacted in situ will be studied.
- upward swelling of buffer material into the backfill will be measured.

10.1.5 Vapour circulation

Questions have arisen as to whether water from rock fractures can be vaporised against the canister in the Spent Fuel Repository and be transported out into the backfill, and through this process cause salt enrichment against the canister. Such enrichment of salt, if extensive, could cause corrosion (sometimes called the sauna effect). However, this issue is not relevant in SFR or SFL, since the temperature in these repositories will always be low.

Current situation

Birgersson and Goudarzi (2017) presented an evaluation of the risk of salt enrichment. A critical inflow rate of 10^{-4} L/min was derived from a theoretical calculation of an enriched mass of sodium chloride over a 1 000-year period, and this inflow rate was then related to the capacity to transport vapour, either in the inner gap or in the outer, pellet-filled gap. The experimental results from the study strongly indicated that the vapour transport capacity in a KBS-3 buffer is not large enough to facilitate significant salt enrichment.

An evaluation of the potential contribution of natural convection to moisture redistribution was presented by Sellin et al. (2017). The reason for this was that vapour diffusion was the only mechanism for moisture transport from warm to cold parts in the water uptake predictions presented in SR-Site. The water retention properties of bentonite, together with the temperature difference between the warm parts closest to the canister and the cold parts in the tunnel roof, define how far the moisture redistribution can potentially proceed. A completely equalised vapour pressure would entail extensive dehydration of the buffer. The extent of the actual moisture redistribution compared to the case with vapour diffusion alone depends on the extent of the natural convection.

The question of extensive steam transport has been raised as a result of the results of the so-called BÅT2 experiment (Nord et al. 2020). This was a full-scale installation test with a segmented buffer that was carried out over 90 days with a canister with a thermal capacity of 1 700 W. After the test period, the test was excavated and water ratios and densities were determined. The results of these measurements show that after 90 days there had already been a noticeable moisture redistribution from the rings around the canister to the cylinders above the canister (Figure 10-6). Nord et al. (2020) also point out that more work should be done to investigate how the buffer behaves during time scales longer than the installation phase.

Programme

- Further studies of moisture redistribution from buffer to backfill in order to gain a better understanding of the contribution of natural convection to moisture redistribution from hot to cold parts of the bentonite in the buffer and backfill. The work includes both laboratory tests and model development.
- Prediction of moisture redistribution and salt enrichment. This work will result in an updated description of moisture redistribution and salt enrichment in a deposition hole/tunnel section with low water inflow.

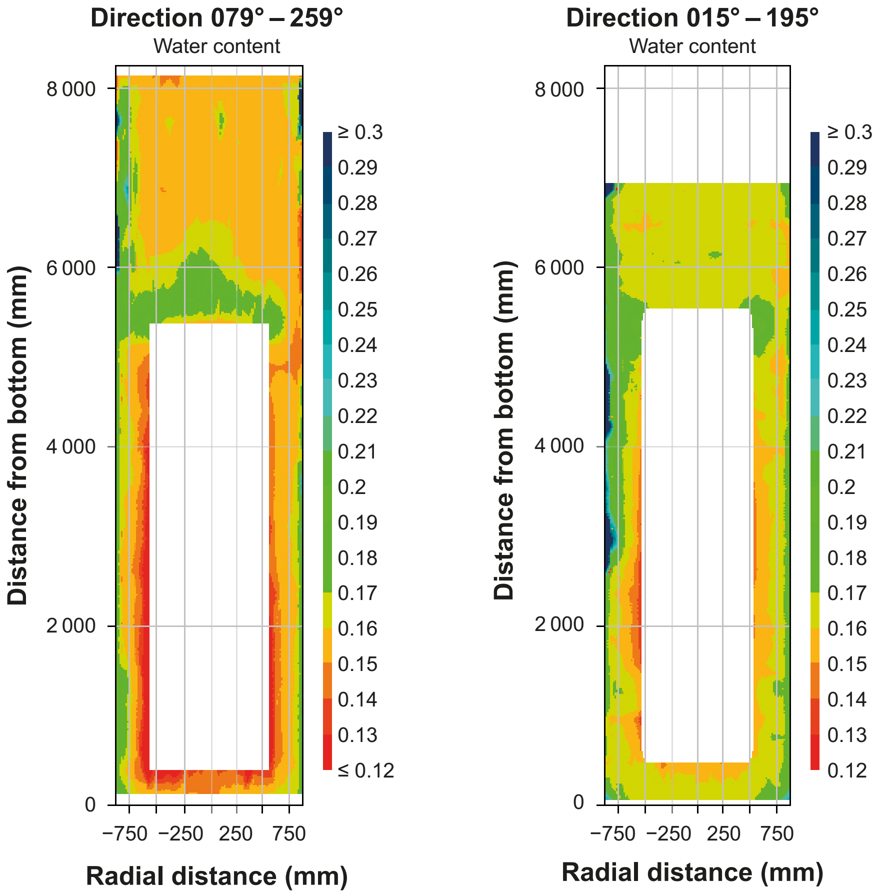

Figure 10-6. *Water ratio distribution in the buffer after 90 days of heating in two installation tests; with segmented blocks (left) and with solid blocks (right). From Nord et al. (2020).*

10.1.6 Microbial sulphide formation under unsaturated conditions

In the Spent Fuel Repository, it is likely that a number of deposition holes will be dry even in a longer time perspective. In a water-saturated buffer, no microbial activity occurs if the density/swelling pressure is sufficiently high. Nor can microbial activity occur in a completely dry bentonite. However, a window can be imagined before full swelling pressure has been reached, but where there is still a certain amount of water in the system. In such an intermediate situation, microbial transformation of sulphate to sulphide could theoretically take place, after which the hydrogen sulphide formed could potentially damage the canister. For the buffer, this can be managed by limiting the amount of sulphate that the buffer can contain, but for the backfill it is not so simple, since the amount of installed material is much greater. Another limitation of the process is the presence of nutrients for the microbes, which may originate in the form of impurities in the bentonite or come with groundwater. A requirement for the buffer bentonite is therefore that a maximum of one per cent of the mass may consist of organic material.

Current situation

A series of experiments have been carried out to investigate the activity of sulphate-reducing bacteria (SRB) in bentonite as a function of the availability of water (liquid or gas phase) (Svensson et al. 2020). The experiments used commercially available *Pseudodesulphovibrio aspoeensis,* and added gypsum (sulphate source), lactate (carbon and energy source) and nutrients. Only in experiments where there was a surplus of liquid water (visible water phase) was hydrogen sulphide formed. In experiments with 100 per cent relative humidity, but without any visible water phase, no hydrogen sulphide could be detected. This means that the hydrate layers on the surface of the montmorillonite (the micropores of the bentonite) that comprise the so-called crystalline swelling are not sufficient for hydrogen sulphide to be formed and detected.

Supplementary experiments with microbes enriched from a borehole in the Äspö HRL have been carried out (Svensson et al. 2020). These showed the same results as experiments with the commercially available *Pseudodesulphovibrio aspoeensis*. Since microbial sulphide production could not be observed experimentally at 100 per cent relative humidity, but only with liquid water, SKB sees no reason at present to further study microbial sulphide formation under unsaturated conditions.

Programme

- The current state of knowledge is judged to be sufficient and no further studies are therefore planned concerning microbial sulphide formation under unsaturated conditions.

10.2 The properties of bentonite material in the saturated state

When the clay barriers are water-saturated, hydraulic conductivity, swelling pressure and shear strength are the most important properties of the barrier function. These properties can be related to the density of a given bentonite. The relation is unique for each bentonite type and for any given type of bentonite the properties vary with its composition, for example the montmorillonite content. Hydraulic conductivity and swelling pressure are the most important properties of the clay barriers in all repositories. The shear strength, on the other hand, is of importance above all for the buffer in the Spent Fuel Repository.

10.2.1 Material composition

The main purpose of the material research laboratory for analysis of and experiments with bentonite material at the Äspö HRL is to serve as an infrastructure for research, development and quality control of bentonite and to stimulate internal competence development. The main methods used in the laboratory are cation exchange capacity (CEC), powder X-ray diffraction (XRD), X-ray fluorescence spectroscopy (XRF), exchangeable cations (EC), water content, bulk density, grain size distribution, swelling pressure, hydraulic conductivity, shear strength, compaction properties and thermal conductivity.

Current situation

Analysis of material from field experiments such as ABM5, LOT A3 and LOT S2 and of a bentonite-like material from the Kiruna mine is in progress. Refinement and testing of the methods, as well as ongoing characterisation of interesting bentonites, have taken place to some extent.

Programme

- Continued evaluation of different bentonite materials and samples from field experiments with respect to hydromechanical and mineralogical properties.

10.2.2 Swelling pressure and hydraulic conductivity

Swelling pressure and hydraulic conductivity of bentonite are two important parameters subject to requirements. The swelling pressure must be high enough to inhibit microbial activity, but not so high that it mechanically damages the copper canister. The hydraulic conductivity should be so low that diffusion becomes the dominant transport process.

More than ten measurement cells have been installed in the research laboratory to measure swelling pressure and hydraulic conductivity. These studies mainly study what the swelling pressure looks like in different bentonites and how the swelling pressure is affected by e.g. the montmorillonite content. Most bentonites behave in a similar way, but there are examples of materials that deviate and may have a considerably higher swelling pressure than other bentonites at a given density.

Current situation

Measurement of swelling pressure and hydraulic conductivity has been performed on a number of different bentonites, as well as on samples from field experiments ABM5, LOT A3 and LOT S2. In ABM5, a number of different bentonites were exposed to temperatures well above the target temperature in the KBS-3 design, and the results indicate that heating had no significant impact on either swelling pressure or hydraulic conductivity.

Programme

- Continued characterisation of reference materials, i.e. bentonites from different sites, mainly through measurement of swelling pressure and hydraulic conductivity.
- Additional measurements of samples from long-term field experiments, including from the Prototype Repository's inner section and a number of packages from the experiments with alternative buffer materials (ABM), as well as LOT S3. The results of ABM5 will be compiled and reported.
- Studies are planned to clarify why investigated bentonite from Bulgaria has a swelling pressure that differs significantly from other investigated bentonites.

10.2.3 Shear strength

Bentonite shear strength is an important parameter for designing the capacity of the canister and for the evaluation of the shear case in the assessment of post-closure safety for the Spent Fuel Repository. The shear strength increases with density and swelling pressure, but the measurements that have been carried out indicate that it may also be affected by other parameters. Since SKB has defined a technical design requirement for shear strength, it is necessary to test all relevant buffer materials in respect of this. Shear strength is best determined by a triaxial test, but can also be determined by the simpler uniaxial test. Since the requirement for strength in the technical design of the buffer is expressed in terms of uniaxial compressive strength, the uniaxial compression test is a suitable test type for more detailed studies of strength.

Current situation

An extensive study of the strength of bentonite has been reported in Dueck et al. (2022b). One conclusion from this study is that shear strength is more related to swelling pressure than to the density of the material. Shear strength is also affected by ion exchange, dehydration and heating, but the effects are small.

The state of knowledge regarding the shear strength of bentonite is considered to be good. A practical measurement method exists that can be applied to check that selected bentonite materials meet requirements, and at present there is no need for further research in this area.

Programme

- In conjunction with the excavation of the Prototype Repository, the uniaxial compressive strength will be measured on samples from the buffer in the deposition hole for the purpose of verifying that the measured values agree with results from previous studies.

10.3 Evolution of bentonite material after water saturation

When the bentonite has reached water saturation and the desired properties have been achieved, it is important that these properties are maintained throughout the time the repository is expected to be functioning. For a better understanding of the evolution of the bentonite material after saturation, work is required above all on buffer loss due to colloid release/erosion and on microbial sulphide formation and sulphide transport in buffer and backfill. Studies are also required on long-term stability with regard to temperature, iron content and cement (BHA in SFL). Smaller-scale work is needed on bentonite/copper interaction. Experiments may also need to be carried out to verify adequate handling of diffusion of solutes, especially sulphide, in bentonite. Another question is how gas, which is formed by corrosion or other processes, can penetrate a bentonite barrier without changing the properties of the bentonite or the other barriers.

10.3.1 Gas transport

If a copper canister in the Spent Fuel Repository is damaged and water comes into contact with the insert, hydrogen gas will be formed inside the damaged canister. Dissolved gas is slowly transported through the bentonite buffer. It is highly likely that a gas phase and gas pressure will build up inside the canister, and it is therefore important to show that this pressure will not result in any negative consequences for the function of the repository. This means that the gas must be able to escape without damaging the buffer or rock. Much the same question applies to the BHA in SFL.

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.

Current situation

Lasgit was commissioned in February 2005 and excavated in 2020–2021. The test was a full-scale test involving a canister in a deposition hole, and the aim was to study gas transport through bentonite. A total of six gas injection tests were carried out in filters on the canister's lateral surface and one test on the bottom of the canister, where the entire canister's empty volume was pressurised. All results from Lasgit are reported in Cuss et al. (2022). The results clearly show that gas penetrates a bentonite buffer at a pressure that is very close to the swelling pressure in the buffer. There is nothing to indicate that the pressure build-up and gas transport would have any impact on either the bentonite buffer or the properties of the other barriers. These conclusions can also be applied to the bentonite in the BHA in SFL.

Programme

- The state of knowledge regarding gas transport in bentonite is assessed to be good, and no further studies are planned in this area at present.

10.3.2 Sulphide formation and sulphide transport

Sulphide dissolved in the pore water in the bentonite can act as a corrosive agent for the copper canister. In order to assess the diffusive transport of sulphide in the bentonite to the canister, it is necessary to understand the concentrations of sulphide that may occur in the pore water. Microbial processes can under certain conditions give rise to the formation of sulphide, and the dry density or swelling pressure of the bentonite has a great impact on microbial activity (Section 10.1.6). Sulphide is basically only a problem for the copper canisters in the Spent Fuel Repository, which means that these processes are not as relevant for SFR and SFL.

Current situation

The focus during the previous RD&D period was on testing, further developing and gathering data using a new test set-up, designed to determine the density limit for microbial sulphide production in bentonite (Figure 10-7). The method is based on that presented in Bengtsson et al. (2017), with the difference that the sulphide produced is analysed by X-ray fluorescence instead of using radiometric technology. Sulphide measurements can be supplemented with measurements of consumed nutrients and sulphate. This makes the tests considerably easier to carry out and allows more tests to be carried out within a given time.

SKB has tested the diffusivity of sulphide in bentonite using ion-selective electrodes (Hedström 2022). The measured effective diffusivity is $\sim 5 \times 10^{-12}$ m^2/s for dry densities in the range 1 500–1 550 kg/m^3.

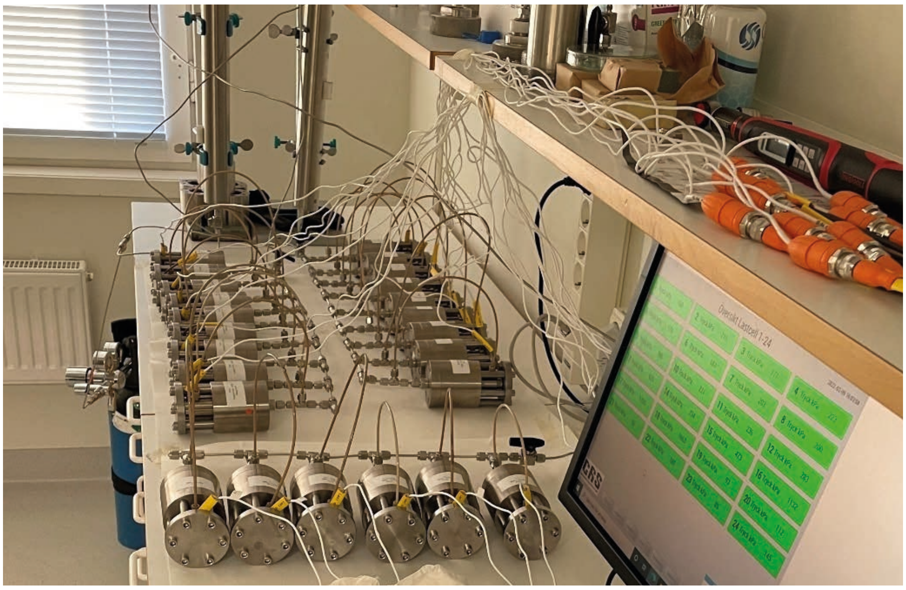

Figure 10-7. *Test cells with bentonite, copper and microbes during water saturation. Following water saturation, nutrients and sulphate are added. Sulphide formation on copper as a function of the density of the bentonite can then be measured.*

Programme

- Investigations to experimentally show the threshold for the survival of microbes expressed in the density or swelling pressure of the buffer will continue. The focus will be on generating data to obtain good statistics, trying to determine which factors are important and refining the actual methodology.

10.3.3 Colloid release/erosion

After installation of the bentonite buffer in the Spent Fuel Repository, it absorbs water and swells. The swelling is restricted by the walls of the deposition hole, causing a swelling pressure to develop. The same applies to the backfill in the BHA in SFL. If fractures intersect the deposition hole or the waste vault, there are no rigid obstacles to swelling at the intersection surfaces. Locally, the swelling then continues into the fractures until equilibrium or steady-state conditions are reached. This unrestricted swelling may lead to separation of individual montmorillonite layers (dispersion) and some of the buffer may be transported away with the groundwater. The maximum unrestricted swelling of bentonite is strongly dependent on charging and concentration of the ions in the interlayer spaces. If the concentrations of solutes in the groundwater are too low, the distance between the individual montmorillonite layers may increase to the extent that the clay/water system takes on a colloidal character, i.e. single or small groups of montmorillonite layers behave like individual colloidal particles. The local salt concentration in the pore water, together with the ratio between monovalent and divalent ions in the montmorillonite at bentonite/groundwater contact, is crucial for possible colloid formation in a repository.

The mass loss through colloid release/erosion can be calculated on the basis of:

- Clay that expands out with swelling in the fractures in the wall of the deposition hole. This mass will be extremely small if the swelling is limited.
- Clay released as colloids at the interface between the swelling clay in a fracture and the groundwater, which is then transported away with the flowing water.
- Clay released as colloids at the interface between the swelling clay in a fracture and the groundwater, which is then transported away by gravity in sloping fractures.

In the post-closure safety assessment, SR-Site, the significance of colloid release and erosion could not be dismissed, and in a few deposition holes the calculated mass loss was so great that advective conditions could not be ruled out.

Current situation

The model used to calculate mass loss in SR-Site (Moreno et al. 2010) was then further developed, and the updated model is documented in Neretnieks et al. (2017). However, the new model also contained a number of uncertainties. These mainly concerned crack expansion, handling of a secondary gel or flocculant formation, friction and handling of gravity. During the previous RD&D period this model was further developed, and the results are documented in Pont et al. (2020) and Pont and Idiart (2022). These sources present a numerical model that handles wall friction, flow together with chemical erosion, and sedimentation due to gravity. The model has been developed and partially validated with experimental data from small-scale tests.

Experimental work to further study the processes has been conducted throughout the period, involving experiments in artificial fractures. Expansion, erosion and sedimentation as a function of e.g. fracture aperture, fracture roughness, horizontal and vertical fractures, bentonite type and flow were studied in the experiments. Figure 10-8 shows an experimental set-up with sedimentation tests.

Programme

- Continued development of the model presented in Pont and Idiart (2022), with a focus on optimising the model numerically, validating the model against laboratory experiments and including flocculation formation and transport of flocculated particles. The intention is also to try to simplify the model to an analytical expression that can be used to calculate mass loss in safety assessments.
- The experimental programme to investigate colloid release/erosion will continue with roughly the same scope and focus as before. The intention is to produce data and results that can support model development and model testing.

10.3.4 Self-healing of bentonite

In the event of mass loss of bentonite from a barrier, for example due to erosion, it is important to be able to understand how the barrier self-heals. This is a part of the mechanical evolution of the bentonite, and the programme for this is described in Section 10.1.4.

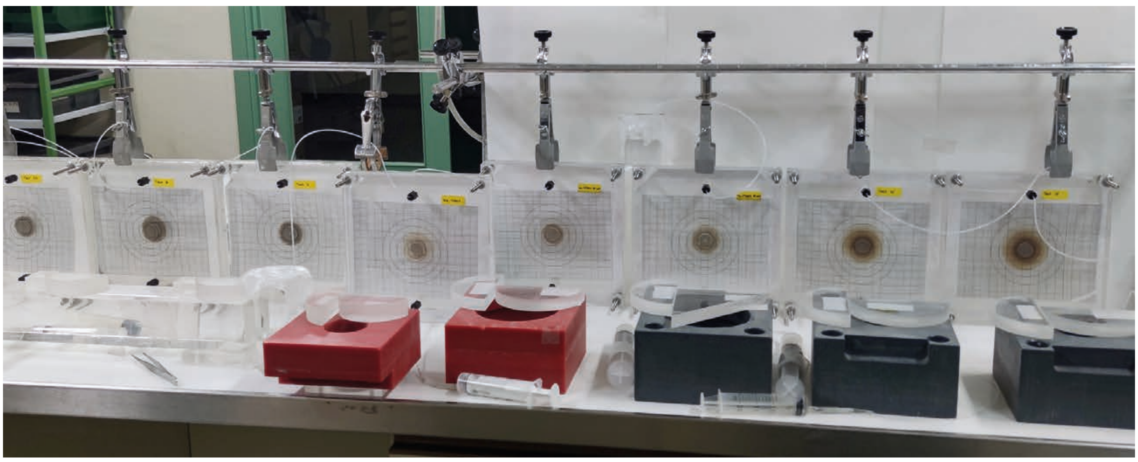

Figure 10-8. *Experimental set-up with sedimentation tests in different types of artificial fractures.*

10.3.5 Mineral stability

Bentonite has been selected as a barrier material because it is expected to be stable in the long term in the environments that exist in the different repositories. Bentonite can be stable for hundreds of millions of years in its formation environment, but changes in the environment can lead to a relatively rapid change in the mineral structure. The factors that primarily determine the stability are temperature, availability of potassium and pH. Redox conditions may also be important. Potassium concentrations in Swedish groundwaters are generally low, but there may be relatively large quantities of potassium in the rock. An increased temperature is expected during a relatively brief period in the buffer in the Spent Fuel Repository, while the backfill and the bentonite barriers in SFR and SFL are never exposed to an elevated temperature. In the Spent Fuel Repository, the interaction between high pH and bentonite is avoided by means of a requirement for low-pH materials, whereas this process is very substantial in both SFR and the BHA in SFL.

Bentonite can be degraded by ionising radiation, but at the relatively low radiation levels that are expected in all repositories, this process can be considered negligible.

Important field experiments for studying mineral stability in bentonite clay include LOT and ABM at the Äspö HRL. In LOT, ageing of MX-80 bentonite is being investigated with a copper heater at 90 °C and 130 °C. In ABM, more than ten different bentonites from different parts of the world are being investigated with a steel heater at 130 °C (except for test package 5 (ABM5), where the effects of higher temperatures were investigated). In both types of experiments, compacted bentonite rings with a diameter of 30 cm are used, stacked around the heater, which is a total of four metres high in LOT and three metres high in ABM.

Current situation

During the previous RD&D period, work involving bentonites from ABM5, LOT A3 and LOT S2 was carried out, and is still in progress. The field experiments ABM2 and ABM5 are planned to be reported together during 2022.

In ABM5, mineral stability at very high temperatures (around 200 °C) was studied, without applied water pressure to counteract boiling. This meant that it was a relatively dry experiment with minor signs of chemical transformation or impact, but all the more physical impact due to the fact that blocks cracked and fell apart. Smectite alteration could not be observed in ABM5, nor were any new clay minerals observed either in SKB's own studies or in Kaufhold et al. (2021). Kumar et al. (2021) used the Rietveld method of evaluation of X-ray diffraction data to show that the smectite content remained constant after the experiment, but SEM/EDX showed some signs of chemical change in the clay minerals in close contact with the iron heater. The chemistry closest to the heater is affected by many factors, such as iron from corrosion or accumulation of magnesium, calcium, sulphur or chloride. This means that this type of interpretation, based solely on chemical data, needs supporting data from diffraction or spectroscopy in order to be confirmed. However, CEC was affected somewhat by the high temperature and dehydration, which was observed both in SKB's own measurements and by Kumar et al. (2021). This differs from other SKB experiments in which the bentonite has been water-saturated and the temperature has been lower (90 °C–130 °C).

After recompaction of the bentonite in ABM5, no significant changes could be seen in key properties such as swelling pressure or hydraulic conductivity (Figure 10-9), confirming the conclusion from the mineralogical investigations that no degradation of smectite has taken place and that the CEC change is either reversible when the smectite is re-saturated during swelling pressure measurement, or that it has no impact on the properties.

During the RD&D period, the experiments LOT A3 and LOT S2 at the Äspö HRL have been excavated. The early analyses focused mainly on the chemical content, water content and other overall parameters of the bentonite, and on copper corrosion in contact with the heater and the copper samples placed in the bentonite (Johansson et al. 2020). As in several other field experiments involving bentonite, magnesium was observed to accumulate against the heater, for unknown reasons. Similarly, gypsum had accumulated at or near the heater, while chlorides showed no signs of accumulating against the heater. In the early overall evaluations of mineralogical content in the packages, no significant changes could be observed (Johansson et al. 2020). The work is, however, ongoing and a report with a greater focus on the mineralogy and chemistry of the bentonite will be compiled when all experimental work has been completed. Unpublished Mössbauer data (Åbo Akademi, Finland) for LOT A3 and LOT S2 indicate that some of the iron in the bentonite has been reduced from Fe(III) to Fe(II). Fe(II) is estimated to account for around 30 per cent of Fe(total) in the reference clay, while in material from the field experiment it was around 50 per cent Fe(II). The change was the same near the canister as close to the rock, and no difference could be seen between the A3 and S2 tests. Similar changes were also observed in bentonite samples from the outer section of the Prototype Repository, but the reduction mechanism is not yet explained (artefacts from the electric heaters cannot be ruled out). No irreversible effects of a partial reduction of Fe(III) have been found on the properties of montmorilllonite, but it cannot be excluded that the reduction may have some effect on the properties under reducing or oxygen-free conditions.

Dioctahedral smectite has been identified in the Kiruna mine, down to a depth of at least 1 200 metres below the surface. The smectite is relatively pure and swells with water, exactly as expected. The Kiruna mine is one of Europe's largest iron ore formations, and transition zones of soft clay up to around 50 metres thick have occurred in close proximity to magnetite-apatite deposits. There are indications that the smectite in Kiruna may be extremely old and therefore potentially very interesting as a natural analogue, if it can be dated. The rock and the hydrological conditions in the Kiruna mine are similar to those in Forsmark, and the occurrences of smectite are of the same order of magnitude as the bentonite barriers, which makes it an interesting analogue to a KBS-3 repository. The work on characterising Kiruna smectite in samples taken from different positions in the mine is ongoing.

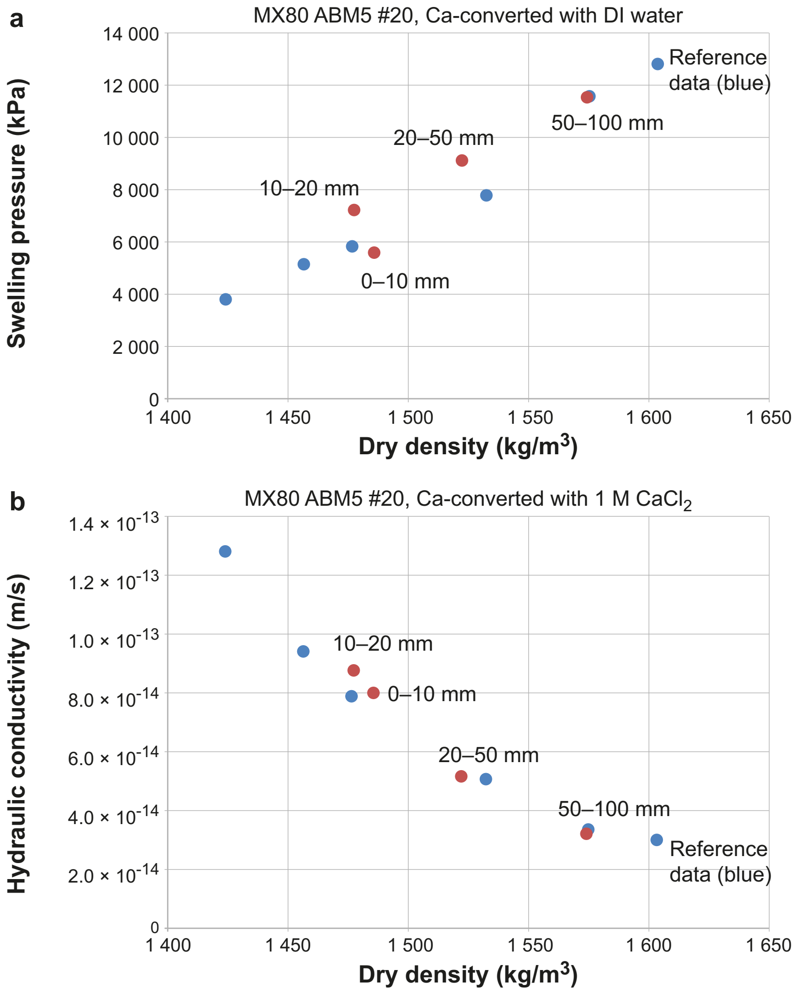

Figure 10-9. *(a) Swelling pressure and (b) hydraulic conductivity at different densities for ABM5 samples of Wyoming bentonite (MX80). The red dots are from the field experiment and the blue dots are reference data from the raw bentonite. The distances indicate the distance between the sample and the heat pipe of the experiment.*

Programme

- Continued analysis and reporting of the experiments LOT A3 and LOT S2.
- Reporting of results from the completed studies on ABM5, LOT A3 and LOT S2.
- Excavation and analysis of remaining field experiments at the Äspö HRL (inner section of the Prototype Repository, LOT S3 and the remaining three ABM packages).
- Mineralogical and hydromechanical studies of material from Kiruna.
- Experiments on different scales to investigate whether different bentonites are chemically neutral against metallic copper and to improve the state of knowledge of the possible corrosion products that are formed.
- Continued development of the THC model in order to better describe bentonite-cement interaction, retrograde dissolution of minerals and redistribution of anions during the unsaturated period. The model will be tested within the Prototype Repository and Task Force EBS.
- SKB plans to apply for time to perform syncrotron measurements on bentonite samples with XANES (X-ray absorption spectroscopy) in order to continue to gain a better understanding of the redox chemistry of the iron in bentonite. If possible, measurement is also planned on samples that have been in contact with copper in order to improve understanding of the corrosion products formed.

10.3.6 Transport of radionuclides

Sorption of radionuclides is one of the most important retarding safety functions in SFR. Sorption takes place primarily on the cement in the barriers and the waste matrix, and is dependent on the chemical composition of the water in the repository. In the silo, however, sorption on the bentonite material in the silo wall may also be of importance for certain radionuclides. If the properties of the silo bentonite change, for example through interaction with cement, the sorption properties will also change.

In the Spent Fuel Repository, sorption of radionuclides on the buffer material is of secondary importance for the function of the repository, since the expected service life of the canister is so long. However, in SFL-BHA, sorption in the bentonite buffer may be of importance for the retardation of certain radionuclides.

Programme

- Supplementary experiments to measure sorption of radionuclides in mixtures of bentonite and transformation products (zeolite) and in pure bentonite.

10.4 Barrier design

10.4.1 Buffer in the Spent Fuel Repository

Current situation

Since the RD&D Programme 2019, SKB has decided to switch to so-called segmented buffer blocks. This means that the buffer blocks have been divided into smaller segments that are much easier to produce (Figure 10-10). The smaller blocks are adapted for production in standard presses used in the ceramic industry. Production of the blocks can thus be automated to a large extent, and therefore be made more efficient. Experiments have been conducted to study how the segmented buffer behaves during the installation phase. These tests have been done on a full scale (Nord et al. 2020) with a heater and natural water inflow from the rock. The results show that the differences compared with the previous design with solid blocks are small. These results apply to the installation phase, and how the blocks behave thereafter is dealt with in Section 10.6 onwards.

Investigations have been carried out of how water uptake occurs with different pellet types (Lundgren and Johannesson 2020) to gain a better understanding of how the pellet type affects the function of the buffer. The results indicate that the same type of pellet that is used in the backfill could be used in the buffer.

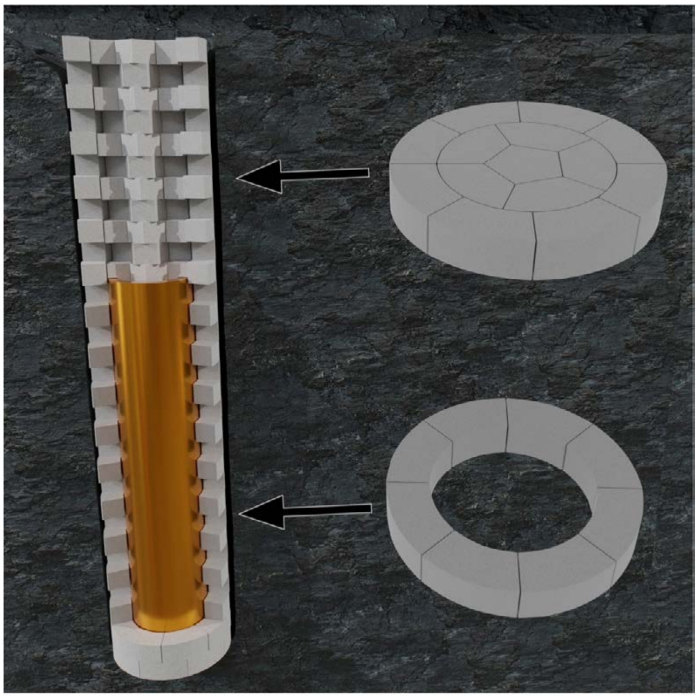

Figure 10-10. Drawing of the design of the segmented blocks.

Development of the buffer design mainly focuses on adapting the components in the buffer in order to be able to manufacture them industrially. This includes setting realistic requirements and tolerances, both for individual components and for the installation. How the water inflow in the deposition hole will affect the installation process also needs to be studied further, both through modelling and in full-scale tests, in order to determine how quickly installation needs to take place. In the process of adapting the products for industrial production, evaluation will also be carried out of whether the same pellet type that is used in the backfill can also be used in the gap between block and rock wall.

Programme

- Full-scale tests to determine limits on how long the installation of buffer can take with different water inflows into the deposition hole. Data from these experiments will then be used to develop and calibrate THM models.
- Establish requirements and tolerances for all constituent components in the buffer and adapt them to production.
- Evaluate the possibility of using extruded pellets instead of roller compacted pellets in the buffer.

10.4.2 Backfill in the Spent Fuel Repository

Current situation

The design of the backfill has been adapted to the latest orientation of the cross-section of the tunnel. How the deposition tunnels are produced and their resulting geometry affect the quantity and geometry of the backfill components and thereby the dry density achieved at installation. The dimensions of the deposition tunnels therefore affect both the design of the backfill and the production of backfill components, and the installation is also affected. After the buffer upward swelling test (Sandén et al. 2017), which was done at the Äspö HRL, a proposal has been prepared for strength requirements for the backfill blocks.

Much of the planned work for backfilling, as well as for the buffer, entails adapting the products in the backfill so that they can be manufactured industrially. This means adapting the tolerances and requirements for the backfill components. This work also includes optimising the design in respect of the updated requirements for stiffness of the backfill that are being developed, in order to ensure that the buffer does not swell up more than is acceptable in the backfill. In addition, further development of methods for handling water inflows during installation of the backfill will be needed.

Alongside continued development of the current design of the backfill, alternative designs will also be studied. One of these alternatives is a so-called granular backfill, where the deposition tunnels are filled with a bentonite granulate that has been processed so that a high dry density can be achieved. This method could simplify installation and manufacture of backfill, but is estimated to need further development in order for SKB to be able to change the design. Granulated design for KBS-3 has been introduced and evaluated by Posiva in Finland (Posiva 2021c).

Programme

- Continued study of the advantages and disadvantages of granular backfilling, where experiments and studies conducted by Posiva will be looked at and evaluated.
- Adapt the design of the backfill according to updated requirements on its stiffness. The requirements for stiffness of the backfill are important in order to ensure that the backfill is able to counteract upward swelling of the buffer.
- Development of methods for handling high water inflows during installation of the backfill.

10.5 Production, inspection and testing of buffer and backfill components

10.5.1 Quality assurance of bentonite material

Current situation

The method for adapted design of buffer and backfill has continued to be developed during the previous RD&D period, and Kronberg et al. (2020) describes the plans for how to proceed with sampling and dimensioning of the dry density of the buffer and backfill.

Measurements of bentonite material will need to continue in order to gain a better understanding of the materials and to obtain a larger body of statistical data. New methods that have been developed will also continue to be developed, for example to measure thermal conductivity of pellets and strength of unsaturated bentonite blocks.

Programme

- Continued development and application of measurement methods for quality assurance of bentonite material.
- Development of a preliminary qualification strategy for bentonite material.
- Preparation of a sampling programme for bentonite deliveries.

10.5.2 Production of buffer components

Current situation

When the decision was made to switch to a segmented buffer, full-scale tests were conducted to investigate how the segmented buffer functions under realistic conditions during the installation phase. For this experiment, more than 300 segmented buffer blocks were produced according to specifications, with good results (Nord et al. 2020). Production of the large blocks that comprised the reference design before segmented buffer blocks were introduced also took place during the period. This was done using different bentonite materials (Johannesson et al. 2020). Problems occurred in production of one of the materials because it consisted of a very fine material, which meant that it was difficult to get al. the trapped air out of the block during compaction. This led to the blocks cracking after compaction, which shows the importance of control of the granular size distribution of the material.

The production system for buffer components has been reviewed in order to try to simplify and optimise the production process.

Programme

- Further tests with segmented blocks to improve understanding of production.
- Manufacture of blocks for demonstration tests and execution of such tests, to test the methodology prior to future qualification and to further demonstrate that the method is able to meet the required specifications for the blocks.

10.5.3 Production of backfill components

Current situation

Since the buffer blocks are now planned to be produced using the same type of presses as the backfill blocks, the experience gained from buffer production can also be applied to backfill production. Exactly as in buffer production, the production system for backfill components has also been examined and efforts have been made to streamline the system.

Programme

- The backfill production system will continue to be developed and optimised in order to be able to build a facility that is as efficient as possible. Blocks will also be produced prior to the planned demonstration tests.

10.6 Disposal and installation of buffer and backfill

10.6.1 General information regarding machine development

Current situation

The machines and equipment related to installations in the deposition tunnels go through a number of developmental steps in accordance with the technological development model that has been developed at SKB. So far, all machines and equipment have gone through the concept phase. During the work with the concept phases, some challenges emerged, while certain conditions in adjacent parts of the final repository design have been changed or may be changed. It is important to take changing conditions into consideration in future phases of technological development.

Programme

- Most of the equipment is ready to enter the design phase during the RD&D period. In this phase, the information received and the tests carried out during the concept phase will be used to start designing the standard equipment required in the final repository. Furthermore, electrification of vehicles and equipment will be an important focus.

10.6.2 Disposal

Current situation

Two prototypes of the deposition machine have been built, both of which have been tested at the Äspö HRL. The tests have shown both strengths and weaknesses, and these experiences will form the basis for the coming design phase. Some surrounding parameters, such as roadways and cross-sectional area of the deposition tunnel, may still affect the design to some extent.

A new concept for covering deposition holes has been developed. The plan was previously to use a shielded floor hatch, which was planned to be fixed in relation to the deposition hole and to offer a fixed point in relation to the position of the hole. The most recent prototype of the deposition machine therefore read the position of the shielded floor hatch for positioning relative to the deposition hole. The new concept for covering the deposition holes entails a loose cover that is not fixed in relation to the deposition hole and which therefore cannot act as a reference point for the position of the deposition machine. A new method for positioning the deposition machine in relation to the deposition hole therefore needs to be developed and verified.

Programme

- The design phase for deposition equipment begins when the final prerequisites have been established.
- Further development of lid handling equipment.

10.6.3 Installation of buffer

Current situation

A concept for buffer handling equipment has been developed. The body consists of a frame structure with a gantry crane and is equipped with lifting boards and conveyor belts for pallet handling. Even though the equipment has been developed for the previous buffer blocks, it is also expected to work well with segmented blocks. Pellet cylinders that can be handled by the buffer handling equipment are used to handle pellets for the buffer. The pellet cylinders have been tested, and function as intended.

If the buffer is designed as sectioned blocks, these must also be laid as a brick wall in order for the stack to be stable. This means that the gaps between the sections should not lie directly above each other, but that each layer should be turned so that the gaps create a pattern similar to a brick wall. This requires that the tool can be turned between each level installed. The previous prototype of the lifting tool does not have this function, and the tool therefore needs to be redesigned and supplemented with both a swivel function and a measuring equipment/control system for this function.

Programme

- Develop the process for installation of segmented buffer.

10.6.4 Installation of backfill

Current situation

The backfill is intended to be installed by an industrial robot that lifts one block at a time from a transport pallet and places it on the stack of blocks. This method has previously been tested on a full scale under realistic conditions (Arvidsson et al. 2015). Since then, development has continued and the method has been refined. Optimisations to increase installation speed are planned.

Programme

- Further develop the method for pellet installation.
- Optimisation of the installation robot and transporter.

10.7 Borehole sealing

Current situation

Development of the method for borehole sealing continued during the previous RD&D period. This includes optimisations and changes in the choice of materials in order to streamline both production of components and practical implementation, while keeping the function unchanged.

Alternative solutions for the copper expander used today have been investigated, as the original design requires high precision in diameter and also requires a drilling rig for it to be installed.

As regards concrete as a closure component, SKB has decided to abandon the use of low-pH concrete, since this has not met the requirements for reliability in installation tests. The concrete now specified in the method is Weber underwater concrete. Because a risk of erosion has been identified in connection with the filling of sections with sand, SKB has also decided to use a somewhat coarser standardised fraction size.

The existing sealing method was mainly developed for subvertical-vertical boreholes, which only covers a part of all boreholes connected to the repository areas. Furthermore, it needs to be ensured that the method is adapted for varying types of boreholes. Continued method development and studies are therefore necessary. In the field experiments that have been carried out, new opportunities for optimisation and efficiency improvements have been identified, for example in respect of technology for casting long sections of concrete.

Programme

- Clarification of requirements for sealing of boreholes that affect post-closure safety in connection with the extended Final Repository for Short-lived Radioactive Waste (SFR).
- Evaluation of long-term properties of concrete to be used for borehole sealing.
- Investigation of the possibility of sealing boreholes with concrete as a permanent closure method.
- Further development of the existing method for borehole sealing to also include gently dipping and horizontal boreholes.
- Investigation of which support material fraction size is to be used in fracture zones with a larger aperture.
- Optimisation of method and work process based on field experiments:
  - Further development of copper expander.
  - Manufacture of supplementary equipment for practical implementation using a drilling rig.
  - Streamlining of work process.

10.8 Closure

10.8.1 Closure of the Spent Fuel Repository

Current situation

No further development has been done regarding closure of the Spent Fuel Repository. The requirements for closure that exist today are not complete and need to be updated before further work is carried out.

Programme

During the RD&D period, the set of requirements will be updated to a level that allows them to be used to review the design of the closure. After this, the closure plan will be updated. Planned activities:

- Update the set of requirements for closure of the Spent Fuel Repository.
- Update closure plan for the Spent Fuel Repository.

11 Bedrock

The main function of the bedrock for SKB's existing and planned final repositories is to ensure stable mechanical and chemical conditions over the period of time that the disposed waste must be kept isolated from humans and the environment. The bedrock will also constitute a barrier that as far as possible prevents or retards the transportation of radionuclides from a final repository to the surface system. In order for this to be achieved, sufficient information about bedrock conditions and of the processes that alter the mechanical and chemical conditions in and around the repository is required. A large proportion of the research questions that relate to the rock and the underground openings in the final repositories apply to all three repositories: the Final Repository for Short-lived Radioactive Waste (SFR), the Spent Fuel Repository, and the Final Repository for Long-lived Waste (SFL). The completed and planned research activities described in this chapter are linked to constructability and design of the final repositories and to detailed site investigations during the construction of the repositories.

11.1 Characterisation and modelling of rock properties and behaviour

Characterising and modelling the properties and behaviour of the rock, especially with respect to its long-term evolution, involves quantifying in situ rock stresses, stress changes, deformations and rock failure caused by different processes (for example rock excavation, glaciation, heating, earthquakes) and determining the effect of these factors on rock stability and water flow. Characterisation of processes relevant to post-closure safety and their effects on rock properties is done by means of both numerical modelling and laboratory studies.

11.1.1 Mechanical properties and behaviour of the rock

The rock mass consists of intact rock that is cut by discontinuities such as fractures, deformation zones, veins, defects and grain boundaries. Coupled models of the rock mass assume that its mechanical properties can be described correctly with respect to the inherent discontinuities on different scales. Empirical methods to characterise the mechanical properties of the rock mass are not suitable for analysing long-term behaviour, where coupled thermal, hydraulic, mechanical and chemical processes (THMC) as well as anisotropy and scale dependence play a role.

Current situation

The classic relationship between the size of a sample of intact rock and its Uniaxial Compressive Strength (UCS) presented by Hoek and Brown (1980) has been questioned in recent years (for example Masoumi et al. 2015, Quiñones et al. 2017), partly because it can potentially affect predictions of the properties of the rock mass and especially the understanding of and ability to predict spalling (Section 11.1.2). Laboratory tests on intact rock samples from Forsmark have shown a very limited size effect for samples larger in diameter than the standard 54 mm (Delgado-Martín et al. 2021a). Another test programme is in progress to determine fracture toughness, and results are now available for samples from different depths for the main rock type in Forsmark (Delgado-Martín et al. 2021b, 2022). These data will be used for crack propagation studies (Section 11.1.2).

SRM (Synthetic rock mass) models consist of numerical simulations of the rock mass, in which the discontinuity network is explicitly represented together with the intercalated, “intact” rock part and where the two components can be deformed and fail. During the previous RD&D period, SKB contributed to the development and efficiency enhancement of the numerical modelling tools used for SRM modelling (3DEC (discontinuum modelling), PFC (particle-based modelling) and FLAC3D (continuum-based modelling). The development efforts have, for example, led to shorter calculation times through parallelised modelling and inclusion of PFC in the FLAC3D platform. In addition, a new type of contact logic has been implemented in PFC, which in the future will facilitate modelling of crack propagation caused by different loads.

Despite the fact that SRM modelling is still limited by numerical computing capacity, which sets a limit for model size (scale), degree of detail and complexity, the models have been able to make a crucial contribution to the development of DFN-based, analytical equations that can describe the properties of the rock mass (DFN – Discrete Fracture Network). The latest development in SRM modelling has led to a quantitative connection between a DFN-based rock mass model and its effective (or equivalent) elastic properties (Davy et al. 2018, Darcel et al. 2021a). A detailed description of the newly developed methodology for estimating the elastic properties of the rock mass is presented in Davy et al. (2018), Darcel et al. (2021a) and Hakami et al. (2022). This is the first step in the development of a common DFN-based framework for estimating rock mass properties. The method will be further developed to include estimation of the strength of the rock mass and hydromechanically coupled properties.

Efforts are also planned to gain a better understanding of the effect of the stress tensor on flow patterns over individual fractures and fracture networks. The hydromechanical properties of individual fractures are of crucial importance for the barrier function of the rock, since they determine how the local fracture transmissivities vary with the stress and the stiffness of the rock mass, as well as how strength is affected on different scales. SKB has, in collaboration with Posiva and NWMO, participated in the Post project (Fracture parameterisation for repository design and post-closure analysis), which aimed to enhance understanding of how to scale up the shear properties of a fracture. Based on recommendations from the Post project (Siren et al. 2017), NWMO and SKB are now collaborating in a second project phase, which aims to gain a fundamental understanding of a number of fracture mechanical aspects, improve existing or develop new constitutive models and drive the development and validation of numerical models that simulate fracture behaviour (Jacobsson et al. 2021).

In conjunction with high normal stresses on fractures, the effect of shear-induced dilatation (volume increase) decreases, but it is still uncertain how much this affects transmissivity (SKB TR-19-24, Section 12.1.1). According to the current understanding of the THM aspects in Forsmark and Laxemar (Hökmark et al. 2010) it is expected that high normal stresses dampen transmissivity effects. During the previous RD&D periods, models for single fractures (with aperture variability) that include rock stress effects have been developed (Stigsson 2019, Zou et al. 2018, Zou and Cvetkovic 2020). These studies indicate that it is not necessarily conservative from a safety assessment perspective to disregard such variation in fracture aperture in modelling of flow and radionuclide transport.

The uncertainties that still exist concerning the hydromechanical properties of the fractures thus mainly concern the scale effect (that is, scale dependence of fracture properties) and changes in transmissivity as a function of shear and normal load.

Programme

The development of a methodology for DFN-based characterisation of the properties of the rock mass based on the latest scientific findings continues and will include:

- Establishment of effective elastic properties on different scales based on DFN realisations and combinations of mechanical properties of intact rock and fractures under different stress conditions.
- Calculation of the coupled hydromechanical properties of the rock mass in models that relate hydraulic fracture aperture (fracture transmissivity) to mechanical fracture aperture and in situ stress. Rules for application of properties in DFN models, for example the transmissivity of fractures, will be analysed and developed so that a relevant initial state can be obtained. Studies will also be carried out of how this can be conceptualised in large-scale models (for example DarcyTools and 3DEC).
- Modelling and in situ and/or laboratory tests will be compared to establish fundamental relationships for describing how the effective hydromechanical properties vary with the stress field and the pore water pressure.

Mechanical properties and parameters for single fractures need to be determined or estimated in order to assess the impact of fractures on constructability and on the safety of the repositories after closure:

- The results of the laboratory tests in the Post project, in combination with theoretical and numerical analysis, will be used to improve existing constitutive models or develop new models, with the purpose of being able to predict fracture mechanical behaviour based on data from field investigations.

In order to improve the hydromechanical understanding of fractures, work is planned to clarify the relationship between fracture aperture and transmissivity and how this is affected by changes in the stress field:

- A literature study linked to flow tests in conjunction with shear in crystalline, hard rock and analogous material.
- Flow experiments during cyclic normal loading/unloading to quantify the effect of normal load on fracture transmissivity.
- Flow tests under shear with different initial normal stress magnitudes are planned to determine whether, and if so to what extent, high normal stresses limit the increase in transmissivity.
- Numerical, hydromechanically coupled modelling of flow change as an effect of shear.

11.1.2 Induced rock mass deformation caused by thermal, seismic or glacial load

Deformation of the rock mass caused by thermal, seismic or glacial loading can lead to changes in the fracture network and fracture properties of the rock. This includes, for example, spalling, development of excavation damage zone and fracturing, fracture deformation, fracture growth, fracture intersection and fracture closure.

Current situation

In the area of excavation damage zone and spalling, the following work has been started and partially completed during the previous two RD&D periods:

- In cooperation with RISE, SKB is conducting an industrial PhD programme at Linnaeus University in Kalmar concerning experimental laboratory-based methods for the assessment of spalling potential. A new laboratory test method has been developed (Jacobsson et al. 2018) to reproduce the spalling process in large core samples under conditions similar to those in a deposition hole in hard rock.
- SKB has participated in the development of micromechanical (particle-based), numerical simulation methods in 3D. The latest advances are presented in Potyondy and Mas Ivars (2020) and Potyondy et al. (2020).
- Within an ongoing PhD programme at Imperial College, London, a fracture mechanics based numerical methodology has been developed to model mechanical spalling. Preliminary results are presented in Saceanu et al. (2020a, b).

Since RD&D Programme 2007 (SKB TR-07-12), SKB has investigated very long-term processes for the mechanical evolution of rock (for example Potyondy 2007, Damjanac and Fairhurst 2010). A number of studies of fracture deformation (opening/closing, dilatation under shearing), fracturing, fracture growth, fracture intersection and fracture closure on the short-term and long-term have been conducted during the previous RD&D period:

- The updated DFN methodology (Selroos et al. 2022) makes it possible to generate DFN geometries based on simplified, rock mechanics-based fracture growth rules. The DFN methodology also includes coupled hydromechanical fracture behaviour, as detailed in Hakami et al. (2022). Development is in progress of a method for analysing the consequences of a stress change on the connectivity of fracture networks (due to potential fracture growth) and on the individual fracture transmissivities and how the effective permeability of the rock mass is affected by different scenarios.
- An ongoing study is investigating how fracture aperture and modus (tension-induced or shear-induced) are affected by geology, topography, water pore pressure and the stress field (Moon et al. 2020). The goal is to try to find rock mechanics-based conditions that can explain the observed dispersion of the occurrence of fractures and their opening and modus in Forsmark.
- A programme is in progress to study the effect of hydromechanical coupling on the flow in hydrogeological models in glaciation scenarios.

Programme

A number of activities aimed at improving understanding of and the ability to estimate the rock excavation damage zone in general, and in particular with regard to spalling around deposition holes and tunnels, are planned. The goal is to investigate the importance of these processes for the post-closure safety of the Spent Fuel Repository and possibilities for optimisation of its layout. Efforts are also focused on gaining a better understanding of the impact of the excavation damage zone on transmissivity as a result of stress changes caused by rock excavation and seismic, thermal and glacial loads. Planned activities include:

- Continuation of the industrial PhD project at Linnaeus University concerning characterisation of the spalling process in a laboratory environment.
- Modelling and further development of particle-based numerical codes to increase 3D modelling capacity (especially for spalling and earthquakes).
- Continuation of a recently initiated MSc project at Dalhousie University, Canada, with a focus on continuum-based methods for modelling of the excavation damage zone and spalling.
- Continuation of a PhD project at Imperial College that uses and, if necessary, further develops a numerical code for spalling modelling based on fracture mechanics.
- Modelling with three different numerical methods (particle-based, continuum-based and fracture mechanics-based) to study spalling and evolution of stress-induced excavation damage zone.
- Development of a probabilistic methodology is planned to estimate the probability of spalling and its geometry. The method will consider variability in the properties of the rock mass and the rock stress field in relation to the direction of the deposition tunnel.

For the estimation of fracture deformation (opening/closing, dilatation under shearing), fracturing, fracture growth, fracture intersection and fracture closure in both the short and long term, the following is planned:

- Development of a theoretical and numerical methodology for the study of the effect on the fracture network and the rock mass stiffness, strength and permeability caused by fracture deformation, fracture initiation, fracture propagation and interconnection in different scenarios (rock excavation, glacial cycle with and without permafrost, thermal load, earthquake) and its consequences for the safety of the final repository. Deformation zone reactivation and deformation zone propagation will also be addressed.
- Refinement of the methodology for simplified, rock mechanics-based DFN generation, based on the results of comparisons between real fracture networks (from field data), DFN realisations based on the simplified geomechanical rules and on fracture mechanics-based methods.
- Continuation of the ongoing study of the relationship between geology, topography, water pore pressure, stress field and fracture aperture and “mode” (tension-induced or shear-induced).
- Supplementary literature survey and data collection regarding the long-term strength of the rock. The study will consider stress corrosion cracking (SCC) at fracture ends in all load 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).

For an update of the basis for thermal dimensioning of the final repository that was used in SR-Site, the following is planned:

- Analyses of the thermal, mechanical, thermomechanical and hydromechanical evolution of the rock in Forsmark (Hökmark et al. 2010).

11.1.3 Rock stresses

Knowledge of the current rock stress field in Forsmark and its evolution under different scenarios constitute a key component of the design of the Spent Fuel Repository, as well as of the assessment of post-closure safety. The uncertainties that exist concerning the rock stress model for Forsmark will be reduced during the construction process by the fact that the model can be refined with the aid of new measurements and indirect observations. Data will be collected for this according to the planned detailed site investigation programme. Furthermore, there are plans to develop a standardised methodology to facilitate updating and verification of the rock stress model in conjunction with the collection of new data during construction of the Spent Fuel Repository. This includes development of a standardised methodology for quality control and classification of new and existing stress measurements, and further development of a tensor-based probabilistic methodology for characterisation and quantification of the stress field and its variability.

Current situation

On the basis of the deformation zone model v 2.3 for Forsmark (Stephens and Simeonov 2015), the rock stress model has been updated in 3D (Hakala et al. 2019), with the purpose of improving the methodology for describing spatial variability in the shape and orientation of the stress ellipsoid in high resolution (block element size 30–100 m). The numerical analysis includes a comparison between a simplified straight-line deformation zone model and a more realistic model with undulating deformation zones. Modelling was carried out with the modelling tool 3DEC, which is based on the distinct element method (Itasca 2019).

The methodology for site modelling regarding rock stresses and rock mechanics properties has been updated (Hakami et al. 2022). A programme for supplementary rock stress measurements during the construction phase, based on the Linear Variable Differential Transformer (LVDT) method, Hakala et al. (2013) and convergence measurements has been developed.

Figueiredo et al. (2020) has used a tensor-based probabilistic method developed by Gao and Harrison (2018a, b) on Forsmark data and on 3D model data from Hakala et al. (2019). The new methodology facilitates mathematical quantification of variability, identification of stress domains and stochastic generation of stress tensors from the distribution.

Development of a methodology for regular updating of the rock stress models has been initiated in the form of a PhD project (StressBay), which is being pursued by SKB in cooperation with NWMO at the University of Toronto. The work aims to further develop the above-mentioned tensor-based probabilistic methodology (Javaid and Harrison 2021, Javaid et al. 2021).

Overcoring without stress measurements performed in spring 2021 on drill cores from a depth of 550 metres from a borehole in the ramp area for the planned Spent Fuel Repository shows good quality cores and no signs of “ring disking” (Hakami and Holmberg 2021). On the basis of this result, SKB is evaluating the possibility of performing future stress measurements in the same borehole before the start of construction of the Spent Fuel Repository, which could contribute to a better characterisation of the stress field at repository level.

Programme

The new updated rock stress model for Forsmark in 3D (Hakala et al. 2019) is one of the components of the updated methodology for continuous processing of geological data and rock stress data that is being developed and will serve as a basis for updated calculations of the risk of spalling and stability within different repository volumes. The updated methodology will also provide boundary conditions for modelling of induced fracture displacements (due to seismic, thermal or glacial load) and coupled hydromechanical processes. The model will then be updated on a continuous basis in conjunction with updates of the structural geology model and when new rock stress data (direct and indirect) are available in conjunction with the construction of the Spent Fuel Repository. The following activities are also planned:

- Revision and reassessment of existing data together with new data regarding the occurrence of ring disking, core disking and borehole breakouts.
- Development and application of a plan for quality control of overcoring and LVDT-based stress field measurements before the start of construction of the Spent Fuel Repository.
- Further development of the tensor-based methodology for characterisation and quantification of the stress field and its variability (StressBay project).
- Estimation of the evolution of the rock stress field in different glaciation scenarios. This will then be used as input in mechanical, hydromechanical and thermohydromechanical process analyses (for example Hökmark et al. 2010).

11.2 Modelling of discrete fracture networks

In a crystalline bedrock, the fractures constitute the main connection for groundwater flow and transportation of solutes between ground surface and a repository. The fractures also constitute planes of weakness in terms of the strength of the rock mass. Understanding the fracturing of the rock mass is thus of importance during design and construction and for conducting credible safety assessments for SKB's different final repositories.

Current situation

SKB has developed a methodology for modelling of discrete fracture networks (Selroos et al. 2021). The methodology constitutes an integrated approach for development of a discrete fracture network concept that can serve as a basis for all users, which, in the development, accounts for both rock mechanics and hydrogeological properties. The modelling methodology for mechanical and hydromechanical properties for individual fractures is presented in (Hakami et al. 2022). The methodology for fracture network modelling, described below, reflects much of the current knowledge.

The development of genetic fracture generation concept, i.e. modelling of fracture networks based on the assumption that fractures grow from fracture seeds, is in progress (FracMan 2022, Lavoine et al. 2020, 2021, LeGoc et al. 2019, Libby et al. 2019). The use of the concept in flow and transport modelling has continued. Specifically, effects of open/closed fractures, internal aperture variability and hydromechanical coupling have been investigated. The results indicate that the hydromechanical coupling together with the relation between fracture size and aperture are important components for reproducing flow characteristics as measured in the field. The focus is now on transport characteristics and is in a reporting phase (Darcel et al. 2021b).

Quantification of uncertainties in geometric data, and how these uncertainties can affect the probabilistic distribution of flow and transport properties through a sheared fracture has been reported in Stigsson (2019). Depending on the stress state, the roughness of the fracture surface, and the direction of flow, the study also shows how the heterogeneous aperture distribution and preferential flow paths change when synthetic fractures are subjected to a simplified numerical shear algorithm. A more advanced numerical model for coupling between mechanical and hydrogeological properties is presented in Li B et al. (2020), where flow change over a heterogeneous fracture is studied as a function of normal stress, and in Zou et al. (2021) where flow change over a heterogeneous fracture is studied as a function of the shear of the fracture.

A study has been carried out to investigate the effect of heterogeneity on different scales in a DFN by evaluating hydraulic tests (Zou and Cvetkovic 2020). Specifically, the effect of heterogeneity on the network scale, and of heterogeneity in aperture (or transmissivity) between fractures and within the same fracture, have been investigated using data from flow logging in boreholes for hydraulic parameterisation of DFN models. Traditionally, variability within a fracture in DFN modelling is often neglected. However, the results indicate that variability within fractures may be a source of uncertainty in evaluated relationships if an evaluation method is used where this heterogeneity is neglected.

A study has been initiated to study alternative and simplified geometric fracture network models, specifically channel or pipe network models, as an alternative to DFN models. The aim is to study whether these alternative models can reproduce the flow in the DFN models at a lower calculation cost, and to study whether these alternative models result in other transport properties and measures of quantity for use in later assessments of post-closure safety.

A PhD project aimed at understanding and quantifying clogging of fractures through accumulation of geological material under different hydrogeological and hydrogeochemical conditions has been concluded (Doolaeghe 2021). The thesis describes how the degree of openness, i.e. whether fractures in a network are open or closed, correlates with other fracture quantities such as depth, fracture direction or normal stress. Furthermore, a graph methodology has been used to simulate network effects of fracture openness, and has shown that such a methodology, which is less expensive than a full DFN model, can be a useful tool for making hydrogeological predictions in a fracture network (Doolaeghe et al. 2020). The results of the study have already been incorporated in part into the new DFN methodology (Selroos et al. 2022) and will be used for further refinement of the methodology applications.

An ongoing PhD project within the EU project Enigma has also been finalized (Molron 2021). The purpose of the project was to investigate whether combined flow, solute transport and radar measurements could be used to condition fracture networks on a deposition hole scale by using pilot holes for deposition holes. Two scientific papers detailing the experimental execution have been published (Molron et al. 2020, 2021). The results show that the method is not industrially applicable as it is time-consuming and because the tested fracture was too low-transmissive for the method to work optimally. However, valuable results were obtained since the latter study indicated that hydraulic experiments should avoid excessive injection pressures in order not to induce temporary aperture changes and thereby transmissivity changes in the fracture under investigation.

In a project linked to the new modelling methodology for discrete fracture networks, it has been tested how deformation zones described as swarms of smaller fractures can be simplified numerically (by upscaling), and how the lost variability as a result of this upscaling can be reproduced in transport simulations through a downscaling algorithm. Traditionally, deformation zones have been described as planes or volumes with homogeneous or heterogeneous properties (for example decreasing transmissivity by depth). However, certain deformation zones can be better described as volumes with increased fracture intensity and thereby be modelled using DFN technology. For larger-scale models, however, it may be numerically heavy to explicitly incorporate all these fractures. A practical solution therefore is to scale up these DFN-described zones to ECPM objects (equivalent continuous porous media) and then project the properties onto flat surfaces. In this way, the deformation zones are re-described as 2D objects in 3D space, but with properties from the underlying DFN model. In order to reproduce the variability lost in upscaling, a methodology has been developed and tested to reintroduce variability in transport simulations. The methodology utilises the functionality of the MARFA code (Section 11.4.1) and has been presented in Williams et al. (2021) for the case of a single deformation zone.

The Äspö Task Force GWFTS has launched a new collaboration project, Task 10, where validation and uncertainties in flow and transport properties will be evaluated. So far, the theoretical framework has been written and presented in Lanyon et al. (2021) and experiments to generate data for the modelling groups have been performed. The possibility of using 3D-printed fractures has also been investigated in Stock (2020) and Stock and Frampton (2021). The collaboration project concerns issues such as flow and transport through a single fracture, a small network of a few fractures and a larger network of fractures.

Programme

- Results and lessons learned from relevant fracture network modelling will be collected in a living document, DFN modelling methodology, volume 2. An initial modelling aims to evaluate the methodology against the ongoing update of the fracture network model of the Forsmark area within the framework of the so-called Baseline work.
- Development of software for modelling of discrete fracture networks, for example FracMan, DFN.lab and MoFrac, takes place continuously in different computer code-specific cooperation forums, such as “DFN studio” for FracMan together with for example Posiva in Finland and NWMO in Canada – see Section 11.4.
- The study of alternative channel network models is continuing with reporting. The aim is a scientific paper in a recognised journal.
- The work on upscaling of deformation zones described with DFN models and subsequent downscaling in MARFA is continuing. In the continued work, the effect of several deformation zones in a model surrounded by background fractures is being studied. This test is necessary before the methodology can be used in safety assessment applications.
- Within the framework of the Äspö Task Force GWFTS Task 10, the possibility of validating flow and transport models is being studied. Based on measurable quantities such as geometry and flow, the possibility of predicting transport properties for individual fractures on a decimetre scale up to fracture networks on a decametre scale is being investigated.
- A PhD project within the framework of the foundation Stiftelsen Bergteknisk Forskning (BeFo, Foundation on Rock Mechanics Research) concerning hydromechanical coupling will be initiated. The goal of the project is to investigate how the stress change due to rock excavation affects the opening distribution and permeability of fractures within the excavation damage zone.

- On the basis of the theories in the modelling methodology for DFN and the results from the DFN Baseline, "HypoSite" will be updated to version 2.0. This hypothetical repository can then be used to assess the need for sampling and additional data capture during the design of the different repositories, to investigate whether the use of conditional fracture models leads to reduced uncertainty in the description of the fracture network, to be used as a guide for the development of DFN concepts, and to estimate the effects of different conceptual assumptions on the geometry of the fractures.

11.3 Seismic impact on repository safety

Under the assumptions made in SR-Site, the effect of earthquakes on post-closure safety constitutes a significant contribution to risk for the Spent Fuel Repository (SKB TR-11-01). The risk contribution is very strongly linked to the frequency-magnitude relationship suggested for the short term (Böðvarsson et al. 2006) and the long term (Hora and Jensen 2005, Fenton et al. 2006). The uncertainties in the long-term predictions, in particular, remain especially with regard to earthquake activity in conjunction with a glacial cycle. Seismic monitoring (Section 11.3.1) and investigations of paleoseismicity (Section 11.3.2) contribute to reducing the uncertainties in the earthquake predictions and to gaining a better understanding of the mechanisms that drive glacially induced earthquakes. The studies planned for the RD&D period are aimed at gaining a better understanding of the connection between contemporary seismicity and glacially induced earthquakes. An underestimation of the seismic activity results in an underestimation of the long-term risk, while an overestimation leads to over-dimensioning of the disposal facility.

Earthquakes in the vicinity of a final repository can cause secondary movements along fractures intersecting the deposition area. Using numerical models, the effects of earthquakes on a KBS-3 repository are investigated and evaluated if the induced shear movement exceeds 5 cm or if the shear velocity exceeds 1 m/s. The modelling methodology has been under development for a long time (Fälth and Hökmark 2006, Fälth et al. 2007, 2008) and has, after extensive testing (Fälth et al. 2014) and case studies (Fälth et al. 2016, 2017, Fälth 2018) reached a high degree of maturity (Hökmark et al. 2019). However, the modelling of secondary movements should be further developed to reduce conservatism in future safety assessments. Development of earthquake modelling methodology and other development efforts that indirectly concern earthquake modelling are presented in Section 11.3.3.

In the safety evaluation for SFL (SKB TR-19-06), no analysis of the effects a possible earthquake could have on the repository's barrier system was carried out. The barrier system's resilience to these processes needs to be assessed in future safety assessments for SFL. The results may contribute to optimisation of the repository design.

No new investigations of earthquake effects on SFR are planned during the RD&D period. However, the planned extension of the local seismic network in Forsmark will make it possible that earthquakes in the area of SFR are also registered.

11.3.1 Seismic monitoring

Seismic monitoring, together with paleoseismic studies, constitutes the foundation for prediction of future earthquake activity in Sweden. Continuous long-term monitoring of earthquakes is crucial for being able to capture patterns of frequency and magnitude, which can vary both in time and space. Monitoring of earthquakes in Sweden is carried out by the Swedish National Seismic Network (SNSN). Since 2008, when the last major expansion of SNSN involving installation of instrumentation (southwestern Sweden) was carried out, SNSN has continuously gathered 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 (Lund et al. 2021) has fundamentally improved the possibilities for interpretation of earthquake activity.

Current situation

The Swedish National Seismic Network (SNSN) has, since the start of the automatic system in 2000, registered, localised and calculated focal mechanisms for more than 10 700 earthquakes of magnitudes between approximately −1 and 4.3 (Figure 11-1). Currently, there are 68 permanent stations and a varying number of temporary stations installed.

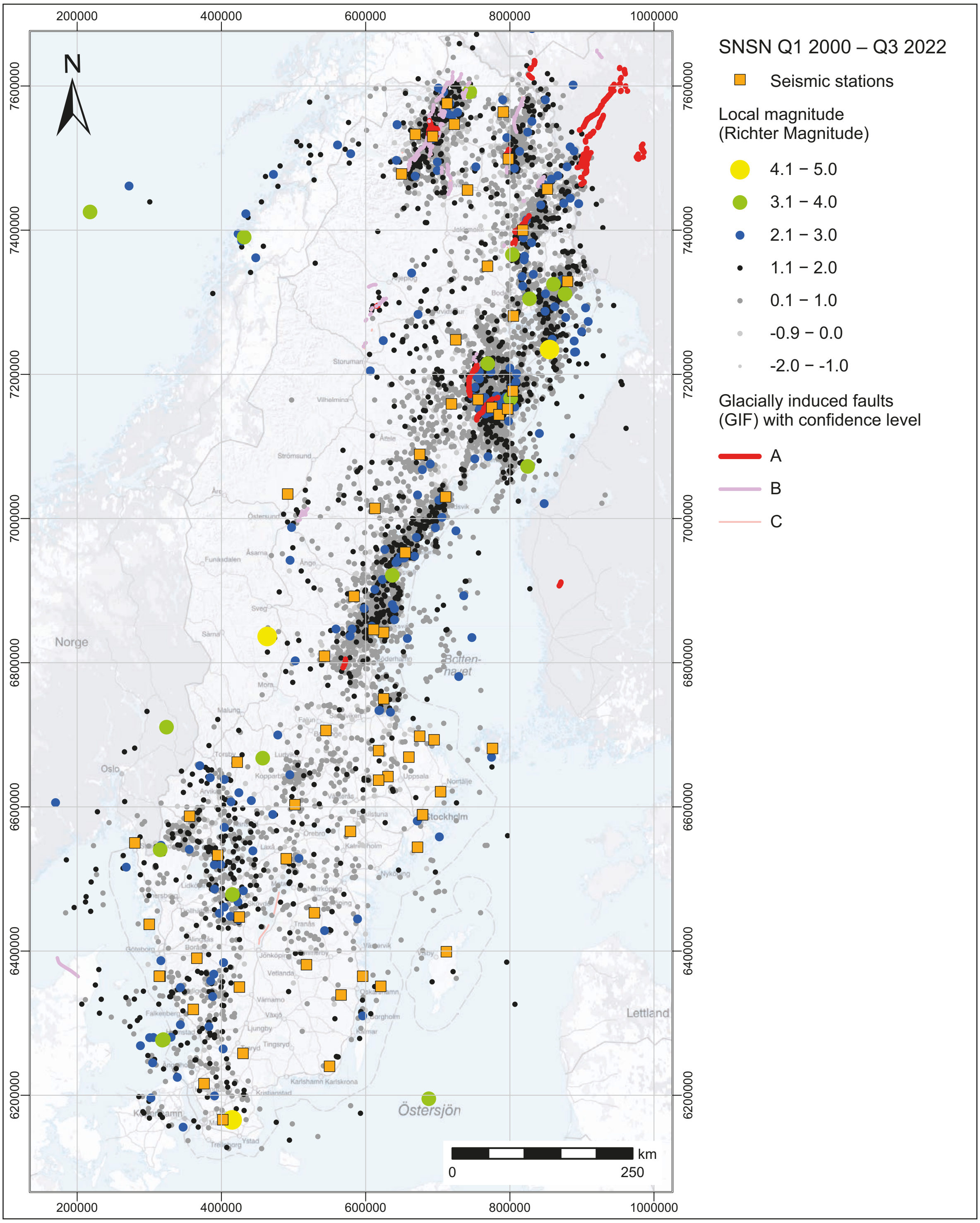

Figure 11-1. Earthquakes registered by the Swedish National Seismic Network (SNSN) during the years 2001–2022. Traces of glacially induced faults (red lines) from the International Database of Glacially Induced Faults (Munier et al. 2020, see also Steffen et al. 2021).

During the previous RD&D period, a number of activities have been in progress that contribute to increasing the precision of seismic data and increase the understanding of mechanisms that drive seismicity at regional and local level.

A seismic tomography study of the Swedish crust has been conducted (e.g. Chan 2014). The resulting three-dimensional velocity model has, together with a multi-event analysis, been used for relocation of the earthquakes, which has considerably improved the precision of the hypocentre locations and calculated magnitudes. Reporting of these activities is in progress.

New calculations of focal plane solutions have been carried out. The results provide information on the stress state in the Earth's crust, and a so-called stress inversion is carried out to determine the stress field at greater depths.

SKB has initiated a project for the construction of a local seismic network in Forsmark to increase the resolution and detection level of the local seismicity. The purpose of the network is, in the initial stage, to measure the undisturbed seismic conditions prior to the extension of SFR and the construction of the Spent Fuel Repository.

Programme

- In addition to a local extension of the fixed network with a number of temporary stations (Section 11.3.2) SKB is not planning any special measures regarding SNSN.
- Data collection with the local seismic network in Forsmark is planned to start before 2024. In subsequent years, data collected prior to and during the construction phase for the extension of SFR and the Spent Fuel Repository will be used to calibrate the network. This data will also serve as the basis for a successive expansion of the network with additional stations.
- Analysis of data recorded with the local seismic network and calculation of focal mechanisms to support models of the stress field and structural geology in Forsmark will be carried out regularly.
- Updating of magnitude-frequency relationships and probability calculations for earthquakes in Forsmark and Oskarshamn with new earthquake data that have been added since the latest calculations (Böðvarsson et al. 2006).

11.3.2 Paleoseismic investigations

In the post-closure safety assessment for the Spent Fuel Repository, SKB assumes that earthquakes may occur in conjunction with future glaciations. The completed and planned investigations and modelling of glacially induced faults increases the understanding of the mechanisms that have caused observed paleoseismicity and of the relationship between glacially induced faults and present-day seismicity. Among other things, it is important to constrain the number and magnitude of earthquakes that can occur within a glacial cycle, since this has implications for the calculation of critical radii (SKB TR-11-01).

Current situation

During the past three years, investigations of the glacially induced Burträsk fault in Sweden's most seismically active area (Figure 11-2b) have continued. Among other things, the existing reflection seismic profile of the fault was re-evaluated and passive seismic methods were tested to get a clearer picture of the fault (Beckel and Juhlin 2019, Beckel et al. 2022) and to improve the location of earthquakes around the fault plane.

One project which has been initiated aims to gain a better understanding of the mechanisms that contributed to and triggered the large, glacially induced earthquakes in northern Scandinavia. High-resolution seismic data from the Burträsk fault area (Lund et al. 2016) is used as the basis for constructing a site-specific background stress field that can be applied in numerical modelling of the fault movement. Through simulation of glacially induced fault movements under varying modelling assumptions (e.g. variations of material properties, stresses, pore pressure and hypocentre position on the fault) and comparisons with observations of fault movements at the ground surface in the Burträsk area, the objective is to investigate which parameters primarily affect the stability and movement of the fault. Using a LiDAR-based elevation model, a profile of the amount of vertical offset along the entire fault scarp has been calculated. Reporting of the Burträsk project is in progress.

Because Öhrling et al. (2018), in their analysis of a high-resolution LiDAR-based elevation model of Uppland, identified two linear land forms that could not definitively be dismissed as glacially induced fault scarps, both sites were investigated during an initial field investigation in the summer of 2019. One of the sites that was judged as being representative of both was chosen for detailed field investigations. In autumn 2019, a strategically located trench was dug across the scarp, and the north wall of the excavation was mapped. The results of the field investigations showed that a seismic event can be ruled out as the cause of the studied landform. Instead, the scarp is interpreted as being the edge of a drumlin reinforced by subglacial meltwater. The results were reported in Öhrling and Smith (2020).

Registrations by SNSN show a very distinct, northeast trending cluster of earthquakes located northwest of Iggesund (Figure 11-2c). Since a clear correlation between present-day seismicity and glacially induced faults has been detected for most of the faults registered so far (Lindblom et al. 2015), this cluster is a strong indication that there may be a so far undetected, glacially induced fault in the Iggesund area. The relative proximity to Forsmark makes it particularly urgent that these findings are investigated in greater detail. SKB has therefore initiated a project aimed at identifying the source of the seismically active band. The project is a collaboration between Uppsala University and SGU, the Geological Survey of Sweden. A dense network of 13 temporary seismic stations was established in 2021, and these will measure for at least three years in order to improve earthquake locations, thus facilitating identification of the structures that are currently seismically active. In 2021, SGU's inventory of geophysical potential field data was also analysed, and a lineament interpretation of existing aeromagnetic data was conducted. A number of structural geological field investigations were also carried out, which among other things resulted in a palaeostress analysis.

SGU's bedrock maps in the Iggesund area, which were created in the period 1987–2008, contained major differences in terms of observation density and scale, which led to tangible quality variations and so-called "map sheet faults". Therefore, SGU's bedrock maps in the Iggesund area have now been harmonised geometrically and semantically, i.e. colours, geometry and designation of different bedrock units and deformation zones have been updated and synchronised between the different map sheets, based on geophysical data, field checks and geological experience. Using a LiDAR-based high-resolution elevation model and its various derivatives, a study has also been conducted to identify linear landforms that could potentially constitute glacially induced faults in the Iggesund area. Interim reporting of this work is in progress.

The results of the Quaternary geological field investigations that were carried out in 2021 have, together with a detailed literature survey, been used to clarify a paleoseismological question concerning erosion surfaces along the Uppland coast, in the Stockholm area and up to the Iggesund area (e.g. McCalpin 2013). This was done to determine whether the erosion surfaces were caused by a palaeotsunami (Mörner et al. 2000) or other erosion processes, such as for example strong, near-coastal bottom flows in the ancient sea (Lagerbäck et al. 2005). The study shows that there is no evidence that the erosion surfaces were created by a paleotsunami.

Programme

- Compilation and publication of results from earthquake analyses and numerical modelling of the Burträsk fault.
- Continued investigations in the Iggesund area to identify the source of observed seismicity. The investigations are planned to include seismic monitoring and analysis, Voxel modelling based on aeromagnetic data from SGU as a basis for the structural geological model, 3D structural geological modelling of the investigation area, and possible excavation across a number of strategically selected linear landforms that cannot be dismissed as glacially induced faults.
- Publication of the study on erosion surfaces in the eastern parts of central Sweden in a scientific journal.
- Scientific monitoring and, if necessary, new investigations concerning the evolution of seismicity and glacially induced faults during a glacial cycle (Ojala et al. 2019, Smith et al. 2021) and concerning very young seismic events, for example in Norway (Olesen et al. 2021).

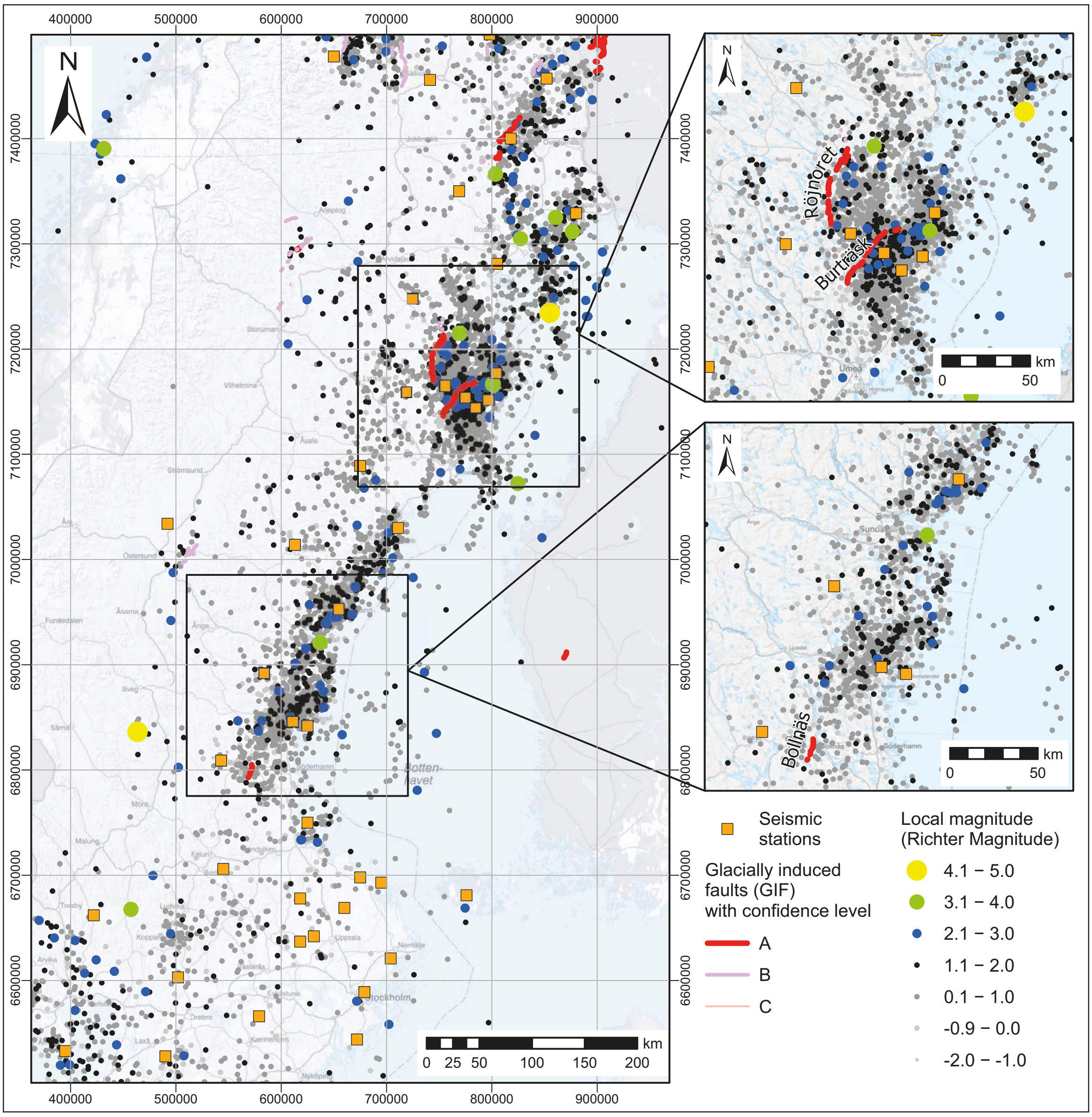

Figure 11-2. *a) Clusters of earthquakes registered by SNSN in b) in connection with the Burträsk fault and c) north–west of Iggesund. Traces of glacially induced faults (red lines) from the International Database of Glacially Induced Faults (Munier et al. 2020, see also Steffen et al. 2021).*

11.3.3 Modelling of seismic impact on the Spent Fuel Repository

Since large uncertainties still exist regarding the frequency/magnitude relationship for both present-day and glacially induced seismic activity, it is difficult to carry out an objection-free probabilistic seismic hazard analysis (PSHA). In the assessment of seismic risk in SR-Site, SKB therefore developed a deterministic method for earthquake modelling that is used to calculate so-called critical radii of target fractures within the repository. These radii change depending on parameters such as distance to an earthquake fault, earthquake magnitude and stress drop (the difference between the stress across a fault before and after an earthquake). Canister positions intersected by target fractures with dimensions exceeding the critical radii should be avoided.

The analysis of the shear load cases in SR-Site was intentionally based on what were judged to be conservative or very conservative conditions and assumptions. With increased process understanding, substantiated by observations, more realistic assumptions can greatly reduce the estimate of induced shear, and thereby of long-term risk, and also provide an opportunity to optimise the repository layout. Previous models have used representative boundary conditions and properties for the repository volume, but not fully explored the possible effect of the natural variability and uncertainty of these conditions and properties on the calculated shear displacements.

Current situation

Development within modelling of seismic impact on the Spent Fuel Repository has been summarised in a report (Hökmark et al. 2019), where also modelling results from recent years are presented together with a list of outstanding questions. During the previous RD&D period, efforts within the field focused mainly on reducing conservatism in the earthquake modelling. In ongoing work that is in the reporting phase, the results of studies regarding the effect of the stress field's variability, in both direction and magnitude, on calculated secondary shear displacements and quantification of these effects to size and place are summarised. Through modelling, target fractures with orientations that maximise shear displacement in different parts of the repository volume have also been investigated. In addition, movements on target fractures close to (100–200 metres) and connected to a fault have been modelled. Furthermore, the stability of steep, local deformation zones in Forsmark has been analysed, including the effect of forebulge on fault stability. Modelling of co-seismic movements of target fractures has also been carried out for earthquakes on steep faults.

Reporting of in-depth studies of the conceptualisation of target fractures (mainly undulation) under static and dynamic loads is in progress.

Work is also in progress on other aspects indirectly related to earthquake modelling. Among other things, the effect of model resolution with respect to the discretisation of the target fractures has been analysed. Similarly, the effect of 2D modelling on fracture stability has been investigated, and the interaction between bentonite, canister and rock has also been included in the calculation of shear displacement. Reporting of this work is in progress.

Programme

- Study of the effect of alternative combinations of Earth- and glaciation models (with respect to the different size, extent, endurance and dynamics of the ice sheet) on the calculated shear displacements of target fractures, in order to verify previous conclusions.
- Application of a refined conceptualisation of fault undulation and fault edges to an earthquake case representative for Forsmark.
- Development and in-depth analysis of repeated earthquakes and their effect on the target fractures, in order to evaluate previous assumptions.
- Modelling of secondary displacements during an earthquake on a gently dipping fault below the planned Spent Fuel Repository, to investigate whether the results presented for Olkiluoto (Fälth et al. 2019) also apply to Forsmark.
- Updating of probability calculations for critical canister positions based on new critical radii, which in turn are based on the results of more realistic earthquake modelling in Hökmark et al. (2019) and in ongoing work.

11.4 Groundwater flow, groundwater chemistry and transport of solutes

11.4.1 Development of computational tools for groundwater flow and transport of solutes

This section describes the research programme for development and maintenance of computational tools for groundwater flow and transport of solutes. Supporting projects that contribute knowledge of hydrogeological questions relevant to the use and development of the computational tools are also dealt with here.

In its review of the RD&D Programme 2019, the Swedish Radiation Safety Authority (SSM) supports SKB's focus on continuously maintaining and developing the computational tools to include new research advances. However, SSM encouraged SKB to assess the benefits of increased complexity in the models in relation to the conceptual uncertainties that exist regarding water flow through fractured rock. Furthermore, SSM pointed out that SKB needs to consider parallel use of simplified models to study uncertainties in both concepts and parameters, and that alternative hydrogeological models, for example alternative geometric fracture network descriptions, could also be used to shed light on remaining uncertainties.

Since explicit DFN models can be used directly for calculation of groundwater flow and transport, certain projects of relevance for this section are also described in Section 11.2.

Current situation

An ongoing project to develop an updated hydrogeological modelling methodology aims to incorporate the development that has taken place in the area and to make the methodology compatible with the modelling methodology that has been developed for DFN modelling (Selroos et al. 2022). The methodology is divided into two parts, one for site modelling and one for safety assessment applications. While the former is at an advanced stage, work on the latter has only just begun. Work to complete the latter has been paused, and will be resumed in good time before the results need to be available.

For modelling of groundwater flow and transport of solutes (but not of radionuclides in safety assessment modelling), SKB's primary tools are still DarcyTools and ConnectFlow. While DarcyTools is a continuum model (Equivalent Continuous Porous Medium, ECPM), ConnectFlow can handle both explicit DFN representation and upscaling to a continuum (ECPM). For both tools, upscaling is done on the basis of an underlying DFN model. Having access to two different computational tools and maintaining modelling competence within both of these provides redundancy and reduces the risks of disturbances in ongoing projects. Furthermore, the two tools have partly different strengths.

Both tools can handle geochemical reactions that enable modelling of groundwater chemistry by means of reactive transport modelling. Further development of ConnectFlow has taken place, so that geochemical reactions can now also be handled in the explicit DFN model (Applegate et al. 2020). The results here show that an ECPM model can be consistent with an underlying DFN model to describe coupled hydrogeochemical modelling, but that this requires a fine numerical resolution that is not always realistic for models on a site scale.

The work on hydraulic acceptance criteria for the Spent Fuel Repository has continued, and a study has been published that tested the concept on the Forsmark model used in SR-Site (Appleyard 2021). The study shows that the concept may be usable, i.e. that a limiting transmissivity value for a pilot hole for a deposition hole can be obtained, and that this limiting value means that deposition holes with properties that may be detrimental to the safety of the repository after closure can be avoided to some extent. Another study based on data from Onkalo in Finland (Hartley et al. 2021) shows, however, that in order for the criterion to be even more stringent, additional parameters may be required, for example distance to the end of the tunnel. Conditioned models (i.e. the actual positions of mapped fractures are honoured) are used in this study, and they show that deposition holes close to tunnel ends have poorer data capture of fractures. (However, conditioned models are not generally a requirement for being able to develop hydraulic acceptance criteria.). Before hydraulic acceptance criteria can be used as a practical tool, further testing and development is required. Work on this has been paused, but will be resumed in good time before the results need to be available.

General development of DarcyTools has continued, with a focus on integrating surface hydrological processes in the tool. This work is ongoing and has not yet been reported. Improvements and testing of how to implement the equivalent flow rate concept (to calculate near-field releases for radionuclide transport) in DarcyTools have been carried out. Public reporting also remains to be done for this project.

Development of the hydrogeochemical computational tools PFLOTRAN and iDP (DarcyTools-PFLOTRAN) has continued. The open source code PFLOTRAN has been further developed for simulation of electromigration processes in fractured media by implementation of the Nernst-Planck equation. This means that geochemical modelling can include multi-component transport of charged species, including electrostatic interactions and anion exclusion. Work is in progress to implement an unstructured grid in iDP in order to be able to refine the resolution in selected parts. The tools have been used for development of a model with microstructural matrix heterogeneity (Trinchero et al. 2020b), for evaluation of an analytical solution for dating of groundwater (Trinchero et al. 2019b), modelling of oxygen penetration and to determine the role of heterogeneity (Trinchero et al. 2018, 2019a) – see further descriptions in Section 11.4.3. Iraola et al. (2019) describes the development that has taken place to model solute transport in a dual porosity medium. Furthermore, iDP is used to model helium transport – see Section 11.4.2.

A PhD project on hydrology and coupled biogeochemical processes has been completed (Jutebring Sterte 2021). The work used hydrological and hydrogeological data from the well investigated and documented catchment Krycklan and showed how hydrological and hydrogeological processes in a catchment affect and control the water chemistry in water discharging from the catchment. Both weathering processes, with subsequent transport of base cations (Jutebring Sterte et al. 2021a), and transport of dissolved organic carbon (DOC) in the catchment have been studied (Jutebring Sterte et al. 2022). Weathering is of particular interest for safety after closure of the Spent Fuel Repository, since these processes can ensure that infiltrating water in the bedrock has a sufficiently high ionic strength not to jeopardise the stability of the bentonite.

The development of the MARFA tool, which is used for modelling of radionuclide transport in the geosphere, is continuing in cooperation with Posiva. The tool is now publicly available due to being an open source code. New functionality has been developed to handle diffusion into stagnant water in the fracture plane, with subsequent matrix diffusion (Trinchero et al. 2020a). Furthermore, development has been initiated to integrate MARFA with DFN models, which is aimed at also being able to handle transient flow conditions. MARFA has also been used to study how microstructural matrix heterogeneity can be up-scaled using effective parameters on a larger scale (Trinchero et al. 2020b). Further discussion of the effect of microstructural heterogeneity and how it is handled in radionuclide transport modelling is covered in Section 11.4.3.

Another study, which focused on evaluation of tracer tests in fractured rock, has estimated retention parameters from breakthrough curves, specifically from the maximum value of the breakthrough curve (Cvetkovic et al. 2020). Being able to utilise the maximum breakthrough renders the information about the breakthrough curve's tail, which was used in previous studies, unnecessary. This makes the practical implementation of this type of tests both easier and cheaper.

Programme

The programme points are aimed at ensuring the research and development need concerning groundwater flow and transport of solutes for SKB's existing and planned repositories.

- Continued development of the computational tool DarcyTools. The focus will be on integration of surface hydrological processes and on development for calculation of quantities used in safety assessment calculations, such as the equivalent flow rate, transport resistance and advective travel time. Up-scaling studies will be planned for evaluation of whether the purely geometric up-scaling used today should be replaced with flow-based up-scaling.
- Continued development of ConnectFlow within the framework of the international collaboration effort iConnect Club, which currently includes SKB and Obayashi (Japan). Studies involving reactive transport modelling in explicit DFN models are continuing, specifically studies of penetration of dilute waters with possible subsequent erosion of bentonite.

- Continued development of iDP with a focus on unstructured grids in order to be able to perform large-scale hydrogeochemical modelling with an increased degree of detail in areas where refined structures are of importance.
- Development and maintenance of MARFA, the computational tool for radionuclide transport, is continuing together with Posiva. The focus during the RD&D period is to integrate MARFA with DFN and groundwater flow codes in a new version called MARFA-DFN.
- A project that is part-financed by BeFo, and which deals with how open boreholes in fractured rock should be implemented in numerical groundwater flow models, is being conducted during the period 2021–2024.
- The Task 10 project is being conducted within SKB's Task Force GWFTS, studying hydromechanical couplings in individual fractures and fracture networks. This work is described in Section 11.2. Task 10 includes a section that deals more generally with the validation of models.
- A project has been initiated to evaluate new modelling tools for integrated modelling of hydrogeology and hydrogeochemistry based on the same site data that was used in SR-Site (part of F-PSAR) and in the PSAR, in preparation for the safety case (FSAR) prior to trial operation of the Spent Fuel Repository. The project will mainly study selected parts of the reference evolution in the post-closure safety assessment and will also serve as a competence transfer activity in the areas of hydrogeology, hydrogeochemistry and safety assessment methodology for the Spent Fuel Repository.

11.4.2 Processes affecting the hydrochemical environment

The hydrogeochemical environment in the Spent Fuel Repository is affected by a number of different processes that may be of importance for the post-closure safety of the repository, for example weathering and microbial activity. Both weathering processes and microbial sulphide formation in the geosphere as well as in the engineered barriers can affect the conditions for corrosion of the copper canister. Questions related to these processes therefore continue to be prioritised.

Current situation

Studies of dissolved organic matter (DOM) in different types of groundwater were conducted during the previous RD&D period. In an experimental study, bioavailable carbon compounds in the groundwater at repository level at Äspö have been characterised and quantified (Osterholz et al. 2022). Conclusions from this study are that the organic matter transported down into the deep continental bedrock is mainly of terrestrial origin. The terrestrial DOM signature appears to be preserved and the material is recalcitrant for degradation, leading to a microbial population that feeds on a small proportion of readily available energy sources, such as geogases and necromass (dead cells), which are found in the oldest saline waters. This supports the hypothesis that the deep biosphere is in metabolic stand-by with extremely long generation times (Osterholz et al. 2022). As a continuation of this study, natural DOM from two of the boreholes on Äspö has been pre-concentrated and used as a substrate for sulphate-reducing bacteria (SRB) to investigate how accessible it is. One conclusion from this study is that DOM at repository level is recalcitrant and will probably not be utilized by microbes for sulphide formation to any great extent during the lifetime of the repository without an external energy source. Reporting of this study is in progress.

The effect of acetate addition on microbial uranium reduction has been investigated. In a study, acetate and a mixture of commercial SRB strains were added to samples of deep groundwater with slightly elevated uranium concentrations from Forsmark, with the purpose of investigating whether the bacteria are able to precipitate uranium when the only available carbon source is acetate (Krall et al. 2020, Suksi et al. 2021). After one month of incubation, uranium concentrations decreased by around 50 percent in the groundwater sample with added acetate and SRB. In the control sample, which only contained groundwater with added acetate and SRB medium (without SRB), uranium concentrations decreased by more than 95 percent. An iron sulphide precipitate was found in both the experimental water and the control water. According to element analysis done with a scanning electron microscope equipped with an energy dispersive X-ray detector (SEM-EDX), the precipitate in the control water contained both microbes and uranium. The experiment shows that acetate precipitates uranium and that the presence of SRB with added acetate reduces precipitation of uranium. Reporting of the results is in progress.

Studies concerning black precipitates observed in the groundwater at several places in the Äspö tunnel have been concluded. The black precipitates contain high concentrations of manganese, and microbes associated with manganese oxidation have been found in the water (Svensson et al. 2021). The studies were mainly aimed at process understanding of a new phenomenon observed in conjunction with the blasting of a new tunnel, which could potentially be of importance for the repositories. Manganese oxides and iron oxides have the potential to adsorb radionuclides and could therefore be positive for post-closure safety in the Spent Fuel Repository. At present, however, the significance is estimated to be negligible.

Studies of the chemical composition and isotope geochemistry of the matrix pore water have been conducted within the site programme for Forsmark. Data from leaching experiments can be used to verify diffusion in the rock matrix, but also to support the palaeohydrogeological evolution of the groundwater. Studies with leaching of neutron-activated rock samples have been conducted at Helsinki University. Results from these indicate the possibility of developing indirect methods that could provide additional information on the salt concentration in the matrix pore water. Supplementary analyses of mineralogy have been carried out and will be reported together with experimental data for anions and for the most important cations in matrix pore water.

Modelling results indicate that the presence of iron(II) counteracts oxygen and sulphide penetration into the rock matrix (Section 11.4.3). To investigate the availability of iron(II), SKB has, together with researchers at Chalmers and Linnaeus University, conducted anaerobic leaching tests with biotite in different water types, including acidic and circum-neutral waters, with and without added potassium, magnesium and other main elements in biotite. Although it was assumed that these additives would inhibit biotite dissolution, iron(II) could be observed in all weathering solutions. Neither microscopic investigation of biotite nor Mössbauer spectroscopy and X-ray diffraction indicate any significant change in the mineral. Reporting of these experiments is in progress. An experiment is in progress to measure stable iron isotopes in groundwater from Forsmark in order to determine the origin of the iron.

All activities aimed at investigating processes that can affect the occurrence and concentrations of sulphide at repository level were brought together in the collaboration project with Posiva, the Integrated Sulfide Project (ISP) (Posiva SKB 2021). Following conclusion of the project, the Sulfide Information Exchange Project (SIEP), which is a four-year information exchange project with Posiva, was initiated in order to gather remaining questions concerning sulphide in the Spent Fuel Repository both in the geosphere (see activities concerning DOM, iron(II) and the effect of added acetate on microbial reduction of uranium above, and activities related to sulphate in buffer and backfill (described in Section 10.3.2)). The project is now at just over the halfway point, with 1.5 years remaining.

In the previous GAP project, which was conducted between 2008 and 2013, groundwater samples were taken from e.g. the borehole DH-GAP04, which goes under the ice and extends down to repository level. However, these samples were of poor quality, as they contained high concentrations of flushing water. In autumn 2022, a new sampling of the groundwater in the borehole in question was carried out.

During the previous RD&D period, the computer programmes PFLOTRAN and DarcyTools were developed to be able to model diffusion of dissolved helium from the rock matrix into groundwater-flowing fractures. The ultimate goal was to be able to support conclusions regarding the dating of deep groundwaters, and preliminary results were published in Trinchero et al. (2019b). Details of other development work on computational tools for groundwater flow and transport of solutes are given in Section 11.4.1.

A dialogue has been initiated with different universities concerning studies of geochemical processes in shallow layers in Forsmark, and a preliminary modelling study concerning calcite has been carried out. Potentially, leaching of cations from soil layers could have a positive impact on the ionic strength of the dilute waters passing through them, before finally reaching repository level and thereby reducing the risk of buffer erosion.

Programme

- A pre-study concerning acetogens and methanogens at repository depth with the aim of assessing the significance of heterotrophic and autotrophic processes for turnover of gases and organic carbon such as acetate.
- Continued studies regarding the effect of adding acetate on microbial sulfate and uranium reduction.
- Studies of matrix pore water, chemical composition and isotope geochemistry are continuing. Supplementary analyses, e.g. of chlorine-36, are needed to ascertain the different sources of salinity that may exist, comparative diffusion calculations and further development of these more indirect methods for analysis of matrix pore water.
- Water samples from the borehole DH-GAP04 in Greenland will be analysed and the results published.
- The results from the development of computational tools for modelling helium flow and thereby supporting the dating of deep groundwaters will be published during the period.
- Model development to describe sulphide flux in the rock, i.e. both sulphate reduction and leaching of iron(II) from minerals and precipitation of iron sulphide, are planned.
- Studies of geochemical processes in shallow layers in Forsmark will continue to create a clearer picture of how weathering processes affect the water chemistry around the Spent Fuel Repository and SFR.

11.4.3 Transport properties and processes affecting solute transport in the bedrock

This section describes the research programme for increased understanding and quantification of processes related to solute transport in the rock.

In its review of the RD&D Programme 2019 regarding sorption, the Swedish Radiation Safety Authority (SSM) pointed out that it is important that sorption databases are kept up-to-date, that some experimental activities are maintained in the long term, and that the development of thermodynamic sorption models is followed. Within the area of diffusion, SSM expressed the view that the planned efforts concerning performance, interpretation and modelling are somewhat unclear and that plans for verifying field experiments should be prepared.

Current situation

Work is in progress to develop an updated methodology for transport modelling for both site-descriptive modelling and safety assessment. The first step has been to compile new development within retardation modelling and to investigate different concepts with transport classes in order to better describe geospatial distribution in retardation models. In connection to this, requirements have been identified for further experimental investigations and method development. The methodology comprises three different parts: interpretation and evaluation of data, extrapolation to safety assessment, and verification/consistency checks, for example with natural analogues. Through large-scale transport modelling with a three-layer matrix (fracture mineral, alteration zone and underlying undisturbed rock), Crawford and Löfgren (2019) studied how the effect of the spatial variability of different layers is reduced during transport along the entire flow path. They showed that fracture-filling mineral coatings have a limited impact on diffusive exchange and retardation, but that the alteration zone is of interest for retardation. This, together with data already available from site investigations, supports an increased focus on the alteration zone in future laboratory and field programme.

Retardation processes, microstructures in the rock matrix and upscaling have been in focus in the international research programme Task Force GWFTS, Task 9 "Increasing the realism in solute transport modelling based on the field experiments REPRO and LTDE-SD", which has now been concluded. The programme included interpretation and realistic modelling of the demonstration experiments WPDE (Poteri et al. 2018a, b) and TDE within Posiva's programme REPRO in Onkalo, and of SKB's previous experiment LTDE-SD in the Äspö HRL. These experiments included tracer experiments from or between boreholes in a tunnel environment, with the goal of demonstrating matrix diffusion and sorption in situ. A basic idea in Task 9 has been to build conceptual understanding of the experiments, the rock matrix and the processes that are manifested in the results, before modelling is started. The programme has resulted in a large number of scientific papers and reports in which heterogeneity has often been in focus (Soler et al. 2021a, b, c, 2022, Crawford et al. 2022, Tachi et al. 2021, Trinchero et al. 2017, Trinchero et al. 2020c, Kekäläinen 2021, Kröhn 2020, Meng et al. 2020, Park and Ji 2018, 2020, Svensson 2020, Svensson et al. 2018, 2019a, b, Iraola et al. 2017) and a PhD thesis (Meng 2020). This has resulted in, among other things, a micro-DFN model of the pore system in the rock matrix and the use of microtomography to better model the matrix microstructure. Task 9 provides access to tools that can handle both chemical and physical variability in the matrix and methods for upscaling the models to scales that are relevant for the safety assessment. In addition, these demonstration tests have been evaluated from a methodological point of view (Löfgren and Nilsson 2019, 2020, Andersson et al. 2020), which is valuable in initial planning of demonstration experiments in the Spent Fuel Repository.

In addition to Task 9, matrix diffusion studies have concerned process understanding, improved modelling on the centimetre to decimetre scale and sampling methodology in site investigations. This has led to a better process understanding of electrostatic interactions between negatively charged mineral surfaces and dissolved ions in the pore water, which has clarified the need to better characterise very small-scale structures in the matrix (down to the nanometre scale). Important characteristics are diffusion-available porosity, aperture distribution and the surface electrostatic properties of the pore walls. Ion transport and electrostatic interactions in rock have been modelled in three parallel projects. Firstly, The Nernst-Planck equation has been implemented in the computational tool PFLOTRAN handling electrostatic interactions with the pore wall via a Donnan approach (Trinchero et al. 2022). This enables more realistic geochemical transport modelling of a multicomponent system during transients, where different ions have different migration rates. Similar model development has also been done with the computational tool COMSOL Multiphysics in order to be able to model so-called electromigration tests, which entails tracer tests under an electric field. Here, electrostatic interactions with the pore wall have been handled via sorption isotherms as well as Poisson-Boltzmann, Grahame and Smoluchowski equations. Electromigration experiments have also been performed by Li X et al. (2020) and modelled by Meng et al. (2020) via a Nernst-Planck approach. These efforts have led to an increased focus on quantification of extrapolation factors from retardation data obtained in the laboratory to repository conditions in future programmes. This applies to in situ load, pore water chemistry, and sample size/granule size. The completed studies also point to the need to verify the extrapolation factors via tracer tests in the repository rock.

In addition to Task 9, sorption studies have focused on in-depth process understanding linked to site-specific properties and extrapolation to repository conditions. SKB and Posiva are funding a PhD project within the EU programme Eurad with a focus on studies of radium sorption with different methods on site-specific material from Forsmark. The purpose of the project is to increase process understanding of radium sorption and link it to site-specific rock properties and mineralogy. Work is in progress on an updated methodology for interpretation of measurement data from crushed material to sorption coefficients, K_d. This methodology includes extrapolation from laboratory to repository conditions and generation of K_d for varying conditions regarding both mineralogy and hydrochemistry. The latter part is based on a dynamic K_d, also called smart-K_d, a concept in which thermodynamic models are used to obtain sets of K_d values for varying conditions, based on experimentally measured distributions and, if necessary, analogues. Furthermore, method development regarding sorption has been initiated with a focus on desorption. A compilation of thermodynamic calculations to determine the magnitude of sorption on corrosion products based on SFL has been prepared. Work is in progress on calculating the formation of corrosion products in order to assess how great the impact of sorption on these may be relative to sorption on cement.

Reactive transport modelling with iDP has been used to study oxygen penetration from glacial meltwater (Trinchero et al. 2019a). The study investigated how a heterogeneous distribution of geochemically reactive minerals in the rock affects oxygen penetration, and what buffer capacity the granitic rock has. This was done by means of a micro-DFN model in DarcyTools and PFLOTRAN. The model includes dissolution of biotite with subsequent reduction of oxygen from released iron(II). The results indicate that oxygen is transported deeper into the matrix than in previous modelling of a single flow path in homogeneous rock. This indicates even more limited oxygen penetration from glacial meltwater.

Model development has taken place for interpretation of diffusion of naturally occurring nuclides from matrix to water-conducting fractures, in order to extract more information and anchor matrix diffusion from natural analogues (Trinchero and Iraola 2020).

Programme

The items below comprise a programme to meet research and development requirements concerning transport of solutes for SKB's existing and planned repositories. The work will be carried out iteratively with method development, predictive modelling, experimental investigations and interpretation and evaluation of results, with the overall purpose of increasing the basis for site-descriptive modelling and assessment of post-closure safety.

- Continued development of methodology for transport modelling with a focus on strategies for modelling and review of computational tools.
- Development of an investigation programme with a focus on the rock volumes around SFR and the Spent Fuel Repository.
- Method development and modification of laboratory and field investigations concerning:
  – Diffusion methods with dose-relevant or geochemically important anions and weakly sorbing cations, under different salinities and loads.
  – Electromigration methods and electrical methods in the laboratory and from boreholes in the repository volume, for quantification of migration data and conductivity of earth currents.
  – Sorption and desorption methods on crushed and monolithic rock and on fracture surfaces, for different dose-relevant tracers at different water compositions.
  – Characterisation methods for near-fracture rock regarding cation exchange capacity, specific surface area, mineralogy, aperture distribution, porosity distribution and electrostatic surface properties.

In order to quantify correlations between retention parameters and better understand coupled processes, most methods will be applied to a small number of samples. The goal is to handle the spatial variability of difficult-to-measure parameters through correlations to parameters that are relatively easy to measure or observe at a large number of sites in the repository rock.

- Development of tools for small-scale modelling of diffusion that can more realistically handle ion migration and also represent a heterogeneous rock matrix.
- Further development of thermodynamic sorption models with smart-K_d concept, the purpose of which is to increase process understanding, supplement data for sorption databases and develop models for generation of K_d for varying conditions (water chemistry, pH, redox and mineralogy). The work will also include experimental investigations.
- Continued studies of sorption and matrix diffusion in natural systems for consistency control and as a complement to the planned verifying demonstration experiments.
- Commencement of a concept study to investigate possible alternatives to future demonstration experiments of diffusion and sorption by means of tracer tests in the Spent Fuel Repository's repository rock.

11.4.4 Climate impact on processes in the geosphere

Current situation

A final sub-study within the hydrogeological modelling that was carried out as a part of the GAP project has been published (Jaquet et al. 2019). In this sub-study, transient simulations over a glacial cycle have been carried out for the GAP site in Greenland. The results show that great care must be taken when subglacial boundary conditions are selected for the different phases during a glacial cycle; specifically, the choice of a pressure boundary condition corresponding to the ice thickness may overestimate the infiltration under certain conditions.

SKB is financing a PhD project in the area of glacial and periglacial hydrology. It is being carried out at Copenhagen University and is part of the CatchNet network initiated by SKB together with other nuclear waste organisations and academic partners. The purpose of the PhD project is to use data from the previous GAP and GRASP projects (the latter stands for Greenland Analogue Surface Project) to increase knowledge and gain a better understanding of the connection between the glacial and periglacial systems, specifically where and how discharge takes place in so-called taliks, i.e. unfrozen parts of the otherwise continuous permafrost. Two scientific papers have been published, the first showing how permafrost aggradation affects the discharge of groundwater on the surface (Hornum et al. 2020). The second article shows how glacially induced fractures in the sediments closest to the surface affect discharge in a so-called pingo (Hornum et al. 2021). A pingo is a phenomenon where groundwater discharge creates an elevation in the permafrost. Monitoring of the GAP and GRASP sites also continued during the previous RD&D period.

Programme

- The PhD project in glacial and periglacial hydrology is continuing and will be concluded during 2023. Two or three more articles are planned. One of these studies is based on the first study (Hornum et al. 2020) but incorporates coupled hydro-thermal processes. Another study will focus on the use of geochemical data to study discharge of deep groundwater, and the last planned work is a modelling study, which will focus on how the deep and shallow groundwater interact at the investigated site in Greenland.

12 Surface ecosystems

SKB's research programme for surface ecosystems is intended primarily to create a basis for calculations of potential radioactive dose to humans and the environment in the assessment of post-closure safety of the repositories. Data and knowledge obtained within the programme will also provide a basis for environmental assessments, environmental monitoring and the assessment of safety in the facilities in operation. The current research questions for the three repositories, the Final Repository for Short-lived Radioactive Waste (SFR), the Spent Fuel Repository and the Final Repository for Long-lived Waste (SFL), largely overlap in the area of surface ecosystems.

The methodology and the models and data used by SKB to represent the biosphere in assessments of repository safety after closure have been developed over several decades. In its review of the safety evaluation of SFL, SE-SFL (SKB TR-19-01), the Swedish Radiation Safety Authority (SSM) expressed a generally positive opinion of the work with the biosphere. It emphasised that the work is comprehensive, and that it appreciated the fact that the report provides an opportunity for SSM to see the progress SKB has made in its work on developing methodology and models (SSM 2021c). There are, however, a number of issues that require further research, either as a result of the regulatory authorities' comments on the review of submitted applications, or because SKB has deemed it necessary in order to reduce uncertainties in future safety assessments. The most important remaining issues in respect of surface ecosystems concern four different areas:

- Uptake pathways and uptake mechanisms for different organisms.
- Temporal and spatial heterogeneity in the landscape.
- Transport and accumulation processes.
- Radiological, biological and chemical properties of important elements in the repositories.

An early overview of SKB's work in this field can be found in the special issue of Ambio published in 2013 (Kautsky et al. 2013). In its work on SR-PSU, the most recent assessment of post-closure safety for SFR, SKB has published reports that describe 1) assumptions in the modelling of the surface ecosystems (SKB R-14-02), 2) data and models used for the dose calculations (Grolander 2013, Saetre et al. 2013, Tröjbom et al. 2013) and 3) the application of the models used in the safety assessment (SKB TR-14-06). In the SE-SFL safety evaluation, several efforts have been made to reduce uncertainties in the analysis and to address weaknesses that were pointed out in the reviews of the safety assessments SR-Site and SR-PSU. These efforts are presented below and in the background reports to SE-SFL (Grolander and Jaeschke 2019, SKB TR-19-05). In the review of SE-SFL, additional opinions were presented, which have been partially incorporated in the coming assessment of post-closure safety included in the PSAR prior to the extension of SFR. Other activities that SKB plans to carry out as a result of these review comments are described below.

Review comments relating to the RD&D Programme 2019 from the Swedish Radiation Safety Authority (SSM), the Swedish National Council for Nuclear Waste, the Royal Swedish Academy of Sciences and Stockholm University confirm that the completed and planned efforts are appropriate, relevant and of good scientific quality. SKB continues to strive to maintain high quality. An important part of fulfilling this goal is collaboration with different research groups, both nationally and internationally. On the national level, SKB collaborates, for example, with groups working in Krycklan and Skogaryd, sites that are part of the Swedish Research Council's national network Sites (http://www.fieldsites.se). Internationally, SKB participates in several different networks, such as IAEA Mereia and BIOPROTA (http://www.bioprota.org/).

This chapter contains a brief description of the issues related to surface ecosystems that are deemed to be most important during the RD&D period, as well as a brief description of SKB's plans for approaching these issues.

12.1 Uptake paths and uptake mechanisms for radionuclides in various organisms

For humans, an important exposure pathway for radionuclides from a final repository consists of intake via food and drink, and exposure via food dominates the dose from many radionuclides. The uptake of radionuclides in organisms is usually crucial for how great the dose contribution is to both humans and biota.

Current situation

Questions concerning uptake pathways and uptake mechanisms for radionuclides include both individual organisms and the ecosystems that the organisms are part of. Dose calculations traditionally use concentration ratios (CR), which describe the concentration of a substance in the organism compared with the concentration in food or in surrounding media (water, soil or sediment). SKB has previously carried out measurements for a large number of elements in relevant ecosystems, but there is a need to expand the measurements to include paired plant and soil samples from certain ecosystems.

Empirically determined CR values are associated with large uncertainties, and SKB has therefore been working for a long time on developing alternative methods for estimating radionuclide uptake in organisms (for example, Kumblad and Kautsky 2004, Konovalenko 2012). For example, in previous safety assessments, SKB has based the carbon uptake in plants and animals on the specific activity of inorganic carbon, i.e. activity relative to the stable substance. In SR-PSU, the uptake of chlorine in plants was assumed to be limited by the plant's nutritional needs, and in SE-SFL, SKB used a model for controlled uptake for two other plant nutrients (potassium and calcium) as well.

For most large animals, the uptake of radionuclides is mainly linked to food ingestion and SKB's work has therefore focused on describing food chains. Initially, uptake into the food chain occurs via plants, and is thus linked to the system's primary production. SKB's previous work concerning uptake mechanisms for aquatic and terrestrial systems has been described in the books on ecosystems (Andersson 2010, Aquilonius 2010, Löfgren 2010), where the descriptions have largely been based on data from SKB's site investigations.

In 2020, SKB initiated cultivation experiments of different agricultural products on different soil types at Byle Farm in Uppland. Byle Farm is a natural analogue to a future Forsmark in about 3 000 years, where a lake has been drained and cultivation takes place on the partially lime-rich soils. In these experiments, the natural uptake of elements in different plants is compared with the content of the soil. It is also possible to link uptake of different substances to evapotranspiration and primary production. The purpose of the study is to supplement the database with site-specific data for agricultural land, but also to gain a better understanding of processes that affect cycling and uptake of radionuclides in a future Forsmark.

SKB is participating in a project at SLU in Umeå called the Kronosekvensprojektet, which involves sampling of soil and plant concentrations of different substances in about 40 marshlands of different ages in Norrland. The project has given rise to several published works that analyse the behaviour of heavy metals depending on the age and biogeochemical properties of marshlands (Wang et al. 2020, 2021). Furthermore, SKB is collaborating in a PhD project in which the element composition in plants and soil in marshlands of different ages and with different properties is being studied. The results are expected to provide a better understanding of how conditions for plant uptake are affected by the environment and change over time in a future Forsmark.

Data from the cultivation experiment and the Kronosekvensprojektet project provide a basis for development of process-oriented uptake models for elements that are either actively absorbed (e.g. nutrient-salt-like substances) or passively (e.g. with water). A distinction between active and passive uptake can facilitate calculations of uptake under changed climate and/or soil conditions that affect the water uptake of plants.

Running water is in many ways similar to lake ecosystems, but differences that exist, e.g. in hydrology, conditions for chemical precipitation of different substances (Section 12.4) and biological uptake mechanisms, may result in different conditions for accumulation of radionuclides. Previous research indicates that certain radionuclides can accumulate near running water (Lidman et al. 2017, Ledesma et al. 2018). A PhD project funded by SKB has simulated transport of different substances depending on hydrology in environments near streams (Jutebring Sterte 2021) – see Section 12.3. Recently, it has also been noted that ditches have different properties than natural watercourses (Peacock et al. 2021) with significantly higher atmospheric releases of carbon dioxide and methane. The processes that control the cycling and transport of radionuclides in running water are directly or indirectly affected by the metabolism of plants or microorganisms, but also by chemical and abiotic processes – see Sections 12.3 and 12.4. The importance of these processes for radionuclide cycling in surface ecosystems in certain future scenarios should be studied more thoroughly.

During the work on the post-closure safety assessment of SFR, factors such as sequestration of organic carbon, gas transport, and uptake of carbon dioxide via roots have been deemed important for describing the flux of carbon-14. The biosphere model was updated in accordance with this in SR-PSU (Saetre et al. 2013). Following comments from the Swedish Radiation Safety Authority (SSM)'s external reviewers (Walke et al. 2017) and comments on SE-SFL (SSM 2021a, c), SKB has, as part of the PSAR prior to the extension of SFR, continued work on developing the integrated modelling of stable and radioactive carbon. SKB is actively participating in the international working group for carbon-14 within BIOPROTA. The post-closure safety assessment includes the cautious assumption that all carbon-14 is available for fixation via photosynthesis (in the form of carbon dioxide or hydrogen carbonate). That this is a reasonable assumption in unsaturated soil layers has been demonstrated experimentally (Hoch et al. 2014), but in oxygen-deficient environments carbon-14 may also reach the biosphere in the form of methane. This means that a smaller fraction of a carbon-14 release is available for uptake in plants and the rest of the food chain, which would give lower calculated doses if it was considered.

SKB has initiated studies to describe the flux of methane in natural ecosystems (Natchimuthu et al. 2015), and through a collaboration with Linköping University, SKB has access to a large and active scientific network. Innovative method development has resulted in new measurement methods (Bastviken et al. 2020, Gålfalk et al. 2021, 2022). This has contributed to valuable knowledge of methane and carbon dioxide flows and their dependence on temperature and the type of biotope involved (Sieczko et al. 2020, Gudasz et al. 2021, Kuhn et al. 2021). Methane reaching the surface ecosystems from the rock can be oxidised by microorganisms and then be converted to biomass or carbon dioxide. The part that is not oxidised may be emitted to the atmosphere in different ways, such as via bubble flow, diffusion through water, or through transport via air channels in aquatic plants.

Carbon-14 from radioactive waste can either reach the atmosphere as methane or carbon dioxide, or be taken up in the food chain via methane-oxidising microorganisms or via plant uptake of carbon dioxide (which in turn may have been formed through methane oxidation). In both running water and ice-covered lakes, methane and carbon dioxide releases exhibit great dynamics and variation (MacIntyre et al. 2021, Rudberg et al. 2021, Sawakuchi et al. 2021, Schenk et al. 2021). The level of understanding of the flux of methane and carbon dioxide in lakes will be compiled in an ongoing project. A literature compilation with a focus on methane releases from geological final repositories shows that methane that reaches lakes and wetlands under certain conditions can be completely oxidised before it reaches the atmosphere (Ikonen 2022). Preliminary results from methane measurements in Forsmark show high methane release from one wetland, however, indicating that both methanogenesis and methane release may be important in environments where radionuclide release from a repository could occur. The uncertainties regarding the flux of methane mean that it remains reasonable to use the cautious assumption that all carbon-14 in methane will be bioavailable, and thus may lead to exposure via food.

The transport and uptake of gas may also be significant for other elements that can evaporate, such as selenium, iodine and chlorine (Hardacre och Heal 2013, Svensson 2019, Thiry et al. 2022). In natural systems, chlorine often occurs as the relatively mobile chloride ion, but chlorine can also be absorbed by plants or be converted into organic chlorine compounds. The conversion and uptake of chlorine is controlled by biological processes in the soil and in plants. The distribution of chlorine between different pools in terrestrial systems has been studied mainly in well-drained environments, where the residence time is surprisingly long (Bastviken et al. 2013, Svensson et al. 2021a).

Investigations in Forsmark have shown that there are also large amounts of chlorine in the ground in damp and wet environments, and that the quantity of inorganic chlorine in wood and ground vegetation can also be significant. The results indicate that the transport of chlorine through these environments is also slowed down via uptake and accumulation, and that the natural load of chlorine may be higher in discharge areas than in drier areas. Furthermore, the amount of chlorine in vegetation does not appear to be related to the concentration in the ground, but rather to the species composition (Svensson et al. 2021b).

The methodology for analysis of doses to non-human organisms that was originally used in SR-Site has been updated (Jaeschke et al. 2013). The updated methodology was used in the analysis for SR-PSU, where it was integrated with the transport and accumulation calculations (SKB TR-14-06). The potential dose to non-human organisms in the assessments of post-closure safety (SR-Site and SR-PSU) performed by SKB has been found to be considerably lower than the screening levels proposed by ICRP and IAEA. This is in spite of the fact that SKB's calculation methodology is more conservative than the methods proposed by the IAEA (IAEA 2018). SKB intends to continue to follow international developments in the field, and to continuously update the methodology when warranted.

SKB has actively participated in the IAEA project MODARIA II and in BIOPROTA in a revision of the IAEA's Biomass methodology (Lindborg 2018, Brown et al. 2022, Griffault et al. 2022, Lindborg et al. 2022) and is continuing to participate in BIOPROTA and IAEA Mereia.

Programme

- Development of alternative or mechanistic uptake models to describe plant uptake.
- Continued work on developing the models for transport and accumulation of chlorine-36.
- Continued investigations to gain a better understanding of the importance of methane conversion for carbon reaching surface ecosystems (streams, lakes and wetlands) via deep groundwater in Forsmark.
- Cultivation of crops in different agricultural soils for sampling and analysis of uptake of different substances (Byle Farm).
- Coordinated sampling of plant and soil samples in Forsmark with a focus on wetlands with discharge of deep groundwater.
- Analysis of data from the Kronosekvensprojektet project to improve understanding of uptake processes.
- Continued development of models and knowledge base for uptake of different substances in organisms that occur in running water.
- Continue to monitor the development of the protection of other organisms in international forums such as IAEA and ICRP.
- Continued active participation in IAEA Mereia and in BIOPROTA's work (including work on carbon-14).

12.2 Temporal and spatial heterogeneity in the landscape

The appearance and future evolution of the landscape are important factors for the calculated dose in a release of radionuclides. Depending on the type of ecosystem in which a release occurs and the properties of the ecosystem, the calculated dose may be affected by several orders of magnitude.

The assessment of post-closure safety identifies potential release areas, biosphere objects, within which doses to humans and biota are calculated. Biosphere objects are delimited with the aid of the geometry of the landscape. The properties of the landscape and biosphere objects affect hydrology, transport and accumulation in different soil layers (regolith), and the conditions for radionuclide uptake and exposure of humans and the environment. The description of the landscape and biosphere objects thus serves as a basis for several different types of analyses within the field of surface ecosystems.

Current situation

As a basis for post-closure safety assessments in Forsmark and Laxemar-Simpevarp, SKB has formulated a description of the events and processes that have determined the evolution of the landscape to date (Söderbäck 2008). The historical description, combined with a description of the present-day landscape and an understanding of its function (SKB TR-08-05, TR-09-01), has been used to describe probable landscape evolution according to different assumptions in respect of future climate and shoreline displacement. In addition to the large-scale and slow changes caused by climate variation and shoreline displacement (Chapter 13) the landscape is also affected by the addition and relocation of sediments, which among other things means that lakes become shallower and infilled (Brydsten and Strömgren 2010). As different ecosystems succeed one another, the chemical and physical properties of land and water change, along with the species composition of plants and animals.

Discharge areas for deep groundwater usually occur in the topographically low points of the landscape, for example within or next to lakes, streams and wetlands – see Section 12.3. The conditions for transport and accumulation of radionuclides in these areas are determined partly by the topography and properties in the local and regional drainage basin, and partly by the properties of the discharge area, e.g. the extent and thickness of different soil layers. The digital height model and the regolith depth model have been updated in the site modelling (Sohlenius et al. 2013b, Petrone et al. 2020, Petrone and Strömgren 2020). These updated geometric models need to be incorporated into the landscape models.

There is a natural co-variation between the sizes and properties of the discharge areas that may be reached by releases, which partly depends on where in the landscape the objects are located (Berglund et al. 2013). SE-SFL described how the doses from radionuclides with different properties are affected by the type of ecosystem in which the releases occur, but also by the properties of the discharge areas in terms of size and layer thicknesses (SKB TR-19-05, TR-19-06).

Variations in local topography, the sequence of soil layers and ecosystem succession can also lead to heterogeneity within a discharge area, as was pointed out by SSM in reviews of SR-Site and the RD&D Programme 2013. In order to investigate how small-scale variation in topography and soil layer thickness affect transport and accumulation of radionuclides reaching a discharge area, the COMSOL tool has been used (von Schenck et al. 2015, Silva et al. 2015, Abarca et al. 2016). The studies include the main discharge area for SFR, a landscape profile in the Krycklan area in Västerbotten (Abarca et al. 2016) and a discharge area in Laxemar (Sáinz-García et al. 2022). The work is planned to continue in the Forsmark area for the purpose of studying, for example, how flow paths, layer strengths and redox zones affect the accumulation of different elements – see Section 12.3.

The continued work in the Krycklan area (Lidman 2013) shows that the landscape's large-scale mosaic of forest and wetland has a great impact on the local hydrology (Section 11.4, Jutebring Sterte 2021) and on the processes that control leaching of the regolith (Section 12.3, Lidman et al. 2019), and thereby also on mass transport of different elements (Laudon et al. 2021).

The landscape evolution model used in SR-Site (Brydsten and Strömgren 2010) and in SR-PSU (Brydsten and Strömgren 2013) should be updated to include new knowledge and current data in the description of future landscape evolution. SKB has started a project to implement the Untamo model for the Forsmark area (Gunia et al. 2021, Gunia and Gunia 2022). The Untamo model is to some extent based on SKB's previous landscape modelling, and the model has been used to describe the landscape evolution in the assessment of post-closure safety for the final repository for spent nuclear fuel in Olkiluoto (Posiva 2013). The first step in the implementation of the model consists of a review and description of Untamo's details compared with the previously used landscape evolution model, in order to identify differences in methodology and results between the two models (Gunia et al. 2021).

New site information from several investigations of wetlands (Sohlenius et al. 2019, 2020, Sohlenius and Svensson 2021) and of discharge areas is available. During the previous RD&D period, SKB worked on developing landscape modelling. This work will continue, which will contribute to improved estimates of uncertainties in the properties of the biosphere objects, and also serve as a basis for detailed modelling of hydrology – see Section 12.3.

In order to understand the effect of the landscape on groundwater flows and transport of substances in a cold-climate domain, an area in Greenland in front of the ice sheet margin has been studied within the GRASP project. The extensive material is still being processed. Some papers have been published (for example Lindborg et al. 2020) and work on further publications is in progress. The new knowledge from GRASP has not yet been fully utilised in the landscape modelling for Forsmark. In its review of SE-SFL (SSM 2021a, c), the Swedish Radiation Safety Authority (SSM) said that future safety assessments can be strengthened by studying future land use and water supply on the site, for example in connection with a warmer climate. New knowledge of the impact of the greenhouse effect on agriculture (e.g. Mattsson et al. 2018) will need to be put into context for a future Forsmark. SKB has also started a project to gather data from a future analogue site, Byle Farm, which is estimated to be able to represent Forsmark in about 3 000 years – see Section 12.1.

Most of SKB's assessments of post-closure safety show that it is the cultivated marshland that gives the highest dose for many radionuclides. Such an object in Forsmark is normally formed by a bay being cut off from the sea and becoming re-vegetated and filled with sediment. If the object has sufficiently thick sediments, the marshland can then be drained for cultivation. Some objects instead see a growth of sloping wetlands, which form so-called hanging marshlands. In an ongoing evaluation, SKB is investigating how topographically based methods, for example the Topographic Wetness Index (TWI), Depth To Watertable Index (DTW) (White et al. 2012) or combinations of similar GIS-based methods (O'Neil et al. 2019), can be used to predict the evolution and occurrence of wetlands.

Programme

- Supplementary site investigations regarding the stratigraphy of the regolith and physical and chemical properties in potential discharge areas.
- Investigations to better describe the infilling process of bays, lakes and marshlands in the Forsmark area.
- Continued development of the landscape evolution model by incorporating new data from site investigations, including elevation and regolith depth models and, by means of continued simulations with Untamo, updating of the description of landscape evolution in Forsmark prior to forthcoming assessments of post-closure safety.
- Evaluation and compilation of data from Byle Farm to illustrate the extent to which the site is a suitable analogue for Forsmark's future cultivation areas.
- Apply data and knowledge from GRASP and Krycklan at the landscape level in the Forsmark area to describe the effects of a future colder climate.
- Compile the state of knowledge of land use and water supply under other climatic conditions.
- Complete the ongoing work on evaluating methods for describing the occurrence of future wetlands in the Forsmark area.
- Continued work with the COMSOL tool to answer and illustrate several questions concerning the description of the landscape, with a focus on the spatial dissolution and demarcation of the biosphere objects.

12.3 Transport and accumulation processes

Transport processes here refer mainly to abiotic transport taking place with water, particles and, to some extent, gas. Accumulation processes refer to processes that describe accumulation of elements in loose deposits (for example via sorption, precipitation or sequestration in the ground's organic material), but not uptake of radionuclides in organisms – see Section 12.1.

A good understanding of radionuclide transport and accumulation in the regolith, and an appropriate model for these processes, is crucial for being able to estimate exposure and dose to humans and other organisms. Hydrology affects both the areas that may be affected by a possible release of radionuclides and the quantities and concentrations of nuclides that reach these areas. Accumulation processes determines how much of an element can accumulate in the soil layers or be associated with particles. The uncertainties in parameter values that describe sorption (K_d) are often large, which affects the dose calculations.

Current situation

The reporting to SE-SFL describes how the current understanding of transport and accumulation processes in the biosphere has been used in the safety evaluation (SKB TR-19-05). The combined transport and exposure model that is currently being used and which has been modified since SR-Site was given the name Biotex in SE-SFL (SKB TR-19-05).

In its review of SR-Site, the Swedish Radiation Safety Authority (SSM) pointed out the importance of how the division of soil layers could affect the modelled transport of radionuclides, and this was then investigated in the subsequent safety evaluation SE-SFL. The analysis showed that discretisation affected accumulation and time, but the effect was strongly dependent on the properties of the biosphere object and radionuclides (SKB TR-19-05). In the review of SE-SFL, SSM and its consultants took a positive view of the further development of Biotex with higher discretisation (SSM 2021a, c), and the finer discretisation has also been used in the recently completed safety assessment for an extension of SFR.

SSM and its reviewers have requested simpler radionuclide transport models as a complement to those used in the assessments of post-closure safety (SSM 2019). In SE-SFL, development of an equilibrium model based on the existing radionuclide transport model was initiated in order to enable quick investigation of how different parameters and assumptions affect the modelling results. A simplified hydrological model was also implemented in the SE-SFL safety evaluation, describing the vertical flows as a function of the drainage (SKB TR-19-05).

The decay products of the uranium chain exhibit transport properties that differ from one other, and in SE-SFL a modelling study was carried out, demonstrating the importance of half-life, sorption and transport rate in the different soil layers (SKB TR-19-05). Work involving decay chains has also been carried out to study and model transport in discharge areas in Forsmark and at sites corresponding to a future Forsmark (e.g. Lidman 2009).

In addition to providing an opportunity to refine temporal and spatial resolution (Section 12.2), the COMSOL tool also constitutes a complement to the modelling carried out with the hydrological tool MIKE SHE. COMSOL enables detailed study of mechanisms driven by physical, chemical and biological processes, while the landscape can be used as a driving factor for water flows. In SE-SFL, radionuclide transport in the object that gave the highest doses was studied in detail by means of calculation of water flows and sorption in different regolith layers in a three-dimensional COMSOL model (SKB TR-19-05, Sáinz-García et al. 2022). Modelling showed that for several layers, flows and concentrations were concentrated into a few areas, while other layers spread radionuclides effectively across a large volume. The detailed COMSOL model was also used for comparisons with a simplified model in Ecolego (SKB TR-19-05). The results show that a simplified model is suitable for calculating average accumulated activity and concentration on a scale of a discharge area with an acceptable accuracy.

In the so-called S-profile in Krycklan (Lidman et al. 2017) further investigations have been conducted. A PhD student funded by SKB recently defended a thesis with a focus on hydrology and solute transport near the surface (Jutebring Sterte 2021). The results show that the variation in discharge and interactions between surface water and groundwater can be linked to the properties of the catchment basin, such as soil properties and the extent of ground frost (Jutebring Sterte et al. 2018). Ground frost was also important for small-scale fluid dynamics and was able to explain variation in stream water quality (Jutebring Sterte et al. 2021a). It was also possible to link water quality, weathering rate and sources of organic matter in stream water to the transport times and transport routes of the drainage (Jutebring Sterte et al. 2021b). The work is of importance for modelling of future transport, since it shows that it may be possible to link drainage flows and interactions between surface water and groundwater directly to areal properties.

In the transport of radionuclides from terrestrial to aquatic ecosystems, the zone close to the watercourse is important. This can lead to the concentration of certain elements, for example thorium, being a hundred times higher in the soil water in the near-stream zone than in normal forest soils. However, there are no clear differences in respect of uptake in vegetation, which is probably connected to the fact that these elements are to a high degree bound to dissolved organic carbon and therefore not available for uptake in plants (Lidman et al. 2017). In order to understand the long-term mass balance in these environments, the lanthanide series, especially europium anomalies, are of particular interest.

A compilation of data from Krycklan (Lidman et al. 2019) showed that europium has the potential to act as a tracer for fixation processes and as a tool for quantifying long-term weathering and accumulation processes in soils.

SSM has pointed out that the hydrological simulations in SR-Site and SR-PSU did not take sufficient account of uncertainties. In SE-SFL, SKB has shown the importance of uncertainties in surface hydrological parameters for transport of radionuclides in a sensitivity analysis (SKB TR-19-05). In order to be able to handle hydrological parameter uncertainties in the Biotex model, a review is also currently in progress of simplified hydrological models (for example Hype, see Lindström et al. 2010) and a study aimed at reducing the calculation time for Mike SHE through model simplifications. The goal of the development work is that uncertainties in surface hydrology in the future can be captured as data uncertainty, which can be propagated to the safety assessment in the form of probabilistic transport calculations with Biotex.

SKB has initiated a validation of near-surface hydrological models in Krycklan by using CFCs (freons) to determine the age of the groundwater (IAEA 2006, Kolbe et al. 2020).

Programme

- High-resolution physical and chemical modelling with the COMSOL tool to understand fixation processes and substantiate K_d values obtained from measurements.
- Continued studies of leaching, mobilisation and accumulation of substances in a landscape perspective in the Krycklan area.
- Continued development of the hydrological model for running water – see Section 12.1.
- Efforts to reduce model complexity in hydrological models while retaining the ability to capture the main features of key processes for transport and accumulation in the biosphere. The work aims to facilitate stochastic applications of numerical 3D models with shortened calculation times.
- Validation of the hydrological model used for Biotex in SR-PSU.
- Field studies of key parameters for surface hydrological modelling, such as hydraulic conductivity and infiltration capacity in the regolith, in order to reduce uncertainties in modelling.
- Continued work on the development of CFC as a validation method for hydrological models and application of the method in Forsmark.
- Continued work to describe how the soil chemical environment will change over time in Forsmark, in particular the effects of calcite leaching, in order to take long-term account of how the mobility and fixation of substances may change in a future landscape.

12.4 Radiological, biological and chemical properties of important elements

In addition to different substances being taken up in different organisms and food chains (Section 12.1) and transported and accumulated in different ways in loose deposits (Section 12.3), they also have different chemical and radiological properties. These properties, in interaction with the chemical environment in which the elements occur, affect both their mobility and their radiotoxicity. This section primarily describes the programme aimed at measuring or compiling these fundamental properties of elements that are potentially important for dose to humans or other organisms.

Current situation

Prior to the assessment of post-closure safety, SR-PSU, the concentration ratios (CR) and sorption data (K_d) that were subsequently used in the analysis of surface ecosystems in Forsmark were compiled (Tröjbom et al. 2013). These values were then partially adapted for Laxemar prior to SE-SFL (Grolander and Jaeschke 2019) and updated for certain analyses in the PSAR prior to the extension of SFR. The values are based primarily on measurements from the site and secondly on available international data. During the period 2009–2019, SKB participated in the IAEA programmes EMRAS

II and MODARIA I and II. In these programmes, aggregated CR and K_d values were compiled (IAEA 2021a, b). Parts of this work have been published in other contexts (Wood et al. 2013, IAEA 2014, 2016). Posiva has also compiled chemical data from surface ecosystems in Finland (for example, Aro 2021, Kirkkala and Mikkilä 2021, Lahdenperä and Kuusisto 2021, Salo 2021, Toivola and Haavisto 2021), which permits comparisons and supplementation with the data used by SKB. There is a need to supplement the current compilation of K_d and CR data with data for some additional elements that are judged to be important for SKB's different final repositories.

Decay chains, for example the uranium chain (including ^{238}U, ^{234}U, ^{230}Th, ^{226}Ra, ^{222}Rn, ^{210}Pb and ^{210}Po) are important for the safety assessment of the Spent Fuel Repository, among other things because uranium constitutes a large part of the inventory and because ^{226}Ra is dose-dominating. The natural occurrence and transport of radionuclides in the uranium chain has previously been studied in a potential discharge area in Klarebäcksmossen in Laxemar (Lidman 2009). Together with a model study in SE-SFL (SKB TR-19-05), it was shown that the different isotopes have different mobilities. The decay chains produce a complex pattern of when and where the highest doses occur.

For several of SKB's repositories, the radionuclides ^{59}Ni and ^{93}Mo have proved to be potentially important for dose. A thorough review and synthesis of available chemical data on nickel and molybdenum from the site investigations was conducted within SE-SFL (Lidman 2022). Molybdenum showed overall high mobility, except in highly reducing environments such as peat soils or organic lake and marine sediments, where strong accumulation of the element could be observed. Nickel generally showed great similarities with many other divalent transition metals, which indicates that this group generally is controlled by similar processes. For both nickel and molybdenum, it was concluded that the site-specific K_d values at least qualitatively reflect the transport and accumulation patterns that can be observed in the site investigation areas.

At Byle Farm, extensive sampling of various arable crops grown in several different types of soil is in progress. The purpose of sampling is to supplement CR and K_d data, and to gain a better understanding of how uptake (Section 12.1) and fixation (Section 12.3) of elements are affected by different soil types and their chemical properties. In Section 12.1, different processes that control chlorine flux have also been mentioned, and continued sampling and chemical analysis of different chlorine compounds are planned, above all for aquatic environments.

The chemical environment of a substance is crucial for its chemical speciation, which determines the substance's properties in this environment (Sohlenius et al. 2013a). The chemical environment can be characterised by pH, redox, concentrations of main constituents (i.e. dominant anions and cations) and the presence of complexing agents, for example dissolved organic carbon and iron colloids.

Through a number of different processes, such as land uplift, climate change, weathering and ecosystem succession, the chemical environment of a site can change over time. One example is the calcite leaching that is expected to take place in Forsmark during the next millennia, which will affect pH values and thereby fixation and uptake of different radionuclides. SKB is working on a compilation of how the occurrence of calcite affects the properties of different elements. As an example, uranium is expected to be less mobile in a future landscape with a lower calcite influence (Lidman et al. 2019). A related study shows the importance of barite precipitations that affect the properties of radium (Jaremalm et al. 2013).

Studies from Krycklan's catchment basin in northern Sweden show that the ratio U-234/U-238 can serve as an indicator of deeper groundwater (Lidman et al. 2016) and in Forsmark, too, there are clear differences between uranium from the bedrock and uranium from the quaternary deposits. An interesting new study from Krycklan uses europium in relation to other lanthanoids as an indicator of leaching (Lidman et al. 2019).

The occurrence of wetlands and other organic soils is another factor that affects the mobility of many radionuclides. This is primarily due to the concentration of dissolved organic carbon (DOC) being higher, which in turn leads to lower pH and increased solubility of iron precipitations (Köhler et al. 2014). This is of importance for the mobility of many insoluble forms of radionuclides, since in the aqueous phase they often occur bound to either colloidal iron or DOC. Discharge of deep groundwater may also have a significant effect on the aquatic environment (Section 12.3) by affecting both pH and DOC concentrations, and thereby the mobility of many radionuclides (Lidman et al. 2016).

Alongside organic matter, iron is potentially important for both sorption and colloidal transport of several radionuclides in oxidising environments (Dahlqvist et al. 2007). The natural landscape evolution may lead to previously oxidising environments becoming waterlogged and reducing, which may destabilise iron precipitations and thereby mobilise substances that are bound to them (Ingri et al. 2018).

Programme

A large part of the activities in this area are coordinated with the sorption and uptake studies presented in previous sections. Sampling and chemical analyses usually take place at the same time and are reported here as one activity:

- Compilation and evaluation of chemical data from SKB's investigations, together with available chemical data from other sites in the Nordic region, to determine K_d and CR and to improve understanding of how the chemical environment affects K_d and CR.
- Review of the methodology for selection of values for sorption and uptake (K_d and CR), and for determination of the uncertainty range of these values, in the assessment of post-closure safety.
- Supplementary sampling (paired samples) of different elements within the sampling and monitoring programme for Forsmark, Byle Farm or in campaigns. Samples should preferably be collected from potential discharge areas and agricultural land. In addition to element concentrations, other physical and chemical soil properties are measured, including pH, organic content, clay content and the presence of complexing agents in water samples.
- Evaluation of milder leaching methods in the analysis of different elements in soil samples, with the aim of obtaining results that are more representative of the chemical environment prevailing in discharge areas.
- In-depth studies and evaluation of distribution patterns for chlorine, in order to be able to link the distribution to the properties of the element and to processes in the ecosystems, and to better model the chlorine cycle in the ecosystem.
- Evaluation of how the soil chemical environment in Forsmark changes over time.
- Continued studies in the Krycklan area in order to gain a better understanding of leaching and enrichment of different elements in a near-stream zone.
- Study of how iron colloids and DOC affect transportation of more insoluble radionuclides.
- The study of how lanthanoids can be used as tracers for leaching that has started in Krycklan will continue in the Forsmark area.
- Review of data from the site investigation areas for isotopes that are included in natural decay chains, and supplementary investigations of isotopes in the uranium chain.

13 Climate and climate-related processes

Within climate and climate-related processes, there are outstanding questions with a bearing on all three final repositories (the Final Repository for Short-lived Radioactive Waste (SFR), the Spent Fuel Repository and the Final Repository for Long-lived Waste (SFL)). For SFR and the Spent Fuel Repository, these are mostly concerned with reducing uncertainties in the assessment of post-closure safety, while for SFL they are concerned with having sufficient and updated data for the coming first assessment of post-closure safety. In addition, research needs to be conducted to evaluate the methodology used to handle climate and climate-related issues in the safety assessments, and work is needed to ensure that the climate scenarios are up-to-date with the current state of knowledge. In addition to providing the safety assessments with information and data, some of the climate-related issues are also relevant to the design of the three repositories. The activities that are planned to be carried out in order to achieve this are described here.

13.1 Climate scenarios and evaluation of extremes

SKB's climate scenarios serve as a basis for many of the analyses that are carried out within the framework of the safety assessments for the three repositories. It is therefore important that the climate scenarios are up-to-date with the current state of knowledge, and that they include and describe the extremes in climate and climate-related processes that are of importance for the function and safety of the repositories.

The methodology SKB uses to handle climate and identify extremes in climate and climate-related processes has so far mainly been based on historical climate data. With a new independent method that simulates the uncertainty in climate for the next one million years, SKB's methodology will be analysed in order to determine whether it has adequately identified necessary extremes in climate with importance for the safety of the repositories after closure. Table 13-1 gives examples of extremes and which issues they affect, particularly in the assessment of post-closure safety of the Spent Fuel Repository.

Table 13-1. Examples of climate-related extremes of importance for the assessment of repository post-closure safety and which issue(s) in the Spent Fuel Repository's assessment of post-closure safety, in particular, each extreme is linked to.

Extreme condition of importance for the assessment of post-closure safety	Linked issue in the safety assessments
Maximum duration of glacial conditions over the next million years	Buffer erosion/canister corrosion
Maximum and minimum number of passages of ice sheet fronts	Elevated groundwater flows/buffer erosion/transport times
Glaciation dynamics	Isostatic load on canister/rock stresses
Maximum length of periods of cold and dry climate conditions without ice sheet	Maximum permafrost depth/groundwater flow/freezing of buffer and backfill
Maximum length of current interglacial (warm period)	Length of initial period of dilute groundwaters and of geochemistry/buffer erosion/landscape evolution

The evaluation is carried out by comparing extremes in climate, as identified with the new method, with equivalent extremes identified with the previous methodology (e.g. SKB TR-10-49, TR-11-01, TR-13-05, TR-20-12).

The question of the length of the current interglacial is also relevant for the modelling of landscape evolution, as well as for the assessment of post-closure safety for SFR, since a longer interglacial entails that a colder climate occurs later in the assessment period, when a potential impact on the repository's barriers is less significant.

Current situation

For the evaluation of methods for identifying extremes in climate, simulations of climate evolution for the next one million years have been carried out in a first phase of an ongoing study. The simulations are based on factors such as different degrees of concentration of greenhouse gases in the atmosphere and known future variations of insolation to the Earth. Since the simulations take these variations into account, the resulting climate developments, unlike those used in the climate scenarios of the safety assessments, include a more realistic variability in climate and glacial cycles for the next one million years. First, a large number of global climate simulations were carried out, after which a downscaling to the Forsmark site was carried out for a selection of these. This phase of the study is described in Lord et al. (2019). In that study, only one (of many conceivable) realisations was scaled down to Forsmark and climate data was presented for this realisation.

The results show that there is considerable natural variability in the climate on these long time scales, with fluctuations in the average annual air temperature of up to ten degrees and recurrent shorter and longer periods of glaciation. Furthermore, the results show that the anthropogenic emissions of greenhouse gases have a very large impact on the climate over the next 100 000 years. In the event of strong emissions, the next glacial period may be postponed by several 100 000 years (Lord et al. 2019).

For evaluation of whether the current methodology has adequately identified and described extremes included in the assessment of post-closure safety, a broader statistical basis is needed than the single climate evolution that was downscaled to the Forsmark site in Lord et al. (2019).

Programme

- All of the existing global climate simulations from Lord et al. (2019) will be downscaled to the Forsmark site to obtain a broader statistical basis for the analysis of extremes.
- An evaluation of climate extremes to be performed based on data from the downscaled climate simulations. The evaluation will serve as a basis for any updating or supplementation of SKB's climate scenarios.
- If the above-mentioned evaluation of climate extremes warrants further studies of climate-related processes (for example supplementary simulations of ice sheet extremes, permafrost/freezing, glacial isostatic change, glacial erosion), this will be assessed and, if necessary, carried out.

13.2 Historical climate change

SKB's climate scenarios used in the assessments of post-closure safety are based partly on how the climate has varied historically and partly on simulations of future climate. Historical climate variations are being studied both with climate models and by means of analysis of data from different types of geological climate archives. For example, SKB's so-called reference glacial cycle is based exclusively on a repetition of climate and environmental conditions reconstructed for the last glacial cycle (including the Weichselian glaciation and the current Holocene interglacial). Studies of geological climate archives are being conducted to obtain the necessary information on how the climate has varied during this glacial cycle. The results of historical changes in air temperature, precipitation and vegetation are used to provide a nuanced picture of the climate change that can occur i) at the climate transition from warm interglacial conditions to cold glacial conditions, ii) during warm periods (interstadials) and cold periods (stadials) embedded in a glacial cycle, and iii) at the transition from glacial conditions back to interglacial conditions. This knowledge is used in the descriptions of and rationale for the climate scenarios used in SKB's different assessments of post-closure safety. Furthermore, they provide necessary information and climate data for other disciplines where the effect of climate change is being studied, for example in hydrogeology and surface ecosystems.

Current situation

Reconstructions of the climate during different periods of the Weichselian glaciation, the current Holocene interglacial period and the previous Eemian interglacial period have been carried out, for example by means of analyses of lake sediments from Sokli in northern Finland. The study analyses unusually thick and fossil-rich sediments of late Quaternary age preserved in situ in the Sokli basin. The studies contribute important information on how quickly the climate in Scandinavia can switch during different phases of a glacial cycle, and also how large temperature, and in some cases, precipitation variations can be during a glacial-interglacial cycle.

The results of the completed studies at Sokli are summarised in Helmens (2019). This provides a detailed reconstruction of the climate and environment of northeastern Fennoscandia from multi-proxy data for i) the Holocene interglacial period (the past 11 000 years), ii) warm and cold periods during the early Weichselian (Marine Isotope Stages (MIS) 5c-d around 115 000 to 90 000 years ago) and iii) the preceding Eemian interglacial period (the period MIS 5e around 130 000 to 115 000 years ago). The results complement previous results, which provide corresponding information for the Mid-Weichselian (MIS3 about 50 000 years ago) (Helmens 2009) and overall for the entire Weichselian glaciation (MIS 5-2, 130 000 to 15 000 years ago) (Helmens 2013).

Following completion of the analysis of the Odderade interstadial period during the early Weichselian (MIS 5a about 85 000 to 74 000 years ago) (Helmens et al. 2021), the analysis of the Eemian interglacial period (Katrantsiotis et al. 2021, Plikk et al. 2021, Salonen et al. 2021), and a supplementary analysis of the current Holocene interglacial period (Rijal et al. 2021), the analysis of warm and cold periods during the last glacial cycle from lake sediments at Sokli is concluded. Additional results from the studies at Sokli are included in Finné et al. (2019) and Felde et al. (2020). For references to reports of previous results, see Chapter 14 in RD&D Programme 2019. The results have shown, among other things, that when the climate went from warm interglacial conditions to cold glacial conditions, this transition took place over a very long time and with marked shifts between warm and cold periods (for example Helmens 2013, 2019, Helmens et al. 2021). The results from Sokli thus partially revise previous reconstructions of glaciation and climate history for the past 130 000 years, above all regarding the transition from warm interglacial conditions to cold glacial conditions (for example Helmens et al. 2021, Dalton et al. 2022).

The results from the study of lake sediments from Sokli in northern Finland have successfully contributed to well-dated quantitative information on how the climate and environment in northern Fennoscandia have varied during the past 130 000 years (including the latest glacial cycle, the Weichselian-Holocene). Conducting these types of studies contributes to improving the dating of terrestrial climate archives in Fennoscandia and Europe, especially when the results are compared with climate archives from other European sites, as in Wohlfarth (2013), Helmens (2019), and Schenk and Wohlfarth (2019).

Programme

- A systematic compilation of all previous results on historical climate and environmental conditions during the last glacial cycle (the Weichselian), the current interglacial (Holocene), and the preceding interglacial (Eemian), obtained from geological studies funded by SKB on the Sokli site in northern Fennoscandia.
- The above compilation is planned to be placed into a larger context and compared with climate reconstructions from other locations in Europe for the purpose of better describing how the reconstructed climate in northern Scandinavia can be used to assess the climate in central Sweden (Forsmark).

13.3 Sea-level variations and shoreline displacement in the short and long term

SKB is conducting work on sea-level changes on different time scales, both until the year 2100 and for the next 10 000 years. The work entails compilations of the contemporary scientific knowledge, which is rapidly developing, and site-specific sea-level simulations for the Forsmark site. In the case of the Spent Fuel Repository, the results of the analysis up to 2100 will be used in the work on dimensioned sea levels so that the repository (access routes and surface facilities) is designed to withstand a maximum rise in sea level (increase in mean sea level plus rise in storm flooding) during the construction and operational phases. The results up to the year 2100 are also relevant for the Final Repository for Short-lived Radioactive Waste (SFR) in order to ensure that the repository can withstand possible sea level rises during the operating period. In addition, the results for the period up to the year 2100 are used as an initial state in the assessments of post-closure safety included in the PSAR for the Spent Fuel Repository and SFR. The sea level in Oskarshamn is also being studied, since it is important for analysis of the hydrology around Clab.

The studies of sea-level variations for the next 10 000 years are used in SKB's safety assessments, where the results are necessary for the analysis of landscape evolution in Forsmark (land/sea distribution and how it changes over time), and for analyses of geochemistry and hydrogeology. For SFR, the analysis of sea levels is also crucial for estimating how long the area above the repository is likely to remain submerged, and for obtaining extreme values of how long/short the submerged period could be.

Current situation

The current state of knowledge of possible future sea level changes in the short and long term has been compiled and presented in SKB (TR-20-12). The compilations include maximum sea level rise for different levels of greenhouse gas emissions. As expected, the results show that the uncertainty regarding the size of future sea level rise is very large, both in the short term up to 2100 (Tables 5-3 and 5-9 and Figure 5-16 in SKB TR-20-12) and for the next 10 000 years (Tables 5-5 and 5-12 and Figures 5-4 and 5-23 in SKB TR-20-12). The uncertainty stems mainly from how great the future climate warming could be and how severely the Earth's cryosphere (especially the Antarctic and Greenland ice sheets) will react to the warmer climate (Sections 5.1.3 and 5.2.3 in SKB TR-20-12).

In large parts of Sweden, an isostatic uplift of the Earth's crust is occurring in response to the latest deglaciation. In Forsmark, this uplift currently amounts to 6.7 millimetres per year (Vestøl et al. 2019) and this rate is expected to remain relatively unchanged until the year 2100. If the sea level rise is not extensive, the isostatic uplift will partially or fully compensate for the future sea level rise. However, if the sea-level rise is extensive, the isostatic uplift will not be able compensate for the entire rise, and the sea level in Forsmark would rise relative to the land.

An update of the state of knowledge of the increase in the mean sea level and in the event of a storm surges in the near future (up to 2100) has been carried out for Forsmark and reported in Pellikka et al. (2020) and in SKB (TR-20-12, Sections 5.1 and 5.2). In Pellikka et al. (2020), the contribution to sea-level rise from storms with recurrence intervals of up to 100 000 years was estimated. The study also includes a probabilistic estimate of what the maximum rise in the total sea level (the rise in mean water level plus the rise during storm surges) at Forsmark will be up to 2050, 2080 and 2100 for the climate scenarios RCP2.6 (low greenhouse gas emissions), RCP4.5 (medium emissions) and RCP8.5 (high emissions). In the short term, up to the year 2100, the study shows that the current isostatic uplift will likely compensate for the sea level rise under low and medium emissions, resulting in a continued marine regression similar to today, where land rises from the sea (Pellikka et al. 2020). However, for high greenhouse gas emissions, or for sea level rises with a lower probability under lower emissions, the results show that warming can result in a marine transgression (where land is submerged by the sea) caused by an increase in the mean sea level up to the year 2100 (Pellikka et al. 2020, Table 5-10 in SKB TR-20-12).

Figure 13-1 shows the sea level rise in Forsmark under high emissions of greenhouse gases (RCP8.5) from Pellikka et al. (2020). Despite the ongoing isostatic uplift, the maximum mean sea level rise in this pessimistic case is around one metre up to 2080 and two metres up to 2100. The results also show that sea-level rise up to 2100 occurs at an accelerating rate (this is not visible in Figure 13-1). In the analysis of possible future maximum sea levels in Forsmark, the additional rise that may occur temporarily in the event of heavy storms should also be added. The study shows that the maximum sea level during heavy storms could increase by about three metres above the current level up to 2080 AD and about four metres up to 2100 AD for an annual probability of 10^{-5} (Figure 13-1). These results are used to plan and design the surface facilities of the Spent Fuel Repository to withstand elevated sea levels during construction and operation.

In the long term, too, the ongoing isostatic uplift means that the lowest projections of sea level rise result in a continued marine regression, similar to the current situation. However, the highest projections of future sea level rise during the next 10 000 years result in a strong marine transgression, followed by a regression. These projections lead to a period of raised sea level (compared with today) for up to 7 500 or 14 000 years, depending on the degree of warming (Sections 5.1.3 and 5.2.3 in SKB TR-20-12).

In order to cover the very large uncertainty in future sea level in the time perspective of 10 000 and 100 000 years, several different possible initially submerged periods are therefore used in SKB's assessments of post-closure safety for the Spent Fuel Repository (SKB TR-20-12) and SFR.

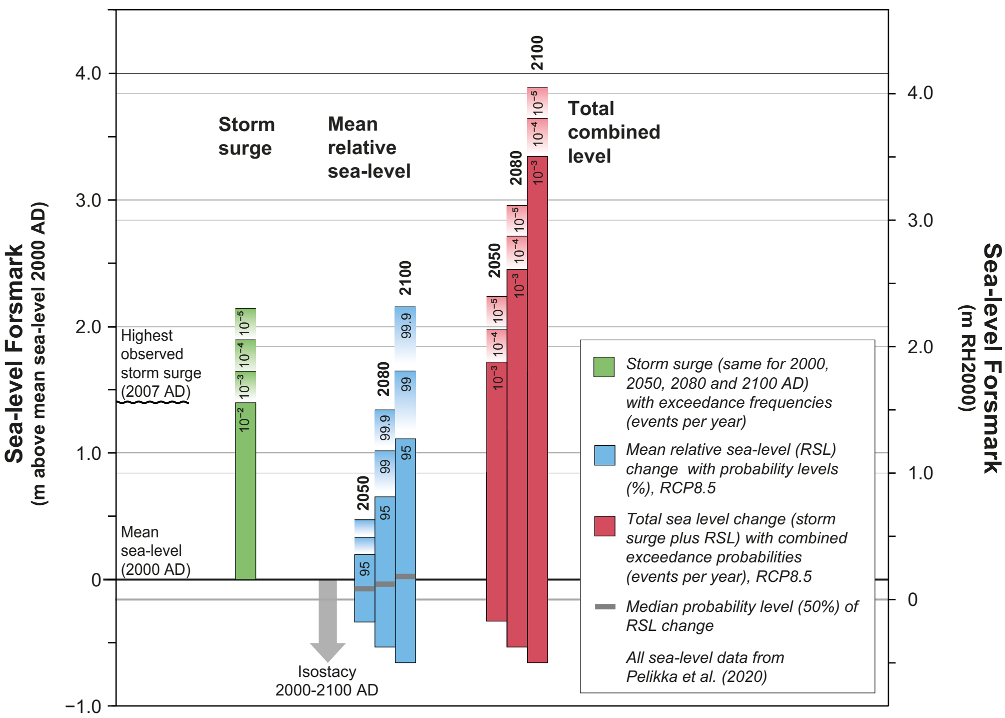

Figure 13-1. *Projections of relative sea level rise in Forsmark for the years 2050, 2080 and 2100 for the pessimistic emissions scenario RCP8.5 (Section 5.2 in SKB TR-20-12), based on data from Pellikka et al. (2020). The total maximum rise (red) is obtained by adding together the mean relative sea level rise (including the effect of isostatic uplift, blue) and the temporary rise during storm surges (green). A significant part of the mean water level rise is compensated for by the ongoing isostatic rise in land level (grey arrow). The left-hand y-axis shows the rise above mean relative sea level in 2000 and the right-hand axis the rise expressed in the RH2000 height system. For details, see Section 5.2 in SKB (TR-20-12).*

Programme

- SKB will continue to follow developments in the research area of future sea-level variations in the short term (up to 2100) and the long term (up to 10 000 years), both with regard to mean sea level rise and rise during storm flooding. If necessary, new compilations of the state of knowledge will be made. In connection with this, the state of knowledge of future global warming and the response of the Antarctic and Greenland ice sheets to the warming will be monitored.
- Development of a model that converts global projections of mean sea level rise to regional conditions in Forsmark by considering isostatic recovery and how much different ice sheet melting contributes to sea level rise in the Baltic Sea.
- Updating of sea-level projections (up to the year 2100) for Forsmark (Pellikka et al. 2020), when new data on global sea level rise warrants this.
- Summary of the state of knowledge of future possible changes in salinity in the Baltic Sea given different scenarios for future climate change.

13.4 Denudation processes affecting the rock surface, including quantification of historical and future glacial erosion

Over longer time perspectives, the rock surface is lowered due to erosion and weathering, processes collectively called denudation. If the denudation is unevenly distributed over initially relatively flat terrain, the topographical relief of the ground surface increases, which in turn can affect hydrological conditions (in the form of ground runoff and groundwater flow in the shallower parts of the rock) and the distribution of rock stresses in the upper hundreds of metres of the rock (Moon et al. 2020). If strong denudation should occur in connection with the Spent Fuel Repository, this affects repository depth and thereby, for example, the possibility of freezing reaching repository depth (SKB TR-20-12, Section 5.5.4). Over very long time periods (presumably several tens of millions of years), denudation in Forsmark would lead to the repository becoming located near or at the ground surface. Most of the denudation in Forsmark occurs through glacial erosion during recurrent glaciations.

A good understanding of the erosion processes that have been active in Forsmark historically and that are assumed to remain so in the future is therefore of great importance, including quantification of maximum glacial erosion over the next 100 000 and one million years. These issues are being studied through a combination of studies such as dating of the exposure age of the rock surface by means of cosmogenic isotopes, geomorphological analysis and numerical modelling.

Current situation

Studies reported in previous RD&D programmes have shown that the extent of total denudation of the crystalline bedrock in Forsmark has generally been low during the past one million years (from a few metres up to about 50 metres) (Olvmo 2010, Hall et al. 2019a). The span is partly due to the fact that there is large variation in the extent of denudation across the site, and partly to the statistical probability used. Regardless of the method used to calculate historical denudation (geomorphological analysis or exposure ages estimated with cosmogenic isotopes), the results show a limited historical denudation of the crystalline bedrock (Olvmo 2010, Hall et al. 2019a), but with a pronounced variability across the site. The limited denudation is largely due to the flat topography of the bedrock surface, which means that the erosion capacity of the ice sheet is lower than that of e.g. more active parts of ice sheets, or alpine glaciers, located at sites with more prominent topographic relief. Glacial erosion has been the most important denudation process in Forsmark during the past million years, and it is expected to remain so during the next one million years (Hall et al. 2019a, SKB TR-20-12).

Lord et al. (2019) presented a preliminary estimate of when and how long ice sheets may cover Forsmark during the next one million years, given different scenarios for greenhouse gas emissions (Section 13.1). If the historical glacial erosion, as determined in Hall et al. (2019a), is assumed to apply also for the future glacial periods described by Lord et al. (2019), the preliminary analysis shows that total denudation in Forsmark during the next one million years will also be limited (Hall et al. 2019a). In this scenario, the estimated depth of the total denudation of solid rock over the coming 100 000 years will in average amount to less than one metre for the Forsmark site, including the site for the Spent Fuel Repository. In the same way, the total denudation of solid rock for the next one million years has been preliminarily estimated from a few metres up to a few tens of metres. The large spread is caused by (expected) large spatial variation across the Forsmark site and nearby investigated areas. Denudation up to of a few tens of metres during the next one million years is not expected to negatively affect the safety of the Spent Fuel Repository, with respect to neither future maximum freezing depth during periglacial periods (SKB TR-20-12, Sections 3.5, 4.5 and 5.5), changes in stresses in the upper part of the rock nor hydrogeological changes.

The uncertainty in the estimated future denudation is assessed to be relatively large, mainly due to the preliminary estimates of the length of the future periods of ice sheet coverage as calculated in Lord et al. (2019). Another, minor, uncertainty is that the calculations are based on the assumption that the effectiveness of glacial erosion during earlier glaciations will also apply to future glaciations. However, this assumption is considered to be reasonable.

During the previous RD&D period, two sub-studies that investigated one of the basic assumptions in the geomorphological analysis in Hall et al. (2019a) were conducted (Goodfellow et al. 2019, Hall et al. 2019b). Neither of these studies, however, alters the overall conclusion in Hall et al. (2019a) that the historical denudation of crystalline bedrock in Forsmark has been low for the past one million years. For details, see Goodfellow et al. (2019) and Hall et al. (2019b).

In Hall et al. (2019a) a new glacial erosion process called glacial ripping was identified. This erosion process could, under certain circumstances, be relatively effective at the sites where it operates, in comparison with the glacial erosion processes of glacial plucking and glacial abrasion. Glacial ripping is a process in which larger layers of the uppermost part of the bedrock are eroded and transported away at the same time. The process is based on the formation of high water pressure under frontal-near parts of an ice sheet during the deglaciation. The high pressure propagates down into existing shallow subhorizontal bedrock fractures, whereby the uppermost layer of the bedrock is lifted by hydraulic jacking. The ice movement then moves and disintegrates the bedrock layer into blocks, which are finally deposited in the form of a block field or a block accumulation behind the retreating ice margin. Figure 13-2a shows a conceptual image of how glacial ripping works and Figure 13-2b shows shallow subhorizontal fractures observed in Forsmark. The process, which requires high water pressures and the presence of shallow subhorizontal fractures a few metres down in the rock, has historically been active in certain parts of Forsmark (Hall et al. 2019a, 2020). Glacial ripping links previously known processes and observations in a time sequence, which explains the occurrence of block fields in Forsmark and other locations.

In order to gain a better process understanding of glacial erosion, with a focus on glacial ripping, a continuation study was initiated during the previous RD&D period. The study uses i) geomorphological analysis of block fields and block accumulations mapped in the field (in Forsmark and in some other areas with relevant terrain), ii) fracture analysis in photographs of vertical cuts of the shallow bedrock in Forsmark (from photographs taken during construction of the cooling water canal to the Forsmark nuclear power plant), iii) analysis of fractures and small-scale geomorphology on rock outcrops, and iv) modelling of block fracturing under glacial conditions.

On the basis of the investigations above, Hall et al. (2020) describe the glacial ripping process in more detail than Hall et al. (2019a) for areas in eastern Sweden with crystalline bedrock with low relief, including Forsmark. The study suggests that the block accumulations that were studied were formed by glacial ripping during the last glacial period. An initial theoretical analysis of the forces needed to move protruding blocks of rock, for example small rock hills, by means of sliding ice at the bottom of an ice sheet, was presented in Krabbendam and Hall (2019). The fracture analysis in photographs of vertical cuts of the shallow bedrock in Forsmark is presented in Krabbendam et al. (2021), which has resulted in a theoretical explanatory model of the formation of new fractures due to the hydromechanical impact that occurs in the uppermost part of the rock by hydraulic jacking. This indicates that hydraulic jacking may result in continued disintegration of the near-surface rock, which reduces in strength and increases in hydraulic conductivity (Krabbendam et al. 2021). Krabbendam et al. (2022) exemplifies ripping with subglacial disintegration of roches moutonnées and subsequent formation of block fields and block caves.

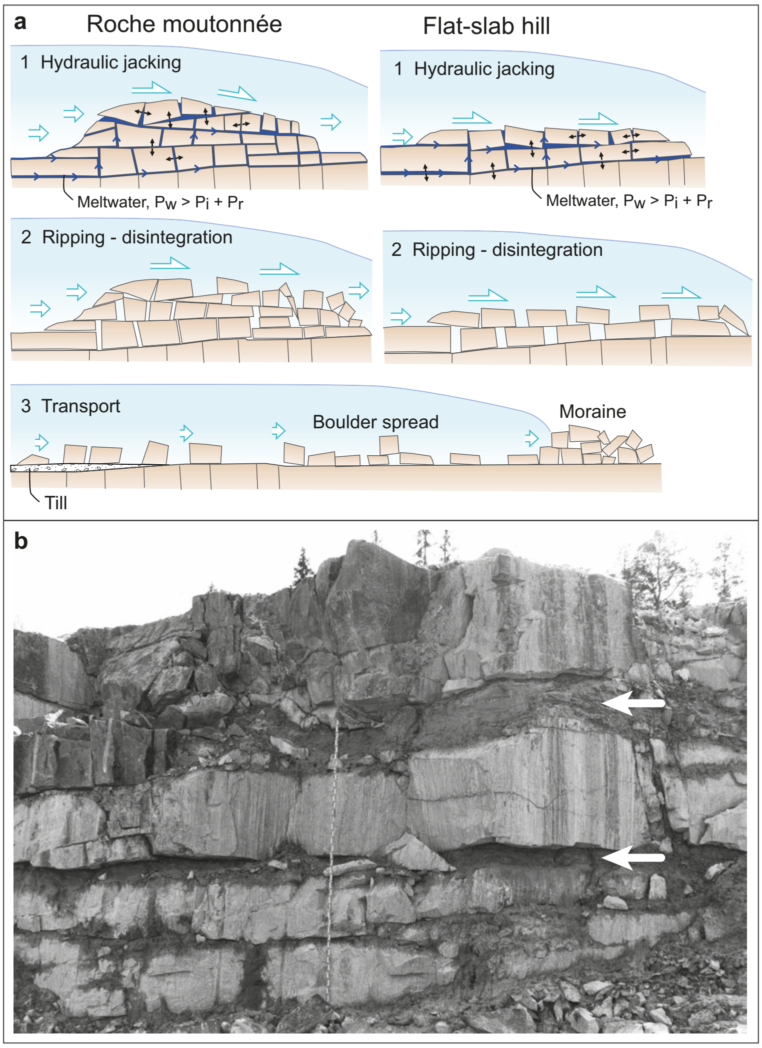

Figure 13-2. *a) Conceptual image of glacial ripping of roches moutonnées and flatter rock outcrops. For details, see Hall et al. (2020). b) Subhorizontal near-surface fractures in Forsmark photographed during construction of the cooling water canal at the nuclear power plant. Modified from Carlsson (1979). Photographer: Göran Hansson.*

One of the investigated sites consists of the Bodagrottorna caves at Iggesund, about 200 kilometres north of Forsmark. The results show that this, and other investigated sites with block accumulations, are most likely glacial landforms (Hall et al. 2020, Krabbendam et al. 2022), in line with how glacial ripping can break up bedrock and form block accumulations where the location and orientation of the blocks reflects the direction of the former ice movement. Krabbendam et al. (2022) show that this interpretation is also supported by observed fracture distribution at depth from previous bedrock drillings at the site (Carlsten and Stråhle 2000). Furthermore, the results show that previous alternative interpretations of the formation of the Bodagrottorna caves, which involved glacially induced earthquakes (see for example Mörner 2005) and/or methane gas explosions (see for example Mörner and Sjöberg 2018) are unlikely. For details, see Hall et al. (2020) and Krabbendam et al. (2022).

For SFR, post-closure safety is assessed for a period of 100 000 years (see for example SKB TR-14-01). Since global warming is expected to postpone the first future glaciation (Lord et al. 2019), the glacial erosion of crystalline rock in Forsmark during the coming 100 000 years is expected to be even more limited than it was during corresponding historical periods (Hall et al. 2019a). In average, it has been estimated to be less than one metre for Forsmark for this future period (Appendix G in SKB TR-20-12) on the basis that non-glacial conditions will prevail at the site during this period (Lord et al. 2019).

How denudation and glacial erosion are distributed over Forsmark during the next one million years is of interest in SKB's assessment of post-closure safety for the Spent Fuel Repository. Specifically for the glacial ripping process, the results in Hall et al. (2019a) show that this process has historically only occurred at certain locations in Forsmark (Hall et al. 2019a). Similarly, it is likely that only certain sites, with suitable conditions, e.g. with a combination of high subglacial water pressures and the presence of shallow subhorizontal fractures, have the potential for this to occur during future glaciations. Regarding the glacial erosion across Forsmark, quantification of historical and future total denudation in Hall et al. (2019a) is based largely on information from isotope analyses carried out for 32 sites in Forsmark and nearby areas in Uppland. The sampling points are few compared with the size of the area, especially in combination with the established large spatial variability in denudation/erosion (see the span of glacial erosion described above). One way of obtaining spatially comprehensive information on the distribution and variability of glacial erosion, to see whether the spot measurements have largely captured the largest expected future denudation at the site, is to carry out high-resolution numerical simulations of glacial erosion over the area. This can be done for both historical and future glaciations. In the former case, the results from the simulations can be compared with the independent quantifications of denudation in Hall et al. (2019a).

Programme

- Conclusion of the ongoing study of glacial erosion concerning i) glacial ripping, ii) mapping of glacial erosion forms on rock outcrops (using previously collected LIDAR data) and iii) analysis of the potential for headward erosion and associated potential future lateral aggradation of area with more forceful historical glacial erosion southeast of Forsmark (see Olvmo 2010).
- Improved estimation of denudation and glacial erosion of the rock surface in Forsmark during the next one million years from cosmogenic isotopes in the rock (^{10}Be, ^{26}Al, ^{14}C, presented in Hall et al. 2019a) based on an extended body of data on possible durations of glacial conditions at Forsmark during this period.
- Numerical modelling of historical and future spatial distribution of glacial erosion across the Forsmark site.

13.5 Ice sheet dynamics and behaviour

Current situation

An ongoing study is investigating how ice sheets with different properties (ice configuration, ice thickness, growth/melting rate) affect the stress field in the bedrock at Forsmark – see Sections 11.1 and 11.3. This work uses both the temporal evolution of the ice sheet included in SKB's reference glaciation and other ice sheet configurations. One of these represents the ice sheet evolution for the period with the largest ice sheet in Eurasia during the past two million years, which occurred during the penultimate glaciation called the Saalian (between 241 000 and 135 000 years ago). A time-dependent ice sheet evolution for the entire Saalian glaciation has therefore been simulated (Colleoni and Liakka 2020) to be used as input data for the study of the bedrock stress field. In the ice sheet simulation, a numerical thermomechanical ice sheet model was used, which was forced by climate data from three equilibrium simulations performed with a fully-coupled global atmosphere-ocean circulation model. The climate evolution for the time periods between these equilibrium simulations was assumed with the aid of geological data. In the study, several sensitivity simulations were performed with different parameter selections and spatial resolutions.

The results of one of the simulations are illustrated in Figure 13-3. In this simulation, the ice sheet builds up slowly during the first approx. 100 000 years of the Saalian glaciation, after which a rapid expansion of the ice sheet takes place up to the glacial maximum around 133 000 years ago. In this simulation, Forsmark is ice-free until about 20 000 years before the glacial maximum, after which the ice rapidly expands across northern Europe. At the glacial maximum, the ice thickness over Forsmark in this simulation is just over 3 500 metres.

The ice sheet modelling study presented in Colleoni and Liakka (2020) is used in the ongoing study of how different ice configurations affect rock stresses (Sections 11.1 and 11.3). It is planned to use the results in future analyses of glacially-induced earthquakes.

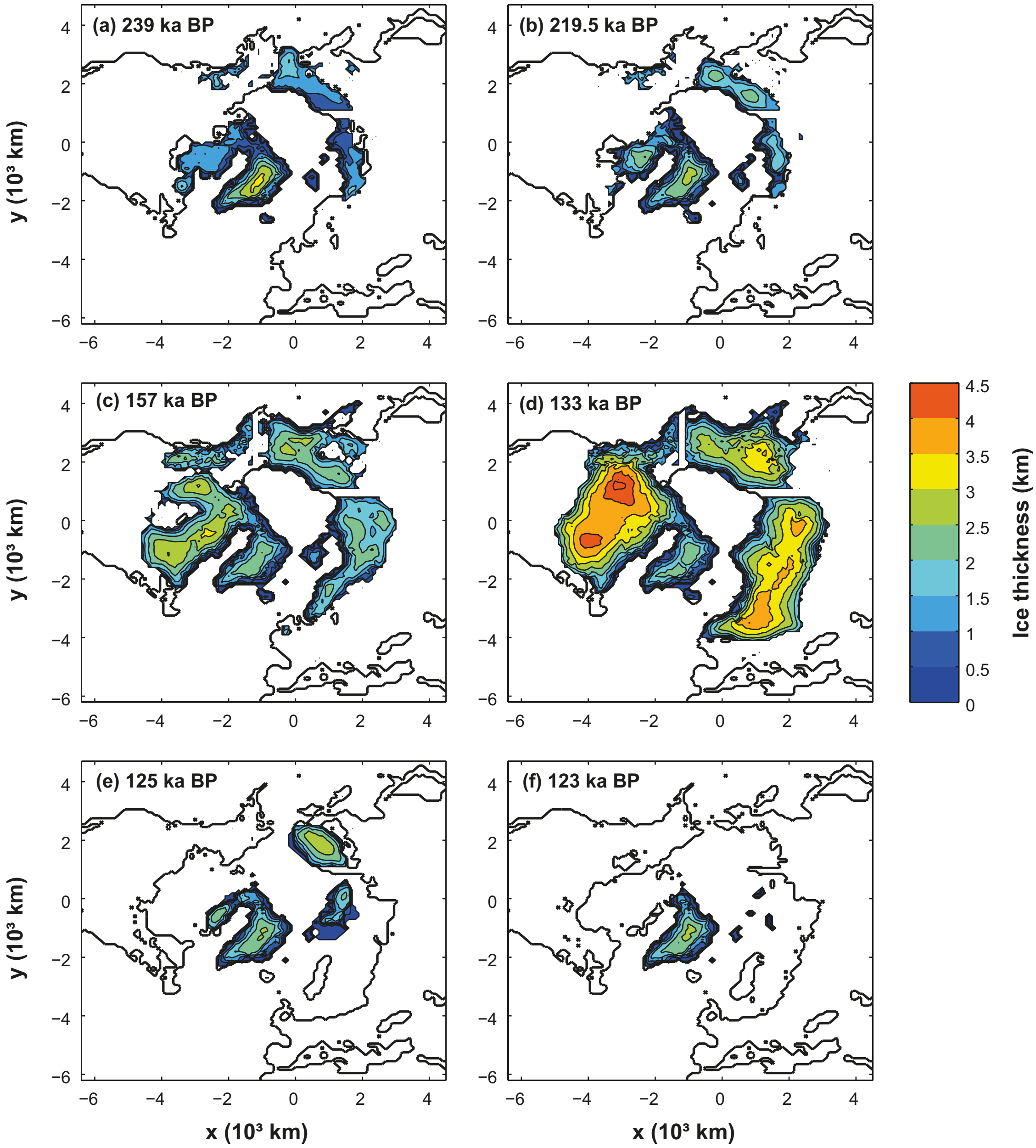

Figure 13-3. *Example of ice sheet evolution in the simulation of the Saalian glaciation (Colleoni and Liakka 2020). The glacial maximum is reached at 133 000 years before present. The coastline differs from that of today (and is different for all reported times) due to variations in the vertical position of the Earth's crust and in the sea level.*

Programme

- An analysis will be performed of how quickly an ice sheet could grow and reach Forsmark from its formation in northern Scandinavia, and how long the ice sheet margin could temporarily stop over Forsmark during a glaciation or deglaciation, which is of importance for the magnitude of groundwater flows and for the extent of glacially-induced stresses in the geosphere (SKB TR-11-01).
- No additional ice sheet simulations are planned during the RD&D period, but may be needed if the evaluation of SKB's climate scenarios (Section 13.1) warrants this.

13.6 Evaluation of permafrost model

Aggradation (growth) and degradation (melting) of permafrost are the most important climate-related processes for a final repository within the periglacial climate domain, regardless of waste type and repository concept. The periglacial climate domain prevails during a significant portion of time (about one-third) in SKB's reference glacial cycle for the Spent Fuel Repository. An analysis of the possibility of freezing of repository facilities and what effects this would have been carried out for both the Spent Fuel Repository (see for example SKB TR-11-01) and the Final Repository for Short-lived Radioactive Waste (SFR) (see for example SKB TR-14-01). Aggradation of permafrost also affects the flow pattern and chemical composition of the groundwater.

Previous RD&D programmes described plans and ongoing work to evaluate the permafrost model used in SKB's safety assessments and safety evaluations. The permafrost model has been used for the Spent Fuel Repository (Hartikainen et al. 2010, SKB TR-10-49, TR-20-12), the Final Repository for Short-lived Radioactive Waste (SFR) (Brandefelt et al. 2013, SKB TR-13-05, TR-14-01) and the Final Repository for Long-lived Waste (SFL) (SKB TR-19-01, TR-19-04).

Current situation

In order to evaluate the reliability of the permafrost model's calculated temperatures in the bedrock, including freezing and permafrost depths, calculated temperatures from the model have been compared with observed temperatures at the site of the GAP study on western Greenland, where permafrost around 400 metres deep exists today (Claesson Liljedahl et al. 2016, Harper at al. 2016). Detailed observations of the bedrock temperature have been carried out in a 650-metre-deep rock borehole (DH-GAP04) at the front of the ice sheet. The borehole penetrates the entire permafrost and reaches down into the unfrozen bedrock. A large amount of input data on the properties of rock, groundwater and surface parameters were compiled for this site, and these data were then used for simulations of temperature and permafrost/freezing depth (for details, see Appendix 2 and 3 in Hartikainen et al. 2022). Simulations of the bedrock temperature were carried out for a 15-kilometre long and 10-kilometre deep profile that passes through the site for temperature measurements in the bedrock (Figure 13-4). The simulations were carried out for a period covering the last 115 000 years, and a large number of simulations were carried out to investigate the sensitivity to uncertainties in input data. Examples of results from the simulations are shown in Figure 13-5. Frozen conditions and permafrost prevail outside the area with an ice sheet, and also extend a few kilometres in under the ice sheet (Figure 13-5a). The bedrock beneath the central parts of the outlier glacier (Figure 13-4) is unfrozen. Figure 13-5b shows simulated temperatures in the bedrock around borehole DH-GAP-04, with the upper part of the borehole located in frozen bedrock and the lower part in unfrozen bedrock. The simulated magnitude and direction of the heat flow is shown in Figure 13-5c.

The results of the evaluation show that in the versions of the permafrost model used in the SR-Site and PSAR safety assessments for the Spent Fuel Repository (Hartikainen et al. 2010, SKB TR-10-49, TR-11-01, TR-20-12), for the safety assessments SR-PSU and PSAR for the Final Repository for Short-lived Radioactive Waste (SFR) (Brandefelt et al. 2013, SKB TR-13-05, TR-14-01), and for the safety evaluation SE-SFL (SKB TR-19-01, TR-19-04), today's temperature distribution in the bedrock is predicted with an error margin within ± 1.5–2 degrees (Hartikainen et al. 2022). The uncertainty interval stems partly from uncertainties in input data and partly from systematic model errors. The latter type of error is judged to be minor, see Hartikainen et al. (2022).

The results of the study also contribute to an understanding of how uncertainties in key parameters affect the temporal and spatial evolution of permafrost and freezing depths – for details see Hartikainen et al. (2022). Parts of the study, concerning the palaeoclimate corrections that need to be done of geothermal data in these types of simulations, are also described in Colgan et al. (2022).

The results of the evaluation will be used to evaluate the assumptions and conclusions drawn in SKB's assessments of post-closure safety from previous simulations of permafrost and bedrock temperatures for future cold periglacial periods in Forsmark. These conclusions concern, for example, the maximum freezing depth at the site for the Spent Fuel Repository (see for example SKB TR-11-01). With the good model predictions that have been demonstrated in this study, the results are expected to confirm the methodology and the conclusions drawn in the safety assessments.

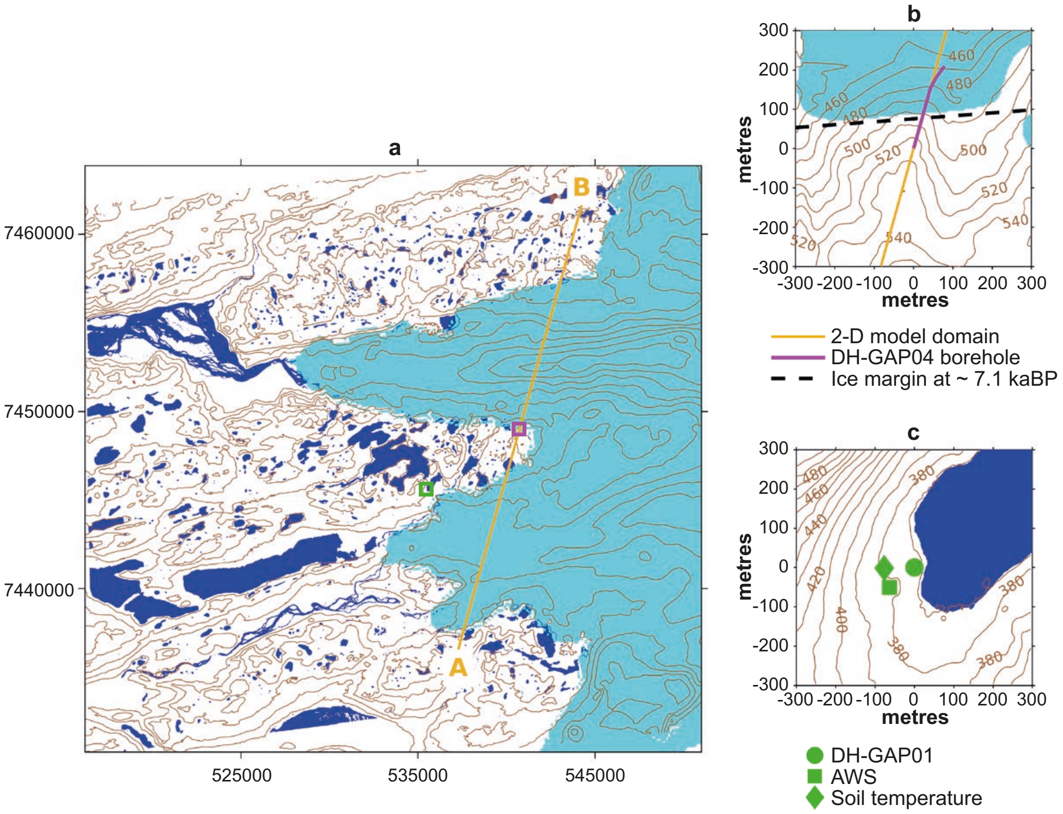

__Figure 13-4.__ a) 2D profile used in the permafrost simulations (yellow line) at the site of the previous GAP study (Claesson Liljedahl et al. 2016, Harper at al. 2016). Cyan: ice sheet, blue: proglacial lakes. b) extent of rock borehole DH-GAP04 (magenta line) in which many years of detailed temperature measurements have been performed. The borehole starts in the ice-free areas near the ice sheet margin and ends in the rock beneath the ice sheet. The 2D model domain is oriented in the same direction as the inclination of the rock borehole. From Hartikainen et al. (2022).

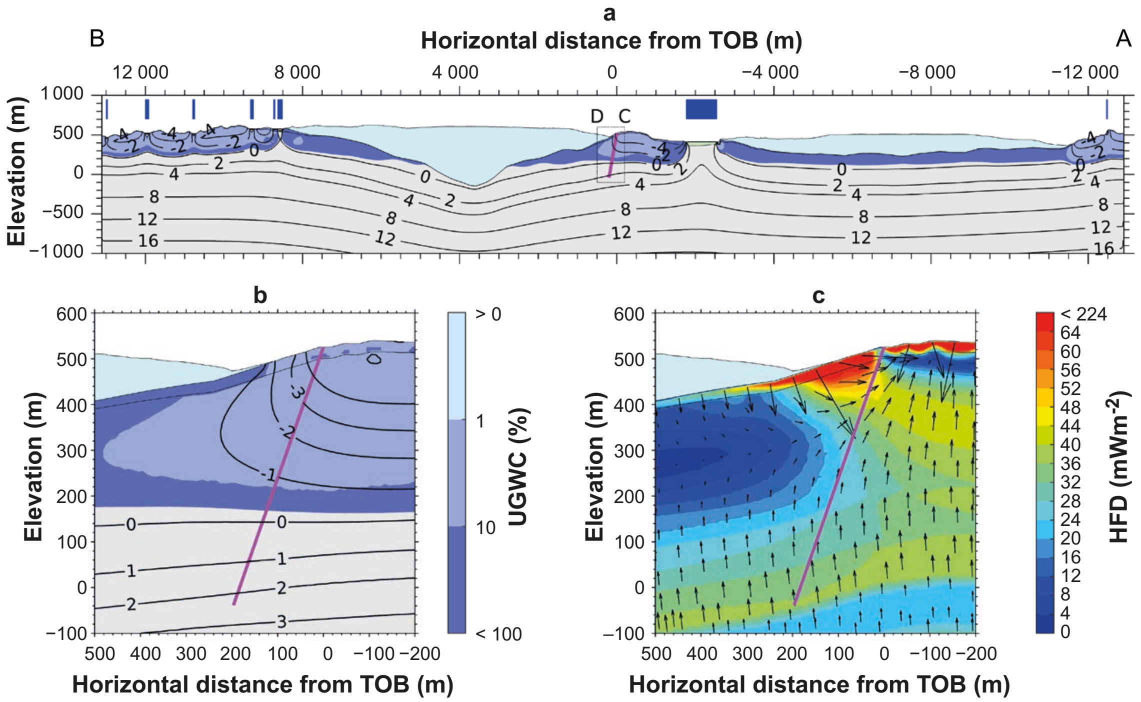

__Figure 13-5.__ a) Example of simulation of the present-day temperature distribution and unfrozen water quantity in the bedrock along the 15 km long profile. b) Enlargement of the area around the borehole DH-GAP04 (rectangle in a). c) Simulated geothermal heat flow and flow direction. The ice sheet is light blue and the DH-GAP04 borehole is shown with a purple line. Dark blue fields in the upper part of Figure a) indicate the location of lakes. The figure uses a 2x vertical exaggeration. TOB = Top of borehole. From Hartikainen et al. (2022).

Programme

- No additional permafrost simulations are planned during the RD&D period, but may be needed if the evaluation of SKB's climate scenarios (Section 13.1) warrants this.

13.7 Analogues for glacial hydrology, hydrogeology and geochemistry under glacial conditions

The GAP study was conducted between 2008 and 2013, after which the results were analysed and published on an ongoing basis. The two final reports from GAP (Claesson Liljedahl et al. 2016, Harper et al. 2016) were published in 2016, simultaneously in both SKB's, Posiva's and NWMO's report series. The two reports summarise the data gathered on glacial hydrology, hydrogeology, glaciology, meteorology, geology and geochemistry from the investigation area in western Greenland, as well as the scientific understanding and the conceptual models resulting from the study.

GAP was initiated to gain a better process understanding of how climate change, and particularly glaciations, may affect the long-term function of a final repository and the evolution of its environment over time. The project has resulted in a better understanding of a number of different processes that are of importance in assessments of post-closure safety, above all with respect to boundary conditions for groundwater modelling, bentonite stability and landscape evolution.

SKB, in collaboration with Posiva, NWMO and Nagra, has complemented the studies in GAP through a minor glacial-hydrological study (ICE) in the same investigation area in Greenland (Harper et al. 2019). The study investigated i) short-term, very high subglacial pressures, ii) gradients in pressure at the base of the ice sheet on a scale corresponding to the ice thickness, and iii) transmissivity and infiltration capability of the bedrock beneath the ice. The results improved the conceptual picture of the hydrology at the transition between ice and bedrock (particularly regarding basal water pressure) obtained from the GAP study (Claesson Liljedahl et al. 2016, Harper et al. 2016). The results for improved surface boundary conditions in future simulations of groundwater flow during future glacial periods at Forsmark.

Current situation

The monitoring of parameters that began during the GAP study in the DH-GAP04 borehole (pressure, temperature and salinity down to KBS-3 repository depth) and the collection of meteorological data from automatic weather stations in a profile up along the ice sheet and at a station located in front of the ice sheet has continued during the previous RD&D period. Data from this monitoring and previously collected data from the GAP study have been used for analysis of how groundwater flow and pressure in the bedrock are affected by the presence of the ice sheet (Claesson Liljedahl et al. 2021). Based on actual observations, the results show for the first time how the hydrogeology of the crystalline bedrock is affected by changes due to a slowly melting ice sheet. The results show that the pressure in the bedrock reacts very quickly and sensitively to variations in the ice sheet's hydrology and mass balance on a daily, seasonal and multi-year timescale (Claesson Liljedahl et al. 2021). Figure 13-6 shows a conceptual image of how the observed pressures in the borehole are controlled by the local geology (local/regional fracture zones in the rock) in relation to the topography of the ice sheet and the bedrock. The lowest part of the borehole (lower section in the figure) seems to be connected to the lowest parts beneath the ice via subhorizontal fractures, while the middle parts of the borehole (upper/middle in the figure) are connected to much shallower parts below the ice via subvertical fractures. The pressure variations at the bottom of the ice sheet differ greatly between these two areas, and this is reflected in the pressures recorded in different parts of the borehole. The results thus show a complex relationship between the pressure at the bottom of the ice sheet, including its spatial and temporal variations, and a clear pressure response in the bedrock.

Data on geothermal heat flow for borehole DH-GAP04 (Claesson Liljedahl et al. 2016, Harper et al. 2016) was calculated by Hartikainen et al. (2022) as part of the work of evaluating the permafrost model described in Section 13.6. These geothermal data (both uncorrected and corrected for variations in historical climates) have been included in a new compilation and analysis of all available geothermal data for Greenland (Colgan et al. 2022). The database will be an important part of future analyses, for example of how the Greenland ice sheet may react to a warmer climate. The in situ measurement results database and the gridded heat-flow model are freely available from GEUS (https://doi.org/10.22008/FK2/F9P03L).

Borehole DH-GAP04 was last sampled in 2015 for geochemical analysis of the groundwater. Today, the borehole has had time to recover for another seven years after the drilling of the hole (when the water chemistry was affected by drilling water). In 2022, water samples were taken from DH-GAP04 at the ice sheet margin for a concluding geochemical analysis (Section 11.4.2). The plan is to transfer SKB's installations in Greenland (rock boreholes, weather stations etc) to new principals.

Programme

- After the activities carried out in 2022, SKB is not planning any further activities at the site of the GAP study in Greenland.

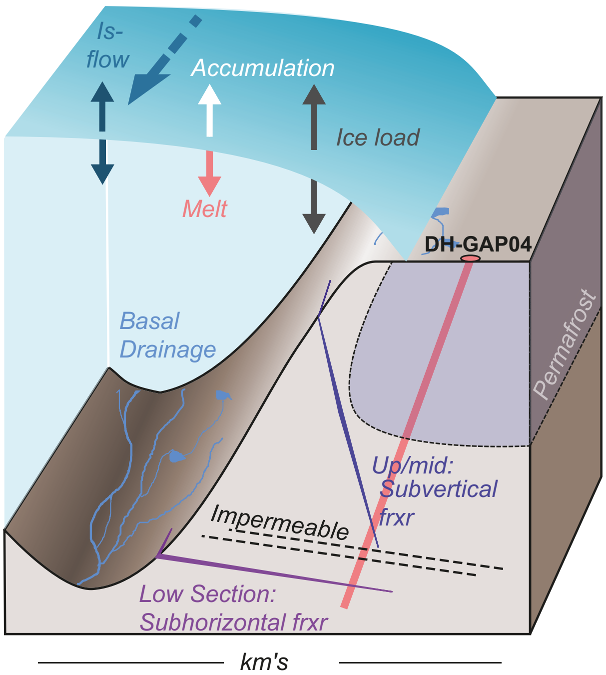

Figure 13-6. *Conceptual image of how ice sheet processes can yield a pressure response in the groundwater system, observed in borehole DH-GAP04 (diagonal red line). The image contains an interpretation of how different parts of the borehole connect to different locations under the ice sheet (with different hydraulic heads at the transition between ice and bedrock). The variations in the hydrology of the ice sheet (from different locations under the ice), with variations on a daily, seasonal, and multi-year timescale, are reflected in the pressure variations observed in the borehole. Adapted from Claesson Liljedahl et al. 2021).*

Part III

Decommissioning of nuclear facilities

Part III of the RD&D Programme 2022 presents the planning and division of responsibilities for implementation of the decommissioning of the Swedish nuclear power reactors and SKB's nuclear facilities. A summary of the development work linked to decommissioning is also given here.

14 Prerequisites for decommissioning of nuclear facilities

This chapter provides an overview of the set of requirements for the decommissioning of nuclear facilities, as set out in the Radiation Safety Authority´s (SSM´s) regulations and in the Swedish Environmental Code. The chapter also describes how the set of requirements affects the structure of a generic decommissioning project in the form of different stages and phases. It also presents the division of roles between the reactor owners and SKB regarding decommissioning and the management and disposal of the resulting radioactive waste. The subsequent chapters provide a more detailed description of how the work will be pursued within each group and within each decommissioning project, with a focus on strategies and planned measures.

14.1 Concepts and requirements

Decommissioning of a nuclear power reactor comprises shutdown operation, possible service operation, and dismantling and demolition. Shutdown operation lasts from final shutdown of the nuclear power reactor until all nuclear fuel has been removed from the facility. In cases where dismantling and demolition cannot begin immediately after shutdown operation, a period of service operation follows, during which the facility is maintained pending the start of dismantling and demolition.

During dismantling and demolition, activities are carried out to dispose of the radioactively contaminated facility components in the form of process systems, buildings and any contaminated soil. The dismantling and demolition phase is concluded when the facility has reached a state that enables it to be released from regulatory control. When the SSM has approved an application for release from regulatory control, the activities at the facility are not governed by Nuclear Activities Act and the Radiation Protection Act. The facility ceases to be a nuclear site, which means that the remaining demolition and site remediation can take place without restrictions under the Nuclear Activities Act and the Radiation Protection Act.

Figure 14-1 shows schematically how the decommissioning of a nuclear power reactor is carried out in relation to the set of requirements that apply to the facility during its life cycle. The upper part of the figure presents the activities planned to take place at the facility, and the lower part shows the requirements set by the Swedish Environmental Code and the SSM´s regulations.

The main licensing processes that govern a decommissioning project are: an environmental permit under the Swedish Environmental Code and approval under the Act on Nuclear Activities and the Radiation Protection Act. According to the Environmental Code, an environmental impact statement (MKB) 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 14-1. In conjunction with final shutdown, it is expected that it will be possible to revise the existing MKB, as shutdown and service operation will largely be similar to previous operation. Prior to dismantling and demolition, on the other hand, a new MKB appropriate for the purpose is required. The MKB, together with consultations, constitutes the basis for an environmental permit under the Environmental Code.

In accordance with the Nuclear Activities Act and the Radiation Protection Act, as well as applicable ordinances, permit conditions and regulations, the following documents must be prepared prior to and in some cases continuously during decommissioning:

- Decommissioning plan and decommissioning strategy.
- Waste management plan.
- SAR.
- Supporting data pursuant to the Euratom Treaty, Article 37.
- Step/sub-project notification.
- Decommissioning report.
- Inspection programme for release from regulatory control.

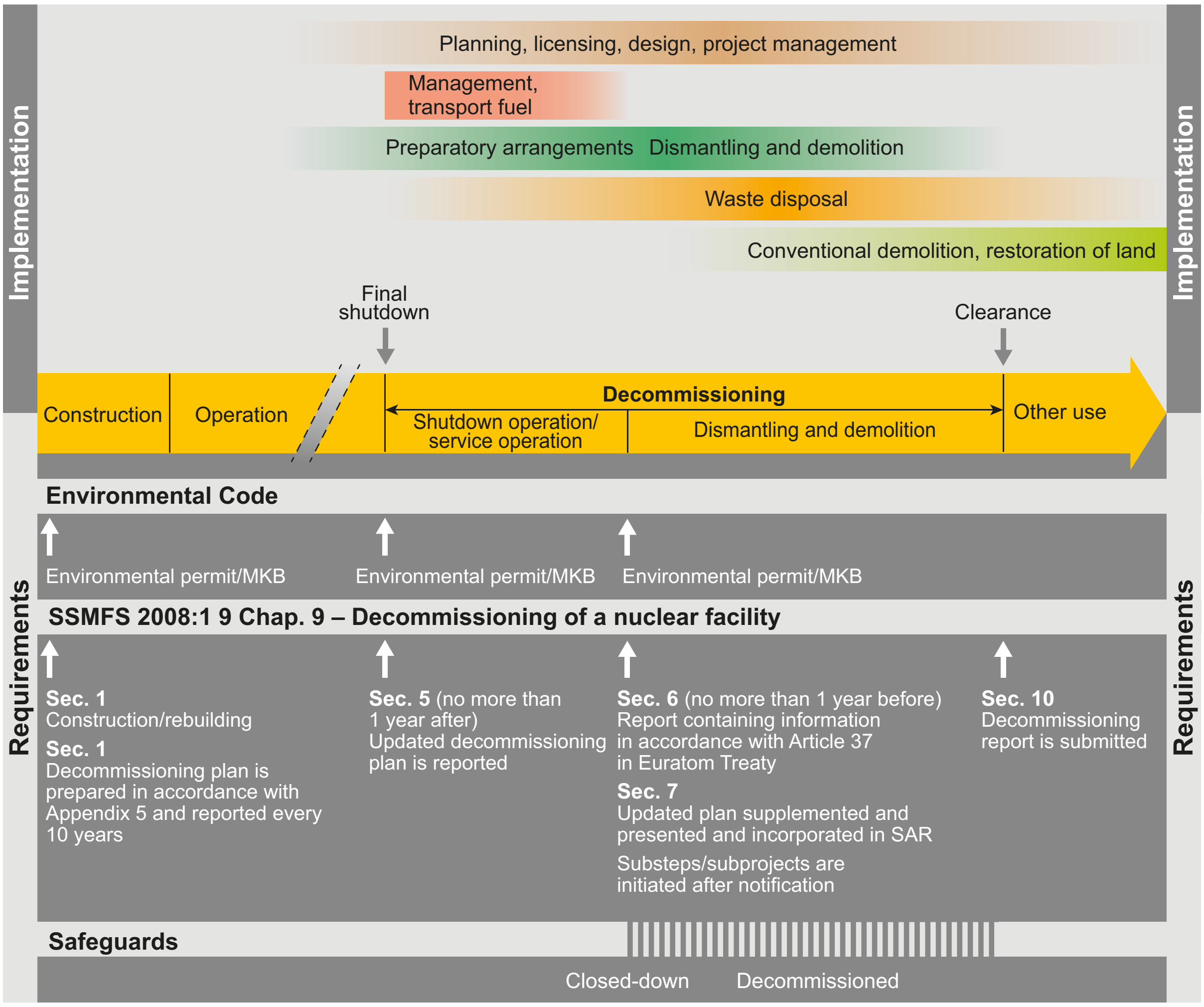

Figure 14-1. *Overview of the different phases of the execution of reactor decommissioning, with the requirements of the SSM and the Environmental Code for decommissioning during the life of a nuclear facility.*

The decommissioning plan must be reported to the SSM, while the waste management plan and SAR must be submitted for formal approval in accordance with SSMFS 2008:1. In addition, supporting material must be prepared for information to the European Commission, in accordance with the Euratom Treaty, Article 37. The formal report is submitted to the European Commission by SSM, but each decommissioning project prepares background reports.

During dismantling and demolition, notification of the measures that will be implemented in the facility is required. These are distributed practically in different steps/sub-projects which are notified successively as decommissioning progresses. Each step notification must include information on matters such as protective measures, choices of technology and risk assessments, and must be subject to a formal safety review.

When dismantling and demolition are completed, a decommissioning report describing experience from decommissioning and the end state of the facility must be submitted to the SSM.

During decommissioning, reporting on nuclear safeguards is also required. The reports include accounting reports, establishment of activity programmes and annual updates of the site description. The reporting requirements cease after written confirmation from the SSM that the facility has achieved “decommissioned installation” status in accordance with Commission Regulation (Euratom) No. 302/2005.

In addition to the above-mentioned requirements, there are also requirements relating to decommissioning issued by other regulators. These requirements, which are not described in detail in this document, include laws and regulations monitored by the Ministry of the Environment, the Swedish Work Environment Authority, the Swedish Civil Contingencies Agency, the Swedish Transport Agency and the Building Committee in the municipality concerned.

Additionally, decommissioning of the Swedish nuclear power plants is affected by the planned operating times of the nuclear power plants and the availability of interim storage facilities and final repositories for decommissioning waste. The overall planning is presented in Section 3.5 and in more detail below.

14.2 Responsibility and division of roles

The licensee of a nuclear facility is responsible for decommissioning under the Nuclear Activities Act, the Radiation Protection Act, the Financing Act and the SSM's regulations. The responsibility for the radioactive waste extends until it has been released from regulatory control or until the SSM has approved final sealing of the final repository in question and the Government has granted exemption from responsibility under Section 10 of the Nuclear Activities Act.

Vattenfall AB is the main owner of Ringhals AB and Forsmarks Kraftgrupp AB. Uniper (legal name Sydkraft Nuclear Power) is the main owner of OKG Aktiebolag and Barsebäck Kraft AB. The licensees of the nuclear power reactors are Barsebäck Kraft AB (Barsebäck 1–2), Forsmarks Kraftgrupp AB (Forsmark 1–3), OKG Aktiebolag (Oskarshamn 1–3) and Ringhals AB (Ringhals 1–4). Vattenfall AB is the licensee of the Ågesta reactor. SKB is responsible for Clab and the final repository for short lived radioactive waste (SFR), and for the future facilities Clink, the Spent Fuel Repository and the repository for long-lived waste (SFL). A government decision is required in order for a nuclear licence to be transferred.

To streamline the work on decommissioning and waste issues, work areas have been divided between actors at both company and group level. The joint waste management commitments are normally coordinated by SKB, while the practice for handling decommissioning-related issues varies slightly within the two industrial groups, Vattenfall and Uniper. This section describes working methods and the distribution of tasks.

14.2.1 Division of roles between the licensees and SKB

The licensees are responsible for decommissioning their nuclear facilities. SKB's principal task is to ensure the availability of a final repository for decommissioning waste in accordance with the needs of the licensees. SKB compiles the development needs identified by the licensees, coordinates general methods and procedures for transportation and final disposal of radioactive waste, and compiles the decommissioning-related costs reported by the licensees.

Under the Nuclear Activities Act, the reactor owners, working in consultation with each other, must arange for a programme for the comprehensive research and development activities and other measures needed to manage and finally dispose of the nuclear waste and the spent nuclear fuel, and to decommission the nuclear power plants. SKB, on behalf of and in cooperation with the reactor owners, prepare the RD&D Programme and submits i to the SSM – see Section 1.2.

General methods and procedures for demolition

SKB's mandate is to coordinate the general methods and procedures for transportation and final disposal of radioactive waste that are needed for decommissioning. SKB's tasks include developing type descriptions that describe how the waste meets the applicable acceptance criteria for each final repository. The waste producer produces type description specifications that serve as a basis for the type description, which also forms part of the waste producers' SAR. They must show that the waste meets the acceptance criteria for production and interim storage at the facility. In addition, SKB has the task of developing new waste containers for all the reactor owners as a group as needed, and of supplementing the transport system with associated transport containers.

In order to achieve optimal national coordination, the reactor owners have agreed on the tasks that SKB is to coordinate in connection with waste management, for example development of guidelines and joint principles for release from regulatory control (SKB R-11-15, Berglund et al. 2016), a common template for reporting of forecast decommissioning waste (Ahlford 2021) and principles for reporting of decommissioning plans (Calderon 2014).

Each reactor owner is responsible for the future decommissioning waste inventory, while SKB is responsible for compiling the inventory in respect of the planning needs for the SFL and the extended SFR. The reactor owners and SKB are collaborating on choosing a safe handling solution that meets regulatory requirements and is overall optimised for the entire waste chain – from demolition through characterisation, packaging, interim storage and transport of the waste to final disposal of different types of decommissioning waste.

In order to further optimise waste management on a national level, measures are being taken to strengthen SKB's coordinating role.

Waste during dismantling and demolition

During dismantling and demolition of a nuclear power plant, large quantities of waste are produced. The waste is of the same kind as the waste produced during operation, with the difference that the volumes will be larger. The waste comes from different parts of the facility and therefore contains great variations in degrees of radioactive contamination. The level of contamination is what determines the subsequent handling and final management route or management of the waste.

Most of the waste can be handled as conventional waste, since it stems from parts of the facility where radioactive material has not been handled, or where no contamination has been detected historically. This waste is referred to as zero-grade waste, which means that handling has been preceded by an initial assessment in which the waste has been categorised as having an extremely low risk of contamination (ELR). The conventional management route also includes waste that has been released from regulatory control. Waste is granted release from regulatory control when the level of contamination in the waste is below the applicable thresholds for release. Categorisation of waste is described in the report SKB (R-11-15) and in Berglund et al. (2016).

Waste that cannot be classified as zero-grade waste or granted release from regulatory control is segregated according to the appropriate management route. A compilation of forecast decommissioning waste from the Swedish nuclear power plants per management route and the main principles for management routes for nuclear waste during dismantling and demolition are presented in Figures 14-2 and 14-3. The compilation in Figure 14-2 gives an overview of the distribution and proportions of waste between different management routes.

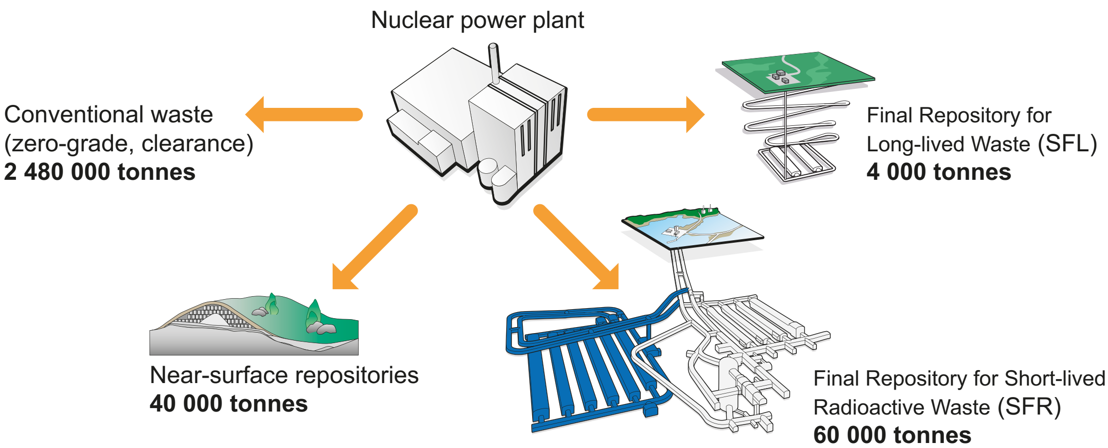

Figure 14-2. *Summary of forecast decommissioning waste from the Swedish nuclear power plants, and different management routes.*

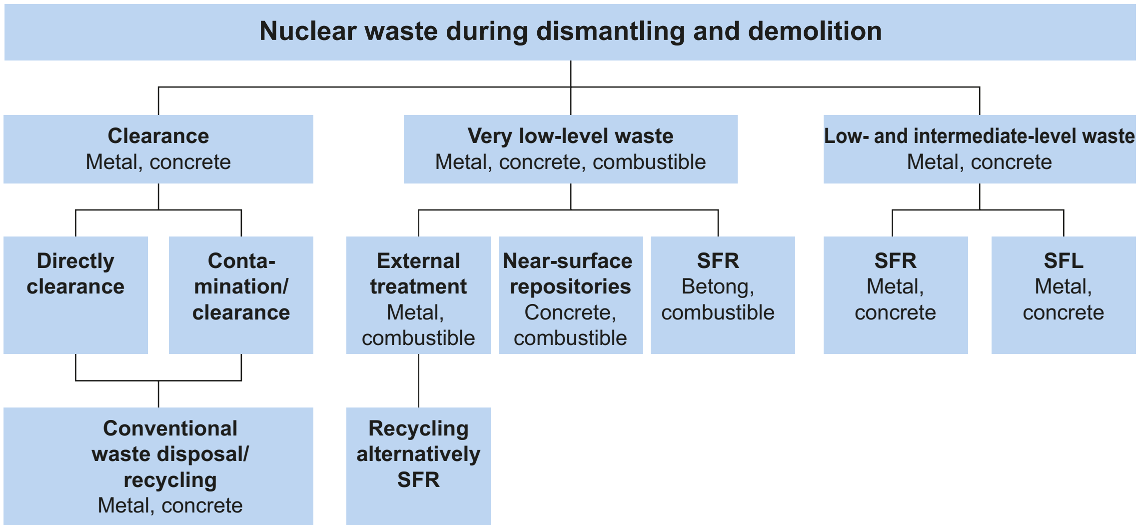

Figure 14-3. *The nuclear power plants' main principles for management routes for nuclear waste during dismantling and demolition.*

Costs of decommissioning

Under the Nuclear Activities Act, the reactor owners are obliged to cover the costs of the measures needed to manage and dispose of the nuclear waste and the spent nuclear fuel and to decommission the facilities. On behalf of the reactor owners, SKB ensures that a cost calculation, known as the Plan report, is prepared every three years, in accordance with the Financing Act – see Section 1.3. Paid-in fees are managed by the state-owned Nuclear Waste Fund.

The reactor owners estimate the cost for the decommissioning of their respective nuclear power reactors and submit the estimates to SKB. SKB compiles the estimates and produces an overall cost estimate and uncertainty analysis, which then serve as a basis for the fees and financing amounts determined by the Government.

Transport system

SKB is responsible for the transportation of spent nuclear fuel and nuclear waste from the nuclear power plants to interim storage facilities and final repositories. The transport system consists of the ship M/s Sigrid, special vehicles and different types of transport casks – see Section 2.3. The ship and the vehicles are used for transportation of low-level waste, intermediate-level waste and spent nuclear fuel. The different transport containers are specifically developed for each waste type. If new transport container need to be developed, SKB is responsible for doing this.

Nuclear waste may, if it is considered justified, be transported outside of SKB's system. This may, for example, apply to larger components and waste that is to be processed by an external party.

14.3 National and international coordination

14.3.1 Industry-wide coordination

From a national perspective, there is a need to coordinate decommissioning, both within the nuclear power companies and between the nuclear power companies and SKB and other licensees, in order to ensure that the whole chain from decommissioning planning to final disposal of the waste is carried out in an optimal manner. There are various forums that support this work, including some important international forums.

One such forum on a national level is the SKB RD&D and Plan group, which includes the reactor owners. The group coordinates SKB's work on the RD&D Programmes and the Plan reports.

To further support line managers within SKB and the reactor owners when it comes to decommissioning and waste management, a forum for waste managers has been established, in which SKB is the convener. This makes it possible to make decisions, within the mandate of the participants, to bring important issues into focus and prioritise among the work carried out by SKB and the nuclear power companies.

In conjunction with the early shutdown of the reactors in Oskarshamn and Ringhals, an increased need for cooperation and coordination has arisen with respect to decommissioning planning, not least within the corporate groups Vattenfall and Uniper, but also between the reactor owners and SKB. This cooperation includes, for example, interim storage of waste while waiting for the planned final repositories to be commissioned, and optimising the management of certain waste fractions.

SKB, the reactor owners and Business Unit Nuclear Decommissioning (BUND) in Vattenfall are also participating in international forums in the field of decommissioning, which are organised by, among others, the IAEA, OECD-NEA and the EU Commission. Collaboration takes place through, among other things, participation in the OECD's collaborative organisation for nuclear energy issues, NEA, and Eurad. Within NEA, participation takes place in the Committee on CDLM, which consists of representatives from regulatory authorities and waste organisations, and in groups under CDLM – see also Section 5.6.3. Participation in Eurad and its activities includes gaining and sharing experience in the waste and decommissioning field between countries in Europe.

The decommissioning programmes within the Vattenfall Group (Vattenfall/Ågesta and Ringhals) are members of the OECD/NEA Co-operative Programme for the Exchange of Scientific and Technical Information concerning Nuclear Installation Decommissioning Projects (CPD) and its subgroup Technical Advisory Group (TAG). The purpose of the programme is to facilitate the exchange of information and experience in connection with decommissioning of nuclear facilities. The exchange promotes the application of safe, environmentally-adapted and cost-effective methods in the decommissioning projects.

14.3.2 Coordination – Uniper

In 2020/2021, Uniper and Fortum developed a joint strategy for demolition and decommissioning of nuclear power plants. A consortium called Fortum-Uniper Nuclear Service was set up, and in the middle of 2021, was awarded the contract for the technical implementation of demolition and decommissioning at OKG Aktiebolag and Barsebäck Kraft AB.

The contract means that OKG Aktiebolag and Barsebäck Kraft AB each have a coherent approach to procurement with the consortium. The consortium, in turn, handles all of the included work packages.

The consortium is responsible for coordinating and carrying out decommissioning of the reactors that have been shut down. The work is being carried out on behalf of the licensees at OKG Aktiebolag and Barsebäck Kraft AB, who are responsible for ensuring that the activities at the facilities comply with the current licence.

Uniper's decommissioning programme covers the decommissioning of OKG Aktiebolag's and Barsebäck Kraft AB's facilities. The division of work is described in each licensee's decommissioning strategy and decommissioning plans. From these it is clear that Uniper's goal is to coordinate and carry out decommissioning of the shutdown reactors Barsebäck 1, Barsebäck 2, Oskarshamn 1 and Oskarshamn 2.

The licensees OKG Aktiebolag and Barsebäck Kraft AB have adapted the organisations to assist the consortium in the demolition and decommissioning work. The nuclear licences remain with OKG Aktiebolag and Barsebäck Kraft AB, and the responsibility for the decommissioning and the nuclear waste remains unchanged with the licence holders.

In addition to the activities related to decommissioning of the shutdown reactors, the owner company within Uniper is actively involved in several collaborative forums to support the development of the industry.

Within Uniper, the owner company, OKG Aktiebolag and Barsebäck Kraft AB meet to manage issues to facilitate the ongoing decommissioning. Outside the Group, Uniper collaborates with Vattenfall, mainly with the aim of supporting SKB in the industry's overall challenges linked, for example, to decommissioning and waste management (back-end). Finally, it should be mentioned that Uniper actively participates in forums where SKB needs the owners' support, for example in the field of waste.

14.3.3 Coordination – Vattenfall

All decommissioning within the Vattenfall Group is coordinated through BUND. This creates good prospects for synergies between the currently completed decommissioning work at the R2 reactor at the Studsvik site, the ongoing decommissioning work at the Ågesta reactor and the imminent work to decommission Ringhals 1 and Ringhals 2. Furthermore, it creates good conditions for development and coordination between the decommissioning in Germany and Sweden.

Decommissioning of Ringhals 1 and Ringhals 2 is coordinated in a decommissioning programme jointly controlled by the licensee Ringhals AB and BUND.

Within Vattenfall, intra-group collaboration is conducted within different areas. A strategic forum for waste management and decommissioning, Strata, has been formed in order to ensure that the strategies and plans for decommissioning and waste management of different organisations are coordinated and optimised. This is based on different aspects such as safety, regulatory requirements, cost-effectiveness and other external and internal stakeholder requirements.

Collaboration also takes place on an operational level in the area of waste and decommissioning, for example, with regard to release from regulatory control, waste documentation and general exchange of experience.

14.4 Skills

Decommissioning of the nuclear facilities in Sweden will continue until the mid-2070s, when decommissioning of the facilities for management and disposal of spent nuclear fuel and radioactive waste will begin. The decommissioning will not take place in a steady flow but, according to current planning, will be carried out in three main stages: one in the 2020s (current), one in the 2040s and the final one in the 2070s. One challenge is thus securing access to skills for all three decommissioning stages, since the need for decommissioning skills between stages will be limited in Sweden. See also Section 5.6.4.

Securing skills within this area will be a strategically important issue for the companies, in both the short and long term. Collaboration with universities and university colleges will form part of long-term skills management in the country, as will the continued encouragement of cooperation within the industry to both retain personnel and attract new personnel in these areas. The establishment of the Nordic Nuclear Trainee Program (NNTP) is another step towards meeting future needs.

Current situation

For the first stage, it is of key importance to take a structured approach to building skills and experience feedback in the field of decommissioning. An important aspect of this is to preserve the key information and skills obtained from early decommissioning projects by documenting them and thereby making them available for use in future stages.

Correspondingly, it is important to have a continuous exchange within and between the companies' German and Swedish activities, in order to ensure transfer of relevant experience and solutions for safe and effective decommissioning (because the decommissioning projects have different preconditions). The companies have established contacts with international suppliers with considerable experience of practical decommissioning activities. In addition, various forms of collaboration have been initiated with the aim of acquiring and building up skills.

This permits experience and skills to be gradually built up and secured within the industry for the final phase of the first decommissioning stage.

Another example is the training activities within the back-end area that the industry has established within KSU.

Programme

An important element of the work to ensure skills for future decommissioning stages is to carefully, and in a structured manner, document the practical experience gained by the companies during decommissioning, to clearly describe how organisation and control should be designed depending on the task, how a facility in operation should best be prepared for demolition, e.g. in respect of documentation, what different forms of decommissioning concepts there are, the experience that exists linked to these, and which concepts work in relation to different types of facilities and in different contexts. This can be summarised in a simplified manner by stating that after this round of decommissioning is completed, the industry will have established a blueprint for how decommissioning is best planned, controlled and implemented with respect to safety and efficiency.

In addition to this, it is critical to secure facility knowledge and formal skills, for example within radiation protection, in the long term. This requires both industry-wide collaboration and collaboration with the educational system. The national strategy of the SSM for skills supply in Sweden in the area of radiation safety is also part of this.

An example of industry-wide collaboration is the Swedish Centre for Nuclear Technology, SKC, a research centre through which the SSM, the nuclear power plants, Westinghouse Electric Sweden AB, KTH, Chalmers University of Technology and Uppsala University have been supporting education, research and development within nuclear applications at universities and university colleges in Sweden for 27 years.

Finally, it is important to note that even if the need for decommissioning skills in Sweden will be limited between decommissioning stages, decommissioning activities will continue globally on a large scale and for a long time. Hence the need for skills will be great over time, which means that the motivation to establish and preserve skills will be significant, so that that relevant skills will most likely be available internationally throughout the Swedish decommissioning period.

15 Planning for decommissioning at Uniper

Since the decision was made to decommission the shutdown reactors Barsebäck 1 (B1), Barsebäck 2 (B2), Oskarshamn 1 (O1) and Oskarshamn 2 (O2), Uniper has further developed the strategy for decommissioning. In 2020 and 2021, Uniper and Fortum established a consortium (Fortum-Uniper Nuclear Service) which halfway through 2021 was awarded the contract for the technical implementation of decommissioning at OKG Aktiebolag and Barsebäck Kraft AB.

The consortium will implement the decommissioning programme in a safe and cost-effective manner by identifying and utilising synergies and allowing the decommissioning sequence to include both facilities. This approach will provide experience that will be of benefit both in terms of continued planning and continued implementation. The technical sequence is designed to create flexibility.

The consortium intends to also carry out decommissioning and provide services to interested parties outside the Group. The activities will take place alongside the contracted decommissioning within the Group. It is Uniper's hope that the external deal will be successful and will make it possible for Oskarshamn 3 (O3) and the remaining Oskarshamn 0 (O0, shared facilities) to also receive support from the consortium.

If O3 and the remaining O0 are decommissioned later than currently planned, or if the consortium ceases to exist before the time for decommissioning, Uniper will work to ensure that experience from the previous demolition is preserved and can be reused when decommissioning begins.

15.1 Barsebäck Kraft AB's planning for decommissioning

The Barsebäck nuclear power plant is situated on the Barsebäck peninsula in Kävlinge municipality, about 20 kilometres north of Malmö. It is owned and managed by Barsebäck Kraft AB. The Barsebäck plant generated electricity between 1975 and 2005. Reactor B1 has been permanently shut down since 1999 and reactor B2 since 2005 – see Figure 15-1.

Overall planning

Barsebäck Kraft AB began the dismantling and demolition of B1 and B2 in 2020. Before work began, preparatory measures had been carried out in accordance with the current licence conditions. The necessary licences were obtained in 2019. During the execution of dismantling and demolition, the different work packages/steps will be reported step by step in accordance with the current licence conditions.

The completed and continued planning for the decommissioning of B1 and B2 is based on using proven technology and proven methods. Technical, safety-related and organisational dependencies have been identified and, following this, a sequence for work packages/steps has been established. Based on this sequence, detailed planning and procurement for each work package/step will be carried out successively. The purpose of the working method is to reduce technical and safety-related risks. During decommissioning, Barsebäck Kraft AB acts as a purchasing organisation.

The overall time plan for decommissioning of the Barsebäck nuclear power plant is presented in Figure 15-2.

Decontamination and release of buildings from regulatory control is planned to be completed in 2028. Conventional demolition and restoration of land will take place in conjunction with the transport of the radioactive waste to the SFR.

Figure 15-1. Barsebäck nuclear power plant viewed from the south with interim storage facilities in the foreground, Barsebäck 1 and Barsebäck 2 in the background. Photograph: Apelöga.

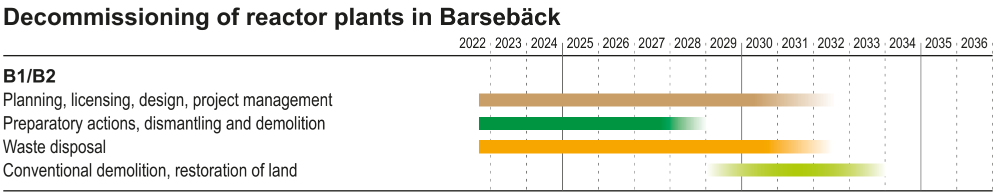

Figure 15-2. Schematic overview of Barsebäck Kraft AB's time plan for decommissioning.

Waste management

Materials and waste produced during decommissioning are of the same type as the waste produced during operation, with the difference that the waste volumes are larger during decommissioning. Handling lines for radiological waste and waste suitable for release from regulatory control have been established. Materials and waste logistics are designed on the basis of the demolition sequence and the radiological survey. There is redundancy in the handling lines in order to reduce the risk of disturbances in the waste logistics – see Figure 14-3. The choice of management route is based on an overall assessment of factors, including cost, risk, environment and the ALARA principle.

Since decommissioning of B1 and B2 will be carried out before the extended SFR is in operation, Barsebäck Kraft AB has built an interim storage facility for the low-level radioactive waste. For the long-lived radioactive waste in interim storage in the existing storage facility pending disposal in the SFL, alternatives for interim storage off-site are being investigated.

Barsebäck Kraft AB does not intend to establish a near-surface repository, since the nuclear activities will cease and the site will be released from regulatory control in its entirety. The possibility of using OKG Aktiebolag's near-surface repository is planned as an alternative to disposing of waste in the extended SFR.

Estimated waste quantities from decommissioning have been presented in the decommissioning plan and waste management plan, and a more detailed reporting is available in each step report. The waste quantities presented per waste type and activity category are based on results from the radiological survey, which will also be updated continuously during demolition.

15.2 OKG Aktiebolag's planning for decommissioning

OKG Aktiebolag owns the nuclear power reactors O1, O2 and O3, as well as a number of shared service facilities called O0. The reactor site is located on the Simpevarp Peninsula on the Baltic coast, about 30 km northeast of Oskarshamn. All three reactors are boiling water reactors (BWR) – see Figure 15-3.

The licence to build O1 was granted in 1966. The facility was connected to the power grid for the first time in 1971 and inaugurated in 1972. In 1969, a permit was granted to build O2, which was commissioned in 1974 and O3 was commissioned in 1985. O1 and O2 are situated adjacent to each other, while O3 is situated slightly further north. O1 was finally shut down in the summer of 2017. Following completion of an extensive modernisation programme at O2, a decision was taken not to restart the facility, and the final shutdown was confirmed for the end of 2015. The O3 reactor is in operation. Decommissioning of OKG Aktiebolag's nuclear facilities is described in the current decommissioning strategy and the decommissioning plans for each facility.

Overall planning

The completed and continued planning for decommissioning of O1 and O2 is managed as large projects, where standardised planning processes based on Lean principles are used (Lean – a methodology for maximising productivity and minimising wastage). Technical, safety-related and organisational dependencies have been identified, following which a sequence for the execution of the different work packages has been determined. Detailed plans for dismantling and demolition are then gradually developed for each work package/step. The purpose of the working method is to reduce technical and safety-related risks. During decommissioning, OKG Aktiebolag intends to act as a purchasing organisation.

Figure 15-3. *View of OKG Aktiebolag's nuclear power plant with the three BWR reactors O1, O2 and O3 from left to right in the picture.*

O3 is planned to remain in operation until 2045. After this, dismantling and demolition are planned to begin in parallel with emptying the facility of fuel. At the time for decommissioning of O3, an industrialisation of the dismantling and demolition work is expected to have taken place, since several decommissioning projects will have been completed by then. This, together with experience from the decommissioning of O1 and O2, provides the conditions for optimised decommissioning of O3. Decommissioning planning for O3 and O0 will continue during ongoing operation in accordance with regular procedures, where experience from decommissioning of O1 and O2, as well as the shared waste facility (0AVF) will be utilised.

Regarding decommissioning of the shared facilities belonging to O0, the plan is that 0AVF will be decommissioned in conjunction with the decommissioning of O1 and O2. Other facilities within O0 will be decommissioned in connection with the decommissioning of O3.

The first stage, 2046, concerns facilities that are not required for the decommissioning of O3. The next stage concerns the remaining facilities, and demolition of these will take place in connection with the demolition of O3. Remediation of all sites will take place between 2050 and 2055, in such a manner and to such an extent that other industrial activities can be established at the sites. Conventional demolition, including the demolition of the reactor buildings for O1 and O2, and restoration of land will take place in connection with the demolition of O3 in 2054–2055 and shared facilities in 2050–2055. Dismantling and demolition of the shared facilities are planned in two stages.

The objective is to remove radioactive material and release the facility from regulatory control.

Figure 15-4 presents the overall time plan for decommissioning of the Oskarshamn nuclear power plant.

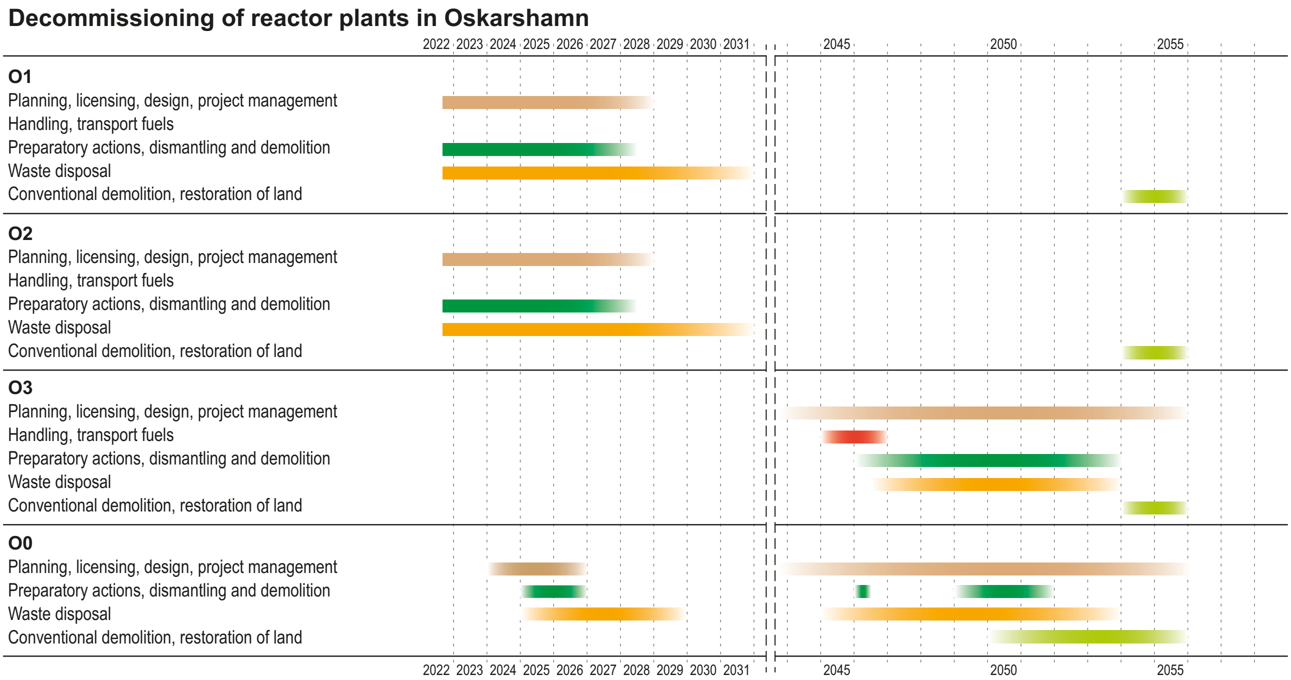

***Figure 15-4.** Schematic overview of OKG Aktiebolag's timeplan for decommissioning (O0 are joint facilities that are reported separately).*

Waste management

O1 and O2 are currently in a dismantling and demolition state. All necessary permits to start dismantling and demolition have been obtained for both reactors. During implementation, the various work packages/steps are reported on an ongoing basis in accordance with the applicable licence conditions.

The large-scale dismantling began in 2020. Prior to that, preparatory measures for example emptying of fuel and segmentation of core components had been carried out,. A radiological survey of the facilities as well as a 3D scan has been completed. Building components and materials from the facilities have been divided into different routes for material and waste management.

Materials and waste produced during decommissioning are of the same type as the waste produced during operation, with the difference that the waste volumes are larger during decommissioning. This means that material and waste management needs to be adapted, but that tried and tested technology and methods can be used. The larger material and waste quantities during decommissioning require well-functioning material and waste logistics. Waste logistics will be built up gradually based on the flow of material/waste per time unit, which can be obtained by linking the demolition sequence to the radiological survey.

The Final Repository for Short-lived Radioactive Waste (SFR) and the Final Repository for Long-lived Waste (SFL) are not available during the decommissioning of O1 and O2. Material and waste management have to be optimised from an overall perspective, where possible and reasonable management routes, such as geological final repositories, release from regulatory control, treatment off-site and near-surface repositories at the nuclear sites, are weighed against each other. This means that redundancy in management routes is desirable where possible, and that the preferred management route is based on an overall assessment of factors such as cost, risk, environment, ALARA principle, etc.

During the decommissioning of O1 and O2 (including 0AVF), the waste that will be disposed of in the extended SFR or the SFL must be kept in interim storage on site, until the final repositories are in operation. This means that the capacity of the existing interim storage facility LLA (storage building for low-level waste) must be increased to accommodate the low-level waste that will be transported to the SFR. However, it is expected that interim storage of intermediate-level waste that will be transported to the SFR and long-lived waste destined for the SFL can be accommodated in the existing interim storage facility BFA.

The existing near-surface repository for final disposal of very low-level waste needs to be extended to receive the very low-level waste from the decommissioning of O1 and O2. An extension is also required in order to accommodate the waste from the remaining operation of Oskarshamn 3. The planned extension of the near-surface repository will also be adapted for waste from decommissioning of Barsebäck Kraft AB's facilities.

The design capacity of both interim storage facilities and near-surface repositories is governed by need based on optimal distribution of the different management routes on the basis of specified factors. The extensions are managed through separate licensing processes.

Estimated waste quantities from decommissioning are based on the quantities from previously completed decommissioning studies reported in the decommissioning plans for each facility. For the facilities undergoing dismantling and demolition, waste quantities are also presented in the current waste management plan and more specifically in each step report. For O1 and O2, more detailed information is available on waste quantities per waste type and activity category, based on results from the radiological survey of the facilities that is being conducted and will continue during the decommissioning projects.

Decontamination and release from regulatory control of buildings belonging to O1 and O2 will be carried out alongside dismantling and demolition. Release of the buildings from regulatory control is planned to be completed in 2028.

16 Planning for decommissioning at Vattenfall

This chapter provides an overview of Vattenfall's decommissioning planning. The plans apply to both the Ågesta reactor and the reactors in Forsmark and Ringhals.

During the shutdown operation, in addition to fuel transport, shutdown of systems and management of operational waste, preparatory activities for dismantling and demolition are also planned. These preparatory activities, for example decontamination of primary systems, are intended to make subsequent work as safe and efficient as possible. This is achieved by, for example, reducing the dose rate for workers and others present in the facility, and by improving the possibilities for flexibility during dismantling by moving the steps from time-critical paths (steps that affect the project's end date).

The organisation of the decommissioning programme includes the task of planning and implementing the constituent projects, in which contractors are used as the main workforce during the implementation phase. The aim is to keep the programme organisation small and efficient.

In order to effectively implement the decommissioning programmes, conditions and scope must be well defined in advance. The time for dismantling and demolition will be optimised, the aim being that decommissioning will begin immediately after shutdown and will take place in a continuous implementation phase. This means that the time for service operation will be minimised and that dismantling and demolition will continue until the end state determined for the facility is achieved.

Implementation of dismantling and demolition will be divided into a number of suitably delineated work packages and stages. In order to achieve efficiency, the activities are planned to be carried out, wherever possible, simultaneously throughout the entire facility. Planning of the execution and chronology of work steps will be optimised throughout decommissioning on the basis of ALARA and BAT in order to minimise dose and maximise efficiency.

Vattenfall's strategy for the reactor pressure vessels from the BWR facilities is to segment them on site at the facility. Segmentation of reactor internals takes place covered by water in the handling pool. Segmentation may take place mechanically (e.g. sawing, cutting, water jet cutting) or thermally (e.g. plasma cutting, laser cutting, electrical discharge machining). For R2, which is a PWR reactor, there are decisions on segmentation of both the pressure vessel and reactor internals. The concentrations of long-lived radionuclides in the reactor internals and the R2 pressure vessel require the waste to be disposed of in the SFL. While awaiting commissioning of the SFL, the waste must be kept in interim storage at Ringhals, and this handling is facilitated by the waste being segmented.

A crucial difference between decommissioning and operation of a facility is the considerably larger amount of waste that is produced during decommissioning. This means that the capacity for managing certain waste streams must increase substantially during decommissioning, so as not to hinder progress in the project. The increase in capacity can be achieved in several different ways, for example by adapting existing buildings or premises that will not be needed after shutdown for waste management. New buildings may also have to be constructed or mobile solutions introduced at the nuclear power plant to meet the needs for capacity.

In general, waste management must be well-founded and effective. This is achieved by classifying the waste before it is produced and sort it immediately as it is produced. Processing of nuclear waste must be minimised and as far as possible carried out as it is generated. Generation of secondary waste should be avoided as far as possible. Large components should be disposed of as whole where this is justified from a cost and safety perspective. Time-consuming waste processing is only carried out if benefits can be ensured.

Waste management, including interim storage and final disposal, is to be optimised from a cohesive operational and decommissioning perspective at group level. The aim is to increase the capacity for management of the waste where it is most appropriate from an overall perspective, for example, a near-surface repository for very low-level waste.

The progress of the decommissioning programmes should, to a reasonable extent, be independent of the capacities for waste treatment, waste transport and completed final repositories. For transportation, SKB's transport system will primarily be used, but the programmes are not exclusively bound by this. Coordination of fuel transport, in particular, should take place at a cooperate group level, with the goal of minimising the impact on the decommissioning programmes.

The conventional waste stream is estimated to account for approximately 95 per cent of the total waste volume, with the majority comprising demolition waste from buildings. In order to minimise the requirements for post-handling, buildings are decontaminated prior to demolition. For the demolition waste that is produced as a result of system dismantling, a procedure for release from regulatory control is established in line with the existing manual for release from regulatory control (Berglund et al. 2016).

The end state of decommissioning is an industrial site released from regulatory control. The end state for the different decommissioning programmes may, however, include an industrial site appropriate for release from regulatory control where buildings and infrastructure that are of use for continued activities, and which can be released from regulatory control without being demolished, are left standing, while other installations are demolished. Conventional demolition of buildings is carried out to about one metre below ground and remaining cavities are backfilled with demolition material. The uppermost ground layer is restored to the state that continued industrial activities on the site require.

16.1 Ringhals AB's planning for decommissioning

Ringhals nuclear power plant is situated on the Värö peninsula in the municipality of Varberg, in the county of Halland. The plant has four reactors, of which Ringhals 1 (R1) is of reactor type BWR and Ringhals 2 (R2), Ringhals 3 (R3) and Ringhals 4 (R4) are of reactor type PWR. The area occupies a total of 2.5 square kilometres and contains, in addition to R1–R4, communal buildings and facilities for waste management, offices, workshops, storage buildings, access roads etc – see Figure 16-1.

Figure 16-1. *Ringhals nuclear power plant, with reactor R1 and R2 above reactor R3 and R4 in the figure. The centre of the site contains office premises and a canteen, etc. Just to the left of R1 in the figure is the plant's waste area, which includes handling, conditioning and storage of nuclear waste.*

Reactors R1 and R2 were built in the 1970s and located in a joint operations area. In the 1980s, R3 and R4 were built in an operations area originally separated from R1 and R2. The operations areas were later connected via a transport route. The reactor locations make it easier to once more separate the reactor pairs and enable parallel operation and decommissioning within the different operations areas after final shutdown of R1 and R2.

A number of additional reactors were originally planned, which means that the area is large. This enables efficient logistics, since there is room, for example, for temporary storage of waste and the possibility of different transport routes within the area. In terms of waste logistics, decommissioning of R1 and R2 will also be facilitated by the existing waste facility being situated adjacent to R1.

Videberg harbour is situated close the site and is used for transportation of fuel and waste.

Overall planning

The decommissioning of the Ringhals reactors is described in two decommissioning plans, one for R1 and R2 and one for R3 and R4. The plans are based on the overall strategy and goals presented in the introduction to the chapter. Operation of reactors R1 and R2 continued until the end of 2020 and 2019 respectively. The shutdown is based on decisions taken in 2015 that both reactors would be taken out of service earlier than originally planned. For reactors R3 and R4, the current planning for operation until 2041 and 2043 respectively remains in place. Shutdown operation took 16 months at R1 and 28 months at R2, considering cooling requirements and capacity for fuel transport. In June 2022, both R1 and R2 transitioned to service operation. The planned start of dismantling and demolition is after summer 2023.

In December 2015, work began on planning the dismantling and demolition of R1 and R2, now compiled in a programme. The focus so far has been on analysing the technical, legal, temporal and commercial preconditions for decommissioning and assessing in detail how the specific steps during decommissioning may be best performed. This includes analysis of demolition methodology, organisation and waste management, for example. The decommissioning plan describes the activities that are analysed or carried out during the different programme phases.

In 2020, Ringhals AB obtained a permit for modification under the Swedish Environmental Code for the reactors R1 and R2 to be allowed to switch from shutdown operation and service operation to complete dismantling and demolition. In conjunction with the decommissioning decision, project STURE – Safe and Secure Phasing Out of R1 and R2, was started with the task of identifying, analysing and implementing measures that are required for R3 and R4 to be able to continue operation when R1 and R2 are decommissioned and to coordinate the portfolio of preparatory measures prior to dismantling and demolition. Project STURE will be completed in 2022 and will be transferred to the Ringhals decommissioning department when the preparatory phase has been completed. The project will, among other things, ensure that the pair of reactors that will remain in regular operation is physically separated from the pair that will be decommissioned (Project Split), organise the transportation of spent fuel from the final cores, drain systems and transport operational waste from the plant, and ensure that the preparatory activities required prior to dismantling and demolition are carried out. System decontamination, with the aim of reducing the dose rate in the facility, has been carried out.

A new joint programme has been created that includes all decommissioning activities, both the preparatory activities, which were previously under project STURE, and the activities for the implementation of decommissioning, which were previously part of the R12D programme.

The overall time plan for decommissioning of the Ringhals nuclear power plant is presented in Figure 16-2.

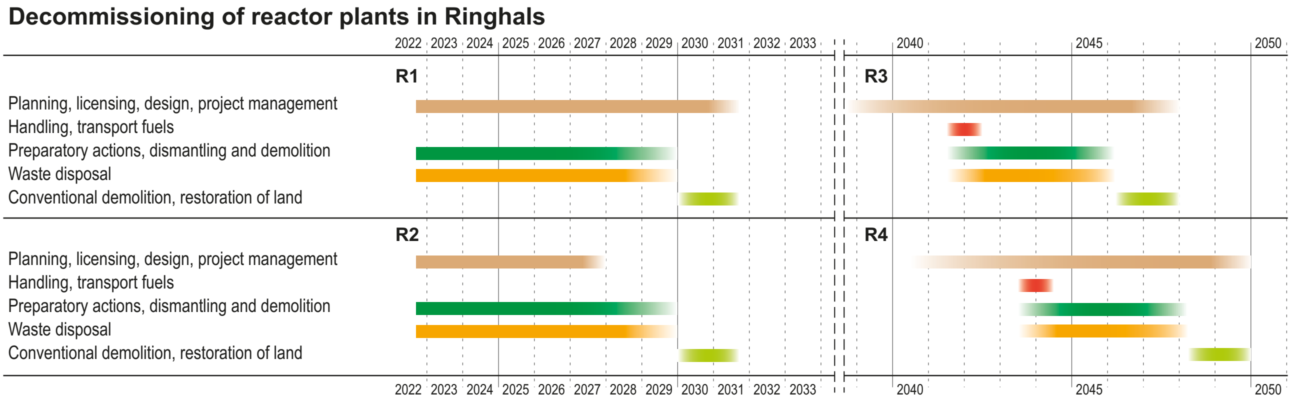

__Figure 16-2.__ Schematic overview of Ringhals AB's time plan for decommissioning.

Waste management

Since the final shutdown of R1 and R2 will occur at a time when the extension of the Final Repository for Short-lived Radioactive Waste (SFR) is not yet in operation, interim storage is required for the radioactive decommissioning waste that is produced. This can take place at Ringhals nuclear power plant and/or at another site. Interim storage at the nuclear power plant would minimise the external dependencies linked to waste management, which means that this option is the preferred choice for most waste streams. Interim storage at the nuclear power plant will take place in rebuilt existing storage facilities for intermediate-level waste and, from 2023, in a new storage facility for low-level waste in the existing waste area.

Reactor pressure vessel segments from R1 can, thanks to their relatively low activity content, be disposed of in the extended SFR. The most neutron-activated internals are segmented and packaged for interim storage prior to future disposal in the Final Repository for Long-lived Waste (SFL). Reactor internals that were some distance from the core region, such as the steam dryer and steam separator, are planned to be segmented and disposed of in the extended SFR.

The reactor pressure vessels and internals of R2, R3 and R4, because of their larger activation, need to be disposed of in SFL. For R2, there is a policy decision on segmentation of the reactor pressure vessel and internals; for R3 and R4 final management has not yet been decided. Interim storage takes place at Ringhals until the extension of SFR is completed and SFL is commissioned.

Estimated waste quantities from decommissioning have been presented in the decommissioning plan and waste management plan, and will be presented in greater detail in each step report. Reported waste quantities per waste type and activity category are based on results from the radiological survey, which will be updated continuously during demolition.

16.2 Forsmarks Kraftgrupp AB's planning for decommissioning

The Forsmark power plant is located on the east coast of Sweden, around four kilometres north of Forsmark village, in the municipality of Östhammar in the county of Uppsala. There are three nuclear power reactors within the facility, Forsmark 1 (F1), Forsmark 2 (F2) and Forsmark 3 (F3) – see Figure 16-3. The power plant also includes buildings for temporary accommodation, storage buildings, workshops and administration. There is also a harbour which is used, for example, by ships transporting spent nuclear fuel and radioactive waste to SKB's facilities.

F1 and F2 are integrated facilities while F3 is situated separately to the northwest of these. Shared facilities such as access roads, harbour, water and sewage treatment plants, water tower and administrative buildings are utilised by all three reactors and by SFR. The size of the area ensures good conditions for parallel operation and decommissioning. Large areas are also available for temporary storage and for the establishment of different transport options.

Figure 16-3. *Forsmark power plant with the three BWR reactors F1, F2 and F3 from left to right in the picture.*

Overall planning

Forsmarks Kraftgrupp AB's current plans are 60 years of operation for all three reactors, which entails final shutdown for F1 in 2040, F2 in 2041 and F3 in 2045. At final shutdown, a period of shutdown operation begins, the length of which must be kept as short as possible. At present, it is estimated that it will be about 12 months.

Decommissioning of F1, F2 and F3 is described in the decommissioning plan, and is based on the overall strategy and the goals presented in the introduction to the chapter. F1 and F2 are expected to be dismantled and demolished in a way that maximises synergy gains and minimises the need for facility separation or service operation. At the end of the decommissioning projects for F1 and F2, shutdown operation will start at F3, which means that decommissioning is expected to continue in the area without interruption from the start of the first project until the last reactor is finally dismantled. Figure 16-4 presents the overall time plan for decommissioning of the Forsmark nuclear power plant.

The basic plan assumes that SFR will be in operation at the time of dismantling and demolition of the Forsmark facilities, which means that the need for interim storage can be limited to the long-lived waste to be disposed of in SFL. The short-lived low- and intermediate-level waste is sent directly to SFR after packaging.

Waste management

When dismantling and demolition starts in 2040, construction of the extension of SFR is expected to have been completed, and the facility is in operation. SFL will not be ready to receive waste, which means that interim storage of long-lived waste will take place, but final conditioning will take place later.

The shutdown will result in multiple waste streams. Fuel will be transported to Clink and then to the Spent Fuel Repository other waste will be sorted into short-lived waste and long-lived waste (to the SFL). The short-lived waste that cannot be released from regulatory control is sorted with respect to activity content and disposed of in near-surface repositories or in SFR.

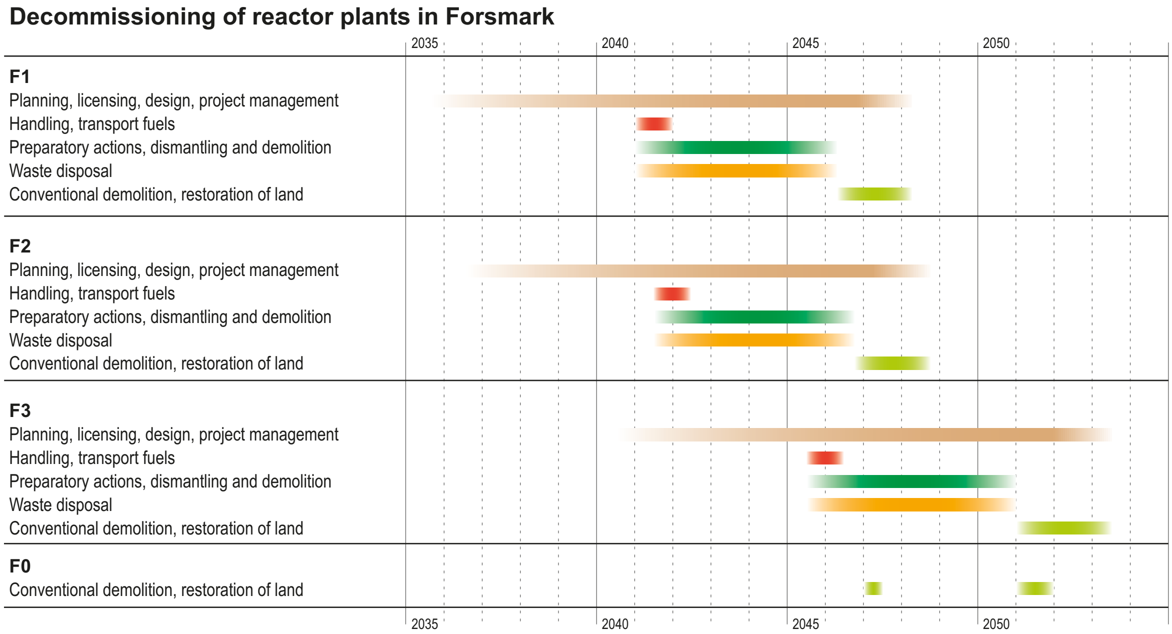

Figure 16-4. *Schematic overview of Forsmarks Kraftgrupp AB's time plan for decommissioning. F0 consists of shared facilities.*

The reactor pressure vessels from F1–F3, together with their internals, will be segmented; some reactor internals, such as the steam dryer, steam separator and reactor pressure vessel will be disposed of in the SFR, and some internals, such as the core grid, core instrumentation, parts of the core shroud, core shroud head and fuel assembly plate will be disposed of in the SFL.

Estimated waste quantities from decommissioning have been presented in the decommissioning plan and waste management plan, and will be presented in greater detail in each step report. Reported waste quantities per waste type and activity category are based on results from the radiological survey, which will be updated continuously during demolition.

16.3 Vattenfall's planning for decommissioning of the Ågesta reactor

The Ågesta plant, which is located about 20 kilometres south of Stockholm, in Huddinge municipality, in the county of Stockholm, was the first nuclear power facility in Sweden used for commercial production of electricity. The Ågesta reactor was a heavy water moderated PWR reactor of 80 MW which provided Farsta with district heating, and the electrical power grid with 10–12 MW of electricity.

The reactor, and a number of other important facility parts, are situated in a rock cavern – see Figure 16-5. The rock cavern, together with a steel lining, acted as the containment. The reactor pressure vessel and two remaining steam generators are situated inside the containment. Inside the rock cavern, but outside the steel lining, are the control room, the control and switchgear building, and the transport tunnel and emergency exit.

Because the Ågesta facility is situated in a rock cavern, there is limited space and opportunity for handling waste and keeping it in interim storage on site. Ågesta's location means that all transport must take place by road. The location near a densely populated area also means that any road transport shipments that need to be carried out will affect nearby residents and facilities to some extent.

Overall planning

Decommissioning of the Ågesta facility is being carried out in the form of a programme that consists of several projects and assignments. The programme is governed by BUND.

Vattenfall's ultimate goal is to remove all radioactive material so that the facility, i.e. the rock cavern, turbine hall, surface facilities and underground facilities can be released from regulatory control and the nuclear licence can be revoked.

In July 2019, Vattenfall obtained a permit to commence dismantling and demolition of the Ågesta facility in accordance with the Swedish Environmental Code, the Nuclear Activities Act and the Radiation Protection Act. In order to meet the requirements for implementation of safe decommissioning, the facility has been upgraded in certain parts. The decommissioning work began in June 2020. Figure 16-6 shows the time plan for the remaining part of the decommissioning of the Ågesta reactor.

The work is divided into two major steps: segmentation of the reactor pressure vessel on site and dismantling of other contaminated systems, components and structures. The work to be carried out in the two steps will to some extent be carried out simultaneously, and is based on joint planning of, for example, logistics, staffing and radiation exposure. The parallel implementation means that skills and resources are utilised rationally, as the personnel available at the facility have the opportunity to contribute to several different tasks. This results in efficient progress, and lessons learned are shared between the various projects. The steps include removal of insulation and remediation of other environmentally hazardous waste, packaging of waste in standard waste containers and scanning and transport to AB SVAFO for interim storage or for further handling by an external party.

After completion of the Bulk project (dismantling of other contaminated systems, components and structures), the rock cavern is judged to be in a condition that meets the requirements for release from regulatory control. An application for release from regulatory control is planned to be submitted to SSM in 2024, with the goal of obtaining a decision at the turn of the year 2024/2025. After a decision on release from regulatory control has been received, all shafts will be sealed with concrete plugs and all connections will be cut and sealed. The facility will then be handed over to the City of Stockholm, which is the property owner.

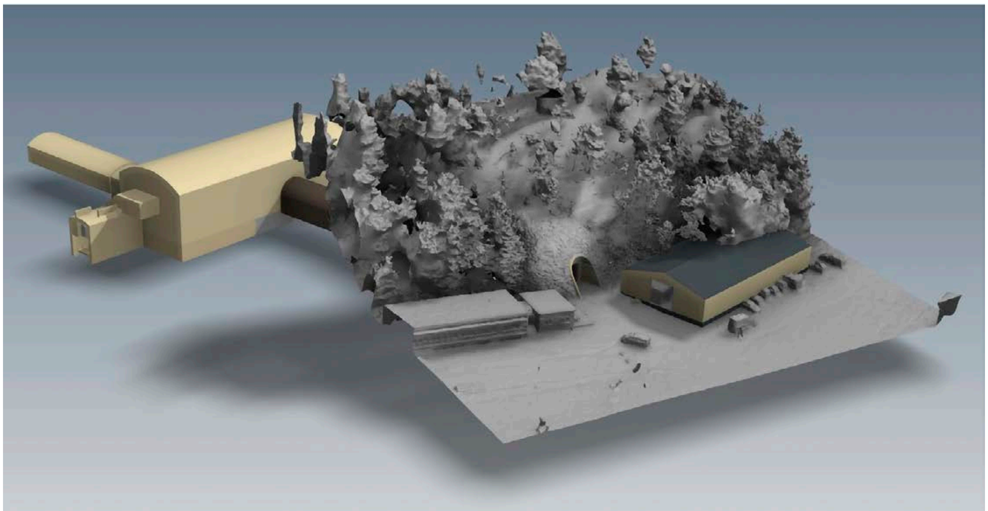

__Figure 16-5.__ 3D view of the Ågesta reactor, which is located in a rock cavern.

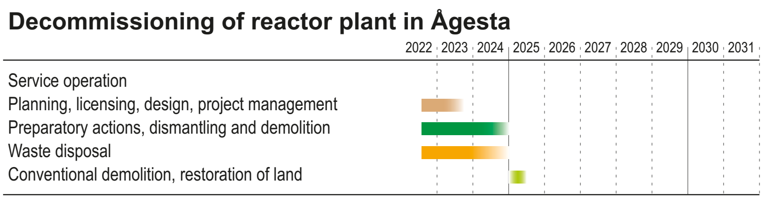

__Figure 16-6.__ Schematic overview of the time plan for the planned decommissioning of the Ågesta reactor.

Waste management

The different waste streams generated in conjunction with the dismantling and demolition of the Ågesta reactor will be managed according to predetermined handling steps. Handling is tailored to result in either release of the material from regulatory control or creation of waste packages suitable for disposal in any of the final repositories within SKB's repository system.

Prior to decommissioning, a smaller waste logistics area was prepared adjacent to the rock facility. This area can be used for temporary storage of produces waste packages with low dose rates awaiting transport from the facility. This ensures that work inside the rock facility can continue without the risk of logistical bottlenecks occurring inside the narrow facility. Waste is transported from the rock cavern to interim storage facilities at the Studsvik site pending commissioning of the SFL and the extended SFR.

17 Planning for decommissioning of SKB's facilities

SKB's facilities are among the last nuclear facilities to be decommissioned in Sweden, and decommissioning will take place around 50 years from now. Decommissioning planning can therefore only be described in general terms and with references to existing decommissioning plans for more detailed information.

17.1 Central facility for interim storage and encapsulation of spent nuclear fuel – Clink

SKB is the licensee for Clab and will remain the licensee when the integration of the planned encapsulation part is finished and the facility has switched to being known as Clink. The decommissioning plan for Clink was updated in 2014, in conjunction with the compilation of supplementary material for the application for Clink. Another update will be made in conjunction with the development of the PSAR prior to construction of the encapsulation part. Decommissioning of Clink will begin after all spent nuclear fuel has been encapsulated and disposed of in the Spent Fuel Repository. The time plan depends on when the last nuclear reactor is taken out of service, but once decommissioning has begun, it is planned to be completed within five to seven years.

The goal of decommissioning is to remove all radioactive material and release the facility from regulatory control. This means that all buildings, including equipment and land, will be granted release from regulatory control to allow subsequent operations at the site.

In 2013, SKB conducted a study for decommissioning of Clink in order to provide waste inventory data as a basis for the extension of the repository for short-lived radioactive waste (SFR) and a cost estimate (Edelborg et al. 2014) for the Plan report. According to current plans, radioactive waste from dismantling and demolition will be sent to SFR for final disposal.

A decommissioning plan has been prepared for the existing Clab facility, and this is currently the applicable plan. It was updated in 2020 with clarification of how dismantling and demolition are planned for the parts of the facility located underground.

After SKB has received an approved PSAR for Clink and an updated decommissioning plan for the facility has been notified to SSM in accordance with Chapter 9, Section 1 of regulation SSMFS 2008:1, the decommissioning plan for Clink can replace the existing decommissioning plan for Clab.

17.2 Final Repository for Short-lived Radioactive Waste (SFR)

SKB is the licensee for SFR, and the decommissioning plan for the existing facility was updated in 2018 when SKB presented the periodic overall assessment of the facility's safety and radiation protection to the SSM.

During the RD&D period, an update of the decommissioning plan for the extended SFR has been prepared. The update has been carried out alongside the work on the PSAR for the facility.

Decommissioning of the SFR will begin when the main operations cease and are not intended to be resumed. Decommissioning will continue until the facility above ground has been released from regulatory control and there are no radiological reasons to prevent the establishment of another industrial activity on the site. Waste from the parts of the facility that may be demolished in connection with decommissioning (the surface facilities) is regarded as conventional waste, since it does not contain any radioactive material. A radiological survey of the facility will need to be carried out in order to rule out possible contamination of building parts that have been in contact with waste containers during operation, for example the terminal building. The goal of decommissioning is a facility released from regulatory control. How far demolition should be carried out thereafter depends mainly on future use of the site area.

The time plan for decommissioning of the SFR is linked to when the last of the current nuclear power plants and SKB's other nuclear facilities have been dismantled and the radioactive waste has been disposed of. The decommissioning of the SFR can then be initiated and is planned to be concluded within five years.

After SKB has obtained an approved PSAR for the extended SFR and an updated decommissioning plan for the facility has been notified to the SSM in accordance with Chapter 9, Section 1 of regulation SSMFS 2008:1, the decommissioning plan for the extended SFR can replace the decommissioning plan for the existing SFR.

17.3 Final Repository for Long-lived Waste (SFL)

No decommissioning plan has yet been prepared for SFL, since the design of this final repository is in the conceptual stage. A decommissioning plan will be prepared in conjunction with the preparatory PSAR (F-PSAR) for the facility. Decommissioning will begin in conjunction with the sealing of the repository, which can only take place when the long-lived waste from the nuclear power plants and from other nuclear facilities has been managed and disposed of.

17.4 Spent Fuel Repository

The current decommissioning plan for the Spent Fuel Repository was prepared in 2017. This was an update to harmonise the plan with current regulations and to follow the industry-wide structure for a decommissioning plan.

Decommissioning of the Spent Fuel Repository will begin after the main operation is concluded, i.e. when all spent nuclear fuel has been disposed of and the deposition tunnels have been backfilled and plugs installed. Decommissioning entails sealing of the remaining parts of the underground facility and demolition of the surface facility. Sealing of the underground facilities is part of the repository's barrier functions and is of importance for post-closure safety.

When decommissioning starts there will be no contamination in the facility above ground. Demolition is therefore carried out as for a conventional facility, and the waste is sorted and recycled as far as possible, or disposed of in landfill. Hazardous waste is managed in compliance with applicable regulations. Thereafter, a ground survey is carried out to serve as a basis for site remediation.

18 Continued activities within decommissioning

This chapter provides an overview of the completed and planned development work related to decommissioning of nuclear facilities. Section 18.1 presents an overview of industry-wide development work. Sections 18.2 and 18.3 present development work at Uniper and Vattenfall.

18.1 Industry-wide development work

The industry-wide development work related to decommissioning of nuclear facilities is conducted to a large extent by SKB together with the nuclear power companies. Most of the development work being done within SKB is linked not only to decommissioning, but to waste management in general. The report on completed and planned development activities is therefore presented in Part II and covers areas such as:

- reference inventory,
- waste casks, waste transport casks,
- acceptance criteria for waste in SFL and the extended SFR.

Examples of activities that have been completed and work that is planned during the RD&D period with bearing on decommissioning are given below.

18.1.1 Non-regular fuels

A prerequisite for being able to start dismantling and demolition is that the facility is free from spent nuclear fuel, including failed fuel that may be present in the facility. In 2015, a project was initiated with the aim of developing a method for managing failed fuel in Forsmark, Ringhals and Oskarshamn. The method involves encapsulation of the failed fuel in specially designed containers. There are two types of containers for encapsulation of failed fuel, called transport boxes and Quivers. Filled containers can be transported to Clab for interim storage. In Clink, they will be placed in copper canisters, sealed and transported to and disposed of in the Spent Fuel Repository. During the previous RD&D period, the project was concluded, and all failed fuel has been managed.

With the introduction of Quivers, SKB has established a practical and safe method that is cost-effective in the long term for managing failed fuel rods at the nuclear power plants.

18.1.2 Harmonised licensing

A fundamental prerequisite for initiating the planned decommissioning projects is that the required permits are granted. Several decommissioning projects are being initiated and are planned to be executed in parallel, and an efficient approach that would increase the possibilities of reaching set milestones according to the established time plan would be if all licensees and the concerned bodies were to have a uniform process.

A review of the entire code of statutes of the SSM has been under way for a number of years. After the review, the code of statutes will be structured in three levels under the current Nuclear Activities Act and Radiation Protection Act. During the previous RD&D period, industry-wide referral work has been carried out via the nuclear power industry's safety coordination group KSKG of the regulations that concern the design, analysis and operation of nuclear power reactors, as well as nuclear material and nuclear waste. There is now a shared interpretation document for the latter.

A shared interpretation of regulations is one element contributing to the formation of uniform processes in the nuclear power industry, and the joint referral procedure will continue throughout the review of the SSM's code of statutes, which is planned to be carried out in the next few years.

18.1.3 International development work

During the previous RD&D period, SKB and the reactor owners followed and participated in the international development work being pursued within decommissioning and technology for dismantling and demolition.

The main exchange takes place within the OECD/NEA collaboration programme but the IAEA's programme is also of importance. The latter focuses to a greater extent on the development of the IAEA's Safety Standards, which form the basis for member countries' sets of requirements.

Participation within OECD/NEA takes place via CDLM. During 2019 and 2020, CDLM evolved to include new working groups and expert groups that will drive the work in several areas of relevance for decommissioning projects, including costs, organisation, safety and environmental aspects.

The exchange of experience and knowledge concerning decommissioning of nuclear facilities has been taking place for a long time between the Swedish nuclear power industry and Enresa in Spain. However, the collaboration was suspended in 2020–2021, due to the travel restrictions imposed by the Covid-19 pandemic.

Programme

SKB and the reactor owners will continue to participate in the international networks within decommissioning, which are both useful and provide opportunities to contribute experience. There will be a need to exchange more in-depth knowledge and experience as the planned decommissioning projects in Sweden begin. More experience will then be built up within the country and the need to gather information will increase.

The collaboration and exchange of knowledge between Enresa of Spain and the Swedish nuclear power industry is planned to be resumed and continue during the RD&D period.

The reactor owners also conduct their own bilateral collaborations and exchanges of experience internationally, and are also engaged in various conferences and workshops.

18.2 Development within Uniper

The Project Performance Center (PPC) was established in 2017 to coordinate Uniper's decommissioning programme. PPC has focused on different scenarios for decommissioning regarding the timing of implementation and how these scenarios relate to technical sequence, organisation, waste management and financing.

At the end of 2018, a strategic decision was made to carry out the decommissioning programme for B1/B2 and O1/O2. The programme follows the critical path that runs sequentially between the facilities.

In 2020 and 2021, Uniper and Fortum established a consortium (Fortum-Uniper Nuclear Service) which in 2021 was awarded the contract for the technical implementation of demolition and decommissioning at OKG Aktiebolag and Barsebäck Kraft AB. The consortium builds on the experience gained during the period in which PPC has been in operation.

In addition to completing the internal decommissioning programme, the consortium intends to offer services outside its own group. Uniper believes that there will be a demand for these services on the market for a long time to come, which will also create an attractive opportunity for continued employment for employees engaged in decommissioning work.

Programme

The large-scale dismantling is under way at all facilities. The decommissioning programme is based on coordination of Barsebäck Kraft AB and OKG Aktiebolag in order to achieve synergies and contribute to safe and efficient decommissioning. During the start of decommissioning, the development need has in principle been the same for both Barsebäck Kraft AB and OKG Aktiebolag. The focus has been on developing processes. Going forward, the main activities will be:

- Refinement of the strategy, considering cooperation with the contracted consortium.
- Planning, detailed design and implementation of remaining work packages as the technical sequence progresses.
- Further development of different functional sub-processes: material and waste management, dismantling and demolition, operation and maintenance, plus protection and security. For example, development and optimisation of the facilities, near-surface repositories, interim storage facilities, release from regulatory control and logistics.
- Collection of experience to achieve learning effects within the technical sequence and with respect to long-term feedback of experience to decommissioning of O3 and O0.

18.3 Development within Vattenfall

Since the RD&D Programme 2019, a decision has been made not to transfer the nuclear licences for R1 and R2 to Vattenfall AB. This change in direction means that Ringhals AB will remain the licensee for these facilities also during decommissioning. However, Vattenfall's overall decision to bring together the Group's decommissioning skills in a separate business area, BUND, remains in place. The decommissioning of R1 and R2 will be carried out in a joint decommissioning programme under the joint control of Ringhals AB and BUND. The programme will coordinate and ensure that decommissioning is carried out in a safe and cost-effective manner.

Vattenfall is constantly working to incorporate experience feedback from decommissioning in a structured manner. Decommissioning skills and experience obtained within the group during the decommissioning of the R2 reactor in Studsvik will be coordinated and transferred to the decommissioning of Ågesta and R1 and R2. Similarly, efforts are under way to exchange experience and utilise synergies between Vattenfall's decommissioning activities in Germany and those in Sweden, both for facilities that are finally shut down and for future decommissioning projects.

Since the RD&D Programme 2019, the Ågesta programme has entered the dismantling and demolition phase. Operations at the facility currently involve dismantling of systems and removal of the contamination that exceeds thresholds for release from regulatory control. Large, complex components such as the fuel handling machine, venting system and heavy water tanks have been dismantled and managed and disposed of. The decommissioning programme, which is planned to be completed in the second half of 2025, has been affected by the Covid-19 pandemic, which among other things has limited work in the narrow rock cavern and hampered access to containers.

Programme

As in the previous RD&D period, a large part of the future development work will be carried out as an integrated part of the decommissioning programmes. Improvements in progress or planned for the period include:

- Efficiency improvements in respect of release of systems, buildings and land from regulatory control, including decontamination and measurement systems. Both refined statistical methods and robotics and artificial intelligence (AI) are being investigated.
- Active efforts to continuously improve already established waste processes, including optimisation of processing, decontamination and waste conditioning measures.
- Continued volume optimisation linked to repository safety and system utilisation (very low-level, short-lived and long-lived waste).
- Develop technical solutions for robotisation/automation of advanced demolition steps with the aim of reducing the dose to personnel.
- Develop an improved governance model together with the supplier market to increase focus on industrial safety and work environment.
- Develop the cost model for calculation and evaluation of costs for decommissioning of a nuclear facility so that risks and uncertainties are clarified, and enable sensitivity analyses to be performed in respect of cost-driving parameters.
- Further development and adaptation of portfolio, programme and project processes. This aims to ensure, among other things, that the decommissioning activities are optimised in respect of both time and implementation, so that the actual work is performed safely and efficiently.

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Appendix

Abbreviations and explanations

0AVF	Shared waste facility at the Oskarshamn Nuclear Power Plant.
ABM	Alternative buffer materials. Alternative buffer materials. Experiments at the Äspö HRL where potential buffer materials are being studied.
ALARA	As Low As Reasonably Achievable. Limitation of radiation doses as far as is reasonably achievable, taking into account both financial and social factors.
Andra	Agence nationale pour la gestion des déchets radioactifs. Organisation responsible for final disposal of radioactive waste in France.
ATB	Waste transport casks.
B1	Nuclear power reactor Barsebäck 1.
B2	Nuclear power reactor Barsebäck 2.
Back-end	Includes decommissioning of nuclear facilities, management of spent nuclear fuel and radioactive waste, as well as remediation and restoration of radioactively contaminated land.
BAT	Best Available Technology. Best Available Technology.
Beacon	Bentonite mechanical evolution. EU project.
BeFo	Stiftelsen Bergteknisk Forskning. Foundation for Rock Engineering Research.
BFA	Waste vaults for interim storage of low-level and intermediate-level waste on the Simpevarp peninsula in Oskarshamn.
BHA	Waste vault for legacy waste in the SFL.
BHK	Waste vault for core components in the SFL.
BLA	Waste vault for low-level waste in the SFR. The SFR contains a waste vault for low-level waste (1BLA) and an additional four waste vaults (2–5BLA) are planned in the extended part of the SFR.
BMA	Waste vault for intermediate-level waste in the SFR. The SFR contains a waste vault for intermediate-level waste (1BMA) and an additional waste vault (2BMA) is planned in the extended part of the SFR.
BMWi	Bundesministerium für Wirtschaft und Energie. Federal Ministry for Economic Affairs and Climate Action in Germany. Responsible for nuclear energy.
BTF	Concrete tank repository in the SFR, mainly intended for dewatered ion exchange resins.
BRIE	Bentonite Rock Interaction Experiment. Experiment at the Äspö HRL.
BUND	Business Unit Nuclear Decommissioning in Vattenfall.
BWR	Boiling Water Reactor. Boiling Water Reactor. The reactors in Forsmark, Oskarshamn and reactor 1 in Ringhals are boiling water reactors.
CDLM	Committee on Decommissioning of Nuclear Installations and Legacy Management.
CEC	Cation Exchange Capacity. Cation Exchange Capacity.
Chalmers	Chalmers University of Technology.
Clab	Central Interim Storage Facility for Spent Nuclear Fuel.
Clink	Central Facility for Interim Storage and Encapsulation of Spent Nuclear Fuel.
COMSOL	Computational tool for modelling and simulation of complex physics-based systems. Comsol Inc.
CORI	Cement-Organic-Radionuclide Interactions.
CR	Concentration Ratios.

DFN	Discrete Fracture Network. Discrete Fracture Network.
DisCo	Modern spent fuel dissolution and chemistry in failed container conditions. EU project.
DOC	Dissolved Organic Carbon. Dissolved Organic Carbon.
Domplu	Dome plug experiment. Full-scale test at the Äspö HRL to test and demonstrate the complete plug system.
Enresa	Empresa Nacional de Residuos Radiactivos S.A. Organisation responsible for final disposal of radioactive waste and decommissioning of the nuclear power plants in Spain.
Eurad	European Joint Research Programme on Radioactive Waste Management. EU programme.
F1	Nuclear power reactor Forsmark 1.
F2	Nuclear power reactor Forsmark 2.
F3	Nuclear power reactor Forsmark 3.
FEBEX	Full-scale Engineered Barriers Experiment. Full-scale demonstration experiment performed in the underground laboratory in Grimsel, Switzerland.
Formas	State research council for sustainable development.
F-PSAR	Preparatory preliminary safety analysis report.
FSW	Friction stir welding. Friction stir welding.
GAP	Greenland Analogue Project.
GIS	Geographic Information System. Geographic Information System.
GRASP	Greenland Analogue Surface Project.
IAEA	International Atomic Energy Agency.
ICE	Greenland ICE project. Glacial-hydrological study that complements the studies in GAP.
ISA	Isosaccharinic acid, or isosaccharinate depending on the pH.
ISO container	Containers in sizes standardised by the International Organization for Standardization (ISO), which can be loaded onto railroad cars, trucks and cargo ships.
KBS-3 method	Method for final disposal of the spent nuclear fuel based on three protective barriers: copper canisters, bentonite clay and the Swedish bedrock. The KBS-3 method is so named because it is based on the third report in the KärnBränsleSäkerhet (Nuclear Fuel Safety) project.
KBS-3 system	The nuclear facilities etc that are needed to carry out final disposal of spent nuclear fuel according to the KBS-3 method. The KBS-3 system consists of a joint facility for interim storage and encapsulation of the spent nuclear fuel, a transport system for transportation of canisters with spent nuclear fuel and a final repository.
K_d	Sorption coefficient, distribution coefficient.
KSKG	The nuclear power industry's safety coordination group.
KSU	Kärnkraftsäkerhet och Utbildning AB (the Swedish Nuclear Training and Safety Centre).
KTB	Transport casks for canisters of spent nuclear fuel.
KTH	The Royal Institute of Technology.
KTL	The Act (1984:3) on Nuclear Activities (the Nuclear Activities Act).
Lasgit	Large scale gas injection test.
LOT	Long-term test of buffer material. Experiment at the Äspö HRL aimed at finding out how bentonite clay primarily behaves in conditions similar to those in a final repository for spent nuclear fuel.

LTDE-SD	Long term diffusion experiment – Sorption-diffusion. Completed experiment at the Äspö HRL.
LVDT	Linear Variable Differential Transformer.
MARFA	Migration Analysis of Radionuclides in the Far Field. Computational tool for modelling of radionuclide transport.
MB	Environmental Code (1998:808).
MiniCan	Miniature Canister Corrosion Experiment. Experiment at the Äspö HRL.
MIS	Marine Isotope Stages.
MMD	The Land and Environment Court.
MODARIA	Modelling and Data for Radiological Impact Assessments. IAEA project.
MODATS	Monitoring Equipment and Data Treatment for Safe Repository Operation and Staged Closure.
Modern2020	Development and Demonstration of monitoring strategies and technologies for geological disposal. EU project.
MoFrac	Computational tool for DFN modelling.
MOX	Mixed Oxide Fuel. Mixed Oxide Fuel.
EIA	Environmental Impact Assessment.
MSB	Swedish Civil Contingencies Agency.
MX-80	Sodium bentonite from Wyoming, USA.
Nagra	Die Nationale Genossenschaft für die Lagerung radioaktiver Abfälle. Organisation responsible for final disposal of radioactive waste in Switzerland.
NEA	Nuclear Energy Agency. A cooperation body for nuclear energy issues in the OECD.
NPT	Non-Proliferation Treaty.
NWMO	Nuclear Waste Management Organisation. Organisation responsible for final disposal of spent nuclear fuel in Canada.
O1	Nuclear power reactor Oskarshamn 1.
O2	Nuclear power reactor Oskarshamn 2.
O3	Nuclear power reactor Oskarshamn 3.
OECD	Organisation for Economic Cooperation and Development.
OKG	OKG Aktiebolag.
ONDRAF/ NIRAS	Organisme national des déchets radioactifs et des matières fissiles enrichies. Organisation responsible for final disposal of radioactive waste in Belgium.
Onkalo	The rock facility that Posiva has been constructing in Olkiluoto since 2004. Onkalo is used for research and development, but will also provide access to the actual final repository.
PAN	Polyacrylonitrile.
Posiva	Posiva Oy. Organisation responsible for final disposal of spent nuclear fuel in Finland.
PPC	Project Performance Center in Uniper.
PSAR	Preliminary Safety Analysis Report prior to construction.
Puram	Public Limited Company for Radioactive Waste Management. Organisation responsible for final disposal of radioactive waste in Hungary.
PWR	Pressurised Water Reactor. Pressurised Water Reactor. The reactors R2, R3 and R4 in Ringhals and the Ågesta reactor are pressurised water reactors.
R1	Nuclear power reactor Ringhals 1.
R2	Nuclear power reactor Ringhals 2.
R3	Nuclear power reactor Ringhals 3.

R4	Nuclear power reactor Ringhals 4.
Redox	A redox reaction is a chemical reaction in which one element is reduced while another element is oxidised.
RISE	Research Institutes of Sweden.
RK&M	Preservation of Records, Knowledge and Memory across Generations. Completed project within OECD/NEA.
RWM	Radioactive Waste Management. Organisation responsible for radioactive waste management in the UK.
RWMC	Radioactive Waste Management Committee. OECD/NEA.
SAR	Safety analysis report.
SFC	Spent Fuel Characterisation.
SFL	Final Repository for Long-lived Waste.
SFR	Final Repository for Short-lived Radioactive Waste.
SKB	Svensk Kärnbränslehantering AB.
SKC	Swedish Centre for Nuclear Technology.
SNSN	Swedish National Seismic Network.
SRM	Synthetic Rock Mass. Model of a rock volume with a probable distribution of fractures on different scales.
SR-PSU	Report on post-closure safety for the SFR extension. Published by SKB in August 2015.
SR-Site	Report on post-closure safety at the Spent Fuel Repository. Published by SKB in March 2011.
SSL	Radiation Protection Act (2018:396).
SSM	Swedish Radiation Safety Authority (SSM).
SSMFS	Code of Statutes of the Swedish Radiation Safety Authority.
STF	Safety-related technical specifications.
STURE	Safe and Secure Phase Out of Reactors 1 and 2 (Ringhals).
Surao	Správa úložišť radioaktivních odpadů. Authority/organisation responsible for radioactive waste in the Czech Republic.
Suus	Safety during construction of the Spent Fuel Repository.
Task Force GWFTS	Task Force on Modelling of Groundwater Flow and Transport of Solutes. International collaboration between specialists and modelling groups on issues concerning groundwater flow and transport of solutes in the bedrock.
Task Force EBS	Task Force on Engineered Barrier Systems. International collaboration between specialists and modelling groups on issues concerning the engineered barriers in the future final repository.
THM	Thermal, Hydraulic, Mechanical (properties, factors or processes).
TB	Transport casks for spent fuel assemblies.
TK	Transport cask for core components.
Vinnova	Sweden's innovation agency.
VTT	Technical Research Centre of Finland Ltd. VTT Technical Research Centre of Finland. Research institute owned and controlled by the Finnish government.
WAC	Waste Acceptance Criteria.

Arkitektkopia AB, Bromma, 2023

SKB is responsible for managing spent nuclear fuel and radioactive waste produced by the Swedish nuclear power plants such that man and the environment are protected in the near and distant future.

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