

Technical Report

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RD&D Programme 2019

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

SVENSK KÄRNBRÄNSLEHANTERING AB

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This report is the English version of SKB RD&D programme 2019. In the event of any discrepancy between the Swedish and English versions, the Swedish original will take precedence.

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Preface

SKB, Svensk Kärnbränslehantering AB, is owned by the licensees for the Swedish nuclear power plants and is responsible for management and disposal of nuclear waste and spent nuclear fuel from the nuclear power reactors. Under the Nuclear Activities Act, the licensees for the reactors must every three years prepare or arrange for a programme for the comprehensive research and development work and other measures needed to safely manage and dispose of the nuclear waste and the spent nuclear fuel and to decommission the nuclear power plants. The licensees have delegated 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 2019, which has been prepared in collaboration with the nuclear power companies.

At the time of submitting the RD&D Programme 2019 to the Swedish Radiation Safety Authority, the licence application for an extension of the Final Repository for Short-lived Radioactive Waste, SFR, is under examination by the Authority, and the main hearing in the Land and Environment Court is in progress. SKB's applications for licences to build a final repository for spent nuclear fuel in Forsmark and an encapsulation plant in Oskarshamn were the subjects of the main hearing in the Land and Environment Court in the autumn of 2017 and in January 2018, both the Court and the Swedish Radiation Safety Authority submitted their statements on the applications to the Government. In the spring of 2019, SKB submitted supplementary documentation to the Government, as requested by the Court in its statement. To avoid redundancy in the present RD&D programme, references are made to the applications or other reports which have been submitted to the Government, the Swedish Radiation Safety Authority and the Land and Environment Court where necessary.

Of SKB's repositories, the Final Repository for Long-lived Waste (SFL) will be commissioned last. In this RD&D Programme, an evaluation of the post-closure safety is presented for the proposed repository concept. The evaluation provides a basis for the continued development of SFL, regarding both siting and barrier design. The programme also presents siting factors and the stepwise siting process for SFL.

Since the publication of the RD&D Programme 2016, two reactors have been shut down. Decommissioning of four reactors is currently under way and within a foreseeable future, two additional reactors will be shut down and decommissioned. The RD&D programme describes the management of very low-level nuclear waste that is produced in connection with decommissioning and demolition. A general description of the capacity of SKB's transportation system is also provided.

This RD&D programme has roughly the same structure as the RD&D Programme 2016 with the exception of the additional sections briefly described above and a section specifically discussing how competence management is to be secured in the long term. Since SKB's activities will proceed for another 70 years until the mission to manage and dispose of radioactive waste and spent nuclear fuel is completed, competence management is a strategically important issue.

SKB's organisation has been adapted during the year in order to safely and efficiently be able to design and construct the facilities in the KBS-3 system and the extended SFR during the coming decade. Research and development will be pursued to the extent needed to ensure that the waste programme in its entirety will meet the requirements for post-closure safety, radiation protection during operation of the facilities and limited impact on the external environment.

Stockholm in September 2019
Svensk Kärnbränslehantering AB



Eva Halldén
CEO



Ella Ekeröth
Purchaser RD&D programme 2019

Summary

The Swedish power industry has been generating electricity by means of nuclear power for more than 50 years. Under the Nuclear Activities Act, the holder of a licence for nuclear activities is responsible for ensuring the safe management and final disposal of nuclear waste and spent nuclear fuel arising from the activities. The holder of a license is also responsible for the comprehensive research and development work necessary for this. In this RD&D Programme 2019, the licensees for the Swedish nuclear power plants and Svensk Kärnbränslehantering (SKB), which is responsible for the management and final disposal of the nuclear waste and the spent nuclear fuel on behalf of its owners, present their plans for research, development and demonstration during the period 2020–2025. 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, Activities and plan of action, describes the activities and plan of action for managing and disposing of nuclear waste and spent nuclear fuel from the operation and decommissioning of the Swedish nuclear power reactors. It also describes the rationales for the research, development and demonstration needed in order to construct and commission new facilities. The facilities that are currently in operation are the Interim Storage Facility for Spent Nuclear Fuel (Clab), the Final Repository for Short-lived Radioactive Waste (SFR), near-surface repositories at the nuclear power plants, and the ship m/s Sigrid.

For the final repository system for spent nuclear fuel, the KBS-3 system, what remains is the construction and commissioning of a new facility for encapsulation of spent nuclear fuel adjacent to Clab and a final repository, the Spent Fuel Repository, as well as development and manufacture of transport casks for canisters of spent nuclear fuel.

SFR needs to be extended for disposal of low-level and intermediate-level nuclear waste. An additional final repository, the Final Repository for Long-lived Waste (SFL), will be constructed and commissioned and transport casks for long-lived waste will be procured. Furthermore, the waste management at the nuclear power plants needs to be adapted so that decommissioning of reactors can be carried out according to plan. This also includes constructing new near-surface repositories and extending existing near-surface repositories for disposal of very low-level waste.

The operating times of the reactors are an important factor in the planning of the nuclear waste programme. After a decision in 2015, Oskarshamn 1 and Oskarshamn 2 have now been shut down, while planning is under way for the shutdown of Ringhals 1 and Ringhals 2 in 2020 and 2019, respectively. For the other six reactors, the planned operating time is 60 years. This applies to the reactors Forsmark 1, Forsmark 2 and Forsmark 3, Oskarshamn 3, and Ringhals 3 and Ringhals 4. The youngest reactors, Forsmark 3 and Oskarshamn 3, will thus be in operation until 2045 according to the reactor owners' current planning.

Plan of action

Construction and commissioning of new nuclear facilities involve a stepwise decision process that is based on the Nuclear Activities Act and SSM's regulations. The process entails, among other things, that increasingly detailed safety analysis reports are successively submitted to SSM.

- First, applications for licences under the Nuclear Activities Act and permissibility under the Environmental Code to construct, own and operate a new nuclear facility will be submitted to SSM and the Land and Environment Court, respectively.
- To start construction of a nuclear facility, an application must be submitted to SSM for approval.

- Prior to trial operation and regular operation, a renewed and supplemented safety analysis report must be submitted to and approved by SSM.

Prior to the RD&D Programme 2019, two licensing matters are being reviewed by the Land and Environment Court and by SSM, one for the final repository system for spent nuclear fuel (SKB is awaiting a government decision for the encapsulation plant and the Spent Fuel Repository) and the other for the extension of SFR for operational and decommissioning waste. For these facilities, there are uncertainties in the timeplans for the licensing process, but plans have been established, given the current conditions.

The plan for the new facility for encapsulation adjacent to Clab is to commence construction around 2024. Regarding the Spent Fuel Repository, the start of construction is planned for 2022. Both facilities are planned to be commissioned in 2031.

The spent nuclear fuel is currently interim-stored in Clab. SKB has a licence to store 8 000 tonnes of spent nuclear fuel, which according to current forecasts will be reached at the end of 2023. SKB has therefore, as a part of the application to construct and operate the encapsulation plant in Clab, also applied for a licence to increase the interim storage capacity to 11 000 tonnes.

According to SKB's current plans, construction of the extension of SFR can start in 2023 and the repository can be commissioned in 2029. The work with dismantling and demolishing the seven reactors that are shut down first is planned to start before the extended SFR is commissioned. This means that the short-lived decommissioning waste must be interim-stored. During the period when construction of the extension is under way, there is also a need for interim storage of short-lived operational waste.

SFL is the final repository that is planned to be commissioned last. Up to commissioning, there are several important milestones that must be reached: site selection, assessment of safety after closure, preparation of licence applications and construction. SKB plans to submit licence applications to construct, own and operate SFL in 2030, so that the repository can be commissioned around 2045.

Research and development

SKB's and the reactor owner's planning of future research and development activities for the final repositories is based on the plan of action, where the stepwise decision process forms a basis for important milestones. The milestones relating to decision steps in the form of applications and safety analysis reports, dictate when knowledge and development of the technology needs to have reached a certain level, while SSM's approval dictates when SKB can commence construction and operation of the facilities. When SKB submits the applications to construct a facility, the purpose is to show that we have the knowledge and ability to construct it in such a way that it will meet the regulatory requirements. Even when SKB has reached the maturity in research and development required to obtain licences under the Nuclear Activities Act, further research and technology development is needed in the coming steps in the licensing process. This entails further development of the technology and systems that is needed to construct and then commission the facility and research to reduce remaining uncertainties in the post-closure safety assessments of the final repositories as support for the continued development and optimisation of the facility. Part I, Activities and plan of action, describes the most important remaining research and development issues, and the planned efforts during the RD&D period are presented in greater detail in Part II, Waste and final disposal.

Issues relating to the preservation of information and knowledge of final repositories and to the development of other concepts for final disposal, especially disposal in deep boreholes, have been recurring during all the years of consultations prior to the applications for the encapsulation plant and the Spent Fuel Repository. SKB is therefore continuing the work on how information and knowledge of the final repositories can be preserved across generations far into the future. SKB intends to continue to follow the development within the areas of drilling of and disposal in deep boreholes.

In order to be able to manage the radioactive waste and the spent nuclear fuel in a safe and cost-effective manner, SKB has developed a systematic approach for carrying out the research, development and demonstration needed to construct and commission new facilities. This includes securing resources and competence in the long term.

Part II Waste and final disposal

The comprehensive research, development and planning conducted over four decades have led to many issues of importance for the nuclear waste programme being resolved. Part II, Waste and final disposal, presents the need for continued research, development and demonstration during the RD&D period for the remaining parts of the nuclear waste programme. It also includes short descriptions of the current situation in each area.

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

- The need for an 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 safety after closure.
- The need for knowledge and competence concerning design, construction, manufacturing and installation of the barriers and components to be used in the facilities.
- The need for knowledge and competence concerning inspection and testing to verify that the system barriers and components satisfy the requirements.

Final Repository for Long-lived Waste

Of SKB's repositories, the Final Repository for Long-lived Waste is planned to be commissioned last. A proposed repository concept has been developed, which during the period 2015–2019 was evaluated with respect to post-closure safety. The safety evaluation constitutes an important basis for the continued development of the repository concept, for formulating waste acceptance criteria and for the siting of the repository. The starting point for the siting process is the fundamental requirements in the Environmental Code, the Nuclear Activities Act and the Radiation Protection Act as well as experience from SKB's previously completed siting processes.

Low-level and intermediate-level waste

Low-level and intermediate-level waste will be disposed of in SFR or SFL depending on the waste's content of short-lived and long-lived radionuclides. Important efforts during the RD&D period are to: i) continue the work concerning how organic compounds affect radionuclide transport in a repository environment, ii) increase the knowledge of the radionuclide inventory, regarding both forecasted waste quantities and the waste's radionuclide content, with specific efforts concerning determination of difficult-to-measure nuclides, and iii) further develop waste containers and waste transport containers for both short-lived and long-lived decommissioning waste.

Spent nuclear fuel

Spent nuclear fuel is long-lived and high-level and requires radiation shielding in all handling, storage and final disposal. Final disposal of the fuel will take place in leak-tight copper canisters in the bedrock. The fuel is insoluble in water, which means that even if a canister is breached in the repository, it would take a considerable amount of time before radionuclides from the fuel could spread in the rock. Knowledge of the fuel's solubility is therefore crucial for the post-closure safety assessment. This is an area where further efforts will be made during the RD&D period.

The spent nuclear fuel is interim-stored in water-filled pools awaiting disposal in the Spent Fuel Repository. In order for the future management of the spent nuclear fuel to take place in a safe manner, inspections and studies concerning how the properties of the fuel are altered during interim storage, transportation, encapsulation and disposal in the Spent Fuel Repository are in progress.

Canister for spent nuclear fuel

In the Spent Fuel Repository, the copper canister provides containment. Further work concerns both research on properties of the copper canister in the repository environment and technology development for manufacturing canisters, verifying them against specified requirements and handling them in the KBS-3 system.

For the assessment of post-closure safety for the Spent Fuel Repository, there are issues regarding corrosion and copper creep that have to be studied further. Sulphide corrosion is the most important corrosion process (for the cases where the buffer has eroded, the process provides a dominant risk contribution in the assessment of post-closure safety), and SKB is continuing the work of gaining a better understanding of the details of the corrosion mechanism. SKB is also conducting work in order to understand and describe radiation-induced corrosion and stress corrosion cracking, and to be able to model localised corrosion, with a special focus on the period with an unsaturated bentonite buffer.

For describing the material properties of copper, the most important research issue is a better understanding of how the addition of phosphorus to the copper leads to favourable creep properties, and SKB will continue the studies both experimentally and theoretically. SKB is also planning initiatives for studying the possible penetration of hydrogen into copper, and if this could affect the material properties to any significant extent.

With respect to the construction, manufacture, inspection and testing of the canister, the following will, among other things, be carried out: i) developing the set of requirements for the canister components and determining whether the requirements can be met in case of, for example, changed technical design requirements, ii) describing how hardening mechanisms in the manufacturing of copper components affect the canister's strength, iii) clarifying both the requirements on the material composition of the canister components and the strategy for quality assurance of the material composition, iv) investigating how casting methods affect the insert material, and v) continuing verification of the welding procedure for sealing the canister and also preparing for qualification of the weld.

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.

Cementitious materials have a central function for post-closure safety in the repositories SFR and SFL. SKB is continuously carrying out work with the purpose of maintaining and strengthening the knowledge and the ability to model the evolution of the properties of cementitious materials over time. Further research will be conducted concerning, among other things, gas transport through cementitious materials.

In preparation for the extension of SFR and prior to the construction of SFL and the Spent Fuel Repository, extensive work is carried out related to the design of concrete structures and materials for the different repositories.

Clay barriers and closure

There are clay barriers in SFR (between the rock and the concrete silo) and there will be clay barriers in the Spent Fuel Repository (buffer and backfill) and SFL (backfill in the waste vault for legacy waste) in order to limit water flow around the copper canisters of spent nuclear fuel and the waste packages with low-level and intermediate-level waste.

The bentonite barriers are in the form of blocks and pellets. For the post-closure function in the repositories, the properties of water-saturated and homogenised bentonite are crucial. Continued efforts are planned to better describe both the homogenisation process and the properties of the barriers after water saturation. Efforts are also needed to better understand gas composition and microbial activity during the unsaturated period, primarily in order to be able to estimate the possible effects on the copper canister. In order to better understand the causes of piping and erosion of material after installation, a programme containing both experiments and modelling has been initiated. This will continue during the RD&D period.

SKB also needs to increase the knowledge of different clay materials and further develop methods for measurement of bentonite properties. Before backfilling tunnels, continued efforts concerning water management during installation are required. To verify that the installation of buffer and backfill functions as intended, full-scale tests will have to be performed under ground.

Prototypes or schematic solutions are available for the deposition process in the Spent Fuel Repository, which include, for example, the deposition machine, the backfill robot and the transportation system for buffer and backfill components. The process will be automated and a supervisory control system is being developed for controlling and monitoring deposition.

Rock

The main function of the bedrock for SKB's existing and planned final repositories is to ensure stable mechanical and chemical conditions over the time that the waste is to be isolated.

In recent years, the models for groundwater flow in the rock have been further developed. Processes for geochemical properties and transport properties are now integrated in these. Continued efforts concern how to further develop and test these new tools and extend the application areas.

Earthquakes close to the Spent Fuel Repository might cause shear movements along fractures intersecting canister positions and the research conducted aims to ensure that the risk for earthquakes and the negative effects on the repository system are not underestimated. When it comes to characterisation and modelling of the rock mass, the fundamental understanding of the mechanical properties of individual fractures and fracture systems needs to be increased, as well as the understanding of how these properties affect, for example, the hydrogeological and geochemical properties of the rock.

According to the chosen reference method, the tunnels in the Spent Fuel Repository will be blasted, but a study of alternative methods, for example mechanical mining, is under way to achieve a more level floor with a smaller excavation damaged zone in the surrounding rock. In order to verify the investigation and excavation methods and to ensure that they give the desired results, underground full-scale tests need to be conducted.

Surface ecosystems

SKB's research programme for surface ecosystems aims primarily at providing a basis for calculations of potential radioactive dose to humans and the environment in the assessment of safety after closure for the different repositories. The programme also provides a basis for environmental monitoring, assessments of any environmental changes and assessments of safety in the facilities in operation.

The most important remaining issues in surface ecosystems concern i) uptake pathways and uptake mechanisms for radionuclides in different organisms, ii) temporal and spatial heterogeneity in the landscape, iii) transport and accumulation processes, iv) radiological, biological and chemical properties of for example carbon, chlorine, molybdenum, nickel and the radionuclides in the decay chain of uranium.

Climate and climate-related processes

The overall purpose of the work on climate issues is to provide the safety assessments with scientifically substantiated scenarios for future climate evolution, as a basis for the evaluation of repository post-closure safety. For the different repository concepts, there are specific issues which are dependent on future climate evolutions and which must be answered in the safety assessments.

The future climate evolutions are based in part on our knowledge regarding the climate history (especially the latest glacial cycle, including the Weichselian glaciation), and in part on modelling of the future climate.

Overall, the work on climate issues is about developing process understanding, compiling climate history, updating safety assessment climate scenarios, and validating the models 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.

There are issues with a bearing on all three repositories, which have to be studied further in order to reduce uncertainties and to improve the reliability of the post-closure safety assessments for the different repositories. They primarily concern: i) historical climate changes during the Weichselian and the Holocene, ii) sea-level variations and shoreline displacement, iii) age and long-term stability of the rock surface in Forsmark, including quantification of glacial erosion, iv) validation of the permafrost

model, and v) variability in climate and ice sheets during the coming one million years. In addition to the above points on future studies, data collection from the bedrock borehole at the Greenland ice sheet and weather station observations on and in front of the ice sheet are planned to continue.

Part III Decommissioning of nuclear facilities

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

The licensee for a nuclear facility is responsible for decommissioning of the facility according to the Nuclear Activities Act, 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 SSM has approved of sealing the repository in question and the Government has granted exemption from responsibility under the Nuclear Activities Act.

Prior to decommissioning, necessary licences must be obtained. According to the Nuclear Activities Act, 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 decommissioning: decommissioning plan and decommissioning strategy, waste management plan, safety analysis report, documentation according to the Euratom treaty Article 37, step or sub-project notification, decommissioning report and inspection programmes for clearance.

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

The timeplans for decommissioning of the Swedish nuclear power plants are governed by the planned operating times for the nuclear power plants. According to current plans, Barsebäck 1, Barsebäck 2, Oskarshamn 1, Oskarshamn 2, Ringhals 1, Ringhals 2 and Ågesta will be released from regulatory control by 2028. The nuclear power companies' plan is to start dismantling and demolition as soon as possible after final shutdown, where conditions like the availability of interim storage facilities and final repositories for decommissioning waste will affect the planning.

For more efficient work with decommissioning and waste issues, work areas have been divided between the different actors on both the company level and the 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 arranges the transportation of spent nuclear fuel and radioactive waste from the nuclear power plants to interim storage facilities and final repositories.

Decommissioning of the nuclear facilities in Sweden will continue up 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 mid-2040s and the final one in the mid-2070s. One of the challenges is thus securing access to competence for all decommissioning stages, since the need for decommissioning competence between stages will be limited in Sweden.

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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, resources and competence

Part I – reading instructions

Part I in the RD&D Programme 2019 describes activities and plan of action to manage and dispose of nuclear waste and spent nuclear fuel from the operation and decommissioning of the Swedish nuclear power reactors. The planned research and development activities needed to implement the remaining parts of the system and to decommission the nuclear power reactors and other nuclear facilities, are justified and summarised based on the plan of action. It also describes the systematic approach that SKB has developed to conduct the research, development and demonstration needed to realise the plan and to dispose of the radioactive waste and the spent nuclear fuel in a safe and cost-effective manner.

Part I also describes the current level of knowledge and the planned activities within two other areas of interest for SKB, namely questions concerning information preservation across generations and the development in the areas of drilling of and disposal in deep boreholes.

1 Introduction

The Swedish power industry has been generating electricity by means of nuclear power for more than 50 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 of the Swedish nuclear power reactors. So far, the system consists of the Central Interim Storage Facility for Spent Nuclear Fuel (Clab), the Final Repository for Short-lived Radioactive Waste (SFR), near-surface repositories at the nuclear sites, as well as the ship m/s Sigrid and transport casks and containers.

For safe management and disposal of the spent nuclear fuel in the long term, what remains to be done is to construct and commission the system of facilities needed for final disposal, the KBS-3 system. Operation of the system then follows, and when all spent nuclear fuel is disposed of, the facilities can be decommissioned and the final repository can be sealed and closed.

The KBS-3 system includes Clab with an encapsulation plant for the spent nuclear fuel, transport casks for shipping canisters of spent nuclear fuel and a final repository for spent nuclear fuel. In addition to these facilities, a system for production of canisters as well as buffer and backfill material is required.

SFR needs to be extended for disposal of low- and intermediate-level operational and decommissioning waste. An additional final repository, the Final Repository for Long-lived Waste (SFL), will be constructed and waste transport containers for long-lived waste is to be manufactured and 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 way that human health and the environment are protected, today and in the future. The RD&D programme also describes the ongoing planning for the reactors that will be decommissioned in the 2020s.

1.1 Prerequisites

1.1.1 Relevant regulatory framework and SKB's mission

Under the Act (1984:3) on Nuclear Activities (the Nuclear Activities Act), the holder of a licence for nuclear activities is responsible for ensuring the safe management and final disposal of nuclear waste and spent nuclear fuel arising from the activities. The holder of a license to operate a nuclear power reactor shall also be responsible for the comprehensive research and development work necessary for this. The licensee is also responsible for safely decommissioning the facilities when the operations have ceased.

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

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

The reactor owners are responsible for decommissioning of their nuclear power reactors. In this context, SKB has been contracted by the reactor owners to participate in the planning of the future decommissioning. SKB is participating by coordinating general methods and procedures for transportation and final disposal of radioactive decommissioning waste.

According to the Nuclear Activities Act, the reactor owners shall, in consultation, prepare or arrange for a programme for the comprehensive research and development work. The programme shall also describe other measures needed to safely manage and dispose of nuclear waste and spent nuclear fuel and to decommission the nuclear power plants. Such a programme for research, development and

demonstration (RD&D programme) shall be submitted to the Swedish Radiation Safety Authority (SSM) every three years. After referral to regulatory authorities and non-governmental organisations, the programmes are reviewed and evaluated by SSM. They are also reviewed by the Swedish National Council for Nuclear Waste. SSM and the Council submit their comments to the Government, which decides whether the programmes meet the requirements of the Nuclear Activities Act and if any guidelines should be provided for the continued activities.

SKB, on behalf of and in cooperation with the reactor owners, prepares the RD&D programmes and submits them to SSM.

Under the Nuclear Activities Act, the reactor owners are also obliged to bear the costs of their commitments. The Act (SFS 2006:647) on Financing of Management of Residual Products from Nuclear Activities (the Financing Act) aims to ensure the financing. In accordance with the Financing Act, the reactor owners are obliged to pay a fee into a fund that is managed by the state Nuclear Waste Fund. On behalf of the reactor owners and pursuant to the Financing Act, SKB presents cost calculations every three years (Section 1.3).

In addition to the nuclear waste received from the nuclear power companies, SKB also receives radioactive waste from research, medicine and industry. This is regulated by commercial agreements between SKB and the companies that are responsible for this waste. SKB currently has agreements with AB SVAFO, Studsvik Nuclear AB, Cyclife Sweden AB¹, European Spallation Source ERIC (ESS) and Westinghouse Electric Sweden AB (WSE).

1.1.2 Fundamental principles

The management of radioactive material is regulated by legislation. Radioactive waste is material emitting harmful ionising radiation that constitutes waste (according to Chapter 15 Section 1 of the Environmental Code) or has no planned and acceptable use. Nuclear waste is radioactive waste that is produced in nuclear power plants and other nuclear facilities. Radioactive waste can also come from research, medicine and industry.

The focus of the work with management of the spent nuclear fuel and the radioactive waste has been determined by a long series of political decisions and statements, which can be summarised in the following points:

- The spent nuclear fuel and the nuclear waste from the Swedish reactors will be disposed of within Sweden's borders with permission from the municipalities concerned.
- Sweden will not dispose of spent fuel or radioactive waste from other countries.
- The spent nuclear fuel will not be reprocessed.
- Final repositories shall be established by the generations that have benefited from Swedish nuclear power.

SKB plans for geological final disposal of the radioactive waste and the spent nuclear fuel. Other strategies have also been studied and discarded, such as launching the spent nuclear fuel into space, disposing of it beneath the deep seabed of the world's oceans or burying it in the continental ice sheet. Several countries and organisations such as the IAEA and the OECD/NEA today agree that geological disposal is a solution that fulfils all requirements on safe final disposal and feasibility. Geological disposal is also supported in the EU's community framework for the responsible and safe management of spent nuclear fuel and radioactive waste².

¹ Previously Studsvik Nuclear Environmental AB.

² The Council's directive 2011/70/Euratom of July 19 2011 establishing a Community framework for the responsible and safe management of spent fuel and radioactive waste.

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

- Repositories shall be located in a long-term stable geological environment.
- 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 barriers shall work passively, i.e. without human intervention and without supply of energy or materials.
- 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 the post-closure safety of a final repository shall be based on multiple barriers whose purpose is to contain, prevent or retard the dispersion of the radioactive elements in the waste. The barriers and other components which are needed in a final repository are largely dependent 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 resistance in the final repository for short-lived radioactive waste are different than those in 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 choose the KBS-3 method³ for final disposal of spent nuclear fuel. Within the framework of the RD&D programmes, SKB has conducted evaluations of different strategies and systems for disposal of spent nuclear fuel on several occasions. In the most recent evaluation (SKB 2014f), SKB explains the background and reasons for the choice of the KBS-3 method in relation to other methods. The evaluation was made against established requirements, both general, societal requirements and environmental, safety and radiation protection requirements.

The method, whose development began at the end of the 1970s, can be summarised as follows:

- The spent nuclear fuel is placed in copper canisters with high resistance to corrosion in the repository environment. The approximately five metres long canisters have an insert of nodular cast iron, which enhances the stability.
- The canisters are surrounded by a buffer of bentonite clay – a naturally occurring mineral that swells in water, protects the canister from minor rock movements and shields the canister from groundwater flow, which limits the amount of corrosive agents in the groundwater that can reach the canister.
- The clay also absorbs radioactive elements that may be released if the canisters were to fail. The canisters with the surrounding bentonite clay are emplaced at a depth of about 500 metres in bedrock with long-term stable conditions.
- If a canister were 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 radionuclides from the repository to the ground surface.

Internationally, the KBS-3 method is one of the methods for final disposal of spent nuclear fuel where development has progressed the furthest. The method is used in Finland (Section 5.5.4) and it, or variants of it, is being considered as a method for final disposal in several other countries, including Canada, South Korea, the UK, Taiwan and the Czech Republic.

³ The KBS-3 method has been given its name because it is based on the third report in the KärnbränsleSäkerhet (Nuclear Fuel Safety) Project (The final stage of the nuclear fuel cycle, Spent nuclear fuel, SKBF/KBS, 1983).

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 planned operating times, forecasts are made of the quantities of nuclear waste and spent nuclear fuel that will be managed and when the need for interim storage and final disposal will arise.

The planning for the waste system is based on the reactor owners' current planning prerequisites. In 2015, a decision was made on premature shutdown of four reactors, Oskarshamn 1, Oskarshamn 2, Ringhals 1 and Ringhals 2, all commissioned during the 1970s. Oskarshamn 1 and Oskarshamn 2 have been shut down, while Ringhals 1 and Ringhals 2 will be shut down at the end of 2020 and 2019, respectively. For the other six reactors, the planned operating time is 60 years. This applies to the reactors Forsmark 1, Forsmark 2 and Forsmark 3, Oskarshamn 3, and Ringhals 3 and Ringhals 4. The youngest reactors, Forsmark 3 and Oskarshamn 3, will thus be in operation until 2045 according to the reactor owners' current planning.

1.1.4 The radioactive waste and the spent nuclear fuel

The plans for disposal 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 (short-lived or long-lived waste). The level of radioactivity determines how the waste is handled before final disposal. The intermediate-level waste and the high-level spent nuclear fuel require radiation-shielded handling, while the very low-level and low-level waste can be handled without radiation shielding. The design of final disposal is largely determined by whether the waste is short-lived or long-lived, as this is of importance for the time period during which barrier performance needs to be maintained.

How much waste that is produced and when it 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. SKB's waste system is designed for the current reactors and is based on the forecasts of the reactor owners.

Very low-level waste

Waste that has a dose rate below 0.5 mSv/h and where the majority of radionuclides are short-lived with a half-life shorter than about 30 years⁴ 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 primarily generated during maintenance outages and maintenance and service work but also as a result of inspections of the facilities. The very low-level part of the decommissioning waste consists of dismantled systems and structural parts, as well as protective and decontamination equipment.

The management of very low-level waste is determined by the material type and activity content and mainly takes place on site at the nuclear power plants. More complex management such as incineration and melting takes place elsewhere. Final disposal of very low-level waste is currently carried out mainly in near-surface repositories. According to the current licence, about 37 000 cubic metres of short-lived operational waste will be disposed of in near-surface repositories at the Forsmark, Oskarshamn and Ringhals nuclear power plants.

Low-level and intermediate-level waste

The low-level and intermediate-level waste can be short-lived or long-lived. Short-lived waste primarily contains radionuclides with a half-life shorter than about 30 years⁴ and only a limited amount of radionuclides with a longer half-life. Long-lived waste mainly contains radionuclides with longer half-lives. When waste is classified as long-lived, it is because it contains a large amount of long-lived nuclides. This is completely independent of the amount of short-lived nuclides in the waste.

⁴ According to the 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.

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

Short-lived waste is presently disposed of in SFR. According to current forecasts, about 180 000 cubic metres of waste will be disposed of in SFR. Most of the short-lived waste is generated in the nuclear power plants. Today, additional short-lived waste comes from Clab and later from Clink⁵, as well as from facilities belonging to Studsvik Nuclear AB, AB SVAFO and Cyclife Sweden AB.

Long-lived waste from the nuclear power plants consists of used core components, reactor pressure vessels from pressurised water reactors (PWRs) and control rods from boiling water reactors (BWRs). The long-lived radionuclides are formed from stable elements in, for example, steel when they are exposed to strong neutron radiation from the reactor core. The total quantity of long-lived low-level and intermediate-level waste is estimated to about 16 000 cubic metres, about one-third of which comes from the nuclear power plants. The rest stems from the legacy waste, which is currently stored by AB SVAFO, and from operations by other companies (Section 1.1.1). SKB plans to dispose of the long-lived waste in SFL.

Spent nuclear fuel

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

The spent fuel generates heat even after it has been removed from the reactor (decay heat), which means that it must be cooled to avoid overheating. The decay heat is above all dependent 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, fuel burnup has increased steadily since the reactors were commissioned. The reason for these modifications is to achieve as efficient utilisation of the fuel as possible. A consequence of increased burnup is increased decay heat, which is of importance for interim storage and final disposal.

The facilities in the KBS-3 system are designed for a total amount of spent nuclear fuel equivalent to about 6 000 canisters. One canister contains about 2 tonnes of fuel. The quantity of spent nuclear fuel is specified as the quantity of uranium that was originally present in the fuel.

Almost all 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. Approximately 20 tonnes of spent nuclear fuel from the Ågesta plant and approximately two tonnes of spent nuclear fuel from Studsvik Nuclear AB's research activities are currently being interim-stored in Clab. 23 tonnes of MOX fuel (mixed oxide fuel) obtained from Germany in exchange for fuel that was sent to France (La Hague) for reprocessing at an early stage are also stored in Clab. Sweden has also sent a small amount of spent nuclear fuel from the first reactor in Oskarshamn to be reprocessed in Sellafield in England. No fuel or radioactive waste from that process will be returned to Sweden.

1.2 Programme for research, development and demonstration

The Nuclear Activities Act regulates the periodicity and scope of the RD&D Programmes. They shall be submitted every three years to SSM and shall describe the measures necessary to manage the radioactive waste. The programmes provide an overview of all necessary measures and a more detailed presentation of the measures that are planned for the coming six years.

⁵ An encapsulation plant for spent nuclear fuel in copper canisters that will be built in connection to Clab and will be operated as an integrated facility, Clink (Central Interim Storage and Encapsulation Plant for Spent Nuclear Fuel).

SSM reviews and evaluates the planned research and development work, presented research and development results, alternative management and disposal methods and planned measures. As a part of the preparation, SSM refers the programme to other regulatory authorities and organisations for consideration. When the review is completed, SSM submits its statement and the RD&D programme to the Government. The Swedish National Council for Nuclear Waste also submits its independent evaluation of the programme to the Government. Finally, the Government decides whether the programme meets the requirements under the Nuclear Activities Act and whether any conditions may be imposed for the continued activities.

The purpose of the RD&D programmes is to show that the reactor owners fulfil their obligations under the Nuclear Activities Act. They are a tool for the reactor owners to continuously present their strategies and plans and have them reviewed.

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

The focus of the RD&D programmes has varied through the years, depending on where the emphasis has been in SKB's activities. A brief summary of the RD&D programmes published by SKB up to and including 2007 was provided in the RD&D programme 2010. Thereafter, a summary has been made of the preceding RD&D Programme.

1.2.1 RD&D Programme 2016

Prior to the RD&D Programme 2016, SKB and the reactor owners participated in consultations with SSM. The purpose was to ensure that the requirements specified by the Government prior to the RD&D Programme 2016 were met. The requirements were related to the development of decommissioning plans and decommissioning studies and a clearer and more structured RD&D Programme, which specifies how research and development initiatives are planned, justified and evaluated in order to satisfy the requirements. To meet the requirements, the RD&D Programme 2016 was given a new structure consisting of four parts.

Part I, Activities and plan of action, described the waste system and the plan of action for commissioning and finally decommissioning the activities that are necessary for disposing of the radioactive waste and the spent nuclear fuel. Based on the plan of action, the planned research and technology development activities during the RD&D period were justified and summarised (2017–2022). This part also described the systematic working methods developed by SKB for conducting the necessary research, development and demonstration.

Part II, Waste and final disposal, described the planned research and technology development activities for the low-level and intermediate-level waste, the spent nuclear fuel and the different parts of the repository system. The current level of knowledge was described in general terms and references were made to more detailed presentations of results in background reports.

Part III, Decommissioning of nuclear facilities, presented the plans for the decommissioning of the nuclear power reactors and SKB's facilities. Here, the dependencies and the flexibility in the system were described, as well as planned development activities.

Part IV, Other issues, described the level of knowledge and the planned activities within two other areas of interest for SKB, questions concerning information preservation across generations and the development in the areas of drilling of and disposal in deep boreholes.

SSM's overall assessment of the RD&D Programme 2016 was that the reported research and development activities were sufficiently comprehensive and the planned measures for management of nuclear waste and spent nuclear fuel, as well as the decommissioning and dismantling of the nuclear facilities, are appropriate to comply with the Nuclear Activities Act. SSM noted that there is still a long-term need for research and development within the management and final disposal of the residual products of nuclear power and the decommissioning and dismantling of the nuclear power plants. Furthermore, SSM stated that SKB has to elaborate the description of safeguards.

The Government decided that the programme met the requirements of the Nuclear Activities Act and required for the continued research and development activities, which entailed that the programme should include research and development regarding decommissioning, dismantling and demolition, as well as management and final disposal of the residual products of nuclear power. The Government decision also stated that SKB in future RD&D programmes shall maintain the structure and division used in the RD&D Programme 2016. Furthermore, the Government requested a description of how competence development and competence management is to be secured in a long-term perspective, the plans and strategies for long-lived waste and an overall system description of the logistics for management of decommissioning waste and transportation.

1.2.2 Milestones and development since the RD&D Programme 2016

Low-level and intermediate-level waste

At the time of presenting the RD&D Programme 2019, a licensing matter is being processed in the Land and Environment Court and at SSM on the extension of SFR for operational and decommissioning waste.

In December 2014, SKB submitted applications under the Nuclear Activities Act and the Environmental Code for the extension of SFR. These applications have been reviewed and SKB has responded to requests for supplementary information from regulatory authorities and reviewing bodies. In January 2019, SSM recommended the approval of SKB's application for authorisation under the Environmental Code to extend SFR. The main hearing in the Land and Environment Court takes place in the autumn of 2019. SSM will be participating in the main hearing in its capacity as a government agency with expertise in the field of nuclear safety and radiation protection.

During 2015–2019, an evaluation on post-closure safety of the proposed final repository concept for SFL was carried out. The results indicate that the repository concept has the potential to meet the requirements on post-closure safety, given that the site exhibits sufficiently favourable properties regarding the limitation of solute transport in the rock.

The RD&D Programme 2016 described the plans for interim storage of long-lived waste in the extended SFR facility. This plan has changed and in the current plans, interim storage will take place either at the nuclear power plant sites or at another site.

Also the plan for how the reactor pressure vessels from BWRs and PWRs will be disposed has changed since the previous RD&D programme. The plans were then to dispose of whole reactor pressure vessels from BWRs and PWRs in the extended SFR and SFL, respectively. According to current planning, the BWR reactor pressure vessels will be segmented. There is also a policy decision on segmenting PWR reactor pressure vessels from Ringhals 2. Parts from the reactor pressure vessels will be interim-stored at the nuclear power plant sites or other. When the final repositories are established, the segmented reactor pressure vessels will be transported to each final repository for disposal. A decision regarding the reactor pressure vessels from Ringhals 3 and 4 will be made closer to the decommissioning of these reactors.

The KBS-3 system

In the autumn of 2017, a main hearing was held in the Land and Environment Court regarding the KBS-3 system. In January 2018, both SSM and the Court issued their statement to the Government. SSM recommended the approval of SKB's application under the Nuclear Activities Act. The Court was also positive in several important aspects, but identified a number of questions on the copper canisters. SKB submitted additional documentation to the Government in April 2019 and is now awaiting the Government's decision regarding permissibility under the Environmental Code and a licence under the Nuclear Activities Act.

1.2.3 RD&D Programme 2019

The RD&D Programme 2019 has essentially the same structure as the RD&D Programme 2016 with the aim to clarify how the planned activities link to the overall goals. Part IV, Other issues, which was included in the RD&D Programme 2016, is left out and is now included in Part I. The RD&D Programme 2019 is thereby divided into three parts:

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

The RD&D Programme 2019 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 is met by references.

The relationship between the content in the RD&D programme and the application documents for new facilities must be clear. SKB has the ambition to avoid redundancy and SSM has stated that they do not want to anticipate statements in any licensing matters in conjunction with the review and evaluation of the RD&D programme. As a consequence of the step-wise licensing, the majority of the research and development needs will be described in documentation requested during that process. Hence, the RD&D programme provides a less detailed description of these needs with references to the submitted documentation within each licensing matter.

In conclusion, redundancy should be avoided in order to avoid double licensing prior to a Government decision and of activities licensed by SSM. A general description of ongoing and planned research is provided in the RD&D programme and a more detailed description of the development prior to construction of new facilities is provided in the plans and programmes included in the application documents for each facility.

In order to avoid redundant reporting to SSM, the reader is referred to the following reports:

- The applications with supplements submitted by SKB regarding a system for final disposal of spent nuclear fuel.
- The applications with supplements submitted by SKB regarding an extension of the Final Repository for Short-lived Radioactive Waste (SFR).
- Studies and plans for decommissioning of the nuclear power reactors and other nuclear facilities.
- Safety analysis reports (SAR) and periodic overall assessments for Clab and SFR.
- The recurrent Plan reports, which provides estimates of future costs.

The applications under the Nuclear Activities Act for final disposal of spent nuclear fuel and under the Environmental Code for the KBS-3-system were submitted in March 2011. They describe the activities that will lead to construction, operation and final disposal. The applications also include results of assessments of safety during operation and after closure. SKB has submitted six supplements to the Land and Environment Court following review statements. SKB has also continuously responded to questions and requests for supplementary information and clarifications from SSM.

An application under the Nuclear Activities Act for the encapsulation plant was submitted in 2006. The application was supplemented in 2009 to integrate the encapsulation plant with Clab to a single facility, Clink. In March 2011, another supplement was submitted with regard to the parts of the applications that concern the KBS-3 system. In its review of the application, SSM requested supplementary information in 2012. SKB responded to this request by submitting an update to the preparatory preliminary safety analysis report (F-PSAR) at the end of 2014 and supplements to the applications to both SSM and the Land and Environment Court in March 2015. The latter supplement also includes an additional application for increasing the interim storage capacity in Clab to 11 000 tonnes of spent nuclear fuel.

An announcement of the applications under the Environmental Code and the Nuclear Activities Act was made in January 2016. The main hearing in the Land and Environment Court was held in the autumn of 2017 and in January 2018 both SSM and the Land and Environment Court issued their statements to the former Ministry of Environment and Energy (the Ministry of Environment since 1 April

2019). In letters from the Ministry on 1 June 2018, SKB was given an opportunity to supplement the case with respect to the questions identified by the Land and Environment Court regarding canister properties and post-closure safety (forms of corrosion and other processes). In June 2018, Oskarshamn Municipality made a positive decision regarding their veto right and approved the construction of Clink. In April 2019, SKB responded to requests for supplements to the Ministry of Environment.

At the end of 2014, SKB submitted applications under the Nuclear Activities Act and the Environmental Code for the extension of SFR and the licensing processes are currently in progress. Supplementary documentation for the matter was submitted in 2015–2017. In January 2019, SSM recommended the approval of SKB's application for authorisation under the Environmental Code to extend SFR. The main hearing in the Land and Environment Court takes place in the autumn of 2019.

As a licensee, SKB has an obligation to submit various reports to SSM for the facilities in operation. This currently applies to Clab and SFR. The reports mainly consist of the safety analysis reports (SARs) for the facilities and periodic overall assessments of the radiation safety that are to be carried out at least every ten years.

Each nuclear facility has a decommissioning plan that is prepared before the facility is constructed and thereafter kept up-to-date until the facility is decommissioned. Each licensee reports significant modifications to SSM and presents an updated plan to the Authority in conjunction with the periodic overall assessment.

Another report linked to the RD&D Programme is the Plan report (Section 1.3). It presents the future cost for disposing of the radioactive waste and the spent nuclear fuel and for decommissioning of the nuclear power reactors. The cost calculation is based on the plans presented in the RD&D programme.

1.3 Financing

The costs for disposing of the operational waste from the nuclear power plants are paid by the reactor owners as they arise, while financing of the rest of the nuclear waste programme is based on the payment of fees into a special fund, the Nuclear Waste Fund. The latter is regulated in the Financing Act and the associated Financing Ordinance.

Every three years, SKB prepares a cost calculation, a Plan report, on behalf of the reactor owners. The report is submitted to the Swedish National Debt Office⁶, who reviews SKB's calculation and makes recommendations for fees and guarantees. The size of the fees and guarantees is decided by the Government. The reactor owners pay the fees to the Nuclear Waste Fund. According to Government regulations, these funds may be invested in interest-bearing accounts at the National Debt Office, in debt instruments issued by the State and covered mortgage bonds. A part of the fund may also be invested in corporate bonds and shares.

At the end of 2018, there was about SEK 69 billion in the reactor owners' shares of the Nuclear Waste Fund (market value). In addition, approximately SEK 44 billion (current price level) has been spent on, among other things, siting, site investigations, implementation and operation of the current system and research and development work. During the period 2018 to 2020, the average fee is SEK 0.05 per kilowatt-hour of electricity produced by the nuclear power plants that are in operation. Barsebäck Kraft AB pays an annual fee of SEK 543 million during the same period.

Besides paying fees, the reactor owners' parent companies pledge guarantees to cover the fees that have not yet been paid. For the reactors that are in operation, a guarantee is also pledged for the eventuality that the Fund proves insufficient due to unplanned events. The guarantee cost amounts to a total of SEK 44 billion.

⁶ The National Debt Office took over SSM's responsibility under the Financing Ordinance on September 1st 2018.

2 Description of the waste system

The Swedish waste system consists of two main parts: the system for management of low-level and intermediate-level waste, and the system for management of spent nuclear fuel (the KBS-3 system). All facilities in the KBS-3 system will be operated by SKB. In the system for management of low-level and intermediate-level waste, there are both facilities that are operated by the reactor owners and other waste producers (near-surface repositories and interim storage facilities) and facilities that are operated, or will be operated, under SKB's auspices (SFR and SFL).

SKB is responsible for the transportation system, which is the same for the low-level and intermediate-level waste and the spent nuclear fuel. Since the nuclear power plants and SKB's facilities are situated on the coast, transportation takes place mainly at sea. The exception is the Ågesta plant, from which transport of decommissioning waste will be on country roads.

Figure 2-1 provides an overview of the complete system for management and disposal of Sweden's radioactive waste and spent nuclear fuel. The illustration shows the flow from the reactor owners and other waste producers via interim storage facilities and treatment plants to different types of final repositories. Solid lines represent transport flows to existing or planned facilities. Dashed lines represent alternative routes of management. The following section gives a description of all existing and planned facilities in the system.

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

The system for short-lived low-level and intermediate-level waste is partially built. SKB's final repository for short-lived radioactive waste, SFR, was commissioned in 1988. Moreover, there are a number of facilities operated by the nuclear power companies, such as near-surface repositories for very low-level waste. There are also facilities situated at the Studsvik site, which are owned by AB SVAFO, Cyclife Sweden AB and Studsvik Nuclear AB. These facilities include treatment plants, interim storage facilities and near-surface repositories.

SKB has submitted licence applications to extend SFR to provide additional space for short-lived operational and decommissioning waste.

The long-lived waste will be disposed of in SFL which will be the last of SKB's facilities to be commissioned.

2.1.1 Facilities for short-lived waste

Treatment of waste

At the nuclear power plants, at the Studsvik site and at Clab, there are treatment plants for short-lived waste. Here the waste is treated and packaged so that it meets the requirements for clearance, disposal in SFR or near-surface repositories. The purpose of the treatment may be to release the material from regulatory control, reduce its volume, concentrate its activity, solidify or condition the material.

Interim storage facilities

At the nuclear power plants, there are facilities for interim storage of short-lived waste. There are buffer storages for operational waste prior to further handling such as treatment and packaging. There are also facilities for buffer storage of finished waste packages before transport to and disposal in SFR.

Dismantling and demolition of the first seven reactors⁷ is planned to start before the extended SFR repository is ready to receive decommissioning waste. This means that the existing interim storage capacity for short-lived waste must be increased. Plans for this are described in Section 3.3.3. A new interim storage facility for low-level waste could consist of a paved surface or a building for storing ISO-containers. Intermediate-level waste requires a building with radiation shielding.

⁷ Barsebäck 1, Barsebäck 2, Oskarshamn 1, Oskarshamn 2, Ringhals 1, Ringhals 2 and the Ågesta reactor.

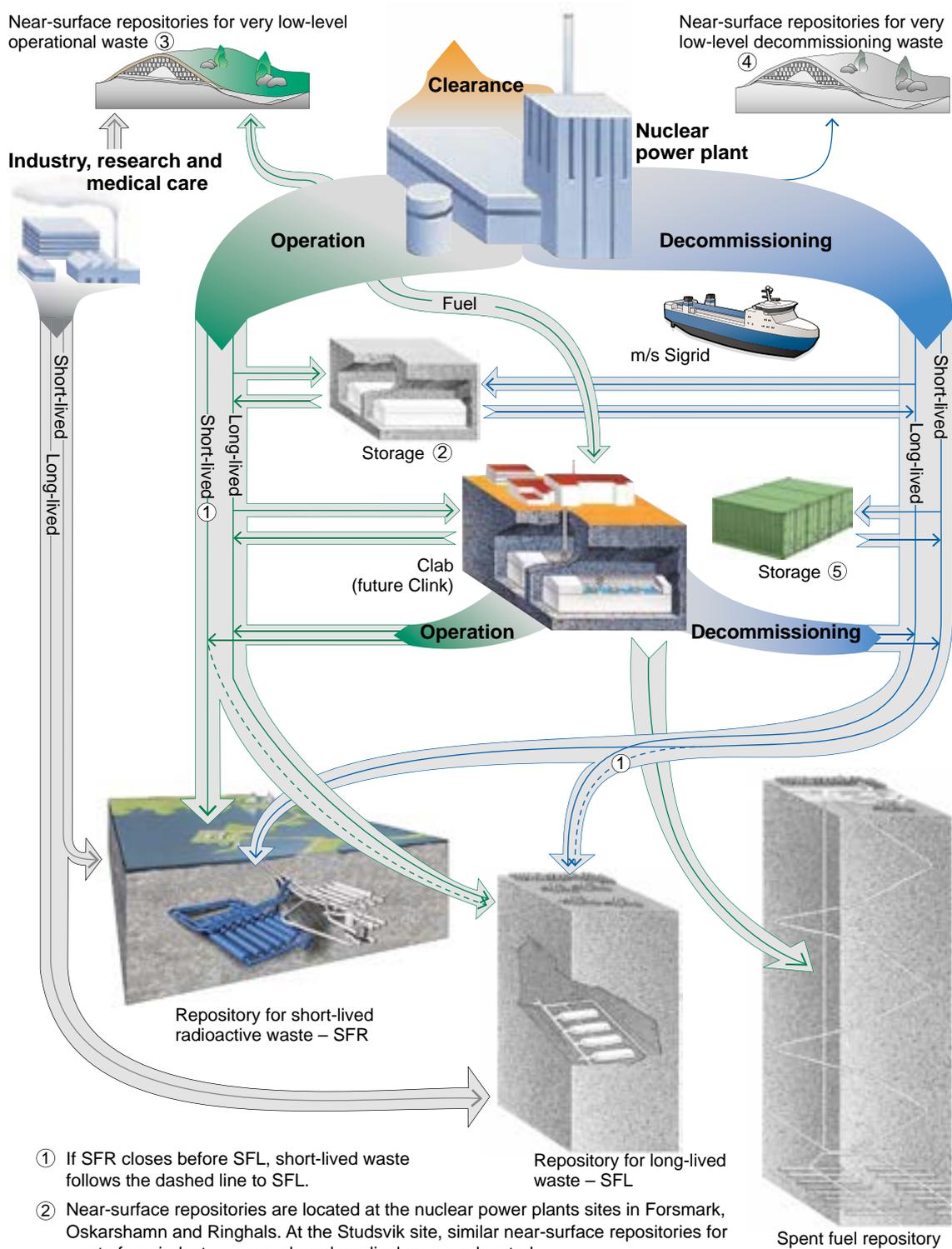


Figure 2-1. The system for management and disposal of Sweden's radioactive waste and spent nuclear fuel. Solid lines represent transport flows to existing or planned facilities. Dashed lines represent alternative routes of management.

Near-surface repositories

Some of the low-level waste contains very low levels of radioactivity. This waste is currently disposed of in the existing near-surface repositories, which are licensed for operational waste. They are located at 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 at the power plant sites today are only licensed for operational waste.

Waste that may be disposed of in near-surface repositories mostly contains short-lived radionuclides with a half-life shorter than about 30 years, i.e. similar to the waste in SFR. Both the activity in individual packages and the total activity allowed in a near-surface repository are, however, considerably lower than in SFR.

The near-surface repository design differs somewhat between sites to suit the local conditions. All near-surface repositories are based on a multi-barrier system, which delays the dispersion of radioactive elements. As for geological repositories, water flow is required for transporting the radionuclides from the near-surface repositories to the biosphere. Water flows through the near-surface repositories are limited primarily in that the repositories are situated above the groundwater level and by providing them with a water-tight cover (Figure 2-2). The near-surface repositories are constructed with draining material around and between the waste packages and a passive drainage to an infiltration bed. The infiltration bed construction, with for example shell gravel, contributes to efficient retention of radionuclides to retard transport. The infiltration bed has a controlled drainage to a recipient to ensure good dilution of radionuclides in the less probable case that the other barriers fail.

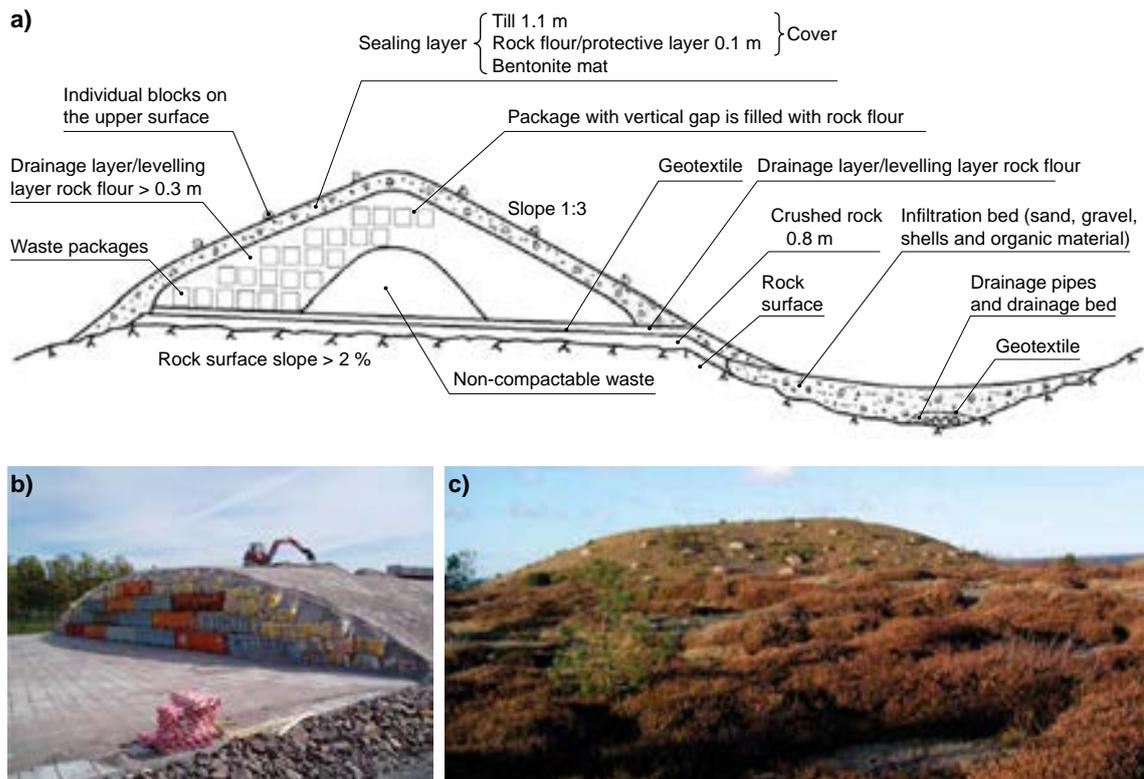


Figure 2-2. a) Schematic description of Ringhals' near-surface repository construction. b) Example of near-surface repository during an ongoing deposition campaign. c) Ringhals' near-surface repository.



Figure 2-3. The final repository for short-lived radioactive waste, SFR, consists of two waste vaults for concrete tanks (1-2BTF), one waste vault for low-level waste (1BLA), one waste vault for intermediate-level waste (1BMA) and a silo for intermediate-level waste. a) View of the surface facility, b) SFR underground, c) Waste vault, d) View of the silo top.

Final repository for short-lived radioactive waste

SFR is located at the Forsmark nuclear power plant (Figure 2-3). The repository is situated beneath the Baltic Sea, covered by about 60 metres of rock. Two one-kilometre-long access tunnels lead from the harbour in Forsmark to the repository area. The facility currently consists of four 160-metre-long waste vaults, plus a 70-metre-high cavern in which a concrete silo has been built. The facility's total storage capacity is 63 000 cubic metres.

Post-closure safety of SFR is based on *limiting* the amount of long-lived radionuclides in the repository and *retardation* of radionuclides in the engineered and natural barriers. The design of each waste vault is adapted to the activity level of the waste that is disposed in it. Low-level waste is disposed of in one of the four waste vaults. Intermediate-level waste with lower activity levels is disposed of in two of the waste vaults. The intermediate-level waste with the highest activity levels is disposed of in the fourth rock vault or in the concrete silo. The silo will contain most of the activity in SFR.

The waste in SFR comes mainly from the nuclear power plants, Clab, Studsvik and Ågesta, whereas a minor part comes from research, medicine and industry. At the end of 2018, about 40 000 cubic metres of waste had been disposed.

When SFR was constructed, the intention was that the facility would receive waste until the year 2010. Due to the prolonged operating time of the nuclear power plants, SFR's operational phase will be prolonged, which entails new demands on the maintenance of the facility. The maintenance programme includes, in addition to remedial and preventive maintenance, identification, handling and prevention of age-related effects.

At present, only operational waste is disposed of in SFR. SFR's storage capacity will be extended to provide space for additional short-lived waste from both operation and decommissioning. SKB has submitted a licence application to extend the facility with additional storage capacity of about 117 000 cubic metres, to hold in total about 180 000 cubic metres of waste. Figure 2-4 illustrates SFR when it is fully extended according to current plans.

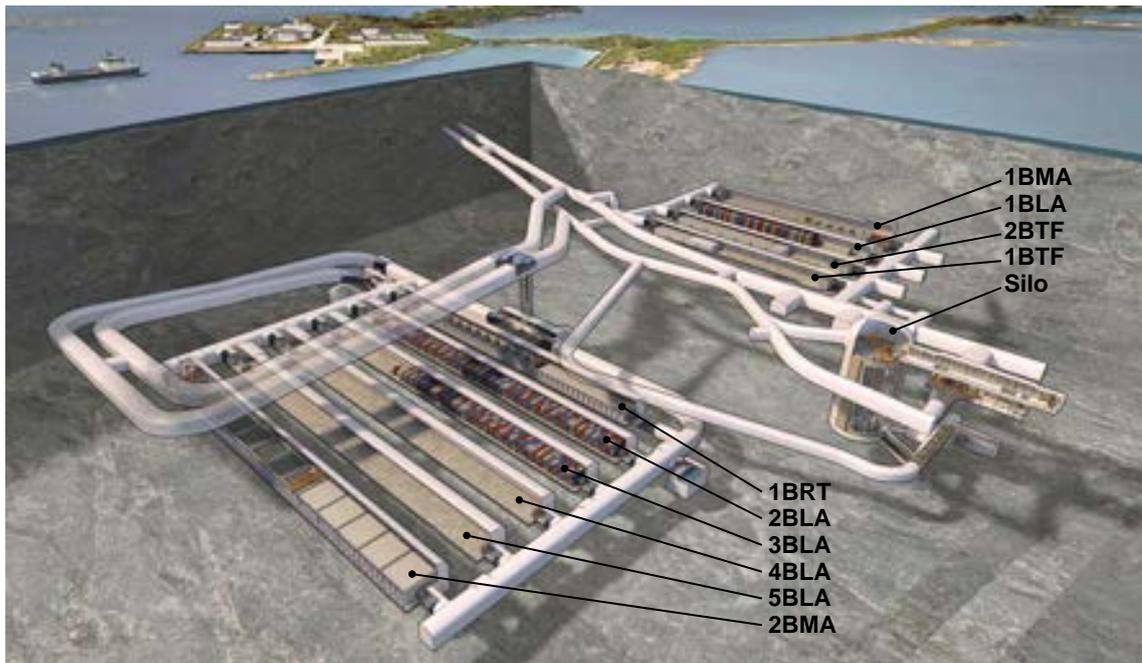


Figure 2-4. When SFR is fully extended, it will contain six additional waste vaults; four for low-level waste (2-5BLA), one for intermediate-level waste (2BMA) and one waste vault for reactor pressure vessels (1BRT).

2.1.2 Facilities for long-lived waste

Treatment of waste

It is currently possible to segment certain used core components at the nuclear sites in order to be able to place these in steel tanks for local interim storage. This has been done previously when upgrading the reactors, but presently it is mainly carried out as a part of the decommissioning projects. The reactor pressure vessel internals from Barsebäck 1 and Barsebäck 2 are segmented. Corresponding work has also been carried out at Oskarshamn 2 while segmentation is under way at Oskarshamn 1. It is planned to be finished in the spring of 2020.

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 handled and what the possibilities are for management and final disposal.

Interim storage facilities

SKB plans to commission SFL around 2045. Until then, the long-lived waste must be interim-stored. Today most of the long-lived waste is interim-stored at the nuclear power plants, in Clab and at the Studsvik site. Clab is mainly intended for interim storage of spent nuclear fuel, but storage canisters with long-lived operational waste (control rods from BWRs and other core components) are also interim-stored in the pools.

Long-lived waste produced during decommissioning of the first reactors is deemed to be possible to accommodate in existing interim storage facilities at the power plants or other sites.

Forsmarks Kraftgrupp AB currently operates an interim storage facility in a building at the power plant site, where both short-lived and long-lived waste produced in conjunction with maintenance outages and power uprates is stored. The long-lived waste consists of segmented reactor internals and has been placed in steel tanks. Today, about 20 steel tanks with long-lived waste are interim-stored and will later be disposed of in SFL.

OKG Aktiebolag operates an interim storage facility for long-lived waste in a special rock vault 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. Waste from the Oskarshamn nuclear power plant and Clab is currently stored in BFA. BFA is deemed to have sufficient capacity for the long-lived waste that will be produced during the 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 from decommissioning of Ringhals 1 and Ringhals 2.

Barsebäck Kraft AB operates an interim storage facility in a building at the power plant site, where long-lived waste from Barsebäck 1 and Barsebäck 2 is stored. The waste consists of segmented reactor internals placed in steel tanks prior to interim storage. In order to be able to release Barsebäck Kraft AB's site from regulatory control before SFL is commissioned, the steel tanks are planned to be transported to an external interim storage facility. The transport is planned to be carried out during the period when dismantling and demolition of the reactors are under way.

Final repository for long-lived waste

SKB plans to dispose of the long-lived waste at a relatively large depth. This final repository (SFL) will be the last final repository in the nuclear waste system to be commissioned. The siting of the repository is yet undecided. The storage capacity of SFL will be relatively small compared with SKB's other final repositories. The necessary storage capacity is estimated to be about 16 000 cubic metres.

The design of the repository is in an early stage. SKB has developed a repository concept that includes two waste vaults, one for core components and segmented PWR reactor pressure vessels from the nuclear power plants and one for legacy waste. Post-closure safety for the proposed repository concept for SFL 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 main part of the radioactivity. SKB plans to design the waste vault for core components with an engineered barrier of concrete.

The legacy waste is stored and managed by AB SVAFO and Studsvik Nuclear AB at the Studsvik site. Additional waste will arise from Cyclife Sweden AB and other Swedish research, industry and medical care. SKB proposes that the engineered barrier in this waste vault will consist of bentonite and concrete. The repository concept is illustrated in Figure 2-5.

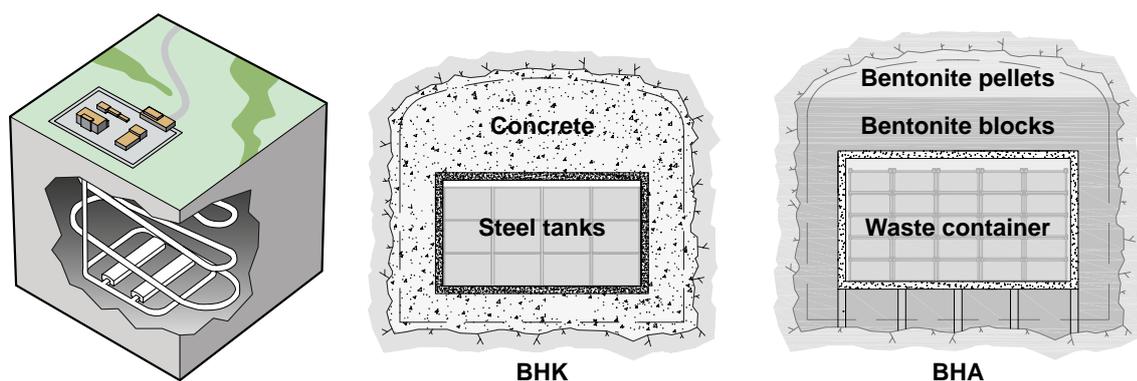


Figure 2-5. Preliminary facility layout (left) and the proposed repository concept for SFL with one waste vault for core components (BHK) and one waste vault for legacy waste (BHA).

2.2 Facilities in the KBS-3 system

SKB's central interim storage facility for spent nuclear fuel in Oskarshamn, Clab, has been in operation since 1985. SKB has applied for licences to build an encapsulation plant for nuclear fuel adjacent to Clab and a final repository, the Spent Fuel Repository, next to Forsmark nuclear power plant. In addition to these facilities, SKB plans to build a facility for machining, assembly and quality assurance of the copper canisters.

Central Interim Storage Facility for Spent Nuclear Fuel

Clab is located adjacent to the Oskarshamn nuclear power plant. The facility consists of a receiving section at ground level and a storage section more than 30 metres below the ground surface. In the receiving section, the transport casks with spent nuclear fuel are received and unloaded under water. The fuel is then placed in storage canisters. The canisters are transported in a fuel elevator down to the storage section where the spent nuclear fuel is interim-stored in water-filled pools, (Figure 2-6).

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

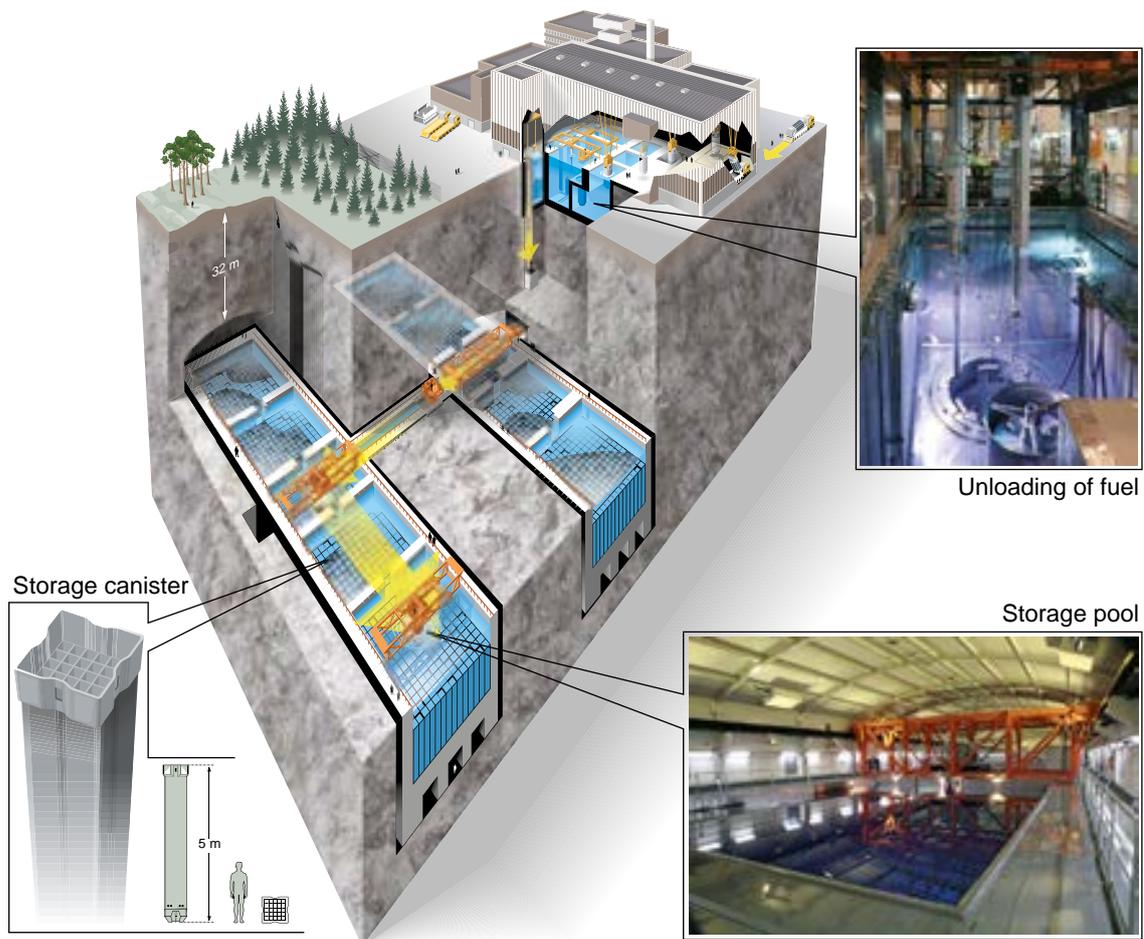


Figure 2-6. The central interim storage facility for spent nuclear fuel, Clab.

Clab has been in operation for more than 30 years and system upgrades and component replacements will be necessary in the future. A number of projects are in progress or have recently been completed, including an upgrade of the cooling chain in order to increase the cooling capacity, replacement of the fire alarm system and modifications in the facility to be able to receive a new type of transport cask.

At the end of 2018, there were 7 002 tonnes of fuel (counted as the original quantity of uranium) in the facility. SKB has a licence to store 8 000 tonnes of fuel in Clab. According to the current forecasts, this storage capacity is projected to be reached in 2023/2024. The pools can accommodate a total of about 11 000 tonnes of fuel provided that the core components that are stored in Clab today are removed from the facility and stored at another site. In 2015, SKB applied for a licence under the Nuclear Activities Act and the Environmental Code to extend the licensed storage capacity to 11 000 tonnes of spent nuclear fuel. This application is processed as a part of the applications for the entire KBS-3 system. The application in its entirety is now with the Government for a decision on permissibility and licences.

Central facility for interim storage and encapsulation of spent nuclear fuel

Before the spent nuclear fuel is disposed of, it will be encapsulated in copper canisters. SKB plans to do this in a new facility adjacent to Clab (Figure 2-7). When this encapsulation plant has been connected with Clab, the two facility sections will be operated as an integrated facility, Clink.

The canister that will be used consists of a copper shell and an insert of nodular cast iron (Figure 2-8). Two types of inserts are required, one that holds twelve fuel assemblies from BWRs and one that holds four fuel assemblies from PWRs. There are also other fuel types to be disposed of, see Section 1.1.4. They can be placed in any of the two insert types.

The canister components, insert, copper shell and lid will be produced by different subcontractors. SKB will need a facility for final machining, assembly and inspection of the canister components. The canister factory will not be a nuclear facility.

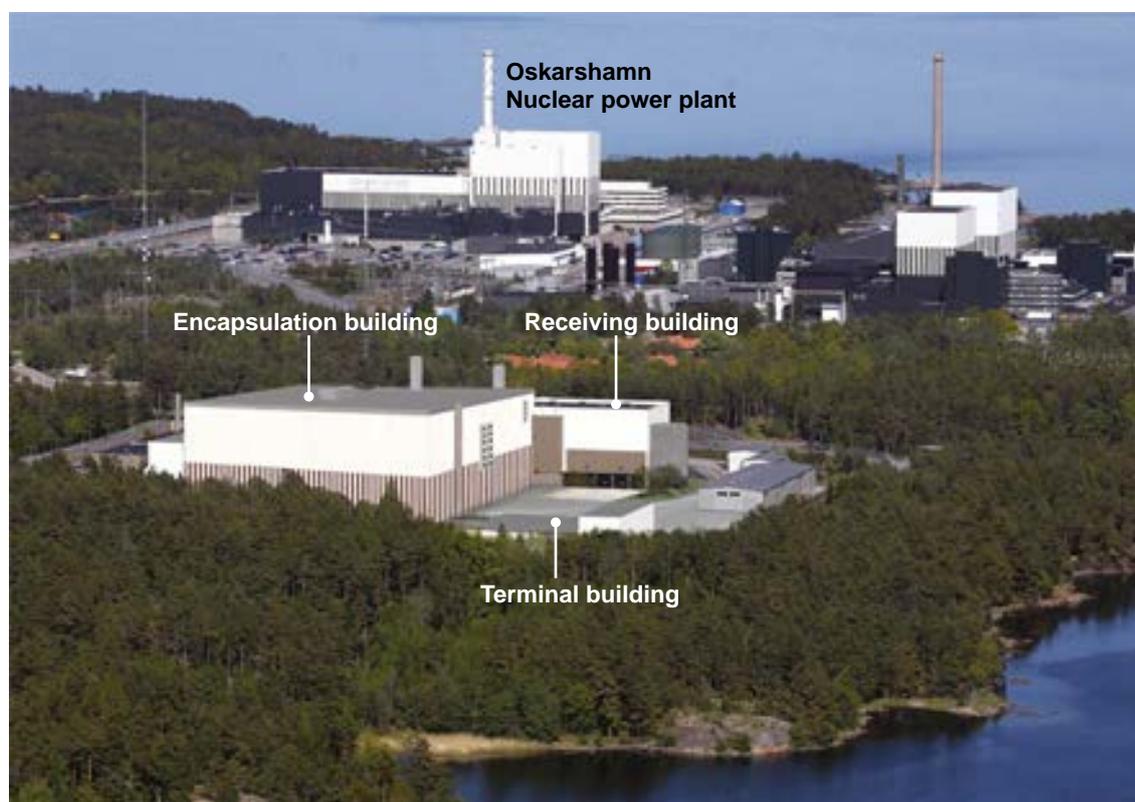


Figure 2-7. Photo-montage demonstrating the integrated facility for interim storage and encapsulation of spent nuclear fuel, Clink.

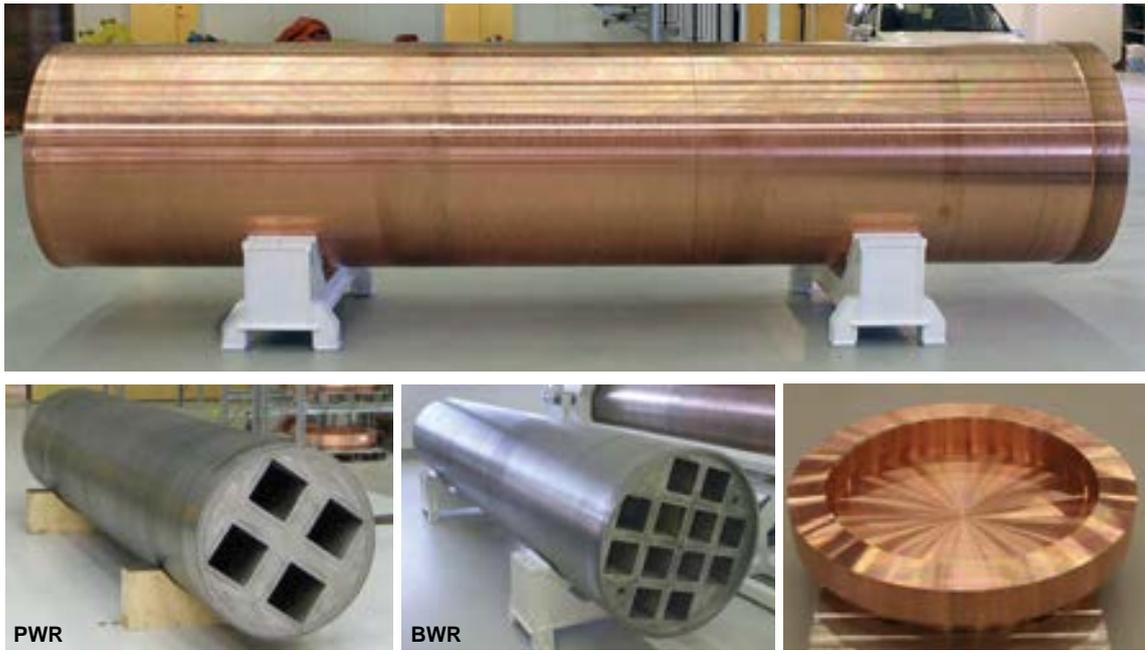


Figure 2-8. Copper canister with inserts of nodular cast iron for PWRs and BWRs and a copper lid. The canister's length, diameter and copper thickness are 4.835 metres, 1.05 metres and 0.05 metres, respectively. A canister with a BWR insert weighs in total about 24.6 tonnes, of which the copper weighs 7.4 tonnes, the insert 13.6 tonnes and the fuel assemblies 3.6 tonnes.

In the encapsulation plant, there will be a number of stations for different operations where all handling of the fuel is remotely operated and radiation shielded. The encapsulation process begins with the fuel being placed in a transport cask and taken up in the fuel elevator from the underground storage pools.

The fuel assemblies to be placed together in a canister are selected in such a way that the total decay heat in the canister will not be too high. The selected fuel assemblies are dried in a radiation-shielded handling cell and lifted over to the canister. The air in the canister is replaced with argon before the canister is sealed. The sealing of the copper canister is done by friction stir welding (FSW). Inspection is made of both the weld surface and that the weld parameters have been kept within applicable limits. If the weld is approved, the canister is taken to the machining station where the canister is machined to its final dimensions. In the next step the canister is moved to the next station where the seal weld is inspected by non-destructive testing. If necessary, the canister is cleaned before being placed in a special transport cask for transport to the Spent Fuel Repository. Clink is designed for encapsulation of 200 canisters per year.

Spent Fuel Repository

Finding a suitable site for a final repository for spent nuclear fuel took several decades. At the end of the site selection process, the choice stood between the Forsmark site in Östhammar Municipality and the Laxemar site in Oskarshamn Municipality. After evaluations of the site investigations, SKB selected Forsmark as the site for the Spent Fuel Repository. The rock at the Forsmark site was considered to provide better conditions post-closure safety. Post-closure safety of the Spent Fuel Repository is based primarily on complete *containment* of the nuclear fuel in copper canisters and secondarily on *retardation* of radionuclides in the surrounding clay and rock barriers, in case of canister failure.

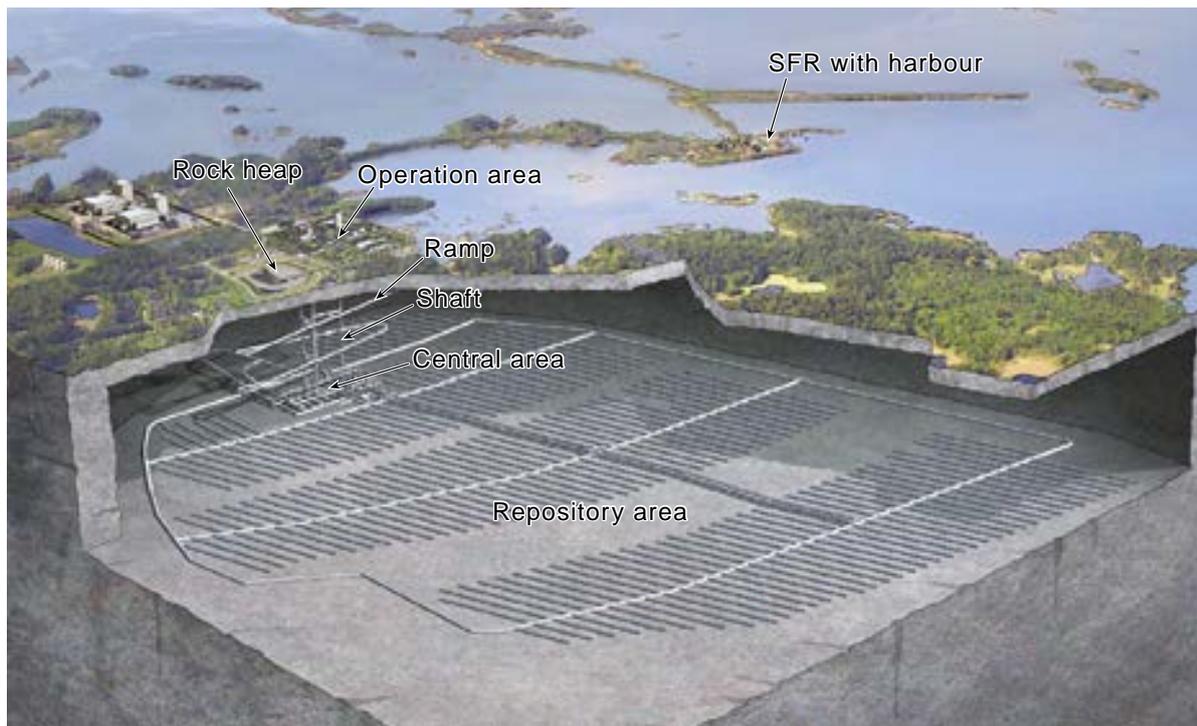


Figure 2-9. Illustration of preliminary layout of the Spent Fuel Repository in Forsmark.

The final repository will consist of a surface facility and an underground facility (Figure 2-9). The underground facility consists of a central area and several deposition areas. Furthermore, there are connections to the surface facility in the form of a ramp for vehicle transport and shafts for elevators and ventilation. The deposition areas, which together constitute the repository area, will be located about 470 metres below the ground level. Each area consists of a large number of deposition tunnels with bored deposition holes in the bottom. The position of the deposition tunnels, as well as the spacing between deposition holes and the design of infrastructure at repository depth, are determined on the basis of rock properties. Important properties are, for example, the location of large deformation zones, the presence of large or highly water-conducting fractures and the thermal conductivity of the rock. The surface facility consists of an operations area, rock heap, possible ventilation stations and storerooms.

The facility is designed for a total quantity of spent nuclear fuel corresponding to approximately 6000 canisters with a deposition capacity of 200 canisters per year. The canisters are transported to the deposition level with a transport vehicle via the ramp. There the canisters are transferred to the deposition machine to be transported to the deposition area and finally deposited. The canisters are placed in the deposition holes, surrounded by bentonite clay. When all canisters in the tunnel have been deposited, 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, other openings are also backfilled and the surface facilities are decommissioned.

2.3 Transportation system

SKB's transportation system was built up during the 1980s. It consists of the ship m/s Sigrid, special vehicles for overland transport and different types of transport casks and containers for fuel and radioactive waste. The ship and the vehicles are used for transportation of both low-level and intermediate-level waste and spent nuclear fuel. The different transport casks and containers are developed specifically for the waste they are intended for.

M/s Sigrid was commissioned in 2014. She replaced m/s Sigyn, which was used for transportation for about 30 years. Like the old ship, the new ship has a double bottom and a double hull. This design protects the cargo in the event of grounding or collision. Typically, the ship makes between 30 and 40 trips per year between the nuclear power plants, Studsvik, SFR and Clab.

Short-lived low-level and intermediate-level waste is shipped from the nuclear power plants, Clab and Studsvik to SFR. Low-level waste does not need any radiation shielding and can therefore be transported in ISO containers. Intermediate-level waste, on the other hand, requires radiation shielding, and most is embedded in concrete or bitumen at the nuclear power plants. The waste is transported in transport casks with 7–20 cm thick walls of steel, depending on how radioactive the waste is (Figure 2-10).

Today part of the long-lived waste, control rods from BWRs, is transported from the nuclear power plants to Clab. They are shipped in a transport cask with approximately 30 cm thick walls of steel. Also the spent nuclear fuel is shipped from the nuclear power plants to Clab in casks with approximately 30 cm thick steel walls. These casks are also equipped with cooling fins to remove the decay heat generated by the fuel.

2.4 Safeguards

Safeguards aim to ensure that nuclear material and nuclear facilities are not used for the production of for example nuclear weapons. Safeguards are a consequence of the fact that Sweden has acceded to the non-proliferation treaty (NPT). Sweden has, through the treaty, committed to submitting its entire nuclear energy programme to international inspection and to implementing a system for nuclear safeguards. In Sweden, nuclear safeguards managed by the International Atomic Energy Agency (IAEA), Euratom and SSM. The safeguards are regulated both nationally and internationally and apply to all facilities where nuclear material is handled.

SKB's facilities must comply with the requirements on 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 shall be reported and it must not be possible to conduct any unauthorised activities in the facilities.

Safeguards are in place at Clab since many years according to established procedures, SSM's regulations and European Commission regulations.



Figure 2-10. M/s Sigrid and current transport casks for short-lived radioactive waste (ATB), for core components (TK) and spent fuel assemblies (TB).

When the entire KBS-3 system is commissioned, all constituent facilities and the transportation system will be covered by an integrated system for 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, to ensure that it is encapsulated and ultimately placed in a final repository.

In addition to spent nuclear fuel, some waste from the Swedish historical nuclear facilities containing small quantities of nuclear material will be disposed of in SFL. Also this nuclear material will be covered by safeguards. Today waste is stored in an interim storage facility for low-level and intermediate-level waste (AM, Active interim storage facility) at the Studsvik site where AB SVAFO is the licensee.

The regulatory authorities and inspection bodies have set up regulatory frameworks for safeguards, but these regulatory frameworks are not fully developed for the encapsulation plants and geological final repositories. SKB has been participating in international collaborations for several years with the IAEA, the European Commission and SSM, among others, in order to contribute to the development of principles and methods for safeguards in these new types of facilities. This collaboration continues in different groups within the different bodies. Preparation of new regulatory frameworks and recommendations are under way, and there are proposals for how safeguards can be managed in an efficient manner.

It is important to develop the technical design in the facilities and the transportation system in parallel so that it is adapted for safe and efficient safeguards during the operational phase, see Sections 3.3.4 and 3.4.5. SKB will not be able to design methods for safeguards after closure of the repositories until the operational and decommissioning phases. SKB believes that the facilities can be designed and activities conducted so that it does not impede the implementation of safeguards after closure.

3 Plan of action

This chapter presents planning for constructing and commissioning new and extended facilities. Furthermore, the chapter describes the reactor owners' and SKB's plans of action regarding decommissioning of nuclear facilities.

The chapter begins with a brief presentation of the plans for implementing the nuclear waste programme according to the current timeplan. Thereafter, alternative courses of action and measures to handle major changes in the planning prerequisites are described.

3.1 Timeplan for the nuclear waste programme

SKB's planning for new facilities involves a stepwise decision process that is based on SSM's regulations. The regulations state, based on international recommendations from e.g. the IAEA and OECD/NEA, that development and licensing of nuclear facilities shall take place through a process in which the requirements on the facility, its design and technical solutions are gradually established. Planning for new facilities is based on the different licences and consents that are required according to this stepwise process and the steps constitute milestones. The most important milestones, which are common to all the planned facilities, are:

- **A licence under the Nuclear Activities Act and permissibility under the Swedish Environmental Code to construct, own and operate a new nuclear facility** – on the basis of a preparatory PSAR (F-PSAR) and an Environmental Impact Statement (EIS). The F-PSAR presents the requirements on safety and radiation protection in the facility and the operations and how they can be fulfilled. During licensing, the applications are reviewed by SSM and the Land and Environment Court. The concerned municipality must approve the activities. Decisions on permissibility under the Swedish Environmental Code and licences under the Nuclear Activities Act are declared by the Government. The Land and the Environment Court thereafter issues licences and conditions according to the Environmental Code.
- **Approval to begin construction** – when a licence has been obtained under the Nuclear Activities Act, an application for construction must be submitted to SSM for approval to begin construction. The application provides SSM with the information that is needed for the Authority to determine whether the designed facility and activities in the construction and operational phase will comply with SSM's regulations and decided licence conditions. For a final repository, a report on post-closure safety is also included. The application for construction must contain the following documents:
 - The application (main document), describing the licensing matter.
 - A description of quality assurance of the application documents.
 - Development and modifications since the previous step in the licensing process.
 - A preliminary safety analysis report (PSAR) that gives an account of the design of the facility, how the operations are organised and how the requirements are fulfilled. The PSAR must be approved by SSM.
 - A safety during construction (Suus) report, which describes how construction will be carried out, including how the activities will be conducted, in order to address the safety and quality issues during construction that are of importance for the operational and post-closure phase.
 - Other documents request by SSM, for example plans for research and development in future steps of the licensing process and an investigation programme for the Spent Fuel Repository.
 - A decommissioning plan.
- **Approval of the safety analysis report (SAR) prior to trial operation and regular operation** – based on continuous presentations of an updated and supplemented safety analysis report (SAR). The safety analysis report is to comprehensively show how the safety of the facility will be achieved and to describe the facility as constructed, analysed and verified, and to demonstrate how the requirements on its construction, function, organisation and operation are fulfilled. The SAR must be approved by SSM.

- **Approval of the SAR prior to closure of the repositories** – based on a revised safety analysis report (SAR) and a plan for closure and decommissioning. This report must also be approved by SSM.

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

The above milestones consist of two parts: firstly SKB's preparation of applications and/or safety analysis reports (SARs), and secondly SSM's and/or other regulatory authorities' approval of these after completed review.

Figure 3-1 shows the general timeplan, including times for coming applications, for the nuclear waste programme. The plans for near-surface repositories are illustrated in Figure 3-6.

3.2 Plan of action for very low-level waste

Facilities for management of very low-level waste currently exist, but in the decommissioning of the nuclear power reactors, the waste volumes that are to be handled will increase substantially. The future decommissioning has made the issue of handling these larger waste volumes pressing. SSM has expressed an expectation of a more specific report on very low-level waste compared with previous RD&D programmes. The RD&D Programme 2019 provides therefore a more detailed description than in previous RD&D programmes of the handling and planning for the part of the low-level waste with very low activity.

3.2.1 Current situation

The very low-level waste produced during operation originates from material that is collected in the controlled area⁸ at the nuclear power plants. After waste separation, some of the material can be released from regulatory control and the fractions are handled in different ways. Waste that cannot be released from regulatory control is deposited in near-surface repositories at the nuclear power plants in Forsmark, Oskarshamn and Ringhals or in SFR. Figure 3-2 shows how 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. The largest volume, however, consists of soft waste.

Metals can often be decontaminated directly at the nuclear power plants with simple techniques where the waste is wiped off, rinsed off or blasted and then released from regulatory control. Metal fractions with a more complex geometry can be melted since the melting process both homogenises and decontaminates the waste. If the total radionuclide content in the molten material is sufficiently low, the waste may be released from regulatory control, otherwise it is deposited in a near-surface repository.

Certain metal fractions are suited neither for decontamination at the nuclear power plant nor for melting. They may consist of components with complex structures and mixed materials, for example galvanised material whose zinc content causes problems with the work environment in the melting plant. In cases where there are no effective treatment alternatives, metals are disposed in near-surface repositories after potential compaction or segmentation to reduce their volume (Figure 3-3).

Very low-level waste in the form of concrete, sand, soil and sludge (inert material) is also difficult to decontaminate and is used, as far as possible, to fill voids in waste packages. An example is shown in Figure 3-3, where sand from blasting is used as top fill in a container with other very low-level waste.

⁸ A building or site where operations are conducted shall constitute a controlled area if an employee may receive such annual doses that the effective dose exceeds 6 mSv or if radioactive contamination of significance with respect to radiation protection may spread to surrounding buildings or work sites (SSMFS 2018:1, Chapter 4, Section 3).

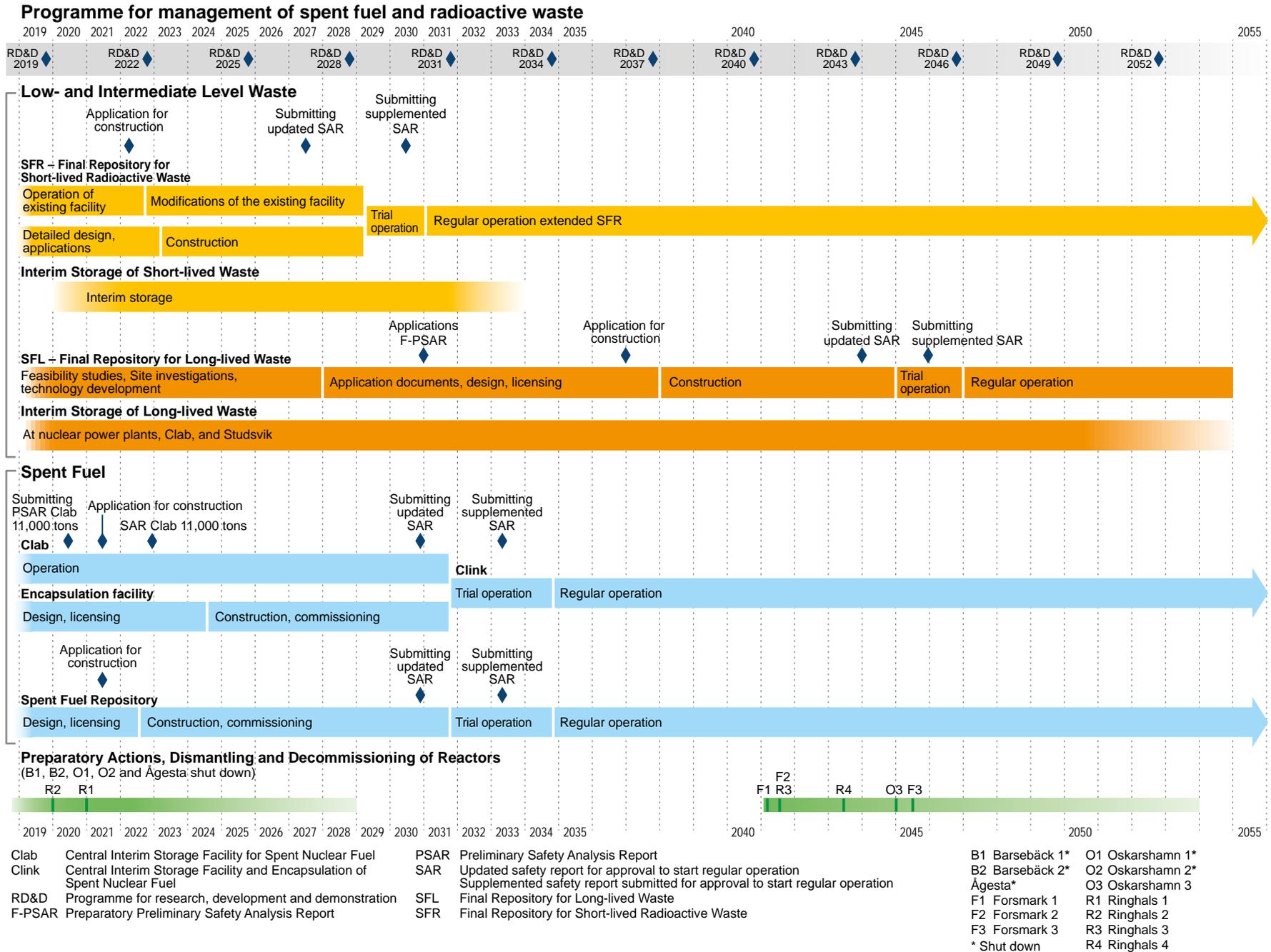


Figure 3-1. General timeplan for SKB's Programme for management of spent fuel and radioactive waste and decommissioning plans for the nuclear power reactors.

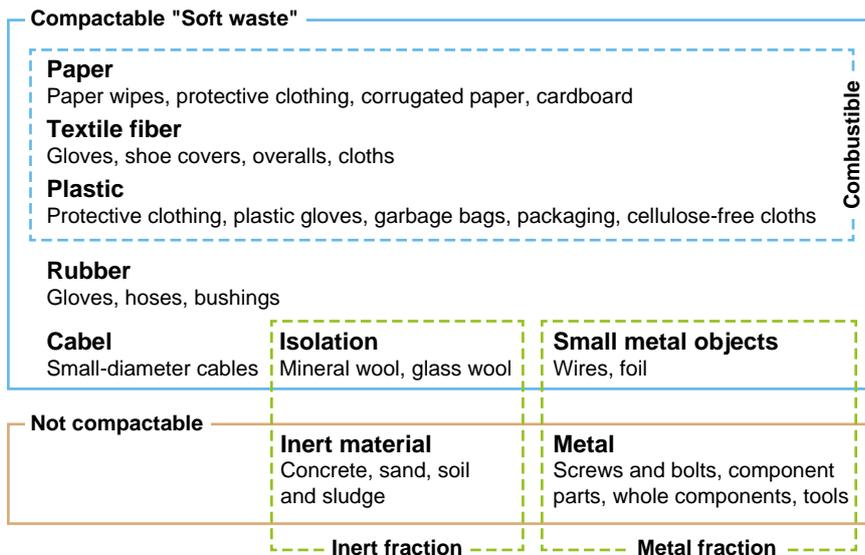


Figure 3-2. Division of very low-level waste into different fractions.



Figure 3-3. Left: metal fraction to a near-surface repository packed in a half-height ISO container; right: top filling of blasting sand to utilise the volume and minimise the void (fitting of lids is done before deposition).

Soft materials can be difficult to decontaminate. Nearly two thirds of the very low-level waste produced during operation consist of plastics in the form of decontamination cloths, gloves or other types of protection for limiting the spread of contamination. A quarter of the waste consists of cellulose or textile, such as decontamination cloths and protective overalls. The remaining part, just below ten percent, consists of insulation, metal foils and cables.

The treatment of soft materials at waste producers with near-surface repositories is based on the compaction of waste bales with the objective of optimising the use of the near-surface repository with respect to volume. The waste bales are packaged in plastic foil (Figure 3-4). Some of the soft material is combustible (Figure 3-2) and can be burned in a facility for controlled incineration of radioactive material at the Studsvik site. The assessments of the best available technology carried out in conjunction with the licensing processes linked to the current near-surface repositories for operational waste indicate, however, that near-surface repositories are the best available technology for large parts of the very low-level waste. The assessment is made with regard to both radiation protection aspects and financial and environmental aspects. The possibility of incinerating waste is thus mainly used by the waste producers that have no near-surface repositories. Other producers use incineration only when the waste for some reason is less suited for near-surface repositories.



Figure 3-4. Left: bale press for compaction of soft waste; right: compacted waste bale.

Waste producers without access to near-surface repositories use external or local treatment of the waste prior to disposal in the waste vault for low-level waste (BLA) in SFR. These waste producers produce very small volumes of waste during operation.

At the nuclear power plants, there are dedicated areas and buildings for storage of very low-level waste. These have been established so that disposal in near-surface repositories can be conducted on separate occasions, which improves safety and provides better financial conditions for operation of near-surface repositories.

3.2.2 Overall planning

During a normal operation year, 50–100 tonnes of very low-level waste, per reactor, is disposed of in near-surface repositories. Decommissioning of a reactor roughly generates 250–500 tonnes of very low-level waste per year during a period of about 10 years. 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.

During decommissioning of nuclear facilities, dismantling of systems and components takes place in the order that provides an effective waste management. This means that the waste will have a more homogeneous composition in terms of radioactivity than the corresponding operational waste. Furthermore, the dismantling takes place in a facility where some of the most contaminated systems are decontaminated and where short-lived radionuclides have often decayed during a couple of years. The very low-level decommissioning waste will thereby be more predictable than the operational waste that is produced at the corresponding place in the facility. Figure 3-5 summarises the sources, mainly consumable supplies and facility components, of the very low-level waste.

In order to handle the large volumes of very low-level waste that are produced during dismantling and decommissioning of the nuclear power reactors, the waste producers have identified a need for near-surface repositories for disposal of large parts of this waste.

The current near-surface repositories are only licensed for operational waste. At present, Ringhals has an ongoing application for a licence to extend their near-surface repository for operational waste. When it comes to the decommissioning waste, feasibility studies are under way for analysing the possibility of extending the near-surface repositories at the Oskarshamn and Ringhals nuclear power plants. The extended repositories would then be licensed for both operational and decommissioning waste.

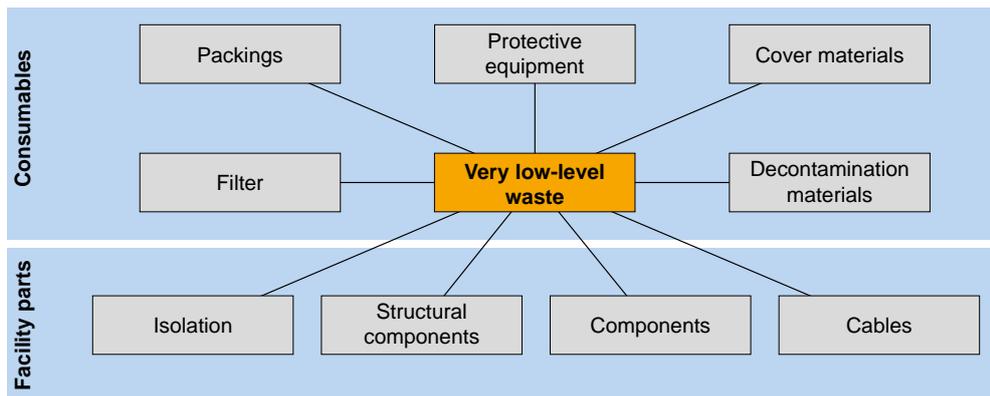


Figure 3-5. Sources of very low-level waste distributed over consumable supplies and facility components.

Timeplans for planned extensions/new establishments of near-surface repositories are presented in Figure 3-6.

Alternative to near-surface repositories is disposal of the very low-level waste in the waste vault for low-level waste (BLA) in SFR. BLA is more advanced than a near-surface repository and can receive waste with higher activity content than the very low-level waste. Another alternative that is being studied by the waste producers is if parts of the waste can be incinerated in conventional waste incineration facilities. A prerequisite for the procedure is that the residual products that are generated can be released from regulatory control.

A meeting series between SSM and the waste producers was initiated in 2017 with the purpose of creating a common platform for discussing issues related to the management and disposal of very low-level waste. The dialogue has been broadened gradually and more actors that potentially could receive parts of the waste have been involved (Cyclife Sweden AB, different incineration facilities and the industry organisation for conventional waste management, Avfall Sweden).

The planned initiatives for management of the very low-level waste from operation and decommissioning can presently be summarised in the following points:

- Continuous work at the waste producers to reduce the amount of waste, for example by better recycling, more efficient sorting of material, unpacking and repacking of incoming goods and zoning of work areas during maintenance outages.
- Follow-up of the application to expand the existing near-surface repository at the Ringhals power plant and follow-up of an updated assessment of the best available technology for Svalören near-surface repository at the Forsmark power plant.
- Expansion of the near-surface repository at the Oskarshamn Nuclear Power Plant to accommodate the remaining operation and decommissioning waste. The possibility to dispose of waste from the Barsebäck Nuclear Power Plant in Oskarshamns Nuclear Power Plant's near-surface repository is being studied (Section 16.3).
- A review of the possibility to establish a new surface repository for decommissioning waste at the Ringhals Nuclear Power Plant.
- Extension of the SFR facility, where parts of the very low-level waste can be disposed of.

3.2.3 Treatment of very low-level waste

The activity level of the very low-level waste is between material cleared from regulatory control, i.e. material returned to the conventional cycle, and material that requires final disposal due to its activity content.

The set of requirements for very low-level waste has changed over time due to, among other things, a different view of recycling, an updated set of requirements related to radiation risks, and technical progress regarding new handling options. There is therefore a need for a consensus between waste producers and SSM regarding how the set of requirements for very low-level waste is to be interpreted. The reactor owners will gather information in collaboration to shed light on the bigger picture when it comes to treatment and other handling of very low-level waste. The goal is a robust and flexible system that is sanctioned by current regulatory authorities.

Overall plan for deposition of very low level radioactive waste

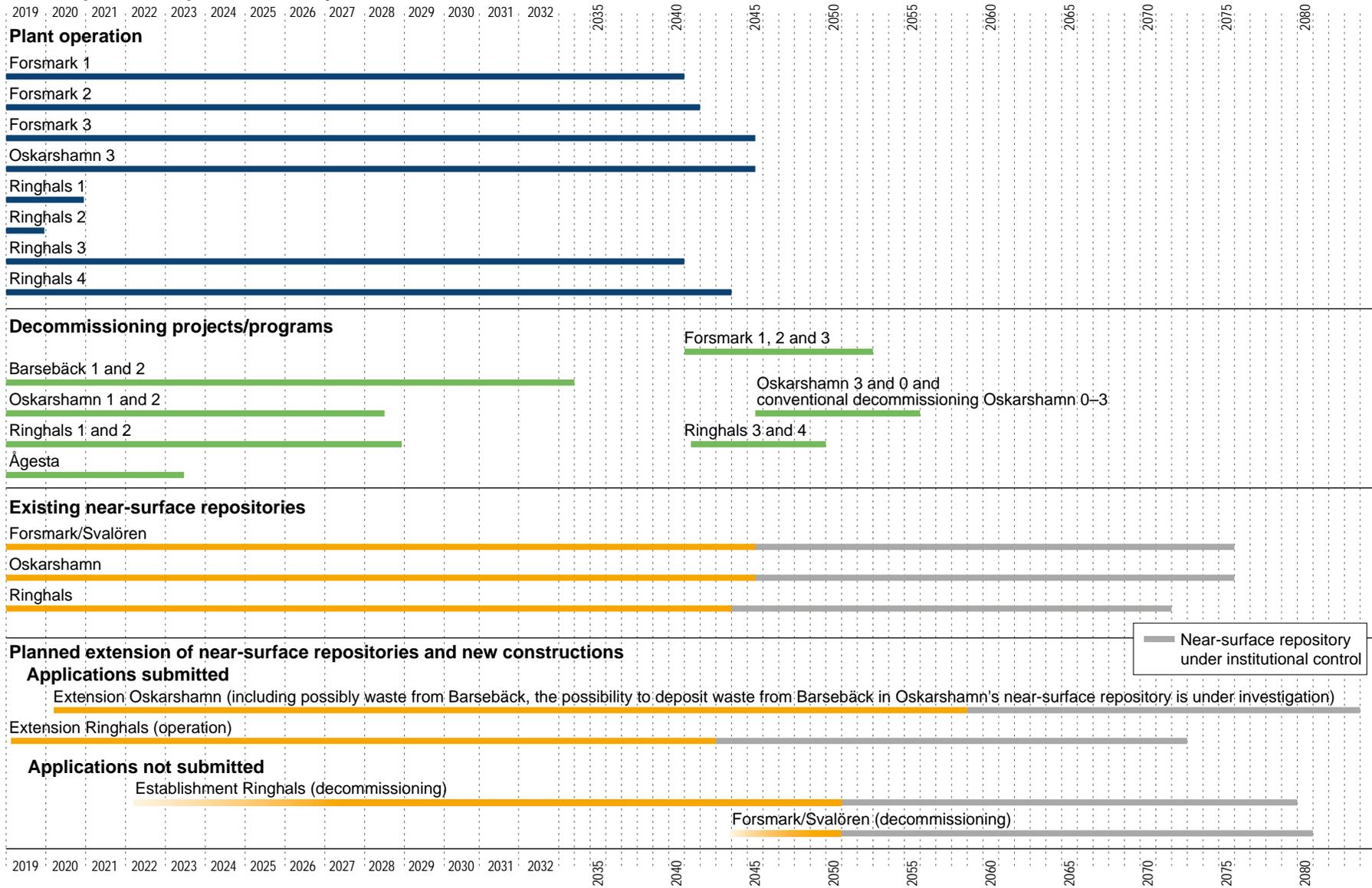


Figure 3-6. Overall plan for disposal of very low-level waste in existing and planned expansions/new establishments of near-surface repositories. The times specified in the figure are schematic, for exact times, the reader is referred to the planning for each facility. Oskarshamn 0 refers to several shared service facilities.

In this context, it should be noted that a changed treatment and handling of the very low-level waste also has an impact on SFR, since parts of the very low-level waste are currently included in the volumes that are forecasted for 2-5BLA, see further in Section 3.7.3.

During the RD&D period, the waste producers will work on developing different aspects connected to the production and treatment of very low-level waste.

- Modified/extended sorting could potentially be a part of improved management of very low-level waste. In order to be able to evaluate what it would entail, consequences in the form of for example investments, dose budget and educational needs will be studied.
- Techniques for decontaminating very low-level waste to material ready for clearance are improved over time. This fact, together with the increasing need for effective clearance in connection with the decommissioning of the early commercial nuclear power reactors entails a need for development within the area of clearance. Developed clearance is therefore considered an important step for improving the management of very low-level waste. An industry-wide group has been initiated, for this work, with the purpose of developing different aspects of the clearance process. The conclusions from the group will serve as a basis for strategic choices and for investment decisions.
- Tests for replacing disposable plastic protection (plastic cover) with protective plates or reusable plastic protection (similar to tarpaulin) have been carried out in order to limit the amount of waste that is produced, to reduce the use of plastics and to increase the degree of reuse. The work will continue during the RD&D period.
- The waste producers are continuously developing the clearance process by reviewing resource and competence requirements, processes and procedures for sorting/separation and prerequisites in the form of buildings, equipment and methodology development. The waste producers are continuously exchanging experiences. Development work for making measurement techniques used for clearance of soft waste fractions more effective will be carried out.
- By estimating waste quantities in different fractions, it becomes possible to relate them to potential environmental benefits, risks and costs for introducing specific measures. Knowledge of waste quantities in different fractions is also a prerequisite for measuring and following up the effect of measures taken. The waste producers therefore intend to work on updating and refining existing forecasts for both operational waste and decommissioning waste.

3.2.4 Interim storage and temporary storage of very low-level waste

The reactor owners that have access to near-surface repositories also have a system for temporary storage of waste since disposal in the near-surface repositories is carried out episodically. Existing temporary storage areas have sufficient capacity to accommodate the waste volume required for efficient handling when disposal takes place. This is equivalent to the waste quantity produced during five to ten normal operation seasons for the reactors at each facility.

Disposal in near-surface repositories will continue to be done episodically. The capacity of the existing temporary storage areas are therefore expected to be sufficient for the waste that is produced during the continued operation of all nuclear power plants. Weather-protected temporary storage areas may be necessary at the sites where decommissioning projects will begin in the near future for handling the large quantity of waste that is expected to be produced.

Figure 3-7 shows the temporary storage area at the Ringhals nuclear power plant. It consists of a robust weather protection, which ensures that the waste and packaging retains their integrity until disposal in the near-surface repository.

Waste producers without access to near-surface repositories are planning for interim storage of the very low-level waste together with other short-lived low-level waste (Section 3.3.3).



Figure 3-7. Oves Lada, temporary storage area for, among other things, very low-level waste at Ringhals nuclear power plant.

3.2.5 Near-surface repositories

Near-surface repositories are an important part of the current system for disposal of waste whose activity content means that it cannot be returned to society's conventional cycle, but where radiation protection aspects do not warrant disposal in a geological repository. Licensing for disposal in near-surface repositories and for extension of existing near-surface repositories have addressed issues regarding for example the mechanical stability of the repositories and which fractions (e.g. combustible waste) that can be disposed of in near-surface repositories and under what conditions.

During the RD&D period, the best available technology for all waste fractions within the category of very low-level waste will be described by the waste producers. This assessment includes a review of issues regarding the ability of near-surface repositories to maintain stability and function over time, as well as further studies of alternative management methods for the different waste fractions. The assessment of the best available technology for the very low-level waste will be kept up-to-date and updated in conjunction with future RD&D programmes. The primary goal is to clarify whether the waste producers and regulatory authorities can agree on the conditions for a broader use of near-surface repositories. In conjunction with this, the part of the very low-level waste that will not be possible to dispose of in a near-surface repository during the remaining operation and decommissioning should be defined.

Since it is unclear at present what the final state of the near-surface repositories will be, development work will be carried out during the RD&D period in order to study legal issues and which regulatory framework will be applied in conjunction with completed institutional control. The focus of the work will be determined in consultation with SSM through continued dialogue in the meeting series between SSM and the waste producers that was initiated in 2017.

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

3.3.1 Current situation

The programme for low-level and intermediate-level waste includes both day-to-day management of existing waste and work to realise the remaining parts of the system required for safe management and final disposal of low-level and intermediate-level waste in the long term. The activities are primarily conducted by SKB, but in some respects also by the reactor owners. The final repositories that SKB plans to establish for low-level and intermediate-level waste include an extension of SFR and construction of SFL.

The current situation for the work with realising the remaining parts of the system for low-level and intermediate-level waste can be summarised in the following points:

- The applications under the Nuclear Activities Act and the Environmental Code to extend SFR were submitted at the end of 2014 and the licensing processes are currently in progress. Supplementary documentation in the licensing matter was submitted in 2015–2017. In December 2017, SSM and the Land and Environment Court at Nacka District Court announced the licence application to construct an extension of the Final Repository for Short-lived Radioactive Waste, SFR, in Forsmark with a new repository part mainly for decommissioning waste. In January 2019, SSM recommended the approval of SKB's application for authorisation under the Environmental Code to extend SFR. The main hearing in the Land and Environment Court takes place in the autumn of 2019.
- In preparation for the extension of SFR, issues concerning licensing dealt with in parallel with continued technology development, design and construction preparations.
- An evaluation of the post-closure safety for the proposed repository concept for SFL has been made during the period 2015–2019. The safety evaluation constitutes an important basis for the continued development and is presented in Chapter 6.
- Studies have been carried out for planning of the siting process that will lead to a selected site for SFL. The starting point for the process is the fundamental requirements on siting in the Environmental Code, the Nuclear Activities Act and the Radiation Protection Act. The siting process will take experience from SKB's previously completed siting processes into account. Planning of the siting process is presented in Chapter 6.

3.3.2 Overall planning

The final repositories that SKB plans to establish for low-level and intermediate-level waste include an extension of SFR and construction of SFL. The owners of the reactors that are decommissioned before the extended SFR facility is commissioned intend to arrange temporary interim storage facilities for the short-lived decommissioning waste. The reactor owners are planning for interim storage of the long-lived decommissioning waste either at the nuclear power plants or at another site.

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

Figure 3-8 shows a general timeplan for management of low-level and intermediate-level waste, together with important milestones. In order to visualise the link to the decommissioning of the reactors, Clink, SFL and SFR are also included in the figure.

3.3.3 Short-lived waste

Interim storage of short-lived waste

The work with dismantling and decommissioning the first seven reactors will be initiated before the extended SFR facility is commissioned. Barsebäck Kraft AB, OKG Aktiebolag and Ringhals AB therefore plan to interim-store the short-lived decommissioning waste primarily at the power plant sites but also other sites may be considered. Interim storage of operational waste will also be necessary during the period when construction of the extended SFR facility is under way, since no disposal will take place in the facility during this period. If disposal is required by the power plants during the construction, a short disposal period could be planned after the completed excavation, i.e. after about half of the total construction time.

Furthermore, one waste vault in the existing SFR, the waste vault for low-level waste, is expected to reach its capacity within a couple of years. This means that additional quantities of this type of waste must be interim-stored at the nuclear power plants until the extended SFR facility is commissioned.

In the RD&D Programme 2016, the reactor owners planned to dispose of reactor pressure vessels from BWRs whole. This has changed and in the current plans, the reactor pressure vessels will be segmented (Chapter 17). The segmented reactor pressure vessels will be interim-stored with other intermediate-level short-lived decommissioning waste.

Low- and Intermediate Level Waste

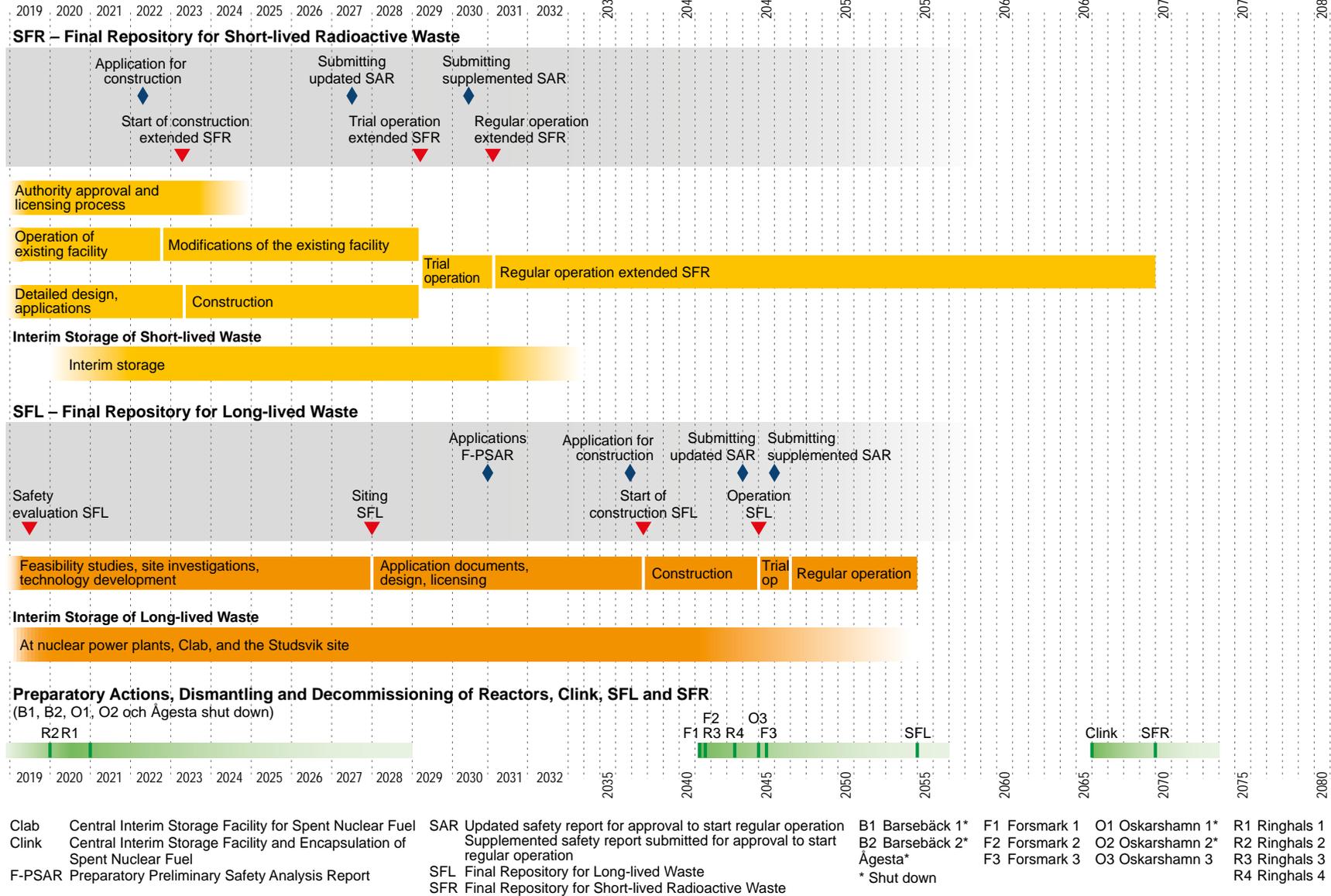


Figure 3-8. Timeplan for low-level and intermediate-level waste and decommissioning of the nuclear power plants.

Barsebäck Kraft AB has existing storage facilities that can be used for interim storage, but the capacity needs to be increased to accommodate the short-lived waste that will be produced during decommissioning of Barsebäck 1 and Barsebäck 2. Barsebäck Kraft AB has investigated the need and the results show that the storage capacity can be increased in their own facility. The new interim storage is planned to be established during the beginning of dismantling and decommissioning of the reactors.

For interim storage of short-lived waste from continued operation of Oskarshamn 3 and decommissioning of Oskarshamn 1 and Oskarshamn 2, produced before the extended SFR facility can be commissioned, OKG Aktiebolag believes that the storage capacity at the facility must be increased.

Ringhals AB has existing storage facilities that can be used for interim storage and the starting point is that the capacity needs to be increased to handle also the short-lived waste from decommissioning of Ringhals 1 and Ringhals 2.

Forsmarks Kraftgrupp AB has existing storage facilities that can be used for interim storage. They are deemed to have sufficient capacity for interim storage of operational waste. This is valid provided that the extension of SFR follows the current timeplan. Since the nuclear power plant will still be in operation when the extended SFR is commissioned there is no need to plan for interim storage of decommissioning waste.

AB SVAFO is planning to construct a new building for interim storage of low-level and intermediate-level waste from their operations. The interim storage facility will be located at the Studsvik site and will be commissioned around 2021. An environmental permit for the interim storage facility was granted by the Land and Environment Court in January 2017. A licence under the Nuclear Activities Act is not required since the interim storage facility is considered to be a facility modification according to SSM. A notification of the facility modification and a preliminary safety analysis report was submitted to SSM in June 2019.

When the extended SFR facility is commissioned, the operational and decommissioning waste will be transported from all interim storage facilities to SFR for final disposal.

Extension of SFR

An application for a licence to extend SFR and also for final disposal of decommissioning waste in the facility was submitted in December 2014. According to SKB's current planning, which has been adapted to the fact that licensing is estimated to take longer than previously assumed, the construction of the extension is expected to start in 2023, with planned trial operation in 2029. A general timeplan for the extension of SFR is presented in Figure 3-9.

Licensing and design of the extended SFR facility

The licensing process for the applications under the Environmental Code and the Nuclear Activities Act is currently under way. The Land and Environment Court at Nacka District Court and SSM are responsible for processing of the applications, after which Östhammar Municipality and the Government will take decisions. The initiative lies largely with these bodies, and the length of the process is dependent on how long they take for processing and decisions. SKB's responsibility is to submit supplementary information as requested and to prepare and participate in the main hearing in the Land and Environment Court. At the same time, SKB continues the development work on the repository barriers and procurement for detailed design. After permits and licences are obtained under the Environmental Code and the Nuclear Activities Act, procurement of contractors is planned and when all licences have become legally binding, the construction of the extended SFR facility will begin.

During the RD&D period, a PSAR will be prepared. It is a part of the application for construction and is submitted to SSM after a Government licence has been obtained under the Nuclear Activities Act and after decisions on permissibility under the Environmental Code.

During the latter part of the licensing processes, building preparations and preparatory investigations are carried out prior to the start of construction. Provided that the licensing process is expeditious, the necessary licences are expected to be obtained so that construction of the extended SFR facility can commence according to the timeplan, in 2023. A licence for construction must be obtained from SSM in order to begin the construction of the extension.

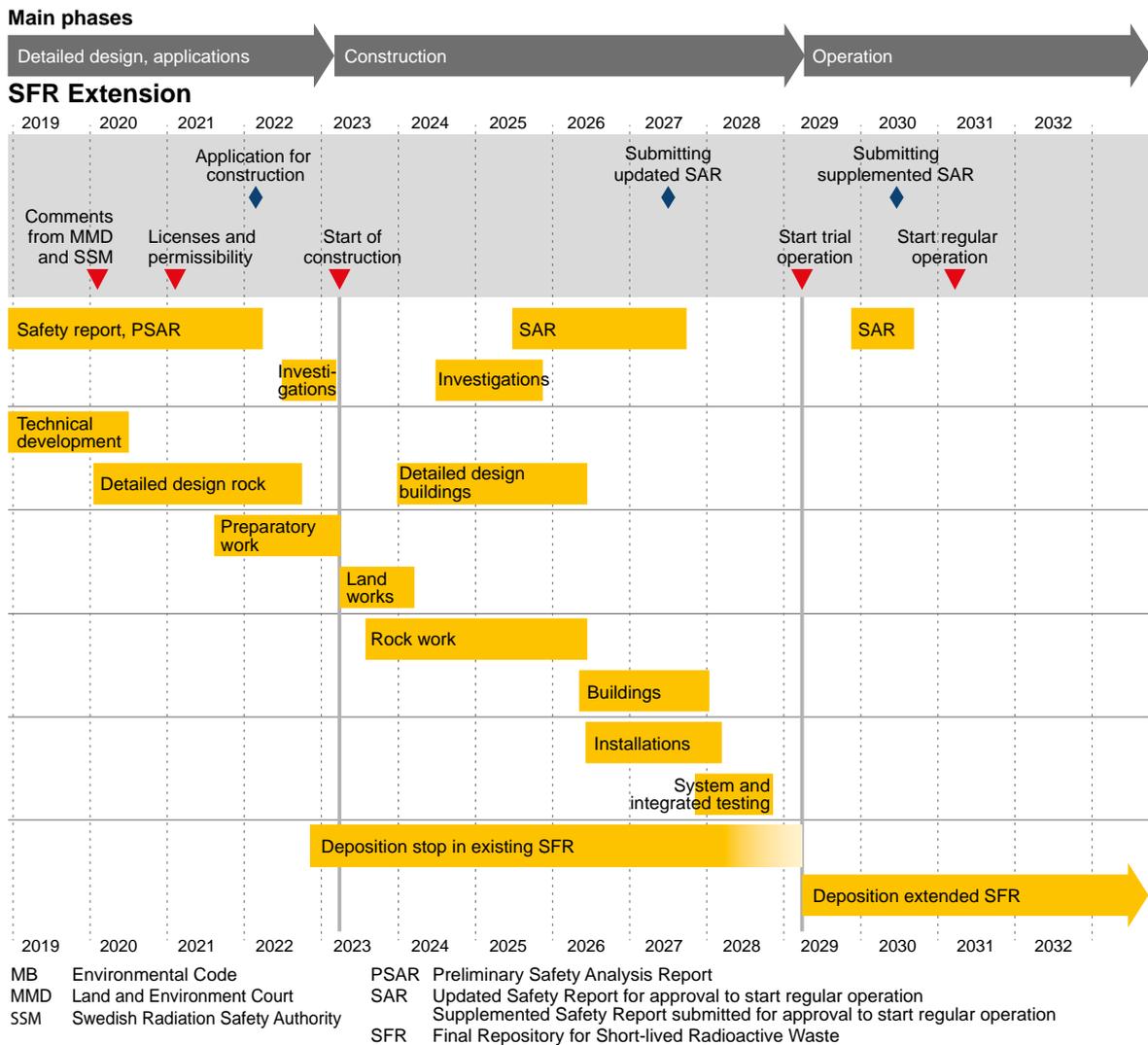


Figure 3-9. General timeplan for extension of SFR.

Construction and commissioning of the extended SFR facility

This phase includes the activities construction, trial operation and handover to regular operation. During some of the building preparations and during construction, SKB will suspend all disposal in the facility. At the same time as SFR is extended, upgrading of the existing facility will be conducted, considering among other things that the operating time has been prolonged in relation to the original plan. SKB plans to submit an updated SAR in 2027. Trial operation with disposal of waste in the extended part of SFR is expected to start about two years later. In 2030, SKB intends to submit a supplemented SAR. According to SKB's assessment, an approval for regular operation can be obtained at the beginning of 2031.

3.3.4 Long-lived waste

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

- Highly neutron-irradiated core components. The waste is produced in connection with both maintenance and dismantling and demolition of reactors.
- Control rods from BWR reactors. Used control rods arise both during operation of the reactors and during dismantling of the final core when the reactors are decommissioned.
- Reactor pressure vessels from PWR reactors. The waste is produced during decommissioning of reactors.

- Long-lived waste from activities at the Studsvik site and from research, medicine and industry. The waste from activities that generate small waste volumes are usually managed by Studsvik Nuclear AB and Cyclife Sweden AB. This waste is produced continuously and is not associated with the operation or decommissioning of the nuclear power plants.
- Legacy waste from research and development in the Swedish nuclear research programmes. Waste from the Swedish Armed Forces, for example night vision sights. This waste is managed and interim-stored by AB SVAFO.

The plans for management and disposal of the long-lived low-level and intermediate-level waste aim at an integrated system for management and final disposal. The planning for the different parts of the system is presented under the respective heading below.

Interim storage of long-lived waste

SKB plans to commission SFL around 2045. Since several reactors will be decommissioned before the final repository is commissioned, long-lived waste from decommissioning will need to be interim-stored. The reactor owners today judge that there is sufficient interim storage capacity at the nuclear power plants.

In order to be able to release Barsebäck Nuclear Power Plant from regulatory control before SFL is commissioned, a new interim storage facility is needed at another site to which the steel tanks that are currently stored at the Nuclear Power Plant can be transported. Barsebäck Kraft AB is considering several different alternative sites for interim storage. The transport is planned to be carried out during the period when decommissioning of the reactors are under way.

AB SVAFO today operates an interim storage facility for low-level and intermediate-level waste (AM, Active interim storage facility) that holds both their own long-lived waste and waste from other licensees, e.g. Studsvik Nuclear AB. This interim storage facility does not have capacity to receive more waste. AB SVAFO is therefore planning to construct a new building for interim storage of low-level and intermediate-level waste. The interim storage facility will be located at the Studsvik site and will be commissioned around 2021.

Existing and planned interim storage facilities are used until it is possible to transport the waste to SFL. For this, in addition to a commissioned SFL, a licensed transport cask is also needed.

Treatment of long-lived waste

Conditioning of the long-lived waste may be necessary before disposal in SFL. Before final conditioning can be carried out, acceptance criteria for the long-lived waste must be formulated. The development of preliminary acceptance criteria for the long-lived waste will continue during the RD&D period based on, among other things, the results of the completed safety evaluation. Two main alternatives for the long-lived waste are included in the current planning. One is stabilisation of metallic waste from the nuclear power plants in steel tanks and the other is transfer and conditioning of waste from AB SVAFO and Studsvik Nuclear AB in waste containers adapted for SFL (Pettersson 2013). Conditioning cannot be carried out until the acceptance criteria have been established. According to current plans, final conditioning of the long-lived waste will be made in conjunction with disposal in SFL, if necessary. The time is also advantageous from a radiation safety perspective, since the waste has decayed for a longer time by then.

The final repository for long-lived waste, SFL

SFL is the repository that SKB is planning to commission last. Up to commissioning, there are several important milestones that must be reached, such as site investigations and site selection, assessment of safety after closure, preparation of licence applications and construction. A general timeplan for the work with SFL is presented in Figure 3-10. The timeplan 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 were required, SKB believes that the time of commissioning of SFL will be postponed. Such a scenario is further described in Section 3.7.4. SKB plans to submit applications under the Nuclear Activities Act and the Environmental Code to construct, own and operate SFL around 2030. According to the planning, the repository will be commissioned around 2045. To provide for the needs of the nuclear power companies, it is judged necessary to keep the repository in operation for about 10 years.

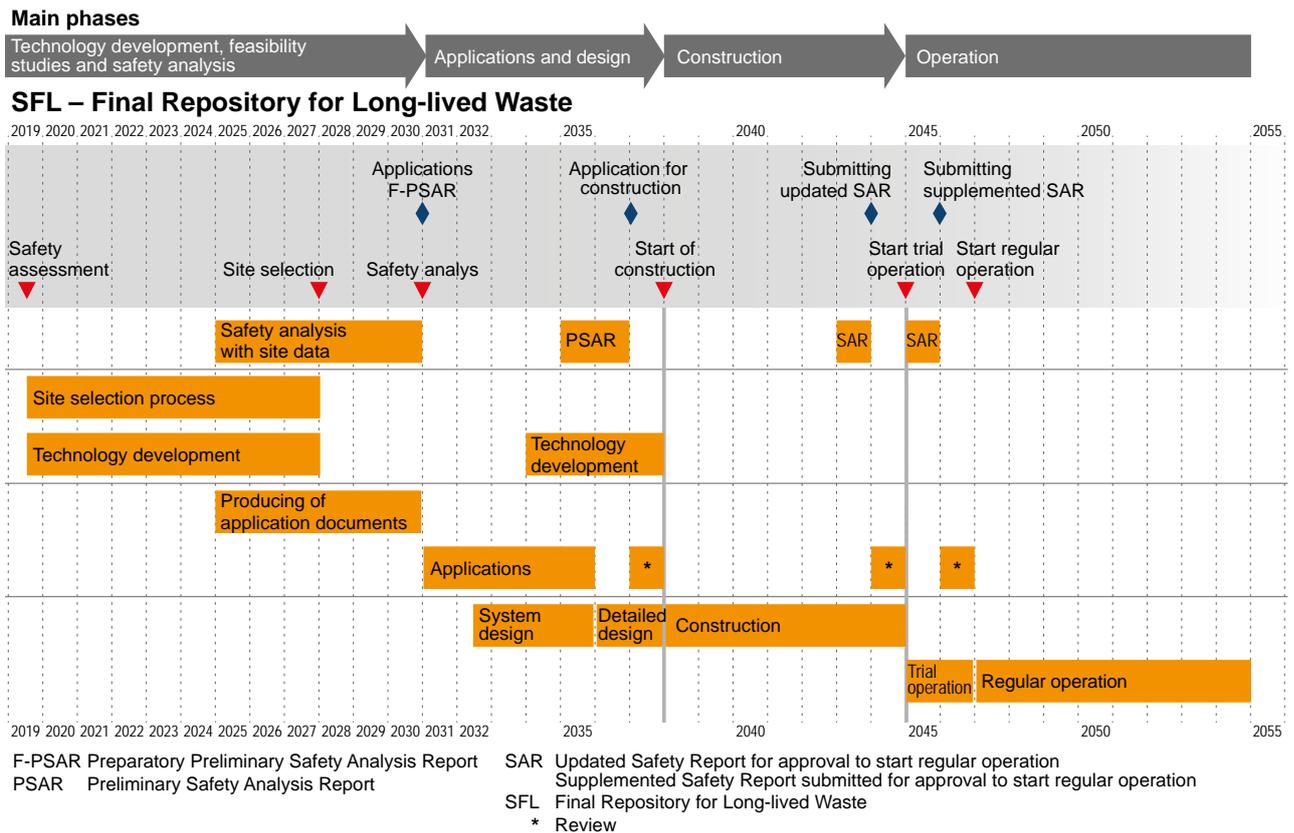


Figure 3-10. Estimated timeplan for the work to commission SFL.

Evaluation of safety after closure of SFL

In 2013, SKB presented a study where different repository designs were evaluated (Elfving et al. 2013). The study describes a proposal for waste packaging, transportation system and facilities for conditioning, interim storage and final disposal of waste. Based on the evaluation, a conceptual repository design for the long-lived waste has been presented, see Section 2.1.2.

During 2015–2019, an evaluation of the proposed concept with respect to post-closure safety was carried out. The safety evaluation was based on the safety assessment methodology that has been developed by SKB during the past decades and that was applied in the most recent safety assessments for the Final Repository for Spent Nuclear Fuel – SR-Site (SKB 2011b) and for SFR – SR-PSU (SKB 2015b). The safety evaluation is, however, not a complete safety assessment. The evaluation is, wherever possible, based on existing knowledge and data gathered in previous site investigations, safety assessments and technology development at SKB.

The results of the safety evaluation indicate that the repository concept has the potential to meet requirements on post-closure safety, given that the future site has sufficiently low groundwater flows to minimise the outward transport of radionuclides from the repository. The results further show which waste categories contribute most to the calculated post-closure radiation safety consequences, yielding a basis for continued work with formulating acceptance criteria for the long-lived waste. The analysed sensitivity cases for different barrier solutions in the safety evaluation form a basis for continued work developing the engineered barriers.

The safety evaluation now comprises the basis for identifying areas for further research prior to future complete safety assessments. Furthermore, the evaluation forms a basis for starting the siting process.

A more detailed account of the safety evaluation’s scope, methodology, results and conclusions is given in Chapter 6.

Siting of SFL

SKB has previously established the basic prerequisites for siting of final repositories for radioactive waste:

- The safety during operation and after closure and the impact on the environment must comply with the requirements in the Nuclear Activities Act and the 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 at the end of the 2020s. The goal is to conduct an open and transparent process in consultation with SSM and the concerned municipalities, where the premises for different stakeholders are clarified early on and where the different steps in the process have been agreed upon and communicated.

The understanding of Sweden's geology that has been gained in SKB's previous siting processes comprises the basis for the work. The basis consists of all areas in Sweden with data related to evaluation of safety at a depth of 400–700 metres, e.g. Forsmark, Simpevarp/Laxemar and all so-called study sites from the siting process for the Spent Fuel Repository. A systematic approach based on a division into siting factors has previously been developed and used by SKB in the siting of both the Spent Fuel Repository and the extension of SFR. This basis and the systematic approach with siting factors, which SKB now has access to and experience of, will be possible to use to a large extent also in the siting process for SFL. As in previous siting processes, SKB will use the following main groups of factors during evaluation and comparisons of siting alternatives.

- Safety after closure.
- Technology for execution.
- Human health and the environment.
- Societal aspects.

The safety evaluation results form the basis for safety-related siting factors. Furthermore, the statements to the Government from SSM and the Land and Environment Court in the KBS-3 matter have constituted important documentation for the design of the process.

Preliminary siting factors and the planned siting process for SFL is presented in Chapter 6.

Applications, construction, operation and closure of SFL

SKB plans to submit licence applications under the Nuclear Activities Act and the Environmental Code for SFL around 2030. After the submission, work will continue with for example basic design and detailed design. Construction and trial operation will be followed by regular operation. Closure of SFL takes place when all interim-stored long-lived waste and the long-lived waste from decommissioning of the last nuclear power plant have been disposed of. Prior to a decision on closure, SKB must ensure that the waste from decommissioning of Clink is suitable for SFR and does not require final disposal in SFL.

Safeguards

There is waste that will be disposed of in SFL, which contains nuclear materials and therefore is subject to safeguards. This legacy waste is currently interim-stored in the rock vault AM in Studsvik for which AB SVAFO is the licensee. The basic principles for safeguards in connection with final disposal of nuclear material from the legacy waste are different compared with the principles for final disposal of spent nuclear fuel. A significant difference is that the nuclear materials cannot be counted and handled as separate units in the same way as nuclear fuel assemblies, since it occurs in different forms. This means that it will not be possible to verify the content of nuclear materials more than to a certain level and that there will be uncertainties in the results.

Verification and treatment of the waste is carried out by AB SVAFO, after which the waste containers are transported to SKB for final disposal in SFL. The interface for delivery needs to be clarified, and the responsibility for ensuring that requirements on safeguards are met both by AB SVAFO and SKB.

Methods for verification, systems for marking, packaging, seal handling, transportation and final disposal, taking into account the specific nature of the waste, must be established.

For the final repository, the control of nuclear materials should be considered already during construction and the facility will, in the international inspection, be verified against drawings during the construction and during operation.

3.4 Plan of action for spent nuclear fuel

This section describes the current situation and the overall planning for future activities for the different facilities in the KBS-3 system. Figure 3-11 shows a general timeplan with important milestones.

SKB’s programme regarding the spent nuclear fuel consists mainly of the following parts:

- Completion of the licensing of the KBS-3 system.
- Technology development of the KBS-3 system to enable commissioning.
- Safety analysis reports (SARs) for the facilities in the KBS-3 system.
- Planning, design, construction and commissioning of the final repository in Forsmark.
- Planning, design, construction and commissioning of the integrated facility for interim storage and encapsulation in Oskarshamn.
- Planning, design, construction and commissioning of the production system for canisters.
- Planning, design, construction and commissioning of the production system for buffer and backfill material.
- Planning and preparations for increasing the interim storage capacity in Clab to more than 8000 tonnes of fuel.

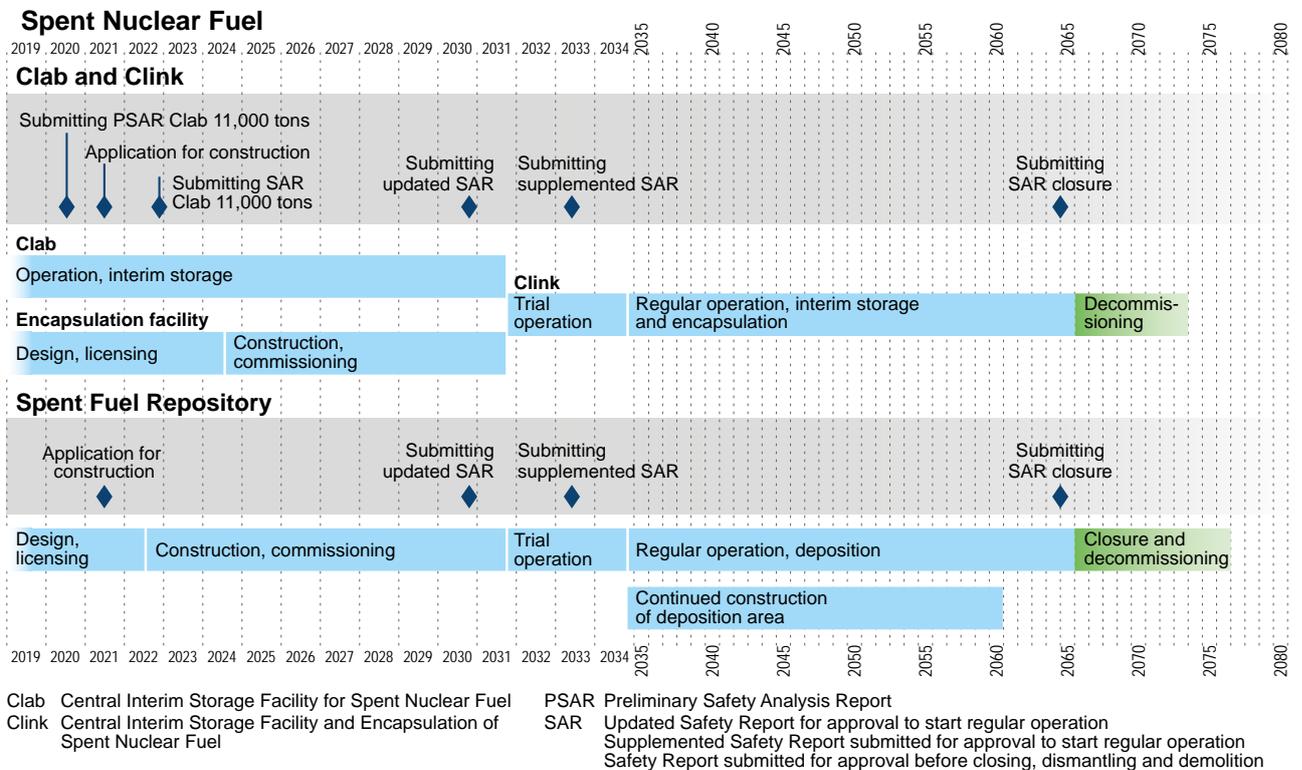


Figure 3-11. General timeplan for the different facilities in the KBS-3 system.

The two construction projects and the work with safety analysis reports (SARs) for the facilities in the KBS-3 system are primary beneficiaries of the research and technology development that is performed for the KBS-3 system.

The planning is based on experience and assessments of the time required for licensing, construction and commissioning and aims at starting operation of the system as soon as possible.

3.4.1 Current situation

The current situation for the work on the remaining parts and facilities in the KBS-3 system can be summarised in the following points:

- Based on the current licensing documentation, with the 2011 applications as a basis, SKB has planned and structured the work that remains until the start of trial operation of the facilities in the KBS-3 system.
- The licensing processes continue, some important milestones that have been reached are:
 - An announcement of the applications under the Environmental Code and the Nuclear Activities Act was made in January 2016.
 - The main hearing in the Land and Environment Court was held in the autumn of 2017 and in January 2018 SSM and the Land and Environment Court both submitted their statements to the Ministry of Environment and Energy (the Ministry of Environment since 1 April 2019).
 - In June 2018 Oskarshamn Municipality made a positive decision regarding their veto right and approved the construction of Clink.
 - SKB responded, in April 2019, to requests for supplements to the Ministry of Environment.
- The preparatory work prior to start of construction is now under way, which, among other things, includes supplementary geotechnical investigations, preliminary design and studies as well as the formulation of requirements.
- The work with safety analysis reports (SARs) for the facilities in the KBS-3 system is under way. This work is based on experience from the preparation of reports on post-closure safety (SR-Site) and safety during operation (SR-Drift). The work of preparing the plans requested by SSM in its review reports concerning SAR-related studies up to the operational phase is also in progress.
- Modernisation of the SAR for Clab has been completed and further updates are under way to provide favourable conditions for increasing the interim storage capacity in Clab to 11 000 tonnes and for Clink.
- The increased storage capacity of spent nuclear fuel in Clab, from today's licensed amount of 8 000 tonnes to 11 000 tonnes, requires that updated design and safety assessments are made for the facility. The performed assessments will be documented in a preliminary safety analysis report (PSAR) for Clab 11 000 tonnes. The preliminary safety analysis report shall be submitted to SSM. The safety analysis report will serve as a basis for the Authority's decision on a licence to extend the storage capacity in Clab. The report can be submitted, at the earliest, after the Government has granted a licence for increased storage capacity according to the application submitted by SKB under the Nuclear Activities Act.
- Work on basic design of the integrated interim storage and encapsulation facility, Clink, is ongoing and is expected to be finished in 2020.

Technology development projects for the Spent Fuel Repository are structured in accordance with the production lines which are linked to the repository's barriers and parts (canister, buffer, backfill, closure and rock). The projects are governed by a strategic technology development plan that links the deliveries from technology development to construction projects and future SARs.

3.4.2 Overall planning

The establishment of the facilities in the KBS-3 system is divided into the following main phases: licensing (and design), construction, commissioning and operation. The activities planned during different phases are summarised for each facility in Sections 3.4.3 to 3.4.5. Some of the milestones shown in Figure 3-12 refer to times for delivery of results from technology development, i.e. points in time when technology components and solutions should be ready to use or have reached a certain development phase (Section 4.1.3). Other milestones involve the development in the licensing matter based on SKB's assessments of times for the Government's decision and the Land and Environment Court's and SSM's continued processing.

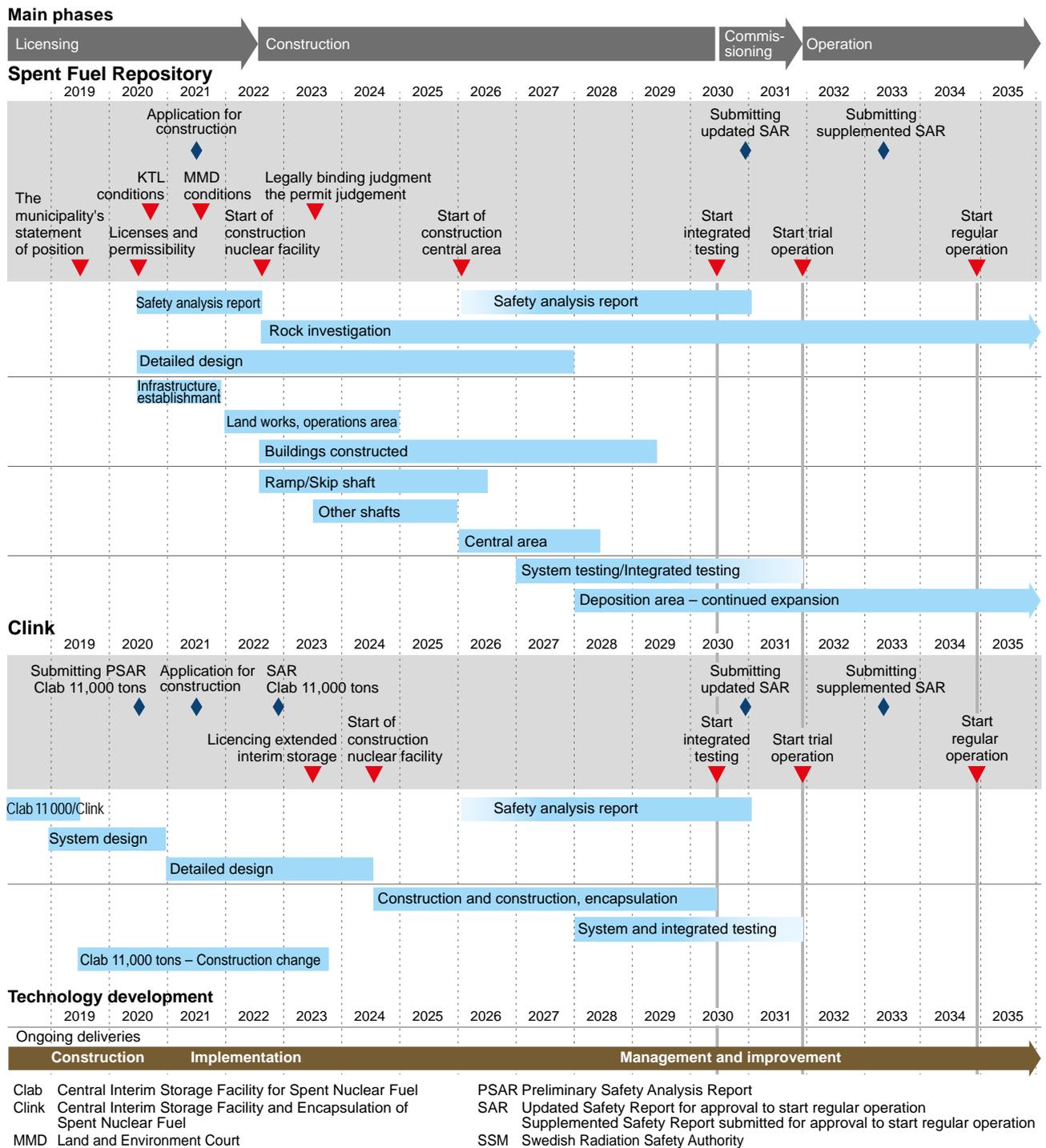


Figure 3-12. Estimated general timeplan for establishment of the Spent Fuel Repository and Clink based on the current progress in the licensing for KBS-3. The technology development needed for the specified milestones is presented in Chapter 4.

According to the plans, construction of the Spent Fuel Repository’s accesses will begin around 2022. For Clink, construction will commence around 2024 so that the facilities can be commissioned simultaneously, around 2031.

Compared with the plans presented in the RD&D Programme 2016, the time for the start of construction of the Spent Fuel Repository and Clink has been delayed by approximately two years. The reason is that the main hearing in the Land and Environment Court was postponed and that the Court requested supplementary information mainly regarding the canister as a barrier. SKB submitted the supplement to the Ministry of Environment in April 2019.

During licensing, the progress of the projects is adapted to the supplementary information that SKB needs to provide and any new information from regulatory authorities. The nearby milestones are

- Östhammar Municipality's decision,
- the Government's decisions on licences and permissibility,
- main hearing on licences and conditions in the Land and Environment Court,
- submission of the application for construction of the Spent Fuel Repository and Clink to SSM.

As soon as the above milestones are passed, SKB will increase the pace of the preparatory work. For example, extensive work with regard to detailed design of facility parts and technical systems will commence.

Before construction of the parts of the repository that are of importance for the safety of the Spent Fuel Repository can be commenced, an application for construction must be submitted to and approved by SSM. The application includes, among other things, the following reports:

- The preliminary safety analysis report (PSAR).
- A description that shows that there is potential for SKB to carry out the planned activities with the required quality with respect to safety and radiation protection during the entire life cycle of the facility. The report is called Suus (Safety during the construction phase).
- Plans for further research and development prior for future steps in the licence application (prior to the operational phase). The purpose of this document is to summarise the more extensive plans for research and development that SSM in review reports has identified as necessary for the continued licensing (up until the SAR prior to the operational phase). Among other things, it contains detailed site investigation and monitoring programmes. The plans and programmes should describe how SKB intends to handle the questions raised by SSM in the review report that was submitted to the Ministry of Environment and Energy in January 2018 (from 1 April 2019 the Ministry of Environment).

The reports will take into account the results from the research, technology development and design activities performed since the applications were prepared, and issues of relevance that have emerged during licensing of the applications.

3.4.3 Interim storage

The spent nuclear fuel is currently interim-stored in Clab. SKB is planning to construct a new facility adjacent to Clab for encapsulation of the spent fuel. The two facility parts will then be operated as an integrated facility, Clink.

The storage capacity in Clab has two limitations: The permissible quantity of spent nuclear fuel in the facility, and the number of physical storage positions in the pools. Clab has a licence to store 8000 tonnes of spent nuclear fuel in the facility. According to forecasts from the nuclear power plants, the quantity of spent nuclear fuel in Clab will reach 8000 tonnes at the end of 2023. SKB therefore plans to, by that time at the latest, obtain a licence for increased interim storage in Clab.

In the spring of 2015, SKB supplemented the applications for Clab and Clink. The supplement includes an additional application to increase the interim storage capacity in Clab to 11 000 tonnes.

SKB's strategy is to obtain a licence from the Government under the Nuclear Activities Act for increasing the interim storage capacity to 11 000 tonnes of spent nuclear fuel in Clab (Clab 11 000 tonnes) and a licence for the encapsulation plant (Clink) at the same time. SKB will, however, want to utilise the licences for the different activities at different times. The planning for licences for Clab 11 000 tonnes and Clink is carried out in relation to the stepwise licensing process that follows SSM's regulations.

In order to obtain a licence for increased interim storage of spent nuclear fuel under the Nuclear Activities Act, SKB must, after a Government decision, submit a preliminary safety analysis report (PSAR Clab 11 000 tonnes) to SSM for approval. The upgrading of the facility and submission of the SAR are planned to take place at the latest in 2023.

The total cooling requirement for storage of 11 000 tonnes of fuel amounts to 12 MW. An upgrade of the existing cooling capacity in Clab has been made and the SAR for the facility will be updated with respect to this.

According to current plans, trial operation of the Spent Fuel Repository and Clink will commence around year 2031. With this, transport of the fuel in Clab can also commence. In order to be able to receive the spent fuel produced until then, additional measures are necessary to free more storage space for fuel, besides increasing the licensed storage capacity. If no measures are taken, the storage positions will be filled around year 2028. SKB therefore plans to segment the control rods from the BWR reactors that require large storage volumes at present. After segmentation, the control rods can be packed more tightly in new storage canisters and then be returned to Clab's storage pools. SKB plans to commence segmentation in the middle of the 2020s. The work will take about five years for the control rods that are stored in Clab today. For new control rods, segmentation will take place when they are received for interim storage. By this measure, the storage capacity is expected to be sufficient until around 2034.

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

3.4.4 Encapsulation

In 2006, SKB applied for a licence under the Nuclear Activities Act to construct, own and operate an encapsulation plant. The application was updated in 2009 for constructing an encapsulation plant adjacent to Clab as an integrated facility, Clink. Since then SKB has submitted a number of supplements and reports to SSM (Section 1.2.3). In March 2011, an application was submitted by SKB under the Environmental Code for the entire final repository system, including Clink and the Spent Fuel Repository.

The current and planned technology development mainly concerns the processes for handling and monitoring of the spent nuclear fuel, manufacturing of canister components, and welding of the seal and bottom welds of the canister.

Basic design

The basic design process is under way and the results will serve as a basis for the construction phase, which follows. In parallel with the basic design, documents are prepared that prior to the construction phase comprises a basis for the application to SSM for approval to start construction of the facility. The application includes a preliminary safety analysis report (PSAR) and other documentation in accordance with the stepwise licensing process. During the basic design, a review of the controlling documents that are specific for Clab is also carried out. If necessary, supplements are made with respect to the control that is specifically required for the construction of Clink. This must be available for implementation prior to the construction phase.

Construction

The construction phase will begin when SSM has approved the application, with the associated PSAR, for commencing construction. The construction phase includes detailed design, establishment of the construction site, preparation of land, construction of an encapsulation plant, installations in the encapsulation plant, facility modifications in Clab, connection of new and existing facility parts and preparations for trial operation. Non-nuclear testing is carried out during the construction phase including testing of components and systems, testing of the entire encapsulation process and finally testing of the entire KBS-3 system. Some of the non-nuclear testing will be carried out with the future operating personnel present as a part of the training prior to the nuclear operation.

The encapsulation plant will be constructed adjacent to Clab and the construction will also include facility modifications in Clab by reason of the future Clink facility. The main safety issue during the construction of Clink is ensuring that the safety of Clab with respect to the facility and the activities can be maintained throughout the construction phase. The operation of Clab proceeds throughout the construction of the encapsulation plant, but reception of fuel may be limited during certain times after planning with the nuclear power plants.

The construction phase will be concluded when the two facility parts are connected, physically and process-wise. This will take place by opening walls between the interim storage part and the encapsulation part and connecting the installations of both facility parts. During the construction phase, training will be held for future operating personnel and controlling documents will be prepared in order to be able to carry out trial operation.

Commissioning

Commissioning takes place in two steps with approval from SSM for each step. First, trial operation is carried out after an application to SSM, which contains an updated safety analysis report (SAR). This safety analysis report is based on the actual facility after construction and the experiences that have been acquired during the non-nuclear testing. Prior to the trial operation, safety-related technical specifications, STF, and other documents with instructions and control of the operation are prepared.

In connection with commissioning, the systems will be tested, first separately and then gradually linked together. At the same time as the facility parts are commissioned, the operating organisation is built up. Personnel is educated and trained for their duties. Fine tuning of technology and organisation will be concluded with a commissioning test of the whole facility under realistic conditions.

When trial operation has been completed and evaluated, an application for a licence to commence regular operation is prepared. A supplemented safety analysis report will be appended to the application, based on the experience gained and measures that have been taken during the trial operation. This safety analysis report will then be continuously managed and kept up-to-date in accordance with SSM's regulations.

Production system for canisters

SKB's production system for canisters should ensure the long-term supply of canisters to Clink. The production system will include several external suppliers and a canister factory for assembly and quality assurance of canister components. The suppliers manufacture nodular cast iron inserts and copper components for the canisters according to SKB's specifications. These are delivered to SKB's canister factory where machining, verification against established criteria and assembly of the components are performed. The result will be a canister that is delivered to Clink. The canister factory will not be a nuclear facility, since spent nuclear fuel will not be handled there.

The work with the canister factory is presently in an early stage compared with the Spent Fuel Repository and Clink. This is because the timeplan for construction and commissioning of the canister factory is substantially shorter than for the other facilities in the KBS-3 system. A feasibility study describing for example the factory's functions, layout and machinery has been completed. One of the purposes of the feasibility study was to update previous studies and clarify strategic decisions regarding for example boundaries to external suppliers and the scope of the quality work related to canister production.

Updated assessments and planning of the production system for canisters will be completed before submission of the PSAR for Clink. As a step in this planning, SKB and Posiva are investigating, among other things, the prospects for a mutual production system for canisters. A mutual system could give coordination advantages, but also means that the production must start considerably earlier than what SKB has planned in order to meet Posiva's needs.

The current and planned technology development mainly concerns the processes for management and quality control of the production of canister components, welding of copper bottom and seal and development of inspection and testing of components and welds. This is described in Chapter 9.

3.4.5 Final disposal

Final disposal of the spent nuclear fuel is planned to take place in the Spent Fuel Repository in Forsmark. Applications for construction and operation of the Spent Fuel Repository were submitted in March 2011 and examination of the applications is under way. Statements by SSM and the Land and Environment Court were submitted to the Government in January 2018.

Licensing and design

Adapted to the progress of licensing, the preparations required to begin construction of the Spent Fuel Repository's accesses are made shortly after all licences are obtained. The construction phase will entail new conditions for SKB's organisation and activities. This applies, for example, to management of the project based on the information flow between the construction works and investigations, modelling, design and safety assessment. During the ongoing licensing process, the project organisation will be successively staffed according to the work tasks in the project.

In parallel with the licensing, the facility is being designed. SKB has completed the system phase and is now preparing and performing detailed design. Deliveries from the technology development are implemented as they arrive. The preparations also include constructional and geological investigations. Ground surveys will be done as a basis for the placement of buildings and foundation engineering. The rock will also be investigated, mainly for the planned locations of accesses. The local infrastructure for the facilities will be prepared. This mainly involves working together with Forsmarks Kraftgrupp AB to adapt the infrastructure already in place in Forsmark to meet the needs of the Spent Fuel Repository and the extended SFR facility.

Studies of the environmental impact of the Spent Fuel Repository will also be carried out during the licensing process. SKB has applied to the County Administrative Board of Uppsala for a species protection exemption in this matter. The exemption granted by the County Administrative Board has been appealed and the case is now in the Land and Environment Court. The species protection matter will only be dealt with after the Government has granted permissibility under the Environmental Code for the Spent Fuel Repository. Overall government conditions for the Spent Fuel Repository's activities will be issued in conjunction with the decision on a licence under the Nuclear Activities Act and permissibility under the Environmental Code. After this, SSM may issue specific conditions under the Nuclear Activities Act. According to the Environmental Code, licences and conditions will be issued after an additional main hearing has been held in the Land and Environment Court.

Detailed design is carried out gradually as the facility is extended and takes into account results from technology development. The results of detailed design are needed as a basis for example for in-depth planning, procurements and construction works. During licensing, detailed design is therefore carried out primarily for facility parts that will be built early. It is above all establishment areas, a site office, accesses to the repository, i.e. ramp, shafts and central area, as well as some of the surface facilities. In order to implement the detailed design related to the repository area, especially deposition tunnels and deposition holes, results and documentation are also required from technology development concerning the rock, see Chapter 12. When the facility has been commissioned, detailed design of the parts that are gradually extended will continue in parallel with disposal.

Construction

The construction phase will begin when SKB has obtained all licences and conditions needed to start construction of the final repository. This means that an application for construction must have been submitted to and approved by SSM.

In the first phase, an operational area is created partly by filling a minor bay with rock spoil, handling area is prepared and temporary construction arrangements are established. The surface and underground facility is extended in parallel, with infrastructure and buildings that are completed above ground.

Construction of the underground facilities is divided into three main parts: The first is when accesses (shafts and ramp) are driven down to the repository level, the second when the central area's underground openings are constructed and technical systems are installed, and the third when the first deposition area is established and the facility is tested and commissioned. Construction of the accesses is time-critical for the progress of the entire project. The ramp and the shaft used for transporting rock spoil to the surface are excavated in parallel, from the surface downward. Until the shaft reaches the repository depth, excavation is limited to these two fronts. When the repository depth has been reached, construction of the central area begins. When the rock loading station and the rock hoist (skip) have been commissioned, the capacity for rock handling increases radically and several driving fronts can be established successively. The excavation of accesses and the central area are followed by installation of the equipment that is needed to operate the facility. The surface facilities are constructed at a pace that is adapted to the underground works.

Excavation of the accesses 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.

As the central area is constructed, investigations of the first deposition area are made and a tunnel is driven providing access to this area. From this tunnel, a few deposition tunnels are excavated in which deposition holes are bored. The purpose of preparing a deposition area at this early stage is firstly to use a part of the area for integration testing and commissioning tests, and secondly to gather the necessary geoscientific data to substantiate an updated safety analysis report prior to trial operation. During the trial operation, the first canisters with spent nuclear fuel will be deposited in the part of the deposition area that is not used for testing.

Commissioning

Operation of the final repository includes both gradual construction and completion of site-adapted deposition tunnels and deposition, backfilling and plugging of deposition tunnels. Commissioning of the subsystems of the final repository takes place successively, as the systems are built and installed. For example, the haulage system for rock spoil, ventilation, control systems etc will be commissioned before the first deposition tunnels are constructed.

In connection with commissioning, the systems will be tested, first separately and then gradually linked together. At the same time as the facility parts are commissioned, the operating organisation is built up. Personnel is educated and trained for their duties. Fine tuning of technology and organisation will be concluded with a commissioning test of the whole facility under realistic conditions. All operational activities will then be carried out, including deposition of a number of canisters, but without any spent nuclear fuel. Deposition takes place in the first deposition area, which is constructed during the latter part of the construction phase. Finally, a commissioning test of the KBS-3 system is conducted, which entails testing of the interfaces between the Spent Fuel Repository, Clink and the transportation system. The technology development needed to complete the system for waste disposal including buffer, backfill, plugs and methodology and machines for installation is presented in Chapter 11. The corresponding technology development needed for rock production and detailed site investigations is presented in Chapter 12.

Commissioning is concluded when SKB obtains SSM's permit for trial operation of the final repository system. All functions and resources, as well as the first deposition area, should then be available so that trial operation can be commenced.

When systems and processes in the facility have been tested and work as intended, SKB submits an updated SAR to SSM that reflects the facility as it is constructed. SSM is expected to be able to approve the SAR around 2031. Before a facility can be put into regular operation, the SAR must be supplemented on the basis of experience gained from trial operation and be approved by SSM.

Safeguards

The purpose of safeguards is that inspection bodies can make sure that all the spent nuclear fuel is encapsulated and placed in the Spent Fuel Repository. The implementation of safeguards ensures continuous knowledge that the fuel, after final verification and encapsulation, is placed in the Spent Fuel Repository and will not leave the repository.

The safeguards system will contain confirmed information on which fuel assemblies the canisters contain, when the fuel was encapsulated, transported and arrived at the Spent Fuel Repository, where the canisters are deposited, and the total quantity of nuclear material in the final repository.

Prior to construction of new nuclear facilities, safeguards must be taken into account already in the design stage so that inspection and control are facilitated during the operational phase. The work within this area will be intensified during the period, and the main work will take place in the construction projects. The work has the purpose of ensuring that stakeholders are aware of the requirements on safeguards so that activities, construction and facility construction can be planned efficiently for the entire KBS-3 system in an integrated manner.

An important part in the international inspection of the Spent Fuel Repository is being able to verify that the facility has been constructed in accordance with the specified drawings. This is done so that the inspection bodies can ensure that there are no routes out of the facility that have not been reported and that there are no areas where unauthorised activities can be carried out.

3.5 Plan of action for decommissioning of nuclear facilities

The plan for decommissioning of the nuclear power plants in Barsebäck, Forsmark, Oskarshamn and Ringhals, the Ågesta reactor and SKB's nuclear facilities is presented in Part III. It describes how work has been divided between the nuclear power companies and SKB, and internally in the two corporate groups Uniper and Vattenfall. Furthermore, the development work that remains to permit decommissioning of the facilities concerned is presented.

The RD&D Programme 2019, 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 complement each other in terms of content, where the RD&D Programme presents the development activities and other measures required 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 decommissioning plans. The main documents are based on supporting documentation, where a central common reference consists of the decommissioning studies that were carried out for each site prior to the Plan report 2013 and updated planning regarding the reactors that will be decommissioned in the 2020s. The decommissioning studies are prepared with a focus on costs but they also describe the technical approach and thereby cover aspects of everything from organisation to storage volumes. The content of the decommissioning studies and other supporting documentation is presented summarily when they are referenced in the three main documents.

3.5.1 Decommissioning overview

Decommissioning of a nuclear facility to release the facility from regulatory control includes a number of activities. Prior to decommissioning, necessary licences must be obtained. When a facility is decommissioned, defuelling operation follows, where all fuel is transported from the reactor to Clab for interim storage. If necessary, shutdown operation follows thereafter until dismantling and demolition can begin. The nuclear power companies plan is to start decommissioning as soon as possible after final shutdown. When the facility/facility parts have been released from regulatory control, conventional demolition and restoration of land will be carried out.

Since dismantling and demolition of Barsebäck 1, Barsebäck 2, Oskarshamn 1, Oskarshamn 2, Ringhals 1, Ringhals 2 and Ågesta will commence before the extended SFR facility is ready to receive short-lived operational and decommissioning waste, the licensees must provide interim storage of this waste at the site or externally. See Section 3.3.3 for interim storage of the short-lived waste and Section 3.3.4 for the long-lived waste.

3.5.2 Current situation and overall planning

Figure 3-13 shows the general timeplan for decommissioning of all nuclear power plants and SKB's facilities. Below is a summary of the current situation for each nuclear power plant and the planning for the future decommissioning of these facilities and SKB's facilities.

Barsebäck Kraft AB

The internals of the reactor pressure vessels are segmented. They are interim-stored in steel tanks in a storage building at the site. The necessary licences have been obtained for dismantling and demolition. Preparatory measures required for dismantling and demolition are completed.

Dismantling and demolition of the nuclear power plant begins in 2020 and clearance is planned to start in the end of the 2020s and will be finished in the beginning of the 2030s.

OKG Aktiebolag

At Oskarshamn 1 and Oskarshamn 2, dismantling and demolition is under way. A number of preparatory measures have been taken. The reactor internals at Oskarshamn 2 have been segmented, while segmentation for Oskarshamn 1 is under way. The segmentation is expected to be finished in the spring of 2020. Final clearance of the buildings is planned to take place around 2028.

Oskarshamn 3 will be in operation until 2045, when decommissioning begins. Clearance of Oskarshamn 3 is expected to be completed around 2053.

The reactors have a number of shared service facilities called block 0. They will be decommissioned in conjunction with the decommissioning of Oskarshamn 3, with the exception of the existing shared waste facility, which will be decommissioned in conjunction with the decommissioning of Oskarshamn 1 and Oskarshamn 2.

Decommissioning of reactors and SKB facilities

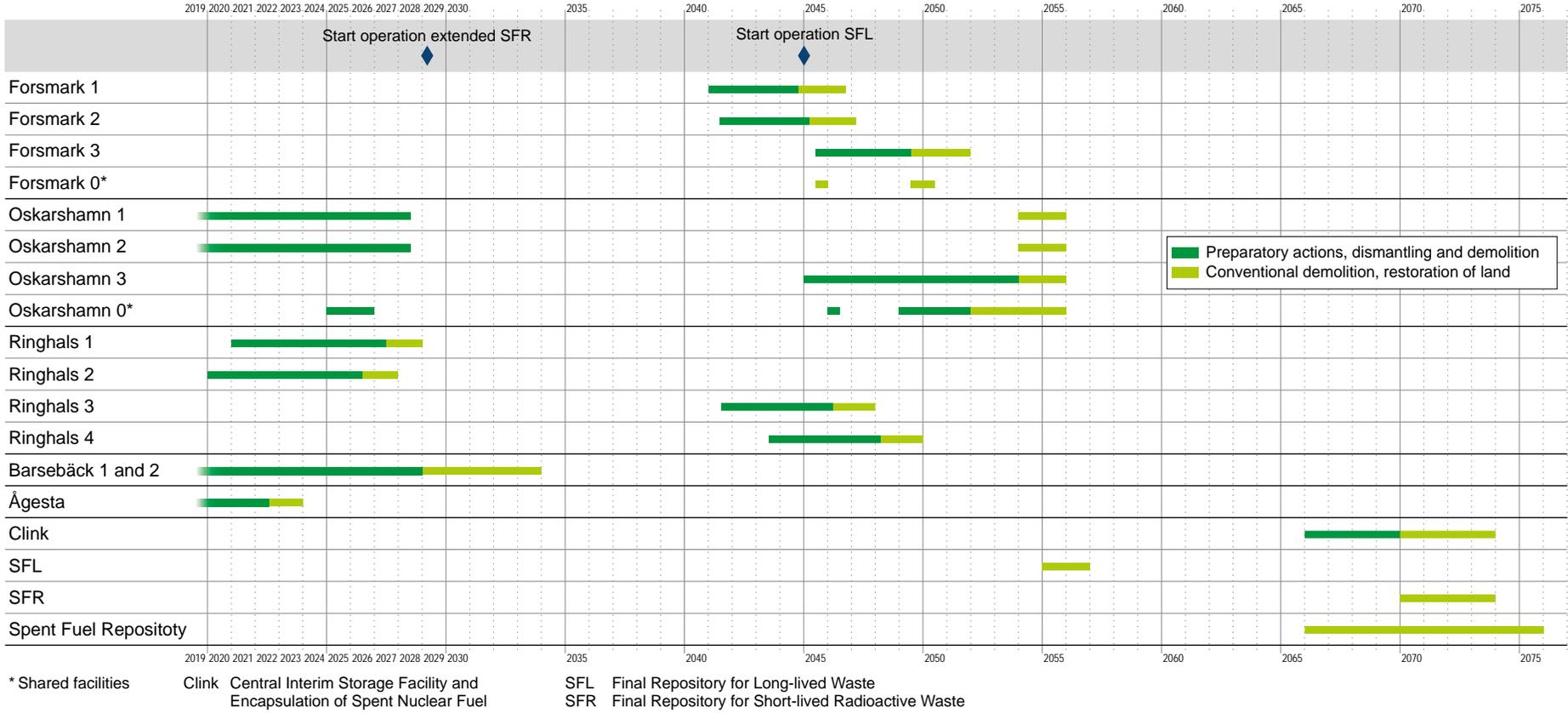


Figure 3-13. Overview of the nuclear power companies' and SKB's timeplans for decommissioning.

The remaining shared facilities that are not required for the decommissioning of Oskarshamn 3 will be dismantled and demolished in parallel with Oskarshamn 3. Other shared facilities will be dismantled and demolished after the completed decommissioning of Oskarshamn 3, or left to other actors. Final clearance of the shared facilities is estimated to take place around 2051.

Ringhals AB

In 2015, a unit was added within Vattenfall AB with the task and responsibility of coordinating and pursuing decommissioning within the Vattenfall Group, the Business Unit Nuclear Decommissioning (BUND). Analysis and planning for operational decommissioning measures are carried out by BUND on behalf of, and in close collaboration with, the licensee Ringhals AB. The focus is presently to analyse and plan for the decommissioning of Ringhals 1 and Ringhals 2 and to evaluate how the specific decommissioning steps should best be resolved.

In conjunction with the shutdown decision for Ringhals 1 and Ringhals 2, the project STURE (Safe and Secure Phase-out of Reactor 1 and 2) was initiated, with the purpose of preparing for decommissioning of the two reactors while operation of Ringhals 3 and Ringhals 4 continues.

Ringhals 1 and Ringhals 2 are planned to be finally shut down in the end of 2020 and 2019, respectively, and they will be decommissioned during the period 2028–2030. Ringhals 3 and Ringhals 4 will be in operation until 2041 and 2043, respectively. According to current planning, the BWR pressure vessel from Ringhals 1 will be segmented. There is also a policy decision to segment the PWR pressure vessel from Ringhals 2. Segmentation is still not decided for Ringhals 3 and Ringhals 4.

Forsmarks Kraftgrupp AB

Forsmarks Kraftgrupp AB's reactors are planned to be 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.

The Ågesta plant

The nuclear power plant in Ågesta has been in shutdown operation since 1974. A decommissioning project was initiated in November 2015. Provided that the necessary licences have been obtained, dismantling and demolition are planned to start during the autumn of 2019. The decommissioning is planned to be completed by 2023.

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. SFL can, however, be decommissioned when the long-lived waste from the last reactor has been managed and disposed of. The closure of SFL assumes that the decommissioning waste from Clink does not contain any long-lived waste.

3.6 Plan of action for transportation

3.6.1 Current situation

The transportation system's tasks are to provide transportation of spent nuclear fuel from the Swedish nuclear power plants to Clab in Oskarshamn and transportation of waste from the nuclear power plants and Studsvik Nuclear AB to SFR in Forsmark.

The annual transport volume is currently on average 90 casks with spent nuclear fuel and about ten casks or containers with radioactive waste to SFR. Most of the transportation is carried out with the ship m/s Sigrid and the remainder with terminal vehicles by land. At present, m/s Sigrid makes about 20 trips per year, which means that there is an overcapacity in the transportation system.

Every year SKB also conducts a number of transport assignments on the international market, such as transport of steam dryers, turbine parts and other project loads.

3.6.2 Overall planning

After 2030, the need for transportation of spent nuclear fuel and radioactive waste is expected to gradually increase when several of SKB's new facilities have been commissioned. The additional volumes, compared with the current situation, are primarily encapsulated spent nuclear fuel, which will be regularly transported from Clink to the Spent Fuel Repository, and decommissioning waste from the nuclear power plant destined for SFR and SFL, when this repository is commissioned. Canister transport casks, waste transport casks and ISO-containers will be used for these transports. Transportation of operational waste to SFR and spent nuclear fuel to Clab for interim storage will continue as long as there are nuclear power plants in operation.

The transportation system's planned future capacity can be simply described as the ship m/s Sigrid, fuel transport casks, waste transport casks, canister transport casks and the presently applicable and known restrictions and conditions regarding terminal vehicles, time for each transport including loading and unloading, service requirements for ship and containers, ice conditions etc.

The transportation need will be large throughout the 2030s. This period has therefore been used to estimate an average and representative annual volume that shall be possible to transport during this time. According to this estimate, approximately 40 shipments by sea must be carried out per year, 14 shipments of radioactive waste, 15 shipments of encapsulated spent nuclear fuel in canister transport casks and 11 shipments of spent nuclear fuel in fuel transport casks.

A comparison of the annual transportation need in time during this period and the transport capacity expected to be available shows that the capacity of the transportation system is sufficient also in the long term (Hanström 2019). The ship's capacity is not deemed to be a restriction, but the completed survey shows that the terminal vehicles, service of transport casks, and the time when the ship is still in the harbour for unloading could constitute restrictions, if they are not appropriately handled. The prerequisites for the study of the annual transportation need are based in part on unconfirmed assumptions. In the coming work with developing and adapting the transportation system, these assumptions will be revised.

When SFL is commissioned in 2045, the transportation system will be subjected to the heaviest loads. Transportation of copper canisters to the Spent Fuel Repository takes place at the same time as transportation of short-lived and long-lived decommissioning waste from decommissioning of the last reactors and from AB SVAFO to the final repositories SFR and SFL. There is currently an overcapacity in the transportation system and it is expected to be able to manage the increased transport volume.

3.6.3 Transportation of low-level and intermediate-level waste

Control rods to Clab

Control rods from BWRs are long-lived waste, which is transported to Clab for interim storage. During transport, a transport cask intended for core components is used.

Short-lived operational and decommissioning waste to SFR

Short-lived low-level and intermediate-level waste is transported from the nuclear power plants, Clab and the Studsvik site to SFR for disposal. Low-level waste can be transported in ISO-containers while intermediate-level waste is transported in waste transport containers.

The volumes of low-level waste in ISO-containers that will be disposed of in SFR are expected to increase in the future when the nuclear power plants are decommissioned. The waste will be transported with m/s Sigrid at separate occasions.

During the period when the extension of SFR is under way, no waste disposal will be possible in the facility. This means that the waste produced will be interim-stored at the waste producers until the extension of SFR is completed and the facility can receive waste again. This also includes segmented reactor pressure vessels from Barsebäck 1 and 2, Ringhals 1 and Oskarshamn 1 and 2. When the extension of SFR is completed, there will be a great need for transportation of waste from the waste producers during the first years, especially waste in ISO-containers.

Operational and decommissioning waste from AB SVAFO, Studsvik Nuclear AB, and Cyclife Sweden AB in Studsvik and decommissioning waste from Ågesta will also require transport to SFR. The forecasts for these facilities are uncertain but the volume is limited in comparison with the waste from the nuclear power plants.

Long-lived operational and decommissioning waste to SFL

Long-lived waste that is produced at the nuclear power plants during operation and decommissioning must be interim-stored until SFL is commissioned.

All long-lived waste intended for SFL will be transported over a ten-year period. The waste consists of both the volumes in interim storage and the waste that is produced during the operating time of SFL.

Shipments of long-lived waste from the Barsebäck nuclear power plant to interim storage at another site are also expected before the commissioning of SFL. Long-lived waste will also be produced in Studsvik Nuclear AB's activities and in research, medicine and industry. During decommissioning of the PWR reactors, the reactor pressure vessels will be transported to SFL for disposal.

The transportation system will be supplemented with a new type of transport container for long-lived intermediate-level waste placed in steel tanks. The transport container is called ATB 1T. It will, due to the activity content, be designed according to the IAEA requirements of Type B(U).

A contract was signed in 2014 with an American supplier, Holtec International Power Division Inc, for the design, licensing and manufacture of ATB 1T. The transport container is licensed by the United States Nuclear Regulatory Commission (U.S. NRC) and a certificate is provided by United States Department of Transportation (U.S. DOT). The approved certificate, Type B(U), is then reviewed and validated by SSM. Application documents for licensing were submitted to the U.S. NRC in September 2015 and the licensing process is ongoing. The U.S. NRC has requested some supplementary tests and updated application documents are planned to be submitted at the end of 2019. Delivery of the first new transport container is planned for the summer of 2020.

3.6.4 Transportation of spent fuel

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

Current transports of spent nuclear fuel in fuel transport casks to Clab will continue in a similar manner to Clink, as long as there are reactors in operation.

The work with replacing SKB's current fuel transport casks with new casks that comply with modern requirements is progressing according to plan. A contract for fuel transport casks was signed in October 2013 with the American supplier, Holtec International Power Division, Inc. The contract includes design, licensing and manufacture of five transport casks with auxiliary equipment. There is an option for manufacturing of an additional cask. The casks have larger capacity than existing casks, which means that fewer casks are required for meeting the transportation demands in the Swedish system. During 2019, it will be decided whether a sixth transport cask is to be ordered. The U.S. NRC's approval of the new cask HI-STAR 80 was obtained in the end of September 2018 and U.S. DOT has issued a licence. An application for validation of the licence was submitted to SSM in early December 2018. Manufacturing of the first HI-STAR 80 has been initiated in Holtec's factory in Pittsburgh and delivery of the first new fuel transport cask is planned for the spring of 2021.

In 2015, new requirements made it necessary for SKB to obtain new bottom shock absorbers for existing fuel transport casks (Type TN17/2). The new bottom shock absorbers have been delivered and the certificate for the transport casks is valid to March 2020. The transport cask supplier TN International has updated the safety analysis report and applied for a new certificate from the French Autorite de Surete Nucleaire, which applies after 2020. After obtaining the French certificate, SKB will apply for a Swedish validation in the spring of 2020.

Encapsulated spent nuclear fuel from Clink to the Spent Fuel Repository

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

The transportation system will be supplemented with a new type of transport cask for transportation of spent nuclear fuel in copper canisters from Clink to the Spent Fuel Repository. This transport cask is called the canister transport cask (KTB). Due to its content of radioactivity, it will be certified as packaging Type B according to IAEA's transport recommendations. A study is under way concerning the quantity of canister transport casks that will be required.

SKB has previously conducted feasibility studies concerning the possible design of the canister transport cask. SKB has, together with suppliers, formulated a proposal for the design and functionality of the casks. The final design will be determined in cooperation with the chosen supplier. The design will follow an iterative process in order to satisfy the regulatory requirements and SKB's own prerequisites, specific requirements and expressed preferences. The design and safety-related properties of the cask are described in a safety report as a basis for obtaining a licence from the authorities in the country where it is manufactured. Before the cask may be used in Sweden, SSM must validate the licence. The time required for design and licensing is estimated to about seven years.

The first canister transport cask will be delivered to Clink and the Spent Fuel Repository prior to the testing of individual systems. According to the current timeplan, this testing can begin around 2029. The initial system-specific tests will be conducted one year before the commissioning test of the entire KBS-3 system. The remaining casks will be manufactured and delivered as they are produced during the period 2030–2034, in parallel with commissioning tests and trial operation.

3.6.5 Special transportation

SKB's transportation system with m/s Sigrid has the ability to carry out external transportation and there is staff with the required competence. The experience gained from external assignments increases the competence for conducting special transportation for the owners during decommissioning of nuclear facilities.

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

3.7 Alternative strategies in case of changed conditions

SKB's and the licensees' 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. There are of course uncertainties of various types but the activities permit relatively large flexibility. Generally, points in time and content of deliveries may be affected by the ongoing licensing. The following is a number of possible changes to conditions and what their consequences may be.

3.7.1 Operating times of the nuclear power reactors

Since the RD&D Programme 2016, the reactors Oskarshamn 1 and Oskarshamn 2 have been shut down, while Ringhals 1 and Ringhals 2 will be shut down at the end of 2020 and 2019, respectively. The planned operating time for the other six reactors is, as in the RD&D Programme 2016, 60 years. This applies to the reactors Forsmark 1, Forsmark 2 and Forsmark 3, Oskarshamn 3, and Ringhals 3 and Ringhals 4. This means that these reactors will be finally shut down during the period 2040 to 2045. How the waste system would be affected by a possible change of the planned operating times for the reactors is described below.

Extension of planned operating times

SKB's facilities in the KBS-3 system will be designed to manage and dispose of 6 000 canisters of spent nuclear fuel. The current forecasts of the nuclear power companies, taking into account the premature shutdowns according to the above, contain about 5 600 canisters. This provides a margin for possible extended operating times for the reactors from the 1980s. For example, the design-basis canister quantity for the Spent Fuel Repository of 6 000 canisters will be reached if the remaining reactors extend their operating times by about six years, i.e. to a total operating time of about 66 years. If the operating times are extended further, the capacity of the Spent Fuel Repository is judged possible to increase, after proper licensing, through the use of unutilised areas at the selected repository depth.

The required interim storage capacity for fuel in Clab is not affected by an extension of the reactor operating times, as the additional spent fuel arises during the 2040s. According to the plans, deposition is then under way in the Spent Fuel Repository and thereby capacity is released in the storage pools. In case of a delay of more than ten years in the commissioning of Clink and the Spent Fuel Repository, it may, however, be necessary to increase Clab's storage capacity for spent nuclear fuel. This is the case regardless of an extension of operating times (Section 3.7.5).

The designed capacity in the extension of SFR is judged to provide a sufficient margin for additional operational waste in case of extended operating times. The designed capacity 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 case of extended operating times. However, it only concerns a small fraction of the total waste amount. The amount of decommissioning waste is not expected to be affected by extended operating times.

In case of extended operating times, the long-lived waste in the form of BWR control rods and other core components will increase. If necessary, it is possible to adjust the final disposal volume in SFL until the start of construction, i.e. until around 2038 with current planning.

Shortening of planned operating times

Conversely, a shortening of the planned operating times would entail a reduced quantity of spent nuclear fuel and operational waste and thus lead to reduced storage requirements in the repository systems. All existing and planned facilities for disposal of nuclear waste and spent nuclear fuel will nevertheless be needed. Since the Spent Fuel Repository is constructed 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 currently forecasted volumes, shortened operating times for the nuclear power reactors will probably mean that the facility will not be fully utilised.

If more reactors, in addition to the four mentioned above, are shut down prematurely, the total quantity of fuel will probably be less than Clab's maximum storage capacity of 11 000 tonnes of fuel.

An increase in the permissible interim storage capacity in Clab is expected, according to SKB's planning, to take place at the latest by the time Clab reaches this capacity. According to current forecasts, 8 000 tonnes of fuel will be reached around 2023/2024. If a license for increased interim storage has not been obtained at this time, the spent nuclear fuel will require interim storage in pools at the nuclear power plants instead. If more reactors than the four mentioned previously are shut down around 2020, Clab's limited capacity to receive and handle fuel may entail that fuel must be interim-stored in pools at the nuclear power plants for several years, before the final cores can be successively transported to Clab for interim storage. Interim storage of spent nuclear fuel at the nuclear power plants after final shutdown is not a desirable scenario, since it entails less safety and increased costs.

Shortened operating times of the reactors, especially if the Forsmark 3 and Oskarshamn 3 reactors shut down earlier than planned, would probably entail an earlier start of decommissioning of the concerned reactors. It could also entail that the entire nuclear waste programme can be concluded earlier than planned. The extent depends on how many reactors would be affected and how much the operating times are shortened. If the last reactor is decommissioned before SFL has been commissioned, the long-lived decommissioning waste will be interim-stored, like other reactor waste, until SFL is commissioned.

3.7.2 Commissioning of the extended Final Repository for Short-lived Radioactive Waste

SKB plans to start trial operation for the extended SFR in 2029. According to the planning, decommissioning of the seven first reactors (including Ågesta) will begin before the extension of SFR is finished. This means that different interim storage facilities for the short-lived and long-lived decommissioning waste are needed (Sections 3.3.3 and 3.3.4).

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 of the extension of SFR thereby also leads to a delay of the clearance of the Barsebäck site, unless the short-lived waste is transported to another site for interim storage as well.

For OKG Aktiebolag and Ringhals AB, the consequences of a delay are not as serious, since there are remaining reactors in operation at the sites. It is deemed possible to increase the interim storage capacity at the power plant sites in order to meet the needs in case of a delay of the extension by a couple of years.

Earlier start

Earlier commissioning of the extended SFR facility entails a shortened and decreased need for interim storage of short-lived decommissioning waste. For Barsebäck Kraft AB, this means that transports can be made earlier and the possibility of releasing the entire facility from regulatory control before the planned time increases.

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 management options that exist today are disposal in near-surface repositories or in the waste vault for low-level waste (BLA) in SFR. 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 content than the very low-level waste to justify both costs and environmental impact (in the form of for example rock excavation and transportation).

The extended SFR facility is designed to dispose of all low-level waste from decommissioning of the reactors, including large quantities of the very low-level waste. The waste forecasts are currently approximate and the waste producers therefore intend to update and refine them. Since there are large uncertainties associated with the forecasts of the amount of very low-level waste, it is important to study several different alternatives for final disposal.

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 the utilisation of the waste vault, however, some kind of extra treatment is preferred. Treatment could involve either moving the activity to another carrier material (by decontamination), or changing the form of the original waste (by for example incineration). As described in Section 3.2.1, 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 are usually technically very difficult to decontaminate, or it is difficult to prove that they are free from radioactivity after decontamination.

For the subset of the waste that is combustible, controlled incineration is an alternative method for concentrating the radioactive content. Controlled incineration requires that the waste has been sorted initially so that the non-combustible, but compactable, fractions are handled separately. Furthermore, controlled incineration requires both transportation of radioactive material and a relatively expensive handling in the incineration facility at the Studsvik site, which greatly limits the reasons for incinerating the waste there.

A development project for identifying opportunities for energy recycling of subsets of the combustible very low-level waste via conventional waste incineration facilities has been initiated (Keith-Roach and Elert 2018). The outcome of the work will be further discussed with SSM during the RD&D period and the project may be extended if there is a consensus between the nuclear industry parties, SSM, and the conventional waste incineration facilities.

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

SKB plans to commission SFL around 2045, which corresponds with the shutdown of the last reactors, Forsmark 3 and Oskarshamn 3 (Section 3.3.4). The timeplan is based on a scenario where SFL is constructed at a site of which SKB has knowledge from previous site investigation programmes.

The siting of SFL is yet undecided. If SFL is constructed at a site where SKB has limited knowledge of the geology from previous investigation programmes, commissioning will be postponed into the future, roughly about five years. Such a siting requires more extensive work, both to identify a site and because it leads to more extensive site investigations. A delay in the commissioning would entail a 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 in a relatively early stage. Up to commissioning of the repository, several important milestones must be reached, such as evaluation and assessment of the safety after closure, siting, preparation of applications, licensing, construction, etc. In order to commission SFL in 2045, according to current planning, SKB must submit applications under the Nuclear Activities Act and the Environmental Code to construct, own and operate SFL around 2030. A delay in any part, or in several parts, which results in a delay of one or several years in the process up to commissioning of the repository cannot be excluded.

Colocation with SFR or the Spent Fuel Repository could entail an earlier commissioning of the repository. The construction phase could potentially be shortened through joint use of ramp and shafts, which shortens the time for excavating to repository depth. A possible earlier start of the commissioning of SFL is not expected to have any major consequences for the waste system. However, it would entail shorter times for interim storage of the long-lived waste.

A possible alternative is that final disposal of the long-lived waste is located at two different sites. According to the plans, SFL consists of two different repository parts with different barrier solutions, which, in principle, could be located at different sites. The consequences of such a solution for the commissioning times are largely dependent on, as in the case with a consolidated SFL, 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 current plans, trial operation of the Spent Fuel Repository and Clink will commence around 2031, which means that SKB will begin unloading of the spent nuclear fuel from Clab's storage pools at that time. For Clab to be able to receive the spent fuel generated up to this point in time, SKB plans to increase the licensed storage capacity to 11 000 tonnes of fuel. In addition, the control rods from BWRs will be segmented to free up storage space for the spent nuclear fuel. The storage capacity is then expected to be sufficient until around 2034 (Section 3.4.3).

Postponement

If further delays in the commissioning of the Spent Fuel Repository and Clink should arise, there is also a possibility to transfer the fuel still stored in normal storage canisters to compact storage canisters. The transfer would then be carried out in Clab's receiving section. After completed transfer, Clab's storage capacity is expected to be sufficient until around 2040.

If additional measures should be required, the core components and control rods stored in Clab could be unloaded and interim-stored at another site. The waste can then be reloaded into steel tanks for dry interim storage. Steel tanks are already used for interim storage of metal parts at the nuclear power

plants in Barsebäck, Forsmark and Oskarshamn. If only fuel is stored in Clab, the storage positions will suffice until around 2043. This provides flexibility also in the event of a larger delay in the commissioning of the Spent Fuel Repository and Clink.

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

Earlier start

SKB judges the likelihood of putting Clink and the Spent Fuel Repository into operation considerably earlier than planned as small. Earlier commissioning is not assessed to have any negative consequences for the waste programme.

3.7.6 Horizontal deposition – KBS-3H

Since 2001, SKB, in collaboration with Posiva, has investigated whether horizontal deposition (KBS-3H) can constitute an alternative to vertical deposition in the Spent Fuel Repository. KBS-3H entails that long horizontal deposition drifts are bored directly from the Spent Fuel Repository's main tunnels. The spent nuclear fuel is deposited in these in supercontainers.

A supercontainer consists of a copper canister surrounded by bentonite buffer and held together by a perforated outer metal shell. Distance blocks of bentonite clay are placed between the supercontainers. The deposition drifts are up to 300 metres long and are divided into two sections by compartment plugs. A drift end plug is installed in the opening of the deposition drift.

The rock volume that needs to be excavated for a KBS-3H repository is smaller than for vertical deposition, which also means that smaller volumes need to be backfilled. The facilities at the operations area above ground and the central area and the accesses underground are marginally affected by a change to horizontal deposition.

An evaluation of KBS-3H has been carried out in collaboration with Posiva. The study covered for example technical maturity and remaining questions regarding technical solutions and operation, status and remaining questions regarding post-closure safety and costs, both the remaining development costs and the calculated operation costs. Despite completed development work, a lot of technical development remains before KBS-3H would be as mature as vertical deposition. For the commissioning of the KBS-3 system to progress according to plan, the remaining development work for horizontal deposition is considered too extensive to be able to justify parallel development efforts even if there are large potential economic advantages of horizontal deposition. Posiva has come to the same conclusion.

In view of the remaining development needs, with associated uncertainties, SKB is not currently planning to pursue any development of KBS-3H during the coming six-year period. This strategy may, however, be revised in conjunction with the RD&D Programme 2022, if SKB comes to the conclusion that the economic advantages of KBS-3H are so great that they would justify continued development efforts.

4 Continued research and development

The system of facilities which so far has been developed for management and final disposal of nuclear waste from the nuclear power plants is described in Chapter 2. Chapter 2 also provides a description of the facilities that remain to be constructed in order to be able to dispose of the spent nuclear fuel and nuclear waste from the decommissioning of the nuclear power reactors.

The plan of action is presented in Chapter 3, and on the basis of that plan, this chapter presents an overview of the research and development needed for implementing the remaining parts of the nuclear waste programme. The details in the form of the current situation and a programme for specific activities and efforts during the RD&D period are presented further in Part II.

In Section 4.1, the foundations for the planning of future research and technology development are described. The section provides an overview of how far the research and technology development must have progressed by the milestones that are relevant for Clink and the final repositories. In order to implement the plan for the very low-level nuclear waste that is described in Chapter 3, specific research and technology development efforts are expected to be limited.

In Sections 4.2 through 4.9, the technology development that is needed to resolve issues concerning the design, construction, and production of the repositories is justified and summarised. An overview is also given of the research that is needed for carrying out post-closure safety assessments of the repositories. The planning for how the repositories are to be monitored during construction and operation is described in Section 4.10.

An overview of the technology development needs based on decommissioning of nuclear facilities is found in Section 4.11, and in Section 4.12 a description is given of other areas that are relevant for SKB's mission.

4.1 Planned activities for each final repository 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, which was presented in Section 3.1, provides 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 Swedish Radiation Safety Authority's approval dictates when SKB can commence construction and operation of the facilities. Hence, it is natural that milestones linked to submission of applications with safety analysis reports will be crucial for the planning of research and development activities, as they dictate when the current level of knowledge and technical solutions will be reported to the authority:

- Preparatory preliminary safety analysis report (F-PSAR) at the time of application for a licence under the Nuclear Activities Act for constructing, owning and operating a facility.
- Preliminary safety analysis report (PSAR) before beginning construction.
- Updated safety analysis report (SAR) prior to trial operation.
- Supplemented safety analysis report (SAR) prior to regular operation.

Prior to the RD&D Programme 2019, two licensing matters are being processed in the Land and Environment Court (MMD) and by the Swedish Radiation Safety Authority (SSM), one for the final repository system for spent nuclear fuel (SKB is awaiting a Government decision for the encapsulation plant and the Spent Fuel Repository) and the other for the extension of SFR for final disposal of short-lived operational and decommissioning waste.

When SKB submits the applications to construct a facility, the purpose is to show that we have the knowledge and ability to construct it in such a way that it meets the regulatory requirements. Even when SKB has reached the level of maturity in research and development that is required to obtain a licence under the Nuclear Activities Act, further research and technology development are needed in the coming stepwise licensing process. This means further development of the technology and the

systems that are required to be able to construct and then commission the facility, and research to reduce remaining uncertainties in the assessments of safety after closure of the final repositories as support for the development and optimisation of the facility. As part of the applications for a licence to construct the Spent Fuel Repository, Clink and the extension of SFR, SKB has presented the level 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 for the Spent Fuel Repository and SR-PSU for SFR). For the KBS-3 system, the main hearing in the Land and Environment Court in 2017 constituted another step in the process. Statements from SSM and the Land and Environment Court arrived in January 2018 and constitute a central basis for the Government's decision to grant approval under the Swedish Environmental Code, as well as an important basis for SKB's continued work with Clink and the Spent Fuel Repository prior to the start of construction. The main hearing in the Land and Environment Court for the extension of SFR will take place in the autumn of 2019.

A safety evaluation of SFL has been carried out for the proposed repository concept. The outcome of the safety evaluation constitutes the basis for continued research and technology development for SFL. It will also provide a basis for specifying site selection criteria and for devising and implementing a siting process.

The reports and studies described above, together with the comments provided by SSM in connection with its review of the applications and reviews of previous RD&D programmes, serve as a basis for the programme for future efforts, which is presented below. The need for research and development activities can be divided into three main categories:

- The need for an increased **process understanding**, i.e. the scientific understanding of the processes that affect the final repositories and thereby the basis for assessing their importance for safety after closure. The work is being carried out according to the management of and strategy for research described in Chapter 5.
- The need for knowledge and competence concerning **design, construction, manufacture, and installation** of the barriers and components to be used in the facilities. The work is carried out according to the technology development process described in Chapter 5.
- The need for knowledge and competence concerning **inspection and testing** to verify that the barriers and components of the system are produced and installed according to approved specifications and thereby meet the requirements. This also includes development of methods and instruments for inspection, testing and monitoring of the final repositories and the sites.

One part of the development work is to demonstrate how developed solutions work in practice. Demonstration experiments will continue, primarily at the Äspö HRL. During the construction of the planned final repositories, some demonstration experiments may come to be undertaken in the different repositories.

As an integral part of the research and development work, studies are made of how the technical solutions may be optimised and made more efficient, without having a negative impact on safety. The prerequisites for such technology optimisation are considered to be especially good when it comes to the interaction between different technical systems and production lines, since the development work so far has focused on finding suitable solutions for individual systems and production lines.

Planning, preparations and execution of decommissioning of the Swedish nuclear power reactors are under way, which is described in Part III. The need for research and development activities for carrying out the decommissioning is relatively limited. Development and research for waste from the decommissioning is included in the presentation for the respective final repository below.

4.1.1 The Final Repository for Short-lived Radioactive Waste

Construction

During the period up to the start of construction, some technology development of the barriers is planned in order to verify the requirements and technical design requirements, which in turn entails an opportunity to optimise the design of the extension of SFR and how excavation will be carried out. The corresponding circumstances also apply for the closure components, for example plugs. The updates of the installation properties of barriers and other repository components that technology development has

led to, are taken into consideration as a basis in the assessment of post-closure safety. Before excavation can begin, a number of investigation boreholes adjacent to the extension will be sealed. The sealing technique will be adapted to the conditions at SFR, and a programme for quality control will be designed and established.

At the start of construction, technical design requirements and requirements for the extension are established. Construction will begin with the establishment of necessary infrastructure. During the construction phase, construction, manufacture and installation of the barriers take place, and preparations for commissioning are made. In the final phase of construction, verification and validation of systems and functions are carried out, and the construction phase is concluded with a commissioning test.

Operation and closure

Before trial operation is initiated, the assessment of post-closure safety will be updated with site-specific information obtained during the construction phase. The additional information is used, among other things, to update the site descriptive model for Forsmark. The new information will provide a more detailed picture of the properties of the rock in the extended area. It will be possible to describe the locations and water-conducting properties of deformation zones with improved precision. Safety-related technical specifications (STF) will also be developed, and the closure plan will be reviewed and updated.

Prior to regular operation of the extended SFR, a supplemented SAR will be developed at the same time as the STF are updated. The documents are then supplemented with experience from trial operation. Trial operation is not expected to require any specific technology development 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, techniques and installation. The decommissioning plan will be supplemented, and an updated assessment of post-closure safety will be included in the revised safety analysis report that is prepared prior to closure.

4.1.2 Final Repository for Long-lived Waste

Development needs following safety evaluation of the repository concept

In an initial phase, the continued development of the Final Repository for Long-lived Waste, SFL, will focus on processing conclusions and experiences from the completed safety evaluation of the proposed repository concept. The conclusions will be the basis for the continued development of the engineered barriers, the preliminary acceptance criteria for the waste and for the siting process.

In 2020, SKB plans to start a programme for the planning and implementation of final disposal of long-lived waste. The purpose is to create a programme organisation with a focus on facilitating the final disposal of long-lived waste and to coordinate the different lines of development for SFL. In the programme for SFL, the following is therefore planned during the RD&D period:

- Carrying out research and studies concerning safety after closure based on the results of the safety evaluation.
- Commencing the siting process.
- Building up an organisation for site investigation.
- Continued technology development, including practical experiments.

The need for an improved level of knowledge, to support future in-depth assessments of post-closure safety, was identified in the safety evaluation, see Chapter 6. This includes needs for research and development aimed at an improved level of knowledge with respect to features, events and processes (FEP) that may affect post-closure safety. Since there are many similarities between SFL, SFR and the Spent Fuel Repository, the needs for an improved level of knowledge for SFL often coincide with needs identified for SFR and/or the Spent Fuel Repository. The needs for research and development identified in the safety evaluation primarily concern:

- In-depth knowledge of the inventory of radionuclides in the waste to be disposed of in SFL. This applies to both forecasted waste quantities and the content of radionuclides in the waste (Section 4.2).

- Increased knowledge of the evolution over time of the concrete barrier in the waste vault for core components, BHK. Continued studies of interactions between groundwater and concrete under repository conditions (Section 4.5).
- Increased knowledge of the evolution over time of the bentonite barrier in the waste vault for legacy waste, BHA (Section 4.6).

Needs for the development of tools and methods used in the evaluation of post-closure safety have also been identified. The plans for meeting these needs are partially dependent on the results of already planned research and development, and will be developed in detail during the RD&D period.

Continued research and safety assessment specific to SFL's needs will be coordinated with technology development and the siting process in the programme. The work will be conducted by iterations between technology development and safety assessment according to established methodology, but with the goal of carrying out faster iterations between them. Work tools developed within the safety evaluation permit faster feedback on technology development, which provides greater flexibility. Thus, SKB does not today foresee a need for a specific report in the form of a safety assessment without site data, as was reported in previous RD&D programmes. The next planned complete safety assessment for SFL is the safety assessment that will be submitted in support of the applications for constructing and operating SFL.

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 that are deemed to require SFL-specific development efforts are:

- Technical methods for the design and construction of waste vaults.
- Technical methods for backfilling the waste vaults with concrete and bentonite.
- Management and final disposal of large components such as PWR reactor pressure vessels.

During the RD&D period, the development of the design of the waste vaults will continue, with the aim of developing the proposed concept to a first layout of the facility. Structures for safe management and storage of waste need to be developed, which simultaneously allow for efficient backfilling of the waste vaults. The work includes further evaluation of which initial state can be achieved given the practical limitations in the waste vaults, and how the initial state can be verified at closure.

During the RD&D period, SKB will also investigate whether it is possible to develop the concept so that it is less sensitive to the site specific properties regarding for instance groundwater flow. In a first phase the development is being conducted to a level that permits relevant comparisons between different conceptual designs, with regard to safety, technology, the environment and costs.

In this RD&D programme, SKB presents fundamental prerequisites and starting points for the siting of SFL, see Chapter 6. During the RD&D period, the process will be developed in consultation with the concerned authorities and municipalities. However, the siting process does not require any specific development efforts.

Licence applications

In support of the applications under the Nuclear Activities Act and the Environmental Code to construct, own and operate SFL, a preparatory preliminary safety analysis report (F-PSAR) is submitted. That means that a site needs to have been selected and described as a basis for the assessment of post-closure safety.

The technology development and the understanding of the barriers of concrete and bentonite and the natural system, i.e. the host rock and the ecosystem, need to have been pursued so far that it is possible to show in the safety assessment that the repository meets the requirements of the Environmental Code (general rules of consideration) as well as the requirements in the Nuclear Activities Act and SSM's regulations. 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. As SFL differs from the other repositories above all when it comes to structures in the waste vaults and technical methods for backfilling, these areas are expected to require development.

Preliminary acceptance criteria for the waste must be formulated, as do technical solutions for treating and conditioning the waste to meet the acceptance criteria.

Construction, operation and closure

Prior to the application for construction, the preparatory preliminary safety analysis report (F-PSAR) will be updated to a preliminary safety analysis report (PSAR). The PSAR will contain an updated presentation of the activities, the design of the facility, and how the specified requirements are met.

The work with technology development will have resulted in a basis in the form of requirements for design of the repository.

The updated safety analysis report (SAR) that is produced before trial operation includes an updated assessment of post-closure safety. There, additional data from construction are included together with new knowledge from technology development and research. Prior to trial operation, some technology development is required in order for SKB to ensure that the barriers and other repository components will meet the stipulated requirements after installation. This basis must be sufficiently detailed to allow design of the closure. The closure plan and the decommissioning plan are updated based on the detailed descriptions.

Prior to decommissioning and closure of the facility, planning and detailed design of the closure components with regard to materials, techniques and installation are carried out. The decommissioning plan will be supplemented and an updated assessment of the safety after closure will be included in the revised SAR.

4.1.3 The 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 acquisition of knowledge for questions concerning post-closure safety is primarily focused on providing a basis for the SAR. The submission of the PSAR is an important checkpoint. More knowledge is primarily required for analysing uncertainties and assessing if they may be reduced. This yields both a more realistic assessment post-closure safety and a basis for optimisation of the repository so that more detailed requirements can be formulated for repository components and layout.

The goal of technology development is to ensure that the technology needed to begin construction of the Spent Fuel Repository and the encapsulation plant in Clink is available prior to the start of construction. This means that parts of the detailed design phase should essentially have been completed, for example for investigation methods and technology for construction of the repository accesses. This basis is needed in order to address issues of importance for radiation safety during the entire life cycle of the facility already during the construction of accesses, the central area and the first deposition area. The basis is presented both in a document on safety during construction of the final repository and in a detailed site investigation programme for the construction phases of the final repository. The purpose is to identify the activities and measures that are essential for verifying the site descriptive model of the Forsmark site and developing it further. Furthermore, the technical systems (for example for deposition and backfilling) that will be present in the repository area need to be developed prior to the PSAR and SAR.

Technology and methods for the production of canisters must be developed and described prior to the construction of Clink. The necessary technical systems in Clink must be specified, which affects the development of the nuclear fuel measurement and the chosen method for drying of fuel assemblies. In the same way, methods for welding and inspection of the canister during encapsulation need to be developed and adapted to the nuclear environment existing in the facility. In the design of the encapsulation plant in Clink and of the final repository, the prerequisites for safeguards must be observed.

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. Thereby construction and installation methods for buffer, backfill and plugs must be in place, as well as methodology for excavation of deposition tunnels and deposition holes. Furthermore, the inspection methods that are to be applied must have been verified, and methods must exist for investigation and modelling of the rock in the deposition area.

In Forsmark, the monitoring of the rock, the groundwater and the ecosystem has continued virtually unchanged after the site investigation and the monitoring is planned to continue until the start of construction of the Spent Fuel Repository. Some adaptation has, however, been made or is 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 the possible environmental impact during repository construction and operation.

The same type of monitoring of geosphere and biosphere parameters is planned to continue during the Spent Fuel Repository's construction and operation. What is new, relative to the site investigation, is predominantly the monitoring that will be performed under ground. Prior to the construction of the Spent Fuel Repository, a monitoring programme will be established where monitoring activities during construction and operation of the final repository are presented (Section 4.10).

Operation

In the Spent Fuel Repository and Clink, integration testing and commissioning tests are planned, which will verify that construction and deposition can be carried out in the Spent Fuel Repository while achieving both safety during operation and after closure. These tests are carried out in a late stage with the equipment and personnel that will operate the facility. This is a final control that operation can take place in the intended manner.

Prior to the commissioning test, all systems for handling and transport of canisters, buffer and backfill must have been manufactured, installed and tested. Process qualification with appurtenant equipment, personnel and suppliers must be completed and documented. At that time, several systems for quality control and inspection of the barriers will be implemented. This applies both to production, handling and installation of canisters, buffer and backfill components and to the rock construction process with associated detailed site investigations.

Prior to the preparation of an updated SAR, efforts will be required to handle additional data from the repository construction in the assessment of post-closure safety.

When the commissioning test has been carried out and the updated SAR has been approved by SSM, trial operation, where spent nuclear fuel is handled and deposited, can be commenced.

Before the KBS-3 system is put into regular operation, the safety analysis report, SAR, is supplemented with experience from the trial operation. As the repository system is commissioned, activities shift to a management phase, where the SAR must be kept up-to-date.

Retrieval of deposited canisters

In Sweden there are no provisions in laws or other statutes requiring that spent nuclear fuel that has been deposited in a final repository shall be possible to retrieve. According to SSM's general advice on the regulations concerning safety in connection with the disposal of nuclear material and nuclear waste (SSMFS 2008:21), measures can be adopted with the primary aim of facilitating the retrieval of disposed canisters. However, such measures must not lead to deterioration in post-closure safety of the repository. There may be situations where retrieval is necessary before closure.

SKB expects it to be possible to retrieve deposited canisters during the operation of the Spent Fuel Repository. This has been demonstrated practically in experiments in the Äspö HRL, most recently in connection with the opening of the outer section of the so-called Prototype Repository (Svemar et al. 2016). In principle it would also be possible to retrieve canisters from a closed repository, but the execution would require significantly more work and resources. The encapsulation plant in Clink is designed so that it will be possible to retrieve canisters containing fuel for re-encapsulation. Retrieval will be possible as a means of dealing with any defects that may arise or be detected during the deposition sequence. These plans will be presented in the PSAR that is submitted as a basis for the construction of the Spent Fuel Repository. To allow for retrieval, some technology development is needed. It must be ensured that the deposition machine can move canisters back. Also practical methods for handling bentonite blocks that have become partially water-saturated need to be developed.

Closure

Prior to closure of the Spent Fuel Repository, a revised SAR will be produced. It will contain an updated assessment of the safety after closure, plus a plan for closure and decommissioning. The updated assessment of the safety after closure will be based on the as-built facility and the planned closure measures. 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 controlling the repository and the operations during closure.

4.2 Low-level and intermediate-level waste

This section briefly describes the needs for research and development regarding low-level and intermediate-level waste that SKB will prioritise during the RD&D period. A deeper process understanding of certain materials in the low-level and intermediate-level waste to be disposed of in SFR or SFL is needed. Furthermore, the understanding of the content of radionuclides in the waste has to be updated and deepened. The current situation and the programme are presented in Chapter 7.

4.2.1 Process understanding

For both SFR and SFL, the retention of radionuclides in the repository is a key safety function. One of the most important processes contributing to retention is the sorption of radionuclides on cement minerals. Chemical degradation of organic matter in the waste can generate products that can form complexes with radionuclides and thereby reduce the degree of sorption on cement minerals.

In the past three years, SKB has studied how complexing degradation products of cellulose affect the sorption of plutonium. During the coming years, SKB will participate in continued studies of how organic substances affect radionuclide transport in the repository environment, with a focus on survey of degradation products of filter aids and how they affect the sorption capacity of the cement.

4.2.2 Radionuclide inventory

Knowledge of the radionuclide inventory for the low-level and intermediate-level waste needs to be improved to reduce uncertainties prior to future assessments of post-closure safety for SFR and SFL. This applies both to forecasted waste quantities and to the content of radionuclides in the waste. The radionuclide inventory is updated continuously, and special effort is made for determination of the so-called difficult-to-measure nuclides, which in many cases are significant for post-closure safety. SKB is also participating in the development of methodology for characterisation of decommissioning waste in consultation with the reactor owners.

During the RD&D period, SKB will develop the methodology for determination of difficult-to-measure nuclides, on the one hand by expanding the set of measurement data used for activity determination, and on the other, by carrying out additional verifying measurements of certain difficult-to-measure nuclides.

4.2.3 Handling of the waste

Acceptance criteria for the long-lived low-level and intermediate-level waste will not be finalised until the design of SFL has been decided. There is already a need, however, for clarifying the planning prerequisites for the handling of the waste produced during operation and dismantling and demolition of the nuclear facilities. Results and conclusions from the safety evaluation for SFL, as well as knowledge of acceptance criteria for final disposal, interim storage, and transport of waste to SFR, will serve as a basis for the continued development of preliminary acceptance criteria for the long-lived waste and for the waste that will be disposed of in the extended SFR. Future acceptance criteria for the long-lived waste may make it necessary to stabilise certain types of waste, for example segmented core components, which today are interim-stored in steel tanks at the nuclear power plants, before final disposal in SFL. A method for stabilisation of waste in steel tanks, for example by grouting with concrete, will then have to be developed. Similarly, future requirements may be imposed on conditioning and packaging of certain types of waste from AB SVAFO, Cyclife Sweden AB and Studsvik Nuclear AB, resulting in a need for technology development.

Besides nuclear fuel, core components are also stored in Clab, mainly BWR control rods. Segmentation of BWR control rods is still regarded as the main alternative before final disposal in SFL. SKB continuously evaluates the need of measures to ensure the interim storage capacity for spent nuclear fuel in Clab, and according to current plans, segmentation of control rods has to commence in 2023 in order to free up storage capacity by packing them more tightly in new storage canisters after segmentation. Development activities related to segmentation of control rods will therefore have to be carried out before then.

According to current planning, reactor pressure vessels from the PWRs at Ringhals will be disposed of in SFL. They may be segmented and handled in smaller parts, or handled whole (with or without internals left in the pressure vessel). There is a policy decision for Ringhals 2 to segment the reactor pressure vessel, while decisions for Ringhals 3 and 4 will be made closer to the decommissioning of these reactors. For the segmented parts, interim storage in waste containers at Ringhals is planned awaiting continued handling, transport and final disposal. No particular technology development is considered to be required for this waste category, beyond that which is already planned generally for handling, transport and final disposal of long-lived waste in steel tanks. In the case of a decision for final disposal of whole reactor pressure vessels from Ringhals 3 and 4, however, some technology development will have to be carried out, related to transport, set-up, backfilling and closure.

4.2.4 Waste containers and waste transport containers

In order to be able to transport waste from the decommissioning of the nuclear facilities in an optimal manner, development of waste containers and waste transport containers for long-lived waste and short-lived decommissioning waste is required. According to current plans, a new waste transport container for long-lived waste (ATB 1T) is going to be certified and commissioned around 2020, and other types of waste transport containers may be developed. The need for waste transport containers is determined by the additional waste packages that will be developed.

Technical issues concerning how long-lived waste from AB SVAFO, Studsvik Nuclear AB and Cyclife Sweden AB is to be treated and packaged have to be resolved when acceptance criteria for long-lived waste have been established. This can for example include development of new types of containers.

4.3 Spent nuclear fuel

Spent nuclear fuel is long-lived and high-level and requires radiation shielding in all handling, storage and final disposal. It constitutes a minor fraction of the total amount of nuclear waste to be disposed of, but contains by far most of the total radioactivity. Spent nuclear fuel can become critical and must be handled in such a way that it does not reach criticality.

If a canister in the Spent Fuel Repository is breached and water enters, the fuel properties are crucial for 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 assessment of safety after closure 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 and thereby reduce the pessimism in future safety assessments.

In the coming years, both research for future assessments of post-closure safety, and technology development regarding handling of the spent nuclear fuel in the different parts of the KBS-3 system are needed.

For the assessment of post-closure safety, there are a number of issues requiring further research, primarily before the SAR, with the level of knowledge in the PSAR and before a decision on start of construction as important cross-check points. Technology for handling of fuel will be developed as it is needed as a basis for basic design and detailed design of Clink, and is 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 8.

4.3.1 Process understanding

Due to radioactive decay, the spent fuel's content of radionuclides decreases with time. In the beginning, during the first centuries after closure of the final repository, gamma radiation from the fission products dominates. However, there are certain fission products with a very long half-life, for instance iodine-129, which are important in the assessment of safety after closure. Other very long-lived radionuclides are various actinides, which generally decay by alpha radiation, and after 300 years this will be the dominating radiation. The risk associated with gamma-emitting radionuclides, which is greatest during handling of the fuel, is avoided through various radiation protection measures such as radiation shielding. The risk associated with alpha-emitting nuclides is linked mainly to direct contact with these substances, for example by ingestion of food. After encapsulation in copper canisters, the radionuclides are contained as long as the copper canister remains leak-tight. The safety assessment presents different scenarios where the integrity of the copper canister can be jeopardised.

Fuel dissolution

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 explaining the observed hydrogen effect. Previous results have shown that reactions at the fuel surface are central, and that the metallic particles that are present in the fuel play an important role in keeping the surface reduced despite alpha radiolysis in the surface layer. Continued research focuses on possible catalytic effects of an oxide surface without metallic particles, and how different substances used as additives in the uranium dioxide can affect the electrochemical properties of the solid phase.

Radionuclide speciation and solubility

The distribution of radionuclides in the different parts of the fuel, especially how large a fraction is situated in the gap between pellet and cladding, is also of importance for the assessment of post-closure safety. Research is therefore in progress concerning how this depends on the fuel type, power history and burnup. There are remaining uncertainties concerning speciation and solubility of released radionuclides, including uranium. As the understanding of uranium chemistry in the repository environment is a cornerstone in the description of the chemical evolution in a broken, water-filled canister, continued research efforts in this area are required.

4.3.2 Fuel integrity, fuel characterisation and fuel information

In order to ensure that all spent fuel will be possible to handle in both Clab and the encapsulation plant, a programme for inspection of the fuel's integrity is being conducted (the so-called ageing programme), where changes in the properties of the fuel during pool storage are being studied. SKB is monitoring international research and experience regarding changes in fuel integrity to use this knowledge in the ageing programme and prior to the handling of the fuel in the encapsulation plant. It is important to ensure that the fuel is not negatively affected by the stresses resulting from the planned handling. One example is the drying of fuel assemblies to be placed in canisters, in which the temperature changes may affect the cladding tubes of Zircaloy surrounding the fuel.

Non-regular fuels are fuels that substantially differ from the regular spent nuclear fuel. Two important examples of non-regular fuel are failed fuel and fuel samples and residues from analyses of various types, above all from Studsvik. Since the properties of these fuel types differ from regular fuel, separate handling and analysis of these are required. The results of the efforts in this area will be used in future safety analysis reports.

An ongoing project on fuel information aims at gathering information on the fuel needed by SKB now and in the future. Within this project, studies are being carried out on how information on the spent nuclear fuel should best be handled and stored prior to commissioning of the complete KBS-3 system.

The decay heat of the spent nuclear fuel is a key property that influences many aspects of the entire handling chain and of the final repository, for instance during encapsulation when the choice of fuel assemblies for encapsulation is largely controlled by the decay heat of individual fuel assemblies. Efforts to improve and update calculations of decay heat in Clab are ongoing. At the same time, work

continues with calorimetric determination and determination by nuclear fuel measurement (particularly gamma and neutrons). Information on decay heat is needed above all for selecting fuel assemblies for the placement in each canister, but another important purpose of this work is to provide a basis for the design of the parts of the encapsulation plant where the measurement equipment will be used. Furthermore, the work with fuel measurement aims at ensuring 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.

4.3.3 Criticality, radiation and safeguards

For criticality calculations, the method for burnup credit has been implemented, a development project that includes validation of a new version of the computer code Scale. Current issues include an analysis of criticality in post-closure scenarios with geometry changes in a broken copper canister, and a strategy for handling of fuel bundles that do not meet the burnup requirements. The calculations made with Scale also provide a basis for assessing the radiation field's impact on the environment. Technology development also takes place in the field of nuclear safeguards, in cooperation with the IAEA, Euratom, and SSM.

4.4 Canister for spent nuclear fuel

The copper canister is an essential barrier in the KBS-3 system as it provides containment of the spent fuel. This section describes the supplementary research needed on the properties of the copper canister (copper shell and nodular cast iron insert) prior to future safety analysis reports 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 to be used in the KBS-3 system. Chapter 9 presents the planned development efforts during the RD&D period.

4.4.1 Process understanding

Before the PSAR for the Spent Fuel Repository, SSM has pointed out areas with additional needs for analyses, and the majority of them concern the canister and its function in the repository. In order to develop the assessment of safety after closure, both more knowledge of the processes and a refined use of this knowledge in the assessment are needed, and this is achieved by means of an iterative process. SKB thus foresees that further studies of the processes are needed prior to the SAR.

Corrosion

Sulphide corrosion is the most important corrosion process for the copper shell, in particular if the buffer has eroded away. By primarily electrochemical experiments, the detailed mechanisms for corrosion can be studied, such as the formation and morphology of the corrosion film, the appearance of the copper surface after corrosion, and the conditions for the corrosion being more localised (which gives deeper attacks for a specific amount of corrosion than if the corrosion is more evenly distributed).

SKB will also continue to study the mechanisms for stress corrosion cracking and radiation-induced corrosion, and to develop the modelling of localised corrosion in the assessment of post-closure safety. A special focus will be 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 for avoiding overestimates of the transport of sulphide to the canister (and thereby of corrosion) as far as possible.

Material properties of canister materials

SKB has chosen an oxygen-free copper with low content of impurities. In order to get favourable creep properties (sufficiently high ductility), phosphorus is added. 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 (down to an atomic level) to show that the material properties do not change over long periods of time. SKB will therefore continue studies of the effect of phosphorus on the creep properties,

both experimentally by testing and microscopy, and theoretically with above all quantum-chemical calculations. SKB also plans investigations for studying the possible intrusion of hydrogen into copper, and whether this could affect the material properties to any significant extent.

4.4.2 Technical design

When it comes to technology development for the canister, the design phase should essentially be completed as a basis for the PSAR. This means that the design and defect analyses are updated based on current technical design requirements. Further knowledge of the metals' behaviour in manufacturing and practical demonstrations is needed to formulate requirements on dimensions, mechanical properties, chemical composition, parameters in manufacture, and acceptance criteria for errors or deviations.

SKB is responsible for the canisters as a product, but has only direct control of certain processes in the production, for instance the sealing by means of friction stir welding in Clink. Through a clear set of requirements and feedback to suppliers, inspection and testing of the canister and its components, and qualification of certain manufacturing processes, inspection procedures, and testing systems, it will be possible to achieve and verify the required canister properties.

4.4.3 Manufacturing, inspection and testing

Updated manufacturing requirements (specifications) and inspection plans will be prepared prior to trial fabrication of canister components and prior to assembly and sealing of complete but empty canisters.

The canister's components and welds are tested to determine mechanical properties, dimensions, and the occurrence of defects. SKB will have to demonstrate that the canister, including its components and the seal weld, is testable. In order to be able to demonstrate this, requirements on testing methods are being formulated, based on an assessment of possible defects, the canister's damage tolerance, and industry-specific norms.

In conjunction with the PSAR for the Spent Fuel Repository, SKB intends to present a strategy for the inspection and testing of the barriers in the final repository, including some elaboration regarding the production of the canister. However, established processes for qualification, inspection and testing will not be in place before the PSAR for Clink or the Spent Fuel Repository, but are instead planned to be finished prior to the commissioning test.

Canister components have been test-manufactured in steel and metal plants and engineering industry in Sweden, Finland and other European countries. Prior to the industrialisation of the production system for canisters, supplier capacity and logistics for canister components have to be studied and measures proposed for ensuring the delivery of canisters with consistent and sufficiently high quality to the encapsulation plant. The goal is that the process for manufacture and inspection of canisters will be tried and tested and possible to carry out on an industrial scale within the RD&D period, whereas qualification of the process is to occur at a later stage.

The equipment that will be used in Clink for the sealing of canisters and inspection of seal welds has to be adapted to the nuclear environment. The adaptation will serve as a basis for the detailed design of Clink.

4.5 Cementitious materials

Cementitious materials occur in the waste matrices, barriers, and structures in SFR and SFL. In all repositories, cementitious materials are also used in plugs, rock support and for grouting.

This section provides a general description 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 that is needed for the design of concrete structures, materials and production methods is described.

For a detailed programme for the RD&D period with regard to cementitious materials in SFR, SFL and the Spent Fuel Repository, see Chapter 10. The programme linked to cementitious materials in the waste matrix is described in Chapter 7.

4.5.1 Process understanding

During the time period covered by an assessment of post-closure safety, the composition and properties of cementitious materials will slowly change. These changes can be caused by chemical processes, such as interactions with groundwater or elements dissolved in the groundwater, or by mechanical processes, such as rock movements or pressure caused by swelling material, internal gas pressure or freezing of the concrete pore water. The properties could also be changed abruptly in case of an earthquake, and above all for SFL, the effects of an earthquake on the repository's protective capability must therefore be assessed.

As cementitious materials have a central function for maintaining the post-closure safety, SKB is conducting continuous work with the purpose of maintaining and strengthening the knowledge and the ability to model the evolution of the material properties over time.

Groundwater impact

By interactions with groundwater and elements dissolved in the groundwater, the cementitious materials' composition and structure will change. The extent of this change depends on the original composition and structure of the material and on the composition of the groundwater.

During the RD&D period, SKB intends to continue to operate and develop its programme for studies of how interactions between cementitious materials and groundwater and solutes in groundwater affect the cementitious materials' properties over the relevant time spans.

Effects of organic and metallic waste and of additives

Organic and metallic materials that degrade in a cement matrix may affect the properties of the engineered barriers both by changing the chemical composition of the pore water and of the cement minerals, and by causing local mechanical stresses. In order to gain a better understanding of how degradation products from organic and metallic materials interact with cement minerals, SKB plans, during the RD&D period, to retrieve and analyse additional samples from experiments in the Concrete and Clay project in the Äspö HRL.

SKB also intends to revise the need for continuing studies of how the different types of additives used for production of cement and cementitious materials can affect the material properties over time. The design of the studies and the time of execution will be affected, however, by the ongoing changes in the industry for cement production, and therefore no programme has been established.

Mechanical loads and gas transport

Mechanical loads that may affect the mechanical and hydraulic properties of the cementitious materials can on the one hand consist of internal, locally acting loads caused for instance by expansive minerals or by freezing of the water contained in the material's pore system, and on the other hand, of loads acting on a larger scale caused for instance by gas pressure or by glacial loads. The origin of the load and the affected area will determine what effects it will have on the properties of the cementitious materials.

During the RD&D period, SKB intends to continue to coordinate research on gas transport through cementitious materials and the loads associated with this.

SKB will also follow up previous calculations and investigations of the concrete structures in the existing SFR with a study of the base slabs in the waste vaults. SKB will study the effects of local loads exceeding the specified design-basis imposed loads, and whether the possible fracturing of any base slab entails that the initial state assumed in the safety assessment is no longer achieved.

4.5.2 Design of concrete structures and materials

In preparation for the extension of SFR and the construction of SFL and the Spent Fuel Repository, SKB is carrying out extensive work linked to the design of concrete structures and materials for the different repositories. The work that was carried out during the past RD&D period will continue in the coming period. The following sections give a summary of the planned development work. The full programmes are presented in Chapter 10.

SFR and SFL

During the RD&D period, SKB plans to continue the work on the design of repository structures and development of materials for these. The programme starts from experience and results from the work carried out during the past RD&D period, with a focus on the main areas

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

The main focus will switch from material development and development of technology for construction to aspects even closer to production.

In order to realise the repository design of SFL, with large amounts of concrete backfill, material needs to be developed and methods for installation be prepared. At present, SKB is not carrying out any specific development work on structural concrete for the two waste vaults, BHK and BHA, in SFL. Experience from previous work indicates that the concrete that has been developed for the caissons in 2BMA is also suitable for the repository structures in SFL. During the RD&D period, the work will instead be focused on studies linked to materials and methods for foundation engineering and backfilling BHK with concrete.

Spent Fuel Repository

Current technical design requirements and requirements on the Spent Fuel Repository assume the use of low-pH materials for ensuring that the material leaching products do not adversely affect the bentonite in buffer and backfill.

During the RD&D period, the development work completed so far related to low-pH cementitious materials for rock grouting and rock support will be compiled, which creates a basis for the planning of any continued work.

SKB intends, in the RD&D period, to complete the evaluation from construction and dismantling of the plug of low-pH concrete that was previously constructed in the underground part of Äspö HRL, and compile these analyses to a basis for further studies.

4.6 Clay barriers and closure

The main purpose of the clay barriers in the Spent Fuel Repository (buffer and backfill), SFR (between the rock and concrete silo) and SFL (backfill in the waste vault for legacy waste) is to restrict the water flow around canisters and containers for low-level 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, fill cavities and seal against the rock and other repository components.

For the assessment of post-closure safety, there are a number of issues concerning the clay barriers that require further research. For the Spent Fuel Repository and the extended SFR, this applies primarily until the SAR, with the level of knowledge in the PSAR and prior to a decision on start of construction as important cross-check points. The recently completed safety evaluation for SFL was primarily based on existing knowledge concerning the properties and function of the bentonite, but for future assessments, the data will have to be strengthened. Most of the processes in the bentonite barriers are common to the different facilities. The results from the research being conducted for the Spent Fuel Repository can for the most part also be used for SFL and for the silo in SFR.

For the Spent Fuel Repository, the design and production process have to be developed further so that there is a sufficient basis for the continued design of the waste vaults and the production system for bentonite components. Methods and procedures for quality assurance during manufacturing, handling and installation have to be developed in further detail. This is done by process surveys and by preparation of preliminary quality plans for the future production of buffer and backfill. To make it possible to formulate clear requirements that are practically verifiable, research is needed regarding the properties of the bentonite material, especially during the period after installation until the bentonite becomes water saturated and the full swelling pressure has developed.

The SFL Safety Evaluation has shown that both concrete barriers and bentonite barriers are potential concepts for the coming final repository from the perspective of safety after closure. Further development is needed in order to produce technically feasible designs of each barrier type so that they can be compared with each other.

For SFR, the level of knowledge is considered essentially satisfactory. Further development of the closure plan is planned, however, in order to make the design and installation more efficient, if possible.

The programme for clay barriers and closure is presented in Chapter 11.

4.6.1 Process understanding

Evolution after installation until saturation

In both the Spent Fuel Repository and the waste vault for legacy waste in SFL (BHA), 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 be developed as the bentonite absorbs water from the surrounding rock.

In the rock in Forsmark, a sizeable fraction of the deposition holes in the Spent Fuel Repository will be partly unsaturated for time periods of 1 000 years or more. This has no great significance for the bentonite's function. However, the chemical environment that is in contact with the canister will be different than the one prevailing under saturated conditions. Transport of elements in gaseous form in an unsaturated buffer can take place much faster than the transport of elements that is possible in a saturated buffer. A better understanding of gas composition during the unsaturated period is therefore necessary for being able to evaluate the significance of corrosion on the copper surface during this period with higher certainty. In order to draw conclusions regarding the long-term function of the bentonite in SFL, its development in 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 piping and erosion of material, in particular in pellet fills. In order to evaluate the consequences of erosion, it is important to understand how water is taken up in pellet fills under different conditions.

In order to more extensively understand and describe the bentonite material's evolution until saturation, further research and investigation efforts are needed, above all concerning the following issues:

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

Bentonite barrier properties in the saturated state

As mentioned above, the clay barriers' main function is to restrict the water flow around the waste in the different repositories. This is achieved by means of low hydraulic conductivity, so that diffusion is the dominant transport mechanism, and by means of a swelling pressure that makes the buffer self-sealing. In the Spent Fuel Repository, the buffer must also keep the canister in position in the deposition hole, limit rock shear movements, limit microbial activity on the canister surface, and filter colloidal particles. It is important that the properties of the buffer are maintained throughout the period being assessed.

The most important properties of the bentonite, i.e. low hydraulic conductivity, high swelling pressure and high shear strength, can all be related to its density. The relation is unique for each bentonite type and the properties vary with its composition (the montmorillonite content, for example). A fundamental understanding of the connection between the bentonite material's composition and its properties is important. Therefore, SKB will continue with detailed studies of the connection between the material's composition and its properties for different bentonite materials.

For the Spent Fuel Repository and SFL, continued efforts concerning bentonite loss due to colloid release/erosion are also needed. The results of such efforts can directly affect the outcome of the assessment of post-closure safety, for example by the possibility to relax the pessimistic assumptions regarding buffer erosion in the safety assessment SR-Site.

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

4.6.2 Barrier design, manufacturing, inspection and testing

The technology development for mechanical mining of tunnels has progressed in the world, and SKB is investigating the possibility of using this technology for the production of deposition tunnels in the Spent Fuel Repository (Section 4.7.2). As part of this involves investigating what this would mean for the backfill design. The design of the backfill also needs further efforts regarding verification of the ability of the backfill to serve as a constraint against an upward-swelling buffer.

Requirements on the installation sequence and installation of buffer and backfill components will be updated and clarified. An important question is how thermal, hydraulic and mechanical processes in the buffer affect the density from the time the buffer is placed in the deposition hole until the backfill is installed over the deposition hole. The backfill requires continued efforts concerning water management during backfill installation. Before detailed design of the deposition area can begin, installation and inspection methods for buffer, backfill and plugs must be designed in detail and tested. To verify that the installation of buffer and backfill works as intended and yields results within acceptable intervals, full-scale tests have to be conducted.

The prepared process for adapting the density requirements on buffer and backfill to a specific bentonite material has to be applied to more bentonite materials, both to improve the understanding of different bentonite materials and to develop competence for measurement methods and measurement design.

As a basis for detailed design of the production building in the Spent Fuel Repository, an evaluation of the possibility of more efficient fabrication of buffer blocks is under way. Manufacturing of entire buffer blocks by uniaxial pressing and the possibility of using segmented buffer blocks are included in the evaluation. In addition to the evaluation of buffer block manufacturing, the following steps must be completed and established prior to detailed design of the production building:

- Manufacturing methods that work on an industrial scale for production of bentonite blocks and bentonite pellets for buffer and backfill.
- Inspection methods that work on an industrial scale.
- Prerequisites for development of quality control and inspection.

4.6.3 Deposition and installation of buffer and backfill

Technical systems for deposition include, for example, deposition machine, backfill robot and transportation system for buffer and backfilling components. Prototypes or at least schematic solutions for the deposition work are available. The deposition process is intended to be automated, and in order to control and monitor such a system, an overall control system is under development. The equipment will be further developed to be available when needed for integration testing and commissioning tests.

For backfilling of BHA in SFL, development of the backfilling concept that has been evaluated in the safety evaluation is planned. The development is focused on technical feasibility.

4.6.4 Closure of boreholes and repositories

Closure

A closure sequence for the Spent Fuel Repository has to be developed, as do the requirements on the layout so that positions for plugs can be prepared when the ramp and central area are being constructed.

Based on the assessment of post-closure safety for SFR and supplementary analyses, the requirements on closure components for SFR will be revised. Completed technology development has also identified a need for further development of design and installation of the seal.

Detailed design of the extension of SFR and expanded knowledge of the rock properties may also affect the design of the closure components. Based on these data, closure will be revised in order to produce a closure that is more adapted also with respect to other aspects (such as environmental impact, flexibility, and cost-effectiveness during installation). As a part of the revision, the need for verifying tests will be inventoried.

Borehole sealing

The methods for sealing of investigation boreholes that have been developed are deemed to be possible to use in Forsmark at the present time. For implementation into regular activities, however, the methods have to be optimised.

4.7 Bedrock

The main function of the bedrock for SKB's existing and planned final repositories is to ensure stable mechanical and chemical conditions and to limit the water flow within and in connection to the repositories during the time that the waste must be isolated in order not to constitute a risk for human health and the environment. In order for this to be achieved, sufficient knowledge is needed of the conditions in the bedrock and of the processes that alter the mechanical and chemical conditions in and around the repository, and the final repositories' underground openings have to be designed in such a way that the long-term stable conditions are not jeopardised. A large part of the research and technology development issues related to the rock and the final repositories' underground openings are common for the three different repositories.

For the post-closure safety assessment, there are several issues that require further research. When it comes to both the Spent Fuel Repository and the extension of SFR, these issues have to be resolved prior to the SAR, with the level of knowledge in the PSAR and before a decision on start of construction as important cross-check points. The recently completed safety evaluation of SFL is based on current knowledge about the barrier function of the rock, and on information previously produced with respect to the Spent Fuel Repository and SFR. The safety evaluation of SFL has pointed to a few areas where further knowledge is needed (see also Chapter 6).

The technology development concerning the rock that is needed for the Spent Fuel Repository as a basis for the PSAR and the document describing safety during construction of the final repository and permitting the start of construction (approved PSAR, accesses designed in detail), is essentially completed. Further development of methods prior to the SAR is expected to lead to more effective selection criteria for deposition holes, which in turn means that fewer holes are rejected unnecessarily.

The areas with remaining issues regarding research, technology development and post-closure safety assessment are summarised below. The programme for the RD&D period is presented in Chapter 12.

4.7.1 Process understanding

Characterisation and modelling of rock properties

Modelling of the mechanical properties of the rock mass with coupled models requires that rock properties can be specified independent 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 groundwater chemical conditions. Integrated modelling is therefore needed to bound the implications of induced deformations, both in the near and far-field, as a result of thermally, seismically or glacially induced loads.

The estimations that have been made so far of rock stresses within the repository volume of the Spent Fuel Repository in Forsmark are associated with great uncertainties, mainly due to scarcity of data. Reducing these uncertainties requires both *in situ* measurements and modelling, which also will result in an improved description of the spatial variability of the stress field with respect to magnitude and orientation.

Discrete fracture networks (DFN) are used in the modelling of rock mechanics, groundwater flow and transport of solutes. The description and parameterisation of individual fractures are important input to the analysis of many issues that play an important role for the safety of the repositories after closure, as well as for issues that have to be answered during the construction and operational phases. The ongoing development of DFN methodology will continue during the next few years.

Seismic impact on repository safety

Earthquakes in the vicinity of the Spent Fuel Repository can induce shear movements along fractures intersecting canister positions. If the earthquakes are sufficiently large and close enough to the repository, induced secondary shear movements may exceed the canister failure criteria, if the slip occurs along inappropriately located and oriented fractures. This was one (out of two) mechanisms for canister failure that could not be ruled out in the safety assessment SR-Site. Understanding of shear movements in the rock is also essential for being able to formulate adequate requirements on the canister's resilience to shear load. For SFL, the possible consequences of earthquakes must be evaluated when the site and concept for the repository have been chosen.

Research on earthquakes being pursued by SKB can roughly be divided into the partially overlapping disciplines paleoseismology, instrumentation and modelling. The main purpose of these research efforts 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 evaluate the potential for a more efficient use of the repository volume.

The largest remaining uncertainty by far 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.

Groundwater flow, groundwater chemistry and transport of solutes

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 to extend their application areas (for example by including microbial processes) for use in site modelling and safety assessment. Furthermore, a development of the hydrogeological simulation tools is planned, so that groundwater flow and radionuclide transport calculations can be made in the same tool without export/import of data files with intermediate results.

In the safety evaluation for SFL, it was identified that the interfaces between concrete backfill and rock and between bentonite backfill and rock are of great importance for the assessment of post-closure safety. Detailed models for calculating the water flow through SFL and the transport of solutes from SFL to the surrounding rock will be developed. The results from more detailed models will serve as a basis for the development of a simplified radionuclide transport model that will be used in future safety assessments for SFL.

Within the field of solute transport, efforts are also needed regarding above all matrix diffusion and sorption, in terms of conceptual understanding, reduced uncertainty in transport parameters and further development of modelling tools. This mainly concerns the Spent Fuel Repository and SFL, and the efforts can primarily lead to less pessimistic assumptions in the assessment of safety after closure through the use of site-specific data with a reduced uncertainty range relative to the data that are used today.

The effects of the hydrogeological system on a final repository during glaciation and during different stages of permafrost aggradation and degradation are site-specific and hence have to be analysed in site models. SKB's Greenland projects GAP (Greenland Analogue Project) and GRASP (Greenland Analogue Surface Project) have provided knowledge on the properties of local catchments under

permafrost conditions, and have contributed to a developed process understanding, particularly with respect to how hydrogeological processes behave in an area with cold, deep and continuous permafrost. There are remaining knowledge gaps, however, linked to the process understanding for time periods when the permafrost grows or thaws. SKB intends to gather information by means of hydrological and biochemical model studies of sites that have different types of permafrost today.

Models that take into account climate effects on site-specific properties related to hydrogeology, geochemistry and transport of solutes, as well as the impact of climate on permafrost and glaciation processes and properties have to be further developed for Forsmark. Corresponding model development and process understanding is also needed for the analysis of how a warmer and wetter climate may affect the site-specific properties and hydrological/hydrogeological and biochemical processes.

4.7.2 Production, verification and inspection

Methodology for detailed site investigations with associated modelling

The detailed site investigations (including monitoring, see Section 4.10) and site-descriptive modelling activities that are carried out in conjunction with construction and operation of a final repository, provide step-by-step information on the properties of the rock, how the repository's underground parts are to be adapted to prevailing conditions and their adequacy for deposition, i.e. how well they satisfy the requirements for safety after closure. In the continued development of the detailed site investigation programme into operational programmes, methods, instruments and modelling methodology for verifying that specified requirements are fulfilled, will be developed and described.

Methodology will be finalised for discipline-specific and integrated geoscientific modelling on different scales and for different purposes. An important area where further efforts have been identified is development of the methodology for DFN modelling, i.e. modelling of the fracture network in the rock. Operational programmes for accesses and the central area and for deposition areas are established with descriptions of strategies for detailed site investigations and site-descriptive modelling, with a special focus on verification of the requirements expressed in formulated technical design requirements. The operational programmes are mutually supportive with applications that are well integrated with current production and deposition cycles.

Tunnel production

Excavation for the extension of SFR can be carried out with existing technology and no special development work is planned in relation to this.

Descriptions of requirements, methodology, execution and verification of results for all excavation carried out in the Spent Fuel Repository needs to be available as a basis for the PSAR and Suus. Further development for creating a more production-efficient method for excavation in the deposition area, such as driving of deposition tunnels and boring of deposition holes, may continue until detailed design of the deposition area begins.

Considering the deposition sequence in the Spent Fuel Repository, where several different machines and pieces of equipment are used, the tunnel floor has to be sufficiently level for the tunnel to be easily accessible and for reducing maintenance and wear of equipment. The current reference method is drill and blast, but a study of alternative methods to achieve a more level floor with less excavation-damaged zone is in progress. The final method is needed first during testing of the entire rock construction methodology at repository depth in Forsmark.

Prior to detailed design of the Spent Fuel Repository's deposition area, functioning methods and sub-processes for excavation under Forsmark conditions need to be devised and be verifiable at repository depth at Forsmark.

This means that the following methods must be determined:

- Investigation method for selection and acceptance of deposition tunnels.
- Excavation methods including methods for rock support and grouting determined for deposition tunnels.
- Method for levelling of the floor in deposition tunnels.

- Investigation method for selection and acceptance of deposition holes.
- Method for boring of deposition holes, levelling the base and constructing bevels.

In order to verify the investigation and excavation methods and ensure that they give the desired results, full-scale tests need to be conducted in underground conditions.

4.8 Surface ecosystems

SKB's research programme for surface ecosystems aims primarily at creating a basis for calculations of potential radioactive dose to humans and the environment in the assessment of safety after closure for the different repositories. The programme also provides a basis for environmental monitoring, assessments of any environmental changes and for assessing safety in the facilities in operation.

The current situation and the programme for the RD&D period are presented in Chapter 13.

Research issues concerning radionuclide cycling and dose calculations in surface ecosystems for the three different repositories overlap each other. SKB's assessment is that there are no remaining critical research issues that need to be resolved prior to the PSAR for the Spent Fuel Repository or the PSAR for the extension of SFR.

Prior to the SAR for the Spent Fuel Repository and for the extended SFR, and prior to the F-PSAR for SFL, there are several issues where further efforts are required, mainly in order to reduce uncertainties in the assessment of post-closure safety. The most important remaining issues are found within four different areas: i) uptake pathways and uptake mechanisms for different organisms, ii) temporal and spatial heterogeneity in the landscape, iii) transport and accumulation processes, iv) radiological, biological and chemical properties of certain substances (for example carbon, chlorine and the decay chain of uranium).

Regarding uptake pathways and uptake mechanisms, SKB will continue its long-term efforts to replace or supplement the concentration ratios for organisms with mechanistic models. SKB has previously developed ecosystem models for land, lake and sea as a basis for modelling of radionuclide transport and organisms' uptake of radionuclides in the surface system. As a continuation of the work with the lake ecosystem model, a model for running waters will be developed. In parallel, work on developing the terrestrial ecosystem model will continue, with a focus on plant uptake via the root system.

4.9 Climate and climate-related processes

The overall purpose of the work on climate issues is to provide the safety assessments with scientifically substantiated and updated scenarios for future climate evolution, as a foundation for the evaluation of repository safety after closure. For the different repository concepts, there are specific issues which are dependent on future climate evolutions and which must be answered in the safety assessments. The current situation and programme for the RD&D period is presented in Chapter 14.

The future climate evolutions are based in part on the current level of knowledge regarding the climate history (especially the latest glacial cycle, including the Weichselian glaciation), and in part on modelling of the future climate. Overall, the work on climate issues therefore entails developing process understanding, compiling climate history, updating safety assessment climate scenarios, and validating the models that are 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. Moreover, other disciplines included in SKB's assessments of post-closure safety will be provided with input data, assumptions and boundary conditions from climate and climate-related processes, such as the development of permafrost and ice sheets, and changes in the sea level.

There are issues with a bearing on all three repositories, which have to be studied further in order to reduce uncertainties and to support the reliability of the assessment of safety after closure for the different repositories. The issues primarily concern i) historical climate changes during the Weichselian and the Holocene, ii) sea-level variations and shoreline displacement, iii) age and long-term stability

of the rock surface in Forsmark, including quantification of glacial erosion, iv) validation of the permafrost model, and v) variability in climate and ice sheets during the coming one million years. In addition to the above points concerning future studies, continued data collection from the rock borehole in the ice sheet on Greenland is planned, as are continued weather station observations on and in front of the ice sheet.

4.10 Monitoring during construction and operation

SKB will prepare monitoring programmes prior to the construction of the Spent Fuel Repository, SFL and prior to the extension of SFR. The programmes aim at providing an overall picture of SKB's future monitoring activities during construction and operation of the final repositories. They describe which data will be collected and what they will be used for. The information that will be collected sometimes has more than one application area and serves as a basis for

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

Regarding data for the second and third point, SKB has previously conducted work within the area. In the monitoring programme, this work will be summarised and the relation to other types of monitoring will be described.

Regarding the first point, supporting safety after closure, new studies will be initiated during the RD&D period. The post-closure safety assessment is based on

- understanding of the thermal, hydraulic, mechanical, chemical and biological processes that control the evolution of the final repositories and constitute the basis for demonstrating the repository's ability to contain or retard the dispersion of radioactive substances
- verification that the installed barriers, tunnels, and rock vaults 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. Regarding verification that the technical design requirements are met, methods and procedures to ensure this will be developed. This will serve as a tool for identifying manufacturing or installation errors and other deviations in material, equipment or handling. The monitoring will contribute to

- verifying SKB's understanding of the evolution of the repository
- supporting assumptions made in the post-closure safety assessment
- identifying 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 considerably the choice of technology, siting, and time frames for carrying out 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 is to install long-term tests, with a focus on the most important aspects of the engineered barriers, down in the rock at representative locations in the repository. Some of these tests can be opened and evaluated during the operational period to provide data for the safety analysis report (SAR). Prior to closure, all such experiments should be concluded and evaluated in order to provide a basis for and confidence in the decision to close and seal the repository.

During the RD&D period, monitoring programmes for the Spent Fuel Repository and for SFR will be prepared and submitted to SSM as a basis for the application to begin construction of these repositories.

SKB has participated in the completed EU project Modern2020 and in the IAEA project “Use of Monitoring Programmes in the Safe Development of Geological Disposal Facilities for Radioactive Waste”. In the upcoming work, parameters that are suitable for inclusion in the monitoring programmes will be identified and their relevance for post-closure safety will be explained. Furthermore, qualitative descriptions of the expected evolution will be produced and, where suitable, exemplified with quantitative results from models. Discussions on which type of measures can be adopted to handle situations where monitoring result deviate from the expectations will be presented. The monitoring programmes will be updated at predetermined times. SKB also follows international work regarding monitoring.

In the upcoming work, monitoring in support of the assessment of safety after closure will be carried out within the following areas:

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

4.11 Decommissioning

Based on development work carried out previously, ongoing decommissioning and on current decommissioning planning, a number of development activities have been identified to make it possible to decommission the Swedish nuclear power plants in a safe and efficient manner. The development areas are largely common among the licensees and relate more to waste management, licensing processes, resources, and coordination than to research and pure technology development, although adaptation of available technologies will be required. This section describes some of these development areas and the challenges associated with them. Part III describes ongoing and planned activities related to decommissioning in more detail.

4.11.1 Waste management

Management of long-lived waste

Long-lived waste from the nuclear power plants will be produced before the planned final repository SFL is commissioned. The long-lived waste cannot be finally conditioned before the final set of requirements for SFL is established, which is expected to be possible at the earliest when a licence to build SFL is obtained in the 2030s.

In order to be able to handle the long-lived waste that is produced during decommissioning as completely as possible, adequate planning premises have to be established. The completed safety evaluation of the proposed repository concept for SFL is a basis for formulating preliminary acceptance criteria for the waste. These intend to provide adequate conditions for the plants’ decommissioning planning, and will allow for some conditioning of the long-lived waste during the decommissioning.

Final conditioning is expected to begin during the design of SFL and will be carried out in parallel with the construction of the facility. It will then be possible to establish a conditioning plant both locally at each facility and centrally. Up to the time of final conditioning, interim storage facilities for the waste will be needed. In several cases, this can be solved locally at the facility. Alternative solutions for interim storage of waste with other licensees are also assessed.

Clearance and management of very low-level waste

The possibilities for handling and removal of the conventional and very low-level waste are of great importance for a decommissioning project. Since material flows are large within these waste categories, there are logistic challenges, which create a need for defining alternatives for disposal of the waste before it is produced, in order to ensure adequate and efficient management. The material that will be cleared during dismantling and demolition can be managed in several ways, for example by recycling the material as backfill in the conventional demolition or by managing certain waste in conventional waste facilities. In some cases, clearance for special purposes would have to be applied, which means that clearance is associated with restrictions regarding how the material may be handled after clearance.

Examples of disposal options for the very low-level waste is disposal in near-surface repositories or SFR, and incineration of material for energy recycling with controlled handling of ash remnants. Materials that can be released from regulatory control, for example metals, are to be recycled for new use to the greatest extent possible.

Current disposal, future challenges and planned work on developing the management of the very low-level waste is described in Section 3.2.

4.11.2 Licensing processes

Clearance of a nuclear facility

In order to finally obtain a decision on clearance, i.e. a decision that the facility and the site will no longer be under the statutory requirements of the Radiation Protection Act and the Nuclear Activities Act, it is required that materials, premises, buildings and land have been inspected regarding the presence of radioactive elements and that the requirements in SSMFS 2018:2 are fulfilled. Furthermore, a decommissioning report on the execution of decommissioning, with descriptions of experience gained and the final state of the facility, must be compiled and submitted to SSM.

Since several of the Swedish nuclear power plants have been shut down or faces final shutdown and are about to initiate dismantling and demolition, there is a need to clarify the set of requirements in the area of revocation of licenses. A clarification is needed regarding when a facility ceases to be nuclear according to the Nuclear Activities Act and the Radiation Protection Act, which documentation has to be provided, and how information is to be archived. A study (SOU 2019:16) has recently been presented, containing proposals for a new Nuclear Activities Act, which among other things addresses revocation of licences. A joint work within the nuclear power companies concerning revocation of licenses has to be initiated within the RD&D period.

4.11.3 Resources and coordination

The planned execution of several parallel decommissioning projects imposes high demands on the availability of competence and personnel. This includes external consultants and contractors as well as internal resources in the nuclear power companies and administrators and experts within the regulatory authorities and the municipalities.

Within the nuclear power companies, a review of competency requirements and the availability of competence is under way in order to create a picture of the situation and initiate measures based on this. In addition, a first dialogue with potential contractors has been initiated. The decommissioning activities that require specific resources are segmenting the reactor internals and pressure vessels, and dismantling and demolishing the biological shield. In addition to these more advanced nuclear decommissioning-specific competencies, a large number of demolition contractors and radiation protection personnel will be required.

Since decommissioning of the first reactors is performed before there is disposal capacity in the final repository system, i.e. before SFR is commissioned after its extension and before SFL has been constructed and commissioned, the radioactive waste must be interim-stored prior to disposal (see the relation between milestones and decommissioning projects in Section 3.3). The earlier start of the decommissioning projects, and the need for interim storage thus arising, mean that the decommissioning projects in the current planning are less sensitive to SKB's timeplans than in the past. Since all nuclear activities are planned to cease at the Barsebäck plant after the end of the decommissioning project, there is still a need for removing the radioactive waste in order to allow for clearance of the facility. This means that some connection still remains between the decommissioning of reactors and the commissioning of the extended SFR facility, and that the decommissioning projects are dependent on an external interim storage solution for long-lived waste. The result of several projects being pursued at the same time and of the need for interim storage facilities due to a later commissioning of the extended SFR, has been better coordination between the nuclear power companies and within the corporate groups.

4.12 Other areas

SKB is following developments within two other areas that are of interest for SKB, namely issues relating to the preservation of information and knowledge through generations, and the development within other methods for final disposal.

4.12.1 Preservation of information and knowledge through generations

For many years SKB has been working with questions concerning archiving and information preservation in the long term, several generations into the future. The purpose of the work has been to try to create as good conditions for information preservation as possible.

In the consultations prior to the applications for the encapsulation plant and the Spent Fuel Repository, questions concerning preservation of information and knowledge about the Spent Fuel Repository, and who is responsible for it after closure, were frequently raised. Among other things, wishes were expressed that SKB should present a proposal for a plan of action on how to preserve information and knowledge for a very long time.

Questions regarding preservation of information and knowledge for future generations may seem most urgent for the Spent Fuel Repository, but also need to be considered for SFR and SFL. Practically, the solutions for information preservation have to be in place when a final repository is sealed, which may take place at the end of this century. It is not possible, either for SKB, regulatory authorities or other parts of society, to determine definitively today how best to proceed with something that is to take place so far into the future.

In the 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 from, among others, SSM and the National Archives. Some of these requirements entail storage with no clear end date. According to SSM's Regulations concerning archiving at nuclear facilities, SSMFS 2008:38, the archives of a final repository must be handed over, organised and indexed, to national or regional archives if activity ceases. In conjunction with permanently sealing a final repository, a review of the final repository's safety analysis report is expected to take place with the aim of determining whether the final repository is safe after closure. In a statement to the Government, SKB has proposed that questions concerning the handover of information and continued preservation and management are to be dealt with in this context.

The structured mode of working SKB is applying today regarding management and preservation of documents, data and information as a part of its regular work with safety analysis reports is a central and valuable starting point for the future assessment and the choice of which information to preserve. In addition to this work, SKB believes that it is important to have a mode of working that aims at keeping the issue alive, developing the work and disseminating the knowledge about the need where SKB's role and responsibility for the issue has to be understood in relation to SKB's other relationships to society. A prerequisite for succeeding is involvement from SKB, regulatory authorities and the municipalities as well as society in general.

SKB has participated in the international initiatives "Preservation of Records, Knowledge and Memory across Generations" and "Assembling Alternative Futures for Heritage", which are presented below.

Preservation of Records, Knowledge and Memory across Generations

Since 2011, SKB has been participating in the OECD/NEA's working group on issues of how to preserve information and knowledge on final repositories for radioactive waste across generations (Preservation of Records, Knowledge and Memory across Generations, RK&M).

A central issue is which type of data and information should be preserved viewed from the different time perspectives. SKB has concretely contributed, among other things, by participating in a practical attempt to produce a common structure for a short summary description of about 40 pages – Key Information File of a final repository. The structure was then tested for three different facilities in Sweden (the planned Spent Fuel Repository), France and the USA. The basis for the description of the Spent Fuel Repository consisted of the applications under the Environmental Code and the Nuclear Activities Act.

The work was carried out in two phases. Phase I was finalised in 2014 with a conference where one of the conclusions was that there is a large interest in the issue even outside those parts of society that work directly with the management of radioactive waste.

The work in phase II was finalised in 2018 and reported in January 2019 at a common workshop together with two other OECD/NEA projects. There the working group presented proposals for principles, methods and tools that can be used to preserve documents, information and knowledge of final repositories for spent nuclear fuel and radioactive waste over time. Among other things, the group is assessing and discussing the prospects for storage in traditional archives and cross archiving, international mechanisms, time capsules, markers (on the surface and underground), creation of traditions and heritage and how and to whom the responsibility for the final repository and the preservation of information and knowledge can be transferred after closure.

A final report is expected to be published in 2019.

Assembling Alternative Futures for Heritage

Assembling Alternative Futures for Heritage (AAFH) was an interdisciplinary research programme aimed at developing a wide, international and multisectoral framework for understanding “heritage” in its most expansive sense. AAFH was a collaboration between University College London, University of Exeter, University of York and Linnaeus University in Kalmar. The programme started in the spring of 2015 and lasted for four years.

The research programme was divided into four themes:

- Preparations for an uncertain future.
- Management at the boundary between nature and culture.
- Management of abundance.
- Preservation of diversity.

SKB participated as an industry partner, above all in the theme Preparations for an uncertain future. The work has among other things addressed the message that the computers of the spacecraft New Horizons was to be provided with based on the question of how to make a message lasting and understandable.

The purpose of SKB’s participation was to establish direct contact with persons who, in one way or another, have as a profession to work with the preservation of objects and knowledge. This was done to benefit from their experience and expertise in relation to the question of how to preserve information and knowledge on final repositories for spent nuclear fuel and radioactive waste, and to disseminate knowledge of the issue in wider circles.

Programme

Only when a final repository is finally sealed do the solutions for information preservation have to be in place. There are still many years left until then. The expectations on and methods for information preservation will be developed and will change during this time. Moreover, information preservation is not only an issue for the nuclear area. The questions and the needs are similar also for other types of hazardous waste and contaminated areas. The need for information preservation is not limited only to Sweden, but is a shared international issue. Based on this, SKB will continue during the operating time of the final repositories to participate in national and international forums and working groups where these issues are discussed and addressed. The purpose is to be aware of the current level of knowledge and to contribute to developing appropriate principles, methods and tools, as well as taking appropriate action, and continuously report to interested parties.

At present, on the basis of this, SKB intends during the period up to the next RD&D programme to participate in the following forums/working groups:

- Project Memory across generations.
- Information, Data and Knowledge Management.

Project Memory across generations

The first phase of the project Memory across generations started in 2019 with a preliminary study. The goal is to contribute to finding solutions to the challenge of preserving knowledge and information on hazardous waste across generations by initiating a collaboration with cultural heritage sectors. Based on the participating parties' needs and expectations, the project will try to develop cultural processes and strategies that facilitate information preservation over time.

Besides SKB, the participants are Linnaeus University (the initiator and coordinator), Östhammar Municipality, SSM, and the National Archives. The work also entails a close cooperation with Oskarshamn Municipality, the Swedish National Council for Nuclear Waste, and the Swedish NGO Office for Nuclear Waste Review (MKG). The ambition is to also involve parties from other sectors that manage hazardous waste.

The project is funded by Vinnova until January 2020, when an application for continued funding will be submitted.

Information, Data and Knowledge Management

In 2018, three different working groups/projects within the framework of the NEA's Radioactive Waste Management Committee (RWMC) with links to the preservation of knowledge and information on final repositories for spent nuclear fuel and radioactive waste were concluded. SKB participated actively in two of them: Radioactive Waste Repository Metadata Management (RepMet) and Preservation of Records, Knowledge and Memory (RK&M) across Generations. The third was the Expert Group on Waste Inventorying and Reporting Methodology (EGIRM).

NEA has recently decided to launch a continuation of the three efforts under a common umbrella: Information, Data and Knowledge Management (IDKM).

SKB will participate in IDKM with regard to the parts that are continuations of RK&M and RepMet. The purpose of SKB's participation is to

- continue sharing experience and developing ideas about practical execution of preparations for preserving knowledge and information on final repositories for spent nuclear fuel and radioactive waste,
- raise questions and explore proposals for solutions for which no time was available before RK&M was concluded.

Examples of questions that are suggested to be explored are ethical aspects of IDKM, possibilities to utilise the internet across generations, and prerequisites for using different types of time capsules.

4.12.2 Other methods for final disposal

Questions relating to the development of other methods for final disposal, especially disposal of spent nuclear fuel in deep boreholes, have been constantly recurring during all years of consultations prior to the applications for the encapsulation plant and the Spent Fuel Repository. In the referral procedure of the application for the KBS-3 system under the Environmental Code, these questions have been continually raised. SKB is continuing to monitor the development of other methods for final disposal of spent nuclear fuel, but is not planning to conduct any research and development in this area during the RD&D period.

General

The spent nuclear fuel can in principle be treated either as a resource, whereby fissile constituents in the fuel are isolated and reused in new fuel, or as waste to be disposed of in a manner that is safe for human health and the environment. Figure 4-1 illustrates the principles, strategies and systems for final disposal of spent nuclear fuel and/or high-level waste that have primarily been considered in earlier discussions.

SKB's assessments (SKB 2014f), as well as the Government's decisions on previous RD&D programmes, support the strategy that spent nuclear fuel in Sweden will be managed as waste. The assessments show that conventional reprocessing or partitioning and transmutation are not economically or technically interesting alternatives for final disposal of spent nuclear fuel from today's reactor programme.

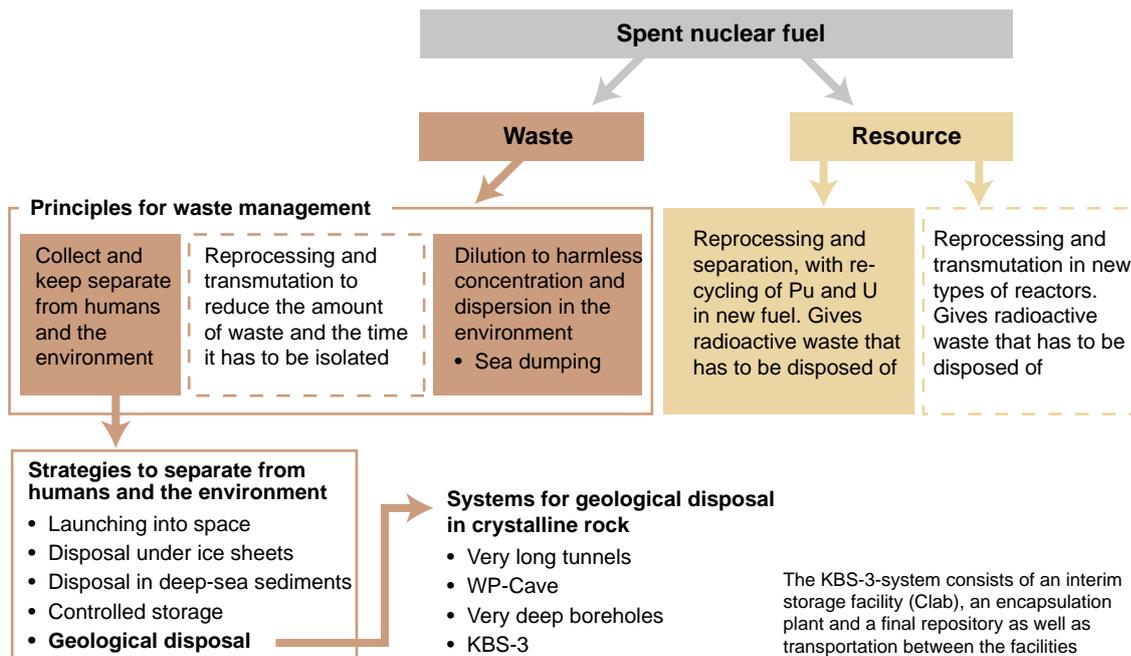


Figure 4-1. Principles, strategies and systems for disposal of spent nuclear fuel. Principles in the dashed boxes are based on technology that is not industrially available today SKB (2014f).

SKB's systematic review also shows that the most realistic strategy for final disposal of the spent nuclear fuel as waste is geological disposal. There is a broad international consensus regarding the principles for geological disposal. In most countries with nuclear power, development of such final repository systems, whose detailed design is adapted to the local nuclear power programme and the available geology, is being conducted. As shown in Figure 4-1, SKB has investigated three systems for geological disposal in addition to the KBS-3 method, namely long tunnels, WP-Cave, and deep boreholes. Of these, disposal in deep boreholes, DBD, is the system that has been put forward by some actors as being of interest.

The opinions regarding DBD that were expressed by SSM and the Land and Environment Court in the statements from these regulatory authorities in January 2018 concerning SKB's application for a licence to build and operate the KBS-3 system are presented below. A general account is also given of the work related to DBD that has been carried out by SKB and others since the RD&D Programme 2016, and of SKB's planned monitoring of efforts in this area.

Disposal in deep boreholes

The concept for disposal of spent nuclear fuel in deep boreholes that has been studied by SKB entails that encapsulated spent nuclear fuel is placed at a depth of 3–5 kilometres in boreholes drilled from the surface, after which the upper part of the borehole is sealed. The most important barrier for isolating the waste and preventing radionuclides from spreading is the rock. The concept is based on the assumption that the groundwater at great depths is stagnant due to stable density stratification, which is a result of groundwater salinity increasing linearly with depth. A more detailed technical description of the system and a comparison of DBD and the KBS-3 method can be found in SKB (2014e).

Opinions from Swedish regulatory authorities

DBD has been a question in SSM's and the Land and Environment Court's processing of SKB's applications for the KBS-3 system. Several parties have expressed the opinion that DBD should be included both in connection with the assessment according to the requirement on using the best available technology, in the Environmental Code's precautionary principle (EC Chapter 2 Section 3), and in connection with the provisions of the Environmental Code regarding presentation of alternatives in conjunction with environmental assessment.

SSM writes in its statement to the Government (January 2018) on SKB's licence application that SKB has made sufficient, albeit limited, efforts concerning alternative methods. SSM's assessment is that even though DBD may turn out to have radiological advantages in comparison to the KBS-3 method, there are large remaining challenges associated with the concept and thereby great uncertainties regarding whether deep boreholes could ultimately meet the requirements on best available technology. The Authority believes that it is not reasonable to continue interim storage of the spent nuclear fuel over a longer time for the purpose of developing DBD. SSM further believes that the EIS meets the fundamental requirements on a general report on alternatives that have been considered. When it comes to DBD, SSM deems that it is not possible to supplement the report to a larger extent without additional extensive investigations. Such investigations are not deemed to be a reasonable requirement for the reporting of alternatives.

In its statement to the Government in January 2018 on SKB's licence application, the Land and Environment Court does not express any specific opinion with regard to DBD. In the summary document, the method is not mentioned at all, while the main document refers to SKB's and other parties' accounts and SKB's supplements and replies in view of other parties' accounts.

International status

In accordance with what was stated about DBD in the RD&D Programme 2016, SKB has continued to follow the international development. A summary of the status in different countries is given below. SKB has also monitored contributions to international conferences and symposiums where, however, the activity decreased substantially after the American research programme was discontinued, see below.

The RD&D Programme 2016 reported that the United States Department of Energy, U.S. DOE, had set aside funds to a programme for drilling a five kilometres deep demonstration borehole and using the borehole for scientific investigations and demonstrations of deposition technology. U.S. DOE announced in December 2016 that it had procured four subcontractors who were to investigate the prerequisites for establishing such a demonstration borehole at one site each, in order to ultimately choose one of the companies who would be allowed to proceed and drill the demonstration borehole. In May 2017, the department announced that, due to new priorities in the budget, it intended to discontinue the financing of the programme for a demonstration borehole. By this decision, in practice the entire American research programme for development of DBD was stopped.

At the University of Sheffield in the UK, research on DBD has been pursued since the beginning of the 21st century. The programme was previously partly financed within the American programme. As this financing has now disappeared, the DBD research is now being conducted without financing, mainly by means of degree projects on the Master's level. The research group has applied for grants from research councils. Such an application to CSIRO (Commonwealth Scientific and Industrial Research Organisation) in Australia is still being processed, while an application to EPSRC (the Engineering and Physical Sciences Research Council) in the UK has been rejected.

In Germany, the focus of the programme for a final disposal of spent nuclear fuel and high-level waste is governed by the recommendations that were formulated in the final report from the "Commission for storage of high-level waste products", which was published in July 2016. The commission believes that since there are large uncertainties as to whether DBD constitutes an appropriate way, research and development efforts related to this concept must not lead to efforts to find a site for the main alternative, which is a built final repository, being curtailed. At present no funded DBD research is conducted in Germany.

The French government organisation for management of radioactive waste, Andra, has in a statement dated to January 2017 declared that DBD as a concept on several points does not conform to French legislation.

Australia has small quantities of low-level and intermediate-level waste from two research reactors and from isotope production. Disposal in deep boreholes is an alternative that is being discussed for the final disposal of the intermediate-level waste. These discussions are, however, still in a very early stage. Contacts exist with research groups mainly in the USA and the UK.

In South Korea, DBD research was pursued during 2013–2017 through Kaeri as an alternative to the programme's main strategy, a built final repository of KBS-3 type. The research, which included collection of data, modelling and development of technology for encapsulation and handling of canisters, was funded within Kaeri's internal budget. No activity on DBD is currently under way in South Korea. Kaeri intends to apply with the Government for a national research programme on DBD, which it hopes to be able to start in 2021.

Japan has no official programme for development of DBD. Japan Agency for Marine-Earth Science and Technology (JAMSTEC) was previously planning to gather geological and topographic data close to the island Minami-Tori-shima in the Pacific ocean about 1 850 kilometres southeast of Tokyo, and proposed that data could be used in conjunction with the development of DBD. It is currently unknown what happened with the proposal.

Exploratory contacts have occurred between Sandia National Laboratories and counterparts in Israel regarding development of a DBD concept adapted to Israeli conditions. At present, the nature of these contacts is unknown.

Evaluation and further work

The international activities for development of disposal in deep boreholes, which previously mainly occurred in the USA and the UK, have been abandoned as a result of U.S. DOE's decision not to continue funding the programme for a demonstration borehole. Although the concept is still being discussed as a partial solution in a couple of countries, SKB has not identified any ongoing research and development in the issue in any country.

SKB's stands by its assessment from previous RD&D programmes: that disposal in deep boreholes is not a realistic method for final disposal of spent nuclear fuel. Radiologically safe operation of a borehole repository cannot be guaranteed, since construction, operation and closure cannot be carried out in a manner that is controlled and verifiable at all stages. The uncertainties regarding the future evolution of a final repository based on the concept of deep boreholes are deemed to be greater than for a repository based on the KBS-3 method.

Despite the aforementioned objections, SKB intends to continue to monitor the development areas drilling of and disposal in deep boreholes. However, there are currently no reasons to conduct a research programme under SKB's own auspices. An example of developments that SKB intends to follow is the Swedish Deep Drilling Program, in order to thereby obtain data and other results that are relevant for disposal in deep boreholes. Furthermore, SKB has the intention to continue to monitor and in relevant contexts participate in international forums regarding disposal in deep boreholes.

5 Procedures, resources and competence

In order to be able to manage the radioactive waste and the spent nuclear fuel in a safe and cost-effective manner, SKB has developed a systematic approach for carrying out the research, development and demonstration needed to construct and commission new facilities. For the facilities in operation, SSM's regulations apply and how these are implemented regarding 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 describes how SKB prioritises research and development activities and gives an overview of how SKB ensures that the necessary competence, resources and tools are available.

SKB's systematic approach for carrying out research, development and demonstration is developed progressively but the fundamentals are the same. Consequently, the presentation in this chapter mainly follows the corresponding chapter in the RD&D Programme 2016. Updates have primarily been made in Section 5.5 Resources and competence. 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 Chapter 15.

5.1 Research

5.1.1 Research goals

SKB's research programme's goal is to ensure availability of the knowledge that is necessary to design, select site, obtain licences, plan, build and operate the planned facilities and to maintain safe operation of SKB's existing facilities. This means that the research activities should

- provide sufficient knowledge of post-closure safety and make sure that safety can be assessed for SKB's existing and planned facilities also in the future,
- provide an adequate basis for the continued technology development and planning that is needed in order to achieve efficient and optimised solutions that simultaneously provide safety both during operation and after closure of SKB's final repositories.

5.1.2 Management of research

The aim of SKB's research programme is largely established in the RD&D program, which, in accordance with the Nuclear Activities Act, is submitted to SSM every three years. In order to initiate, prioritise and support the research, SKB has a research council with representation from different activities with a strong connection to the safety assessments.

In conjunction 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, with detailed development of the plans for the coming years. In this way, there is a continuous review and follow-up of the established research programme. 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 proposals for plans, including estimated resource requirements, for the coming years to the research council and other interested parties within SKB. The research council acquires, through the presentations, an overview of the research needs and makes an overall assessment of the need for efforts during the coming years. The research council's assessment then serves as a foundation for planning of activities and is an important basis in the work with the coming RD&D programme.

5.1.3 Future focus of the research

The purpose of the repositories is to protect human health and the environment from radiological harmful effects of the waste in the long term, and post-closure safety is therefore essential for the design, siting, construction and operation of the repositories. Formally, whether a repository has an acceptable

level of safety is determined by the Government and authorities reviews of SKB's assessments of safety after closure. 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 in each assessment of post-closure safety is the evaluation of the knowledge base, both with regard to 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 extended SFR, as well as the recently completed safety evaluation of 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. Safety assessments are thus fundamental for the research programme.

The stepwise decision process for nuclear facilities is based on continuous updates and decisions regarding approval of the safety analysis report (SAR) during the construction and operation of the different facilities (Section 3.1). Each decision step for a final repository requires an assessment of post-closure safety and prior to each decision SSM is expected, among other things, to assess whether the knowledge base concerning post-closure safety is sufficient for SSM to approve that SKB should proceed to the next step. Prior to a government decision on permissibility under the Environmental Code and to a licence under the Nuclear Activities Act, i.e. the first formal decision on a final repository, SSM makes high demands on the level of knowledge regarding the central parts of the documentation. SSM has reviewed the safety assessments for the Spent Fuel Repository (SR-Site) and for the extended SFR (SR-PSU) and supported the construction of these facilities. A central milestone regarding the level of knowledge can thereby be considered to have been passed. Even when SKB has obtained a licence to construct new facilities, the need remains for being able to make assessments of the safety of the final repositories both during operation and post-closure, entailing requirements for knowledge regarding how both the engineered barriers and the natural processes in the rock and on the ground surface interact and evolve over time. According to the regulations, the SAR should be constantly kept up-to-date and, in addition, a periodic overall assessment of the safety and radiation protection of each facility should be made every ten years according to the requirements of the Nuclear Activities Act.

The future research initiatives that are needed concern primarily SKB's existing and planned final repositories for radioactive waste and spent nuclear fuel (SFR, SFL and the Spent Fuel Repository). The scientific understanding of issues related to Clink, the Spent Fuel Repository and SFR is well developed through decades of research in the Swedish programme, and also in other countries' national programmes and joint international projects. Maintained knowledge, new knowledge and new research are needed, however, to meet the future needs. With the start of construction of the new facilities, the focus of the work will shift from research and schematic solutions to technology development of quality controlled industrialised systems. Research initiatives and safety evaluations will be needed as support for technology development and in particular for the development of useful technical design requirements on the final repositories and for being able to verify that the developed technical solutions fulfil these requirements.

Although there is currently extensive knowledge in all the areas that are of importance for the post-closure safety, new questions can be expected to arise. The origin of questions that may require research activities is of several kinds:

- Questions are generated or identified internally at SKB during development of repository concepts and in conjunction with the execution of safety assessments during operation and after closure. Also site investigations can generate questions, and as sites are being investigated in more detail, additional questions may arise. New or modified waste types may also lead to new questions.
- Questions are also raised in the rest of the world, for example within the scientific community and in SKB's sister organisations in other countries.
- Questions are raised by SSM and reviewing bodies in reviews of the RD&D programmes and in the licensing processes.
- Changes in requirements in other parts of the world may entail that new questions need to be handled.

New questions are analysed and discussed in the research council before a decision is made on focus and scope of any research activities.

The monitoring of knowledge in related areas that may be of importance for the Swedish nuclear waste programme in the future is included in SKB's task. It may, for example, concern the development of methods for treatment and final disposal of radioactive waste, reprocessing of spent nuclear fuel, development of new waste types and new types of reactors. This mainly takes place through the maintenance of international contacts and by following journals from the industry.

5.1.4 Review, openness and transparency

SKB's research is conducted on the basis of the requirement that research results should 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 for use in the assessments, see also Section 5.4.

The fundamental principle is that SKB's research results will be published in the open literature to facilitate external review. Research results have, since the research programme was initiated in the 1970s, 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 results. An independent review of the results is carried out through the peer review that takes place before publication. SKB's research results are also presented and discussed at scientific conferences and are published in the conference proceedings.

SKB is also working to disseminate research results outside the scientific community. For example by publication in popular scientific journals, publication on SKB's website and in SKB's paper Lagerbladet, themed evenings (mainly in Östhammar) and information at schools and universities.

5.2 Technology development

5.2.1 Technology development goals

The goal 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 shall take place in a regulated, controlled and rational manner while the requirements on post-closure safety, low radiation dose during operation of the facilities and limited impact on the external environment are met. The development takes place in steps, taking into account that the duration of the development of different subsystems may differ.

5.2.2 Control of technology development

Technology development is commissioned by the functions in SKB that are responsible for constructing the new facilities. Control of technology development is based on what needs to be finished at the milestones identified for the different construction projects described in Chapter 3. Successive planning is done to clarify and justify the technology development that is needed for the commissioned facility (completed technology development), when it needs to be completed and which resources are required. The plans are based on the technology that needs to be available, implemented and deployed when the trial operation of the facilities begins. Thereafter, an evaluation is made of how far this technology needs to be developed prior to the early milestones within each of the construction projects. A more detailed planning is done for the upcoming years, where goals and needs are presented in the description of well-defined technology development projects.

The technology development projects have primarily been organised as a number of production lines linked to the properties of the spent nuclear fuel and the low-level and intermediate-level waste, as well as the waste system barriers and other components such as canister, buffer, cementitious materials,

backfill, closure and rock. This division is reflected in the structure of Chapter 4. Technology development pertains to systems, i.e. both physical products such as machines, barriers and measurement instruments, and processes that describe how the products are used to achieve a certain function and how the system interacts with its surroundings, including humans, technology and organisation.

In the preparation of the RD&D programme, a review is made of the current status of technology development, which is harmonised with and adapted to the general time plans for the development of the system for management of radioactive waste and spent nuclear fuel, as described in Chapter 3. After preparation, the plans for technology development are incorporated into the RD&D Programme and the activity plans and then presented to the company management and SKB's board for decision-making. Then the plans are harmonised annually between the construction projects, the clients for technology development and the functions at SKB that carry out the development work.

5.2.3 Technology development process

As reported in previous RD&D programmes, SKB has developed a process for management of technology development up to the moment when the systems are commissioned. The basis is that technology development is divided into a number of phases. These are:

- Concept phase.
- Design phase.
- Implementation and handover phase.

For each phase, there is a specification of what should have been achieved and thus what should serve as a basis for a decision to proceed to the next phase of development.

The model specifies on a general level what should be done in each phase and what should be delivered. It also shows on a general level how the work will be carried out. For execution of the activities in the different phases of the technology development, SKB's project management model is normally applied. The scope of the projects is determined on a case by case basis. Normally, a project does not run over several phases but is limited to a certain 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 systems.

The purpose of the concept phase is to specify the requirements on the system, the subsystem or component, evaluate several conceivable solutions and propose a technical solution (or several) to proceed with in the next phase.

The purpose of the design phase, which consists of two parts, the basic design phase and the detailed design phase, is to produce a design of the subsystem or component, to verify that it satisfies the requirements, and to formulate proposals for production, operation, inspection and maintenance of the subsystem or component.

The design phase may be iterative since it may turn out that the proposed solution does not satisfy the requirements or that it cannot be produced or inspected in a cost-effective way. As the development work proceeds, the solution is developed in more detail but more extensive modifications of the technical solution may also be needed.

Basic design includes defining the subsystem and its design and prerequisites, as well as requirements from and requirements on the whole system. This can essentially be considered to correspond to what in design planning is called system planning and will result in a basic design documentation.

The result of the basic design is, as a rule, not sufficiently detailed so that it can be immediately implemented. Therefore, more detailed design work usually remains to be done. Detailed design includes all the documentation that is needed, for example, process descriptions, organisation requirements, construction and manufacturing drawings or other documentation that clearly defines the components of the system, established construction, production and inspection methods, operational safety programmes, etc, to be able to hand over and implement the engineered system.

The implementation and handover phase involves the incorporation of products, processes and methods in the facility's activities. The scope, duration and resource needs of the phase and how the work will proceed differ depending on the system, the product or the method that will be adopted. The phase includes, at least, planning, training/transfer of competence, procurement/purchases, quality controlling measures such as qualification of procedures, equipment, suppliers and personnel that are needed for the operation of the system, as well as handover of all documentation.

5.2.4 Technical design requirements

The technical design requirements are based on internationally accepted and agreed upon radiation protection goals and safety principles, which have been translated into national laws and regulations. Based on these overall goals, safety functions for the post-closure phase of each final repository are defined. A number of technical design requirements are specified for a technical design that can maintain these safety functions. The technical design requirements comprise requirements that the KBS-3 facilities with their barriers must satisfy in order to ensure safety both during operation and after closure. These requirements specify e.g. what mechanical loads the barriers must be able to withstand, limitations concerning the composition and properties of the barrier materials, acceptable deviations in the dimensions of the barriers, and acceptance criteria for the various underground openings.

A set of technical design requirements and other requirements were specified in the applications for construction of the Spent Fuel Repository, the encapsulation plant in Clink and the extension of SFR. Technical design requirements related to the post-closure safety of the Spent Fuel Repository are presented in SKB (2009a) and for the extended SFR facility in SKB (2014b). Technical design requirements and links (relationships) between different requirements, as well as technical specifications, have been collected and structured in a database.

Technical design requirements, technical solutions and assessments of safety during operation and after closure are formulated as the work proceeds. It is an iterative process where preliminary quantitative requirements on the design are initially specified. A technical solution is devised and evaluated with safety assessment methodology with respect to whether it fulfils the requirements on post-closure safety. In parallel, potential production and inspection processes and the requirements that need to be specified for them are evaluated. Altogether, this leads to an update of the technical design requirements and the technical solution that can be taken to the next phase of technology development. A technical solution can for example be the design of a barrier or a method for constructing tunnels without causing a damaged zone that affects the barrier function of the rock.

More detailed specification or re-appraisal of the relative importance of requirements between different systems may also need to be done during detailed design or prior to implementation. The basic principles for evaluating technical design requirements that are related to several barriers are:

- Altogether, the technical design requirements shall result in agreement with requirements concerning safety during operation and after closure of the final repository.
- The technical design requirements must be achievable in practice and verifiable for all concerned barriers.
- Technical design requirements that entail simple, robust and effective solutions are preferred.

Overall assessments of repositories regarding requirements and compliance are done in conjunction with the assessments of safety during operation and after closure that will serve as a basis for each decision in the stepwise licensing process under the Nuclear Activities Act.

Regarding the Spent Fuel Repository's barriers, SKB and Posiva, in cooperation, have formulated revised and harmonised technical design requirements (Posiva SKB 2017). These requirements are based on hitherto conducted technology development work and on the conclusions from the assessments of post-closure safety for the Spent Fuel Repository (SR-Site) and the Finnish repository (TURVA-2012), which comprised a part of Posiva's application in 2012. The technical design requirements are formulated as requirements on the repository's different parts and barriers and must therefore be practically achievable and verifiable so that compliance with the requirements can be inspected during manufacturing and installation of the repository components. The requirements are based on the properties that the various manufactured repository parts should have in order to contribute to the repository's safety functions in both the short and long term and they are formulated so that, if they are fulfilled, completed safety assessments will be able to show that the repository is safe after closure.

The harmonised technical design requirements constitute the basis for the coming PSAR, even if they may need to be revised slightly as a result of SSM's review of the licence application. Further revision of the technical design requirements can also be expected in conjunction with detailed design and the renewal of safety analysis report prior to trial operation.

5.2.5 Quality assurance, control and inspection

An important goal of technology development is that it should be possible to verify that the developed technical solutions meet the specified requirements. Quality assurance, control and inspection refers to the measures that need to be taken in order to ensure and build up confidence that the requirements made on the facilities during operation and after closure of the final repositories are met. The goal is that the results obtained should conform to acceptable values for properties that contribute to safety and radiation protection.

The technical solution that is established in the development work should be possible to produce in such a way that the final product conforms to the established design. Before production can begin, the manufacturing and testing methods that SKB intends to employ must be proven stable.

Measurement and production systems' ability to produce and quality-assure products that conform to the established design is demonstrated in qualifications.

SKB is continuing to establish principles for the qualification process, which will be presented in the PSAR. The qualification of each measurement and production system is adapted to the manufactured or tested component's importance for safety after closure, available proven technology, available standards and norms, and the conditions that will prevail, both physical as organisational, in the execution of the planned production. This means that each qualification is unique, where some, in principle, only reference the standards and norms that are to be employed while others require extensive analyses and execution of demonstrations.

Testing and inspections will also be carried out during construction of the facilities to confirm that the construction works as intended and to ensure that no errors or deviations remain with significance for the safety after closure. Inspections are subsumed under the inspection programmes whose design and contents are dependent on the types of requirements that will be verified, for example inspection programmes for the external environment, programmes for investigations of the rock, programmes for inspection of rock excavations and programmes for inspection of occupational safety.

5.3 SKB's facilities for research, development and demonstration

5.3.1 Äspö HRL

The activities at the Äspö HRL, which was built during the period 1990–1995, are a continuation of the work that was previously pursued in the Stripa Mine in Bergslagen. The laboratory is situated on the island Äspö north of the Oskarshamn Nuclear Power Plant (Figure 5-1). The underground laboratory consists of a tunnel from the Simpevarp Peninsula, where the Oskarshamn Nuclear Power Plant is located, to the southern part of Äspö. On Äspö the main tunnel descends in two spiral turns to a depth of 460 metres. The various experiments and demonstration tests are conducted in niches and short tunnels that branch out from the main tunnel. An illustration of the HRL is shown in Figure 5-2 and current experiments are presented in the Äspö HRL's yearly report (SKB 2019f).

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. These experiences will be of benefit for the coming detailed site investigations for the Spent Fuel Repository, the extension of SFR in Forsmark and the siting, design and construction of SFL.

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



Figure 5-1. The Äspö HRL is situated on the island of Äspö north of the Oskarshamn Nuclear Power Plant.

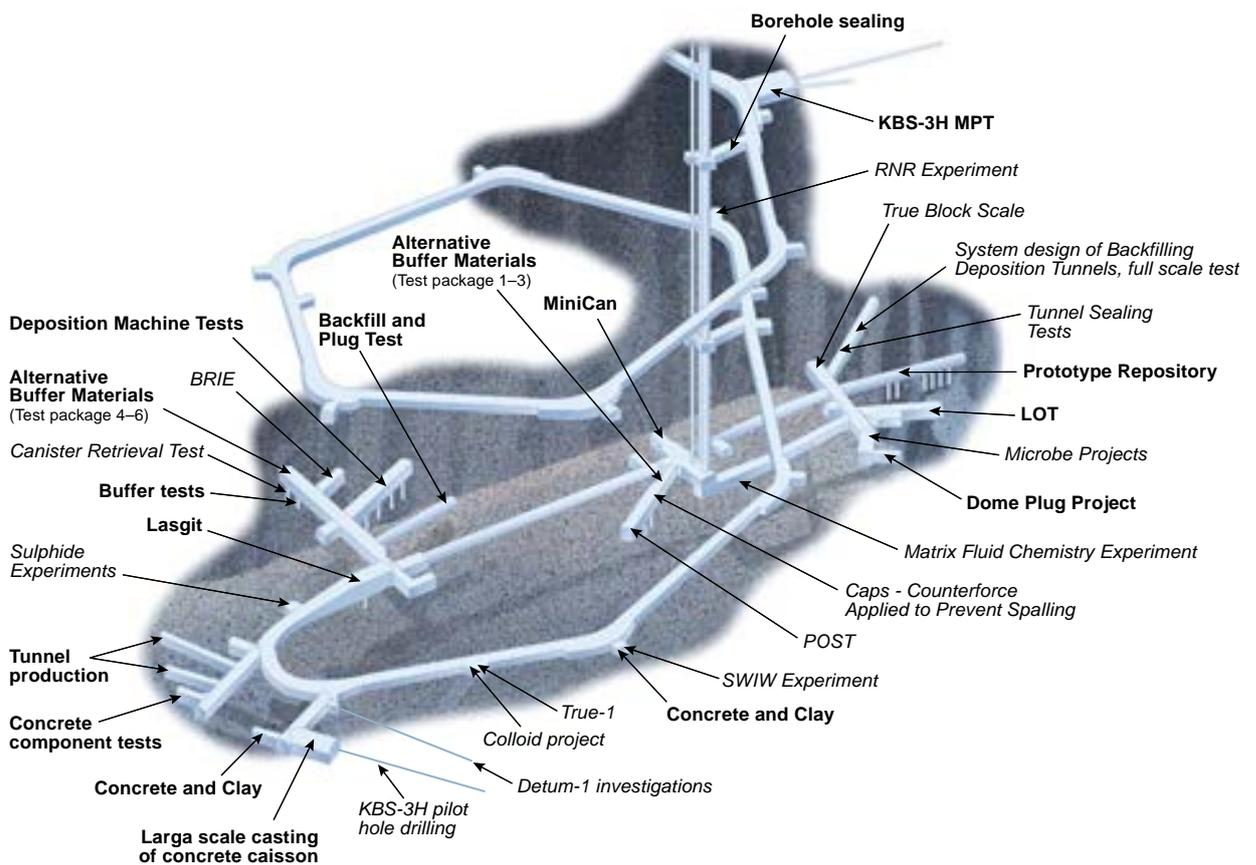


Figure 5-2. The Äspö HRL with ongoing (**bold**) and concluded (*italic*) experiments.

After the start of operation in 1995, experiments were initiated gradually to investigate how the barriers and the other components of the Spent Fuel Repository (canister, buffer, backfill and closure) should be designed and handled in order to provide optimal functionality. Another important purpose is to develop and demonstrate methods for building 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 in full scale. The results from several of these experiments comprised an important basis for SKB's application for the KBS-3 system. In the continued work with the development of the KBS-3 system, the Äspö HRL will play an important role, for example through ongoing long-term tests that will be concluded in the coming years and the full-scale tests that are planned to be carried out there.

Prior to the future construction of SFR and 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. The main focus has been on development studies linked to the concrete caissons in the future 2BMA, in which a caisson in quarter scale has now been built as a concluding production test. The initial studies were performed at different external concrete laboratories while the concluding large-scale test castings were carried out in the Äspö HRL. Prior to future assessments of the safety after closure for SFR and SFL, additional research projects are carried out with a focus on studies of interactions between different types of barrier materials relevant for these repositories and for different types of materials that are representative of low-level and intermediate-level waste.

Today and in the coming years, activities at the Äspö HRL are focused on the engineered barriers. The focus will be on technology development and testing of equipment and systems for use in the Spent Fuel Repository and for SFR and SFL. SKB's long-term needs of the Äspö HRL's underground part are presently being investigated and decisions regarding the scope and timeplan for continued activities will be made in 2020.

The Äspö HRL's role as a centre for research and development of technology for final disposal includes hosting some additional SKB laboratories and experimental facilities.

The water chemistry laboratory

The water chemistry laboratory on Äspö is accredited for analysing the chemical components in groundwater that are of particular significance for final repository performance after closure. During the site investigation phase, the laboratory was responsible for the handling of all analyses and result summaries for the site investigation projects in both Forsmark and Laxemar. The combined expertise in the laboratory is used for planning and construction of the corresponding laboratory in Forsmark.

The 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 to the final repository during the operational phase. The research laboratory comprises relevant infrastructure (resources, methodology, set of instruments and documentation) and the premises used for the activities. 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.

Multi-purpose test facilities

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

One of the barriers in all final repositories consists of bentonite. In the Spent Fuel Repository, copper canisters are surrounded by highly compacted bentonite. Bentonite also surrounds the silo in SFR and is planned as a barrier in SFL. Bentonite will also be used for backfilling the tunnels in the repositories. In the Multi-purpose test facilities, SKB is conducting studies of the properties of the bentonite by, for example, simulating water conditions in a controlled manner. Methods are also developed for backfilling the repository's tunnels and building plugs to seal the deposition tunnels.

The studies performed in the Multi-purpose test facilities are often preparatory tests on various scales and with various scope in preparation for full-scale tests at repository depth in the Äspö HRL. The Multi-purpose test facilities also have equipment and space for reception of bentonite deliveries and mixing of bentonite to the desired water content.

5.3.2 The Canister Laboratory

The Canister Laboratory, situated in the harbour area in Oskarshamn, was built during the period 1996–1998. Among other things, the technology for welding the bottom and sealing the lid on the canister is being tested and developed in the Canister Laboratory. Also the methods that SKB will use to inspect the canister’s components and welds are being developed and demonstrated here. Most experiments are performed in full scale. Development of the methods that will be used to manufacture the canister components is being led from the Canister Laboratory. Inspection and evaluation are largely carried out at the Canister Laboratory, while the manufacturing tests are conducted at external suppliers. The goal is to develop methods for manufacture and inspection that meet the specified quality requirements and have sufficiently high reliability to be used in future canister production and in Clink. Important equipment in the laboratory includes a system for friction welding with a rotating tool, equipment for non-destructive testing and a handling system for full-size canisters. The equipment for friction welding has been modified so that welding can take place in an “oxygen-free” environment and a new X-ray system has been installed for testing the friction welder. Figure 5-3 shows the equipment for friction welding.

5.4 Work tools

Execution of the research, development and design activities needed to decommission the nuclear power plants and dispose of the nuclear waste requires access to a set of work tools. SKB and the reactor owners have, as a part of the RD&D programme, developed or acquired a set of such tools. This section gives a brief overview of essential tools and the efforts made to maintain and further develop these tools.

5.4.1 Databases

Management of radioactive waste and spent nuclear fuel entails management of large quantities of data that is ideally collected and structured in databases. SKB and the reactor owners have databases containing information, SKB uses DARK and PlutoWeb for spent nuclear fuel and long-lived waste that is interim-stored in Clab, and Gadd for all low-level and intermediate-level waste disposed of in SFR. The database Gadd has been developed in stages and now includes nuclear waste from the nuclear facilities that deliver waste to SFR.

In research and in order to conduct safety assessments, a number of databases with basic scientific data are used, mainly taken from public sources with for example radionuclide data and thermodynamic data.

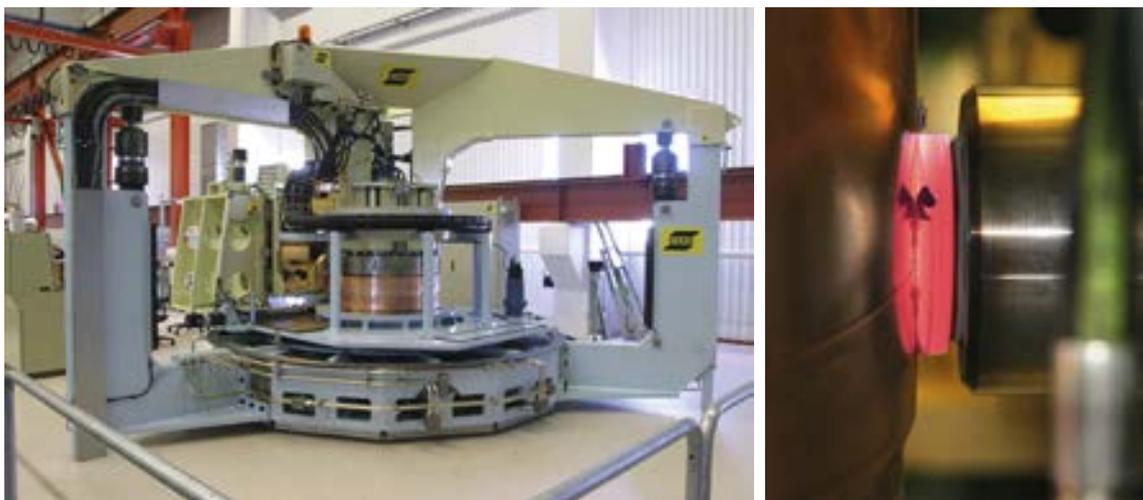


Figure 5-3. Friction welding of copper lids. The left image shows the Canister Laboratory’s equipment for development and the right image shows the rotating tool that is pressed into the joint between the parts that are to be joined.

The results of SKB's research and development work, data for the work with siting of repositories and data for site descriptions, design and safety assessments are managed in a number of different databases, for example Sicada for investigation data from, for example, site investigations and from the analyses performed at SKB's different laboratories and a GIS database for the management of geographic information and modelling.

SKB uses a systematic approach for the management of requirements on final repositories for spent nuclear fuel and radioactive waste. This includes requirements that should be fulfilled during the design, construction, operation and decommissioning of the facilities. Requirements on different levels of detail, from radiation safety principles and overall facility functions to the design of individual components, and the relationships between them, are documented and followed up in the database Doors NG.

There are also databases of a more administrative character such as Bibas, which is SKB's library database, and SKBdoc, which is the document management system.

The databases are updated continuously with information regarding additional quantities of radioactive waste and spent nuclear fuel, new data from research and investigations and new publications and documents. There is also a constant maintenance of these systems and the software is developed in step with the development of computers and operative systems.

5.4.2 Model and calculation tools

Models and calculations are a central part of the work with design, evaluation and assessments of safety during operation and after closure. 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 has access to a number of model and calculation tools. There are both model and calculation tools developed at SKB and commercial tools that have been purchased and adapted when necessary for SKB's application.

For carrying out the assessments of post-closure safety for the Spent Fuel Repository (SR-Site) and for the extended SFR facility (SR-PSU), a large number of models and calculation tools (SKB 2010c, 2014c) were used. The calculation tools that were used include both commercial software with hundreds of thousands of users, and software specially developed for safety assessments with perhaps only a few tens of active users and developers. SKB has quality requirements that the calculation tools must meet in order to be approved for use as a part of the safety assessments. According to these requirements (Chapter 2 in SKB 2010c), for the results to be approved for use in safety assessments, there must be documentation proving that:

- The software is suitable for its task in the safety assessment.
- The software has been developed in an appropriate manner and the calculations yield correct results.
- The software has been used correctly 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 or commercially available databases (see previous section).

Model and calculation tools are maintained and upgraded continuously alongside the general development of computers and operative systems. SKB is also carrying out development to optimise some of the commercial tools for the safety assessments and continuous development of own calculation tools (for example DarcyTools) is also carried out. SKB strives to have competence regarding all software used in safety assessments and to as far as possible use parallel sets of software and modelling teams to thereby create independent verification of results.

Development of a model database is under way to ensure a systematic and traceable use of version-controlled models, both with respect to site models and models used in the safety assessments.

5.4.3 Site investigations and site models

In order to gather data that are needed to build and evaluate the safety of a final repository, SKB has, in many cases, together with sister organisations and cooperation partners developed special investigation methods and instruments. Thus, SKB manages a set of measurement instruments for execution of site

investigations and is further developing such methods and instruments as needed. These instruments are kept in an instrument storehouse located adjacent to the Canister Laboratory in Oskarshamn. From the storehouse, 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 until today. A site descriptive model is one of the cornerstones for repository design, environmental impact assessments and for the assessment of post-closure safety for a repository. SKB has developed site descriptive models for Forsmark (SKB 2008, 2013c) and Laxemar (SKB 2009b).

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 presently only deemed justified to maintain the models for Forsmark. Updates of the model for Laxemar might be relevant as a part of the work with the siting of SFL, see Chapter 6. In Forsmark, a monitoring programme where data are collected on groundwater pressure, groundwater composition, seismic events, precipitation, temperature, development of ecosystems etc has been in progress since the site investigations were concluded. The monitoring will continue and hence new data are gradually supplied, which can be used to update the site descriptive models for Forsmark. The models are used as a basis for design and will be updated in conjunction with the gradual extension of the repositories. Larger updates will be carried out if there is substantial new information prior to the updated SAR in accordance with the timeplan, as discussed in Chapter 3.

5.4.4 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 procedures include procedures for procurement, approval of the fact that suppliers have the correct skills and can live up to SKB's requirements, approval of content in databases, approval of model and calculation tools and approval of input to models and calculation results.

Results from SKB's research and development are presented, as a rule, in SKB's report series or in scientific journals. The reports and other documents of importance for safety, which are produced by SKB, undergo a documented review process. The review is to ensure that the documents meet specified requirements regarding scope and content and that the information that has been submitted is methodically correct and based on approved sources and calculation tools.

5.5 Resources and expertise

5.5.1 Competence and resource needs

SKB's activities are broad and multifaceted, which means that SKB needs competence in several different areas. A large part of the knowledge and technology required by SKB is publicly available. Other parts are specifically linked to management and final disposal of nuclear waste.

Parts of the necessary knowledge for SKB are included in the general scientific knowledge base and are developed within the scientific community. SKB needs to have sufficient competence, however, to maintain its ability to assimilate the knowledge from the scientific community that is significant for the management and final disposal of nuclear waste, and to be a competent purchaser of research. By conducting its own research, SKB ensures this maintenance of competence. Large parts of the detailed knowledge regarding for example the function of the repository barriers in a geological environment are, however, so specific for nuclear waste that the knowledge has been produced or will need to be produced under SKB's own auspices or in international collaboration. This applies also to some knowledge of the geological environment in itself, of the biosphere where the consequences of potential releases from the repositories arise, and of large-scale environmental changes, primarily climate-related, which may affect the repositories in the future. SKB therefore needs to have a coherent group of persons with knowledge of the methodology for the assessment of post-closure safety with broad and interdisciplinary insight into how the different processes that affect repository

safety interact. The group must also include persons with in-depth knowledge of the disciplines that affect safety, i.e. geoscience (e.g. geology, hydrogeology, geochemistry), materials science (canister materials, clay materials, cementitious materials), waste and spent nuclear fuel (criticality, radiation protection, chemistry, solubility etc), solute transport (engineered barriers, rock), surface ecosystems and climate evolution. Furthermore, competence is needed to combine and integrate knowledge from all these disciplines in order to be able to carry out assessments of safety during operation and after closure, i.e. competence concerning safety assessment methodology.

The need for competence to manage and carry out technology development is based on the plans that are prepared (Section 5.2.2). In order to be able to evaluate the development needs and control development, SKB needs general competence in the production lines linked to the properties of the nuclear waste and the waste system barriers and other components such as canister, buffer, cementitious materials, backfill, closure and rock. The development can in many cases be carried out by different research institutes or consulting companies but in certain areas, where there are few other purchasers of the competence SKB requires, SKB needs to have competence for development within the company. This applies in particular 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 competence management is to ensure that the competence exists and is being developed, in the short and long term, within such areas.

With the imminent construction of new facilities (Clink, canister production system, the Spent Fuel Repository and SFL) and the extension of SFR, competence with respect to execution of large construction projects (e.g. project management, construction management, construction, excavation, rock investigations, installation) is also required. Dismantling and demolition of the nuclear power reactors will require similar competence mainly with respect to project management and construction, but also specific competence regarding for example radiation protection and classification of waste.

In addition to competence within the different disciplines, competence is also needed on different levels within the disciplines. There is a need for generalists with system understanding who can act as project purchaser and/or project managers, manager and supervisor competence as well as expert and research competence. At a later stage, competent and sometimes specialised contractors (for example installers and operators) that can translate knowledge into practical action will be required. Moreover, there is a need for personnel with good facility knowledge to construct, operate and maintain the facilities that are planned and those in operation. When assessing the need for competence, it is important to also consider the number of persons with a certain competence needed to carry out the necessary tasks. This applies in particular during more work-intensive periods such as the preparation of the basis for application documents (for example safety assessments) and the construction of new facilities.

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 5.3).

5.5.2 Building and maintaining competence

As an activity operator, SKB is obliged to ensure that tasks are performed by persons with the necessary competence. Whether the competence requirement is to be met by in-house personnel or by external suppliers or consultants is to some extent a strategic issue. The decision made is based on both an assessment of which tasks are of such strategic or safety importance that they should be performed by SKB's own personnel, the risks of being dependent on external suppliers and financial considerations. The outcome of decisions on competency requirements and trade-offs between having in-house personnel and using external suppliers can vary over time.

SKB's point of view is that SKB should have its own personnel with competence in order to be able to manage and lead the work with research, development and operation of the system for management of radioactive waste and spent nuclear fuel. This also entails that SKB should have the necessary competence to procure and evaluate the services and products linked to management and final disposal of spent nuclear fuel and radioactive waste that SKB orders from external suppliers. SKB's strategy is that its own resources should have the necessary competence and carry out the work tasks of strategic importance for radiation safety instead of engaging consultants. Examples are the safety assessment

methodology and development of canisters for spent nuclear fuel, where SKB has chosen to largely rely on own resources to build and maintain competence in the company in the long term. If other products or services are available commercially or through SKB's owners, the general strategy is to use these.

SKB has established a competence management process in its management system. The process is based on a systematic approach for complying with internal and external requirements to ensure that adequate competence is available for maintaining high safety and achieving the goal of the activities in the short and long term.

In conjunction with the annual planning of activities, a competence and staffing analysis is carried out. The competence analysis shows the competence needed in a position or role in order to perform tasks according to the requirements and needs of the activities. These tasks are identified in order to be able to ensure correct and efficient management of competence assurance and planning for replacements with the final goal of maintaining high safety in the activities. Roles of specific strategic importance or of importance for radiation safety are identified. For changes in the organisation and activities, there is a structured and traceable approach to ensure staffing and competence so that the requirements are met in the future.

The analysis is made on both individual and group level and with a timeframe of four to five years. Given the time it takes to develop competence, the future perspective is always essential. Strategic competence analyses with a timeframe of about ten years are conducted regularly but with slightly longer intervals than the annual planning of activities. The purpose of the recently completed strategic analyses for future construction projects is to identify future staffing needs (competence and number of personnel) and how competence is to be secured for organisation and staffing during the different project phases, mainly in the establishment and construction phases and to some extent in the operational phase.

The analysis shows the competence needed to execute the activities and the need for competence development either by further training of existing personnel or by new recruitment. Training programmes are established for individuals and groups when necessary. This could for example concern positions that require special access, such as operating personnel. The general training programme includes introductory training for new employees. SKB has a competence management system in which competence assurance (documentation of competence and any gaps between requirements and assessed level) is performed for own personnel and consultants.

The personnel's competence is developed for example through mentor programmes and rotation programmes where employees are given an opportunity to work within different areas and in different roles. SKB also has a competence transfer programme to prepare for generation changes and to reduce the vulnerability to loss of competence (for example due to personnel turnover). Competence assurance also includes preparing for future changes in activities and external requirements.

In order to develop and keep SKB's experts, they are given the opportunity to participate in scientific conferences, own research and publications of their results in scientific journals. Some have for example been industrial PhD students and have obtained their PhD degree within the framework of their work at SKB. Several of SKB's experts are associate professors and some have completed sabbaticals at universities or university colleges.

5.5.3 Competence network and suppliers

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 experts are engaged for this, often from research institutes, universities and university colleges. These experts are engaged to a greater or lesser extent depending on the need at the time. Many have been associated with SKB's activities to varying degrees for decades. For temporary needs of resources, for example for large projects concerning design and construction of facilities, external suppliers are generally engaged. SKB's owners are also an important resource.

5.5.4 Collaboration

Collaboration with universities and university colleges

SKB collaborates with universities and university colleges in order to obtain critical knowledge in areas where it is lacking. A collaboration that SKB considers to be very valuable. This is typically research where SKB funds PhD projects while the university supervises the PhD student. SKB-funded PhD students constitute a prospective future competence reserve for SKB and others in the industry.

At present, SKB has about 15 SKB-funded or partly funded PhD projects. Through the years, SKB has funded or partly funded far more than 100 PhD students. Many of these have, during the research programme and after their PhD degree, had key roles in progressing SKB's work.

Collaboration with Posiva

For a number of years, SKB has had an 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 the KBS-3 method. In 2001, the Parliament of Finland ratified the Finnish Government's decision in principle regarding the method and site for the Finnish final repository. The facility is planned to be built at Olkiluoto in Eurajoki. In 2004, Posiva began building a hard rock facility (Onkalo) in Olkiluoto and reached the planned repository depth in 2010. Onkalo is used today for research and development, but will also constitute the accesses to the actual final repository.

At the end of 2012, Posiva submitted an application for a licence to build an encapsulation plant and a final repository for spent nuclear fuel according to the KBS-3 method. In February 2015, the Finnish Radiation and Nuclear Safety Authority (STUK) announced their statement on this application and recommended that the Finnish Government should grant a licence. In its review statement, STUK identified a number of issues that Posiva must resolve and report before it is possible to grant an operational permit. In November 2015, the Finnish Government granted a licence for construction of the final repository and the encapsulation plant. Posiva has now started constructing the facilities and intends to begin trial operation, if permits for this are granted, around 2023/2024.

In 2013, SKB and Posiva began planning an in-depth collaboration with the goal of developing common technical solutions for the final repository system prior to commissioning. Agreements that were signed then were updated at the end of 2018. The collaboration includes canister, bentonite and rock issues, machine design and issues linked to finding financially optimal solutions without compromising safety.

In addition to the benefits in terms of efficiency that this collaboration brings, SKB's ability to carry out research and development is also enhanced. The formulation of common plans and joint projects entails that these will be prepared and reviewed more thoroughly and comprehensively than if SKB carried out this work on its own. Through the collaboration, SKB gets access to Posiva's facilities, especially Onkalo, and also access to the research institutions, institutes and other experts Posiva is collaborating with.

Other international collaboration

A large part of SKB's international collaborations take place within the framework of the EU, the OECD/NEA and the IAEA. SKB is also participating in a number of bilateral and multilateral collaboration projects, often with other nuclear waste organisations.

The EU's work in the field of nuclear energy is regulated by the Euratom treaty. Among other things, the European Commission strives to harmonise the 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.

The IGD-TP is a technology platform that was formed in 2009 at the initiative of SKB. 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, Nagra (Switzerland), RWM (UK), Puram (Hungary), Covra (Netherlands) and Surao (the Czech Republic). Around 130 universities, university colleges, research institutes and consultancy

firms are involved as participants and support the platform's vision that a final repository for high-level waste and/or spent nuclear fuel will be commissioned by 2025. The platform has been important for the focus of the EU's research programme.

During the past RD&D period, SKB has participated in the following EU projects, which now are nearly finished or have been concluded:

Modern2020	Further technological development on monitoring	www.modern2020.eu
Cebama	CEment BAsed Materials: properties, evolution and barrier functions	www.cebama.eu
Mind	Microbiology In Nuclear waste Disposal	www.mind15.eu
Beacon	Bentonite mechanical evolution	www.beacon-h2020.eu
Disco	Modern spent fuel dissolution and chemistry in failed container	www.disco-h2020.eu
Chance	Characterisation of nuclear compounds	www.chance-h2020.eu

The way the EU funds research and development in the nuclear waste area will change. Instead of supporting individual projects, support will be given to a consolidated programme, a Joint Programme, called 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 programme will start in 2019 and runs for five years. It consists of, in its first phase, seven different research projects, two strategic projects and efforts concerning knowledge transfer. SKB will lead a research project on spent nuclear fuel. This has connections to another international collaboration between Euratom, U.S. DOE and SKB, which is called Non Destructive Assay Spent Fuel project.

The NEA, which is a collaborative organisation for nuclear energy issues in the OECD, has organised the Radioactive Waste Management Committee (RWMC), which consists of representatives from authorities and waste organisations. SKB is participating in the following committees and groups within RWMC:

- Forum on Stakeholder Confidence (FCS).
- Integration Group for the Safety Case (IGSC).

In 2011, RWMC initiated the project Preservation of Records, Knowledge and Memory Across Generations (RK&M). The goal of the project was to present a form of "tool box" and an international consensus on proposals for which tools to use to best preserve documents, information and knowledge on final repositories for radioactive waste in different time perspectives. SKB has participated in the project since the beginning until its conclusion in 2018 (Section 4.12.1). SKB also participated in the project Radioactive Waste Repository Metadata Management (RepMet), which was carried out between 2014 and 2017. This project investigated which metadata can support the availability and understandability of the data and information that is gathered for use in the safety assessments.

An international project led by the NEA and Studsvik Nuclear AB is under way, the NEA Studsvik Cladding Integrity Project (Scip), concerning fuel and the interaction between pellet and cladding, which is relevant throughout the fuel's life cycle since it may affect the integrity of the cladding. The project has gone through three phases, of which SKB has participated in the third and is planning to participate also in the coming fourth phase.

SKB is also participating in the work group Working Party on Management of Materials from Decommissioning and Dismantling (WPDD) that in 2019 was transferred from RWMC to the by the NEA newly formed Committee on Decommissioning of Nuclear Installations and Legacy Management (CDLM) for collaboration on decommissioning issues. Furthermore, SKB is participating in a NEA project within the framework of the NEA Data bank (Thermochemical Database Project), where the purpose is to create a quality-assured chemical thermodynamic database that contains data for substances that are of importance for the management of nuclear waste.

The International Atomic Energy Agency, IAEA, is also organising a number of projects and groups where SKB is participating, such as:

- Geosaf II – The project concerns issues about integration of operational and long-term safety for geological final repository facilities.
- Modaria II – The work is aimed at refining models for assessment of radiation doses to humans and the environment.

- Hidra II – Different aspects of intrusion into the repository are being studied within the project.
- Astor – an expert group aimed at developing recommendations for safeguards in geological final repositories.
- Use of Monitoring Programmes in the Safe Development of Geological Disposal Facilities for Radioactive Waste – the working group has the purpose of explaining how a monitoring programme should be developed and implemented in different phases of a final repository programme and how information from monitoring can be integrated with other information in the decision process for final repositories.

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 SSM, in accordance with the convention, produces every three years. The report is reviewed by convention parties in special review meetings.

The Nordic network Husc (Human Performance and Safety Culture) is a network of Nordic companies and organisations that act in the nuclear area. SKB also contributes, together with SKB's owner companies, to the Nordic research collaboration Nordic Nuclear Safety Research (NKS).

SKB also has, through Vattenfall, access to international competence and resources through research organisations such as Eniss (European Nuclear Installations Safety Standards), EPRI (Electric Power Research Institute) and Nugenia. SKB is also actively participating in Nugenia's work.

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

SKB is also collaborating with other waste organisations all over 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 with several organisations. For example, the joint international work that SKB coordinates at the Äspö HRL and the Greenland Analogue Project (GAP), which included studies of how the next ice sheet may affect a final repository for spent nuclear fuel.

In special forums, specialists and modelling groups from several countries are collaborating on selected issues of importance for final disposal of nuclear waste. Two forums are established within 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). The collaboration aims to evaluate different concepts and modelling methods and facilitate collaboration between experimentalists and modellers.

5.5.5 Current challenges and maintenance of competence in the long term

The question of competence management in particular in the longer term, has been raised by SSM and the Swedish National Council for Nuclear Waste. SSM received an assignment in 2016 from the Government to study the prerequisites for maintaining competence in SSM's area of responsibility, which was reported in September 2018 (Hällgren 2018). The Government decided (Government Decision I:7 on the RD&D Programme 2016) at the recommendation of the Swedish National Council for Nuclear Waste that the RD&D Programme 2019 should contain a description of how competence development and competence management is to be secured in a time perspective of 50–100 years. The focus in this section is SKB's activities, while issues related to decommissioning of reactors are dealt with in Section 15.4.

General and fundamental prerequisites

As described above, SKB requires competence within many areas, especially persons educated within natural science and technology. The areas within which SKB will need competence can be expected to remain relatively unchanged over time, but the number of people that are needed in each area will change as the activities change from the current emphasis on research and technology development to construction, operation and decommissioning of facilities.

Generally, the national trends with a declining interest for technical and scientific education are 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 educational programmes in relatively narrow areas such as nuclear technology and maintain important research environments, which, among other things, is illustrated in SSM's study. It is therefore important to increase the interest in natural sciences and technology so that a sufficient number of educated persons can graduate. Maintaining interest in natural sciences and technology is a common issue for many actors where SKB, together with its owners, can contribute. SKB works together with the owners to increase the attractiveness of the industry. SKB intends to continue increasing the confidence in SKB's mission and highlighting the many interesting and challenging tasks and roles needed to accomplish the mission. SKB intends to collaborate with companies and schools, primarily at the locations where SKB is active, and participate in labour market days, fairs and various industry events. A successful example of how new recruitment to the industry can be carried out is Vattenfall's trainee programme, which started in 2018. These initiatives contribute separately and together to SKB's long-term competence management. 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 for e.g. 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 in the country, collaboration with universities and university colleges will be a part of the long-term competence management, as will the continued encouragement of collaboration within the industry in order to both keep existing personnel and attract new personnel in these areas. For example, Vattenfall is funding an engineering programme focused on nuclear power at Uppsala University, as well as Vattenfallgymnasiet in Forsmark, in order to secure access to qualified competence far into the 2040s. Furthermore, Forsmarks Kraftgrupp AB, Ringhals AB, OKG Aktiebolag and Westinghouse Electric Sweden AB are funding the Swedish Centre for Nuclear Technology (SKC). SKC supports education, research and development within nuclear applications at universities and university colleges in Sweden.

In general and in a long time perspective, SKB expects most of the need for personnel to be satisfied by personnel with basic education, who are then further trained by SKB for the company-specific applications. In addition, there is a need for a small number of persons with in-depth competence, for example postgraduates, combined with long experience of areas important for SKB.

In Section 5.5.2, SKB described how SKB organises competence management in current activities. The procedures and the approach established and the strategic competence analyses carried out comprise a good basis for handling competence development and competence management in the long term, although there is always room for improvement. SKB is for example participating 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 with systematic competence analyses and common training activities.

Current challenges

A current challenge for SKB is the new facilities that are to be constructed and then operated for a long time. There are also other challenges regarding competence management in the long term, of which some are discussed below.

When it comes to the construction of the new facilities, SKB plans to build up a purchaser organisation to procure, control and follow up the contracts that the construction of each facility will be divided into. The purchaser organisation will have the SKB-unique competence needed for the construction of the facilities. It will mainly consist of own personnel but consultants may also be eligible. Since the construction of the facilities mainly includes using well established technology, SKB is planning to engage established suppliers (contractors) with the necessary competence. Special requirements apply for the construction of nuclear facilities, which suppliers must comply with during the course of the work. SKB has previous competence and experience, from the construction of Clab 2 and the Äspö HRL and the site investigations in Oskarshamn and Forsmark, of controlling suppliers and ensuring

that they live up to SKB's in many cases specific requirements. An important component in the control is the training programmes, which SKB is presently developing for everyone who will work with the construction of the facilities. The challenges SKB sees in building up the organisation for the construction projects are primarily related to competition for personnel from other large infrastructure projects during the coming decades. A challenge may also be the availability of the senior competence needed to organise and lead the large construction projects, especially considering the generation change that occurs as many of those that have built up the Swedish nuclear waste programme will soon retire or have already retired.

Competence management and challenges in the long term

Regarding competence management in the very long term, i.e. a 50–100 year perspective, there are two important prerequisites that must be considered:

- SKB's activities are long-term and are planned to continue for about another 70 years, i.e. up until around 2090, and there are plans and financing for the focus and scope of the activities during this time.
- SKB is a dominant actor in Sweden when it comes to the management of radioactive waste, but tasks of substantial extent will also be carried out by the owner companies, suppliers and regulatory authorities.

The first point is an advantage as competence development and competence management can be planned in the long term in a way that few other companies can. It should facilitate the process considering that the building of competence takes time. SKB has well developed procedures for competence development and competence management, which have been presented in Section 5.5.2, 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.

The research and technology development conducted by SKB provide an important contribution to the development of competence and competence management with respect to the management of radioactive waste and spent nuclear fuel. The collaboration with universities and university colleges is strategically important for SKB, in part to obtain new knowledge within areas important for SKB, in part to maintain and develop competence. SKB strives for long-term collaboration with universities and university colleges and tries to, if appropriate and if the conditions are right, act so that support for PhD projects can contribute to building and preserving good research environments. As mentioned earlier, SKB currently supports about 15 PhD projects. In addition to developing knowledge in issues of importance for final disposal of waste, the PhD students funded by SKB become a competence reserve for SKB and for other actors in the area. As is evident from Sections 5.1 and 5.2, research and technology development is primarily governed by SKB's needs but the programme as a whole provides significant contributions to competence management, especially expert competence, even in the long term.

One aspect which especially needs to be considered during the entire life of the facilities is facility knowledge, and for final repositories also knowledge of the surrounding rock. Special efforts 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.

SKB's role as the main employer when it comes to the management of radioactive waste gives SKB a special responsibility for its competence management. Being the main employer within an area makes a company more sensitive to the general trend in society where mobility in the labour market is increasing. If someone quits, it may be difficult to find replacements with the appropriate skills. The presumed lack of external competence leads to a need for relatively large efforts for internal training. SKB is, however, not the only employer within the area, and the corresponding competence is also needed by consultant companies and regulatory authorities, which can reduce potential problems with recruitment of replacements, while simultaneously making the industry as a whole more attractive.

SKB is working actively to develop the company as an attractive employer (employer branding). Partly through external activities such as participation in labour market days and recruitment campaigns to increase the interest in working at SKB. Partly internally, by ensuring that work tasks are stimulating and developing, that there are good opportunities for further training and specialisation, and that there are career opportunities within the company.

A risk identified by SKB is if any part of the activities were to be suspended and the problems that might then occur with maintaining competence within different areas.

Suspension of activities can be planned or unplanned. Planned suspensions of activities could for example be the site investigations that have already been made for the Spent Fuel Repository and the extension of SFR, but will be carried out for SFL within a number of years. Unplanned suspensions could for example be a result of not obtaining licences from the authorities.

A suspension within a part of SKB's activities normally leads to a need for relocating personnel. If the suspension is long, it may be difficult to maintain competence and retain personnel. A prerequisite for preserving knowledge during longer suspensions is thorough and systematic documentation that describes the work tasks that have been performed, the equipment used, the results achieved and the competence needed. An example is the method descriptions and other documentation that were prepared in conjunction with the site investigations. When the activities are resumed, time must be allocated for studying the documentation and training any new personnel who are to carry out the work. The possibilities of relocating personnel are naturally dependent on the competence area. SKB sees relatively good prospects for retaining staff and competence of SKB's experts who work with safety assessments. Safety assessments are to be performed periodically as a part of the stepwise licensing process for three final repositories. Those working with research and technology development can also to an increased extent work within international projects (Section 5.5.4) or be lent out to waste organisations in other countries via SKB International. To work in projects for other waste organisations provides variation and is in general considered stimulating and contributes to maintaining and developing competent personnel within the company. For other personnel categories such as operating personnel, it can be difficult to find other suitable tasks within SKB. There might, however, be opportunities to lend out personnel to SKB's owners where similar work tasks occur.

There are also risks with narrow fields of expertise where there might be only one single expert in Sweden or sometimes in the world. In such cases, the international collaboration SKB has with other waste organisations is essential (Section 5.5.4). 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 5.5.4 is, among other things, to maintain competence and support each organisation within narrow fields of expertise for the issue of nuclear waste. It can also be noted that questions relating to the maintenance of competence are thoroughly discussed internationally within for example 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.

SKB has described above the principles, procedures and approaches developed by SKB for competence development and competence management. SKB has developed strategic competence management plans and analysed the risks and problems that may arise when it comes to competence management in the long term. SKB considers potential problems to be manageable and sees it as possible to, by applying and further developing the procedures for competence development and competence management, accomplish SKB's mission of disposing the radioactive waste from the nuclear power plants with retained competence.

Part II

Waste and final disposal

- 6 Final Repository for Long-lived Waste
- 7 Low- and intermediate-level waste
- 8 Spent nuclear fuel
- 9 Canister for spent nuclear fuel
- 10 Cementitious materials
- 11 Clay barriers and closure
- 12 Bedrock
- 13 Surface ecosystems
- 14 Climate and climate-related processes

Part II – reading instructions

Part II of the RD&D Programme 2019 describes planned research and technology development during the RD&D period. The focus is on SKB's identified priority issues for the continued management and final disposal of radioactive waste and spent nuclear fuel. The description of planned activities is presented for low- and intermediate-level waste, spent nuclear fuel and the different parts of the repository system, which means that research and technology development is described in an integrated way for the three repositories. The current level of knowledge is described in general terms and references are made to more detailed presentations of results in background reports.

Part II also describes planning and development activities for the final repository for long-lived waste, SFL, and the points of departure for the siting process for SFL.

6 Final Repository for Long-lived Waste

SKB has developed a repository concept for the Final Repository for Long-lived Waste (SFL). The repository concept (Section 2.1.2) includes two repository sections, one for reactor internals and PWR reactor pressure vessels from the nuclear power plants, the waste vault for reactor internals (BHK), and one for legacy waste, the waste vault for legacy waste (BHA). The reactor internals, which consist of metallic waste, comprise about one-third of the volume but contain (initially) the main part of the radioactivity. SKB plans to design the repository section for reactor internals with an engineered barrier of concrete.

The legacy waste is stored and managed by AB SVAFO and Studsvik Nuclear AB in Studsvik. Waste from Cyclife Sweden AB and other Swedish research, industry and medical care is also included. SKB proposes that this section of the repository is designed with a bentonite barrier. The repository concept is illustrated in Figure 2-5.

The safety principle for the proposed repository concept is *retardation*. The concept is thus based on a system of engineered and natural barriers that reduce the mobility and retard the transport of radionuclides from the repository. The effect of the retardation is that radionuclides are retained and decay, primarily in the repository but also in the surrounding bedrock. The proposed repository concept furthermore includes locating SFL at a depth sufficient to avoid the adverse effects of permafrost on the engineered barriers.

This chapter describes the evaluation of post-closure safety for the proposed repository concept (Section 6.1), which has been carried out during the RD&D period. Furthermore, the siting process for SFL is described (Section 6.2).

6.1 Safety Evaluation for SFL

During the RD&D period, an evaluation of the post-closure safety has been carried out for the proposed repository concept. The purpose of the evaluation is to give SKB a basis for determining if the studied repository concept has the potential to meet SSM's requirements and to provide information for the continued development of SFL with regard to siting and barrier design. Prerequisites, methodology, results and conclusions of the safety evaluation are summarised in this section. The complete documentation can be found in a main report (SKB 2019c), and a set of main references and background reports.

6.1.1 Purpose and level of the evaluation

The purpose of the safety evaluation is to evaluate conditions under which the proposed repository concept has the potential to meet regulatory requirements on post-closure safety. The evaluation serves as a basis for the site selection process and development of engineered barriers as well as for identifying requirements on the waste (WAC, waste acceptance criteria). The evaluation also identifies areas for further efforts for future assessments of the post-closure safety.

The safety evaluation is based on SKB's safety assessment methodology. The methodology has been adapted to provide a basis for the continued development of SFL rather than demonstrating regulatory compliance. There are many commonalities between the proposed repository concept for SFL and the other two repositories, SFR and the Spent Fuel Repository. Waste and waste containers in SFL have similarities with corresponding components in the extended SFR, and the same applies for the engineered barriers. Since SFL is a final repository for long-lived radioactive waste, the regulations require that the assessment of post-closure safety of the repository covers a million years, i.e. the same time period as for the Spent Fuel Repository. The proposed repository concept furthermore requires that SFL, like the Spent Fuel Repository, is placed at a depth sufficient to avoid adverse effects of permafrost on the engineered barriers. On account of these similarities, the evaluation is, wherever possible, based on existing knowledge and data obtained during previous site investigations, safety assessments and technology development at SKB.

6.1.2 Prerequisites for the evaluation

Example site

As the site for SFL has not yet been selected, available data from site investigations for the Spent Fuel Repository and the extended SFR have been used for the safety evaluation. A realistic, detailed and coherent dataset from SKB's site investigation for the Spent Fuel Repository in Laxemar in Oskarshamn municipality has been used. Since Laxemar is considered to be a relevant example of a site for a geological repository in Sweden, these data have constituted the point of departure for the safety evaluation. Based on a first assessment of groundwater flows (Abarca et al. 2016a), a specific position within the site investigation area has been chosen for the evaluation (Figure 6-1). The bedrock in the chosen position has a low frequency of fractures and thereby low permeability compared with other positions within the rock volume that was earlier found suitable for a Spent Fuel Repository in Laxemar. Furthermore, the repository is assumed to be built at a depth of 500 metres, which is deemed to be sufficient to avoid adverse effects of permafrost and to impede future inadvertent intrusion. In addition, data from Forsmark in Östhammar municipality have been used to study sensitivity to site-specific properties.

Radionuclide inventory

An important prerequisite for the evaluation is the estimated inventory of radionuclides and other materials in each waste vault. The inventory has been taken from estimates made by SKB for long-lived intermediate-level waste from the nuclear power plants and long-lived waste from AB Studsvik, Cyclife Sweden AB and AB SVAFO.

The BHK waste vault is designed to hold the long-lived waste from the nuclear power plants. This waste is better characterised than the legacy waste since induced activity can be estimated with relatively good accuracy and the materials of the constituent components in the waste are generally known. The greatest uncertainty is probably associated with the material composition of the irradiated material, especially tracers that are not normally analysed when determining the material composition. In these cases, pessimistic assumptions are made (e.g. the chlorine content in steel).

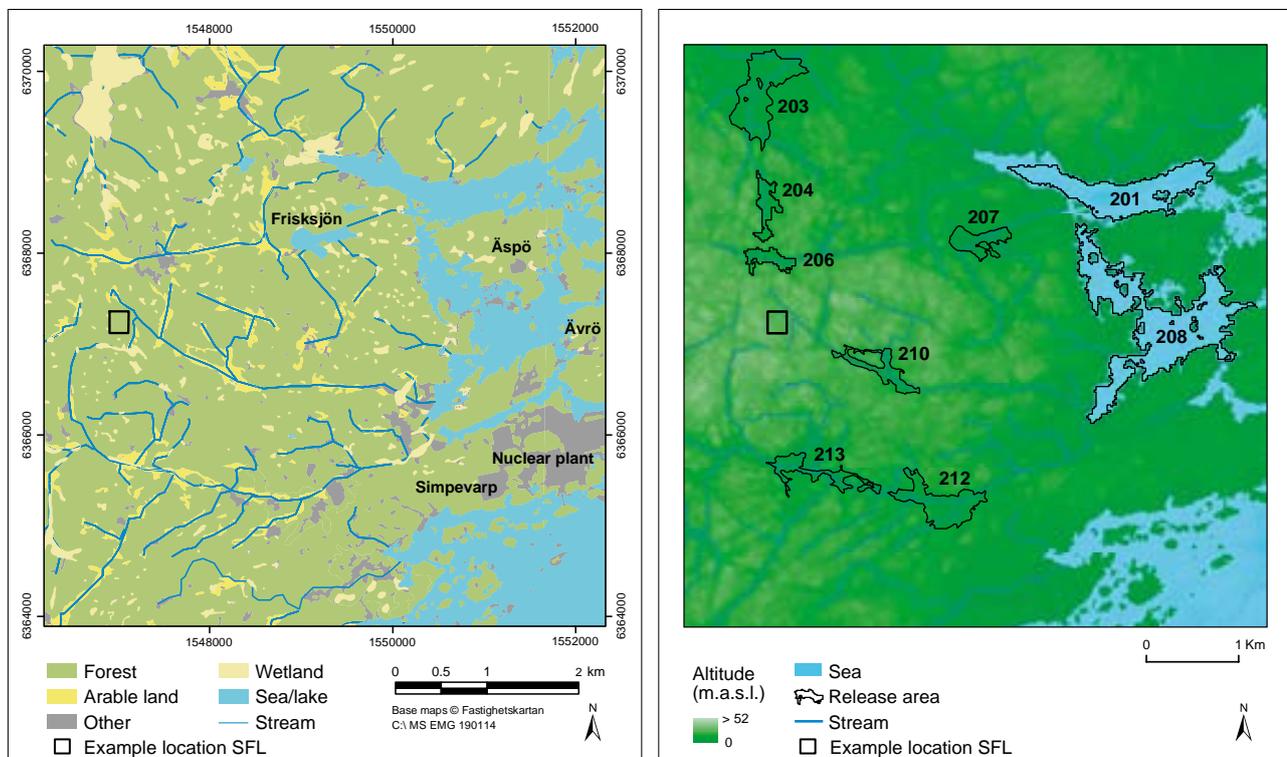


Figure 6-1. Illustration of the use of land in the area around Laxemar today (left panel) and the ground-surface elevation above the current shoreline (right panel). The right panel also shows the discharge areas that have been assessed in the safety evaluation.

BHA is designed to hold waste from Studsvik Nuclear AB, Cyclife Sweden AB and AB SVAFO. The radionuclide inventory of the legacy waste is associated with great uncertainties. For a few waste fractions, there are relatively reliable data for actinides or difficult-to-measure nuclides such as carbon-14. Only gamma spectroscopic measurements are, however available for the majority of the waste. The inventory of difficult-to-measure nuclides is based on a nuclide vector presented by AB SVAFO. Since this nuclide vector was developed with respect to clearance, it generally contains very pessimistic values for difficult-to-measure nuclides, which do usually not constitute limits for clearance. When this nuclide vector is applied to the legacy waste, it leads to an overestimation of some difficult-to-measure nuclides. The material inventory of the legacy waste is also highly uncertain. The material estimate is primarily based on the material composition of a few investigated waste packages and the results have then been extrapolated to all the waste within the same waste class (e.g. trash and scrap metal).

Radioactive waste from the operation of the European Spallation Source (ESS) is planned to be deposited in SFL. Since planning of the construction and operation of ESS is in progress, there is only limited information on the amount and composition of this waste. The waste from ESS was therefore handled with a simplified procedure in the safety evaluation.

At the closure of SFL, the estimated radiotoxicity⁹ is approximately three times higher for the waste from the nuclear power plants than for the legacy waste (Figure 6-2). The inventory of waste from the nuclear power plants is dominated by the relatively short-lived radionuclides nickel-63 and cobalt-60. Already a hundred years after repository closure, the estimated radiotoxicity of the legacy waste will be higher than the radiotoxicity of the waste from the nuclear power plants, and the difference increases with time. In the estimated inventory of waste from the nuclear power plants, the total radiotoxicity after a thousand years has decreased to less than one percent and after a million years it constitutes less than 0.01 percent of the value at closure of the repository. The estimated inventory of the legacy waste contains a larger fraction of the relatively long-lived radionuclides technetium-99 and uranium-238. Together with the decay products in the uranium chain, this entails that the total radiotoxicity of the legacy waste decreases more slowly than the radiotoxicity of reactor internals. After a million years, it has decreased to approximately one percent of that at closure.

The total radiotoxicity in SFL (Figure 6-2) at closure is about ten times higher than in the extended SFR, but slightly lower than the inventory in one of about 6 000 copper canisters in the Spent Fuel Repository¹⁰.

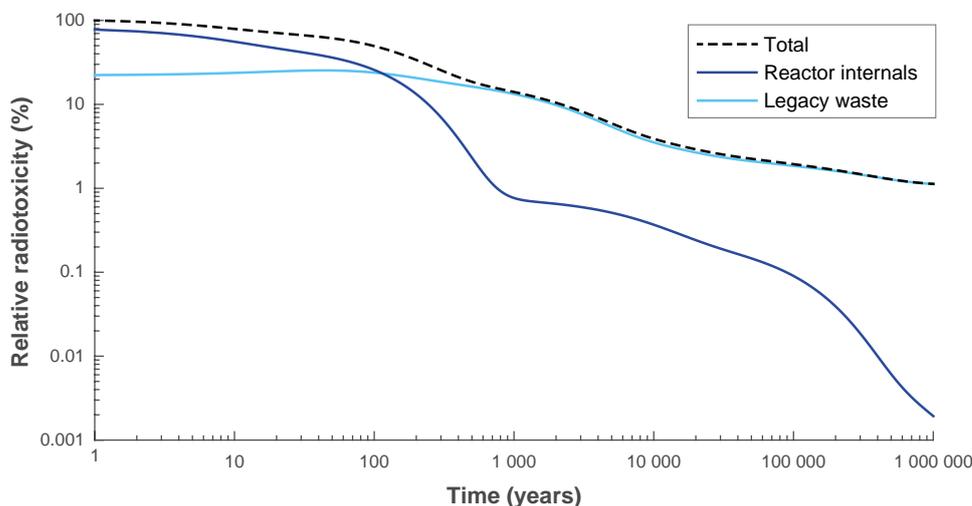


Figure 6-2. Radiotoxicity of the radionuclide inventory that is evaluated in SFL Safety Evaluation for waste from nuclear power plants (dark blue line; reactor internals) and legacy waste (light blue line). The radiotoxicity is shown as a percentage of the total radiotoxicity of the waste at closure.

⁹ The radiotoxicity of the radionuclide inventory has been calculated under the assumption that radionuclides are ingested.

¹⁰ The comparatively high toxicity of the spent nuclear fuel is an important reason for the use of a completely containing barrier for that waste form. The nuclear fuel itself, in contrast to large parts of the SFL waste, is also very insoluble in the repository environment.

Material properties

The proposed concept includes cementitious materials in waste matrices, structures and in the backfill in the waste vault for reactor internals, BHK. As a basis for the safety evaluation, the concrete has been assumed to have a composition that resembles the concrete used in the construction of the existing SFR. The post-closure evolution of the chemical and hydraulic properties of concrete in BHK has been modelled (Idiart and Shafei 2019, Idiart et al. 2019a, b, Idiart and Laviña 2019). Within the programme aimed at developing concrete for the caissons in 2BMA, a type of concrete containing relatively large quantities of finely ground limestone has been developed (Section 10.2.1). The studies carried out during the RD&D period indicate that this concrete has a very low initial porosity, which means that long-term chemical leaching is slowed down (Section 10.1.6). SKB's assessment is that this concrete could also be used in SFL (Section 10.3.1). Preliminary calculations of how the release of radioactivity from BHK would be affected by the use of such denser concrete have therefore been performed as a so-called evaluation case in the safety evaluation.

The proposed concept includes bentonite clay as backfill in the waste vault for legacy waste, BHA, and as a part of plugs and other closure components. Assumptions made in the safety evaluation regarding the properties and function of the bentonite are primarily based on existing knowledge. The backfill bentonite is assumed to have the same properties as the buffer bentonite in the Spent Fuel Repository. Processes that could lead to a change of the properties and function of the bentonite during the period up to and after closure are not expected to have any significant impact on the repository's capability to retard releases of radionuclides. This assumption needs to be revisited in future assessments on the basis of improved data. The point of departure for these future assessments is studies conducted for the Spent Fuel Repository (Section 11.3.2) and an ongoing initial modelling of mineral transformations in the backfill bentonite in BHA due to interactions with cementitious materials in waste matrices and the structure around the waste (Section 11.3.4).

Handling of uncertainties

Uncertainties in climate evolution (scenario uncertainty) are handled in the safety evaluation by including three variants that cover future climate domains expected to occur at a site in Sweden over the next one million years. The reference evolution also handles uncertainties relating to the hydrochemical evolution in the surrounding bedrock and uncertainties associated with the evolution of concrete and bentonite material in the waste vaults. Parameter uncertainties are also handled to some extent, for example in relation to groundwater flow and retention parameters for the surrounding bedrock and properties of the area where the release to the biosphere is assumed to occur.

Overall, efforts have been made in the safety evaluation to demonstrate an understanding of the repository's performance and evolution. Efforts have also been made to ensure the applicability of the models, parameter values and other assumptions used for the description and quantification of repository performance and evolution. There remains a need for a more elaborate analysis and documentation of uncertainties in future safety assessments for SFL.

6.1.3 Methodology and application

The methodology for the safety evaluation was developed based on SKB's methodology for assessments of post-closure safety. The application of the methodology is briefly described in this section. The analysis can be summarised in ten parts or steps (Figure 6-3), which constitute an iterative process.

Features, events and processes that may affect post-closure safety

The assessment is based on identification of all features, events and processes (FEP) that may affect the repository and are significant for the assessment of post-closure safety. FEPs are defined on the basis of the waste, the proposed repository concept and the example site. Due to the many commonalities between the proposed repository concept for SFL and SFR, the FEP catalogue from SR-PSU has been used as a basis for the analysis. The result of the FEP analysis was documented in a newly established FEP catalogue for SFL within the framework of SKB's FEP-database. All identified FEPs are described in the FEP catalogue. Furthermore, if and how each identified FEP has been handled in the safety evaluation is described. For FEPs that are not handled in the safety evaluation, an explanation is provided as well as an indication of whether the FEP is expected to require handling in a future complete safety assessment for SFL.

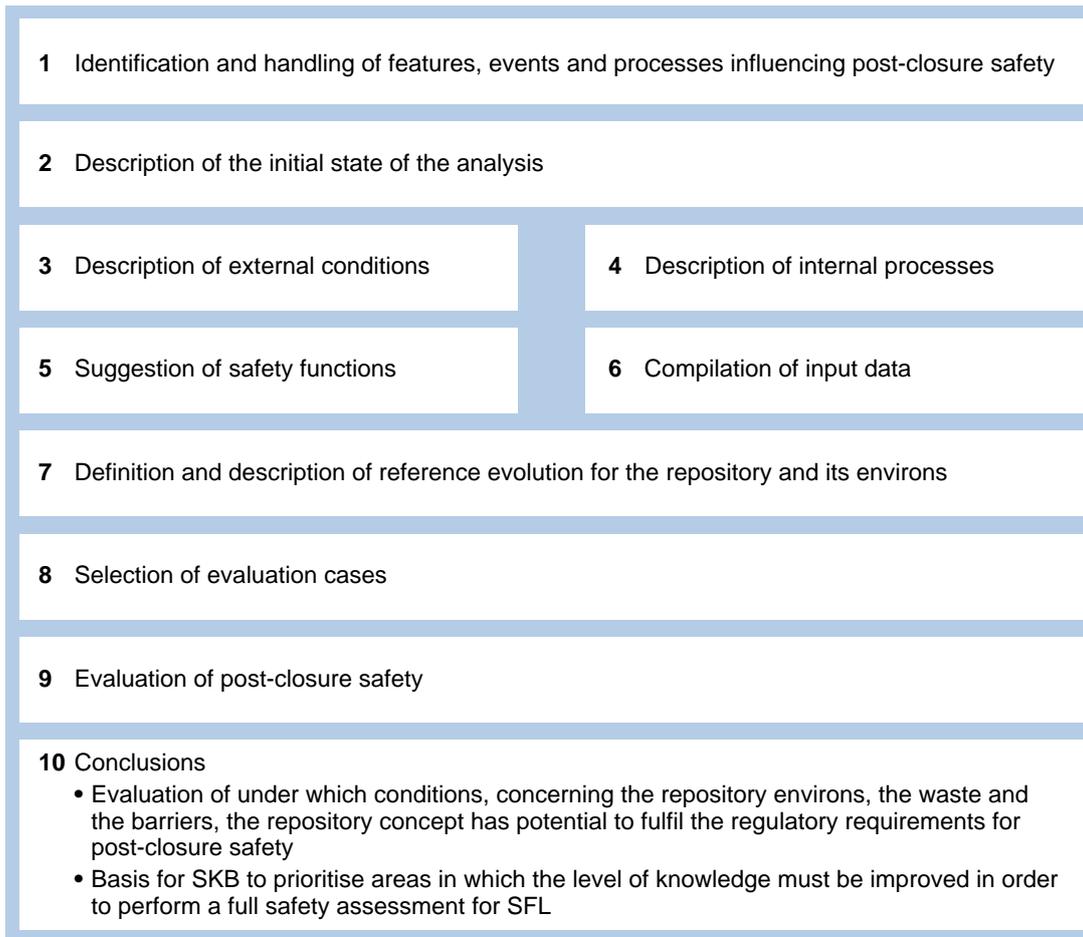


Figure 6-3. Overview of the methodology applied for the evaluation of post-closure safety for the proposed repository concept for SFL.

Initial state of the repository and its environs

The initial state of the analysis is defined as the expected state of the system, i.e. the repository and its environs, at repository closure. As the year for the closure of SFL is not yet determined, the repository has been assumed to be closed in 2075. The conditions in the repository’s surroundings are assumed to be similar to the present-day conditions at the example site in Laxemar (Section 6.1.2). The initial state of the repository was defined under these assumptions and on the basis of estimates of conditions in the waste and other repository components at closure. Some of the assumptions made regarding the initial state have been evaluated in greater detail (see step 8 *Selection of evaluation cases*).

External reference conditions

Changes of the climate and climate-related processes, such as shoreline displacement and development of permafrost and ice sheets, may affect the repository’s protective capability at the time scales covered by the evaluation. In order to be able to evaluate the significance of such changes for the evolution of the repository, a set of external reference conditions are selected as a basis for the evaluation.

The concept of external conditions also includes large-scale geological processes and effects, future human actions as well as other events (for example meteorite impacts). These processes and effects have not been included in the safety evaluation, but will be handled to the extent required in a future complete safety assessment.

The selection of external reference conditions for the safety evaluation is based on the same prerequisites as those considered in SKB’s previous safety assessments for SFR and the Spent Fuel Repository. These are briefly summarised as follows:

- It is highly likely that the current interglacial period, the Holocene, will be significantly longer than previous interglacial periods. Due to future variations in insolation and human emissions of greenhouse gases, this period is expected to extend between 50 000 and 100 000 years, or even longer, into the future.
- The reconstruction of the latest glacial-interglacial cycle, which was the starting point for the analysis in the most recent safety assessment for the Spent Fuel Repository, is a relevant example of the sequence of climate domains during a glacial-interglacial cycle.

Three variants of external reference conditions were selected (Table 6-1).

Table 6-1. Reference external conditions for the safety evaluation for SFL.

Variant	Description
Base variant	A hypothetical case where the current external (climate) conditions are assumed to prevail at the site throughout the assessment period (one million years). This simplification has been selected as a basis for the evaluation in order to build an understanding of how the proposed repository concept functions.
Increased greenhouse effect	Represents a warmer climate than today, with an altered hydrological seasonal cycle compared with current conditions, during the coming 50 000 years. Thereafter, the current external conditions are assumed to prevail during the rest of the assessment period.
Simplified glacial cycle	Represents a trend towards a colder climate than today. Defined based on the overall climate evolution in SKB's reconstruction of the latest glacial cycle (Figure 6-4) Permits evaluation of the effects of a succession of temperate climate, periglacial conditions with permafrost, ice-sheet evolution and submerged conditions. This simplified glacial-interglacial cycle is repeated ten times during the assessment period of one million years.

The selection of reference external conditions will be reconsidered prior to future complete safety assessments for SFL. In particular, additional bounding climate cases, which represent more extreme future climate evolutions than those included in the safety evaluation, may need to be included. The selection of climate cases is made on the basis of results from the safety evaluation, along with supplementary studies of the impact of climate evolution on the system's safety functions.

Internal processes

Relevant internal processes in SFL were identified in step 1 in the methodology and the handling of these in the safety evaluation for SFL has been documented in the FEP catalogue for SFL. The analysis is largely based on existing descriptions of relevant processes. Since the safety evaluation is based on data from another site than the most recent assessments for the extended SFR and the Spent Fuel Repository, both of which evaluate Forsmark, the process descriptions for the surface systems have been updated in the safety evaluation.

Safety functions

A central element in SKB's safety assessment methodology is the definition of a set of safety functions, which are used to clarify in which way a repository component contributes to the post-closure safety. The safety principle for the proposed repository concept is *retardation*. The proposed concept is thus based on a system of engineered and natural barriers that reduce the mobility and retard the transport of radionuclides from the repository (Figure 6-5). The effect of the retardation is that radionuclides are retained and decay, primarily in the repository but also in the surrounding bedrock.

In the safety evaluation for SFL, safety functions for the proposed repository concept are suggested, given the safety principle *retardation* and knowledge of the initial state and the internal and external processes. As the repository design, the barrier system and the siting of SFL have not yet been determined, the selection of safety functions needs to be reconsidered in future safety assessments. A future in-depth FEP analysis may also show that additional processes need to be considered in the selection of safety functions.

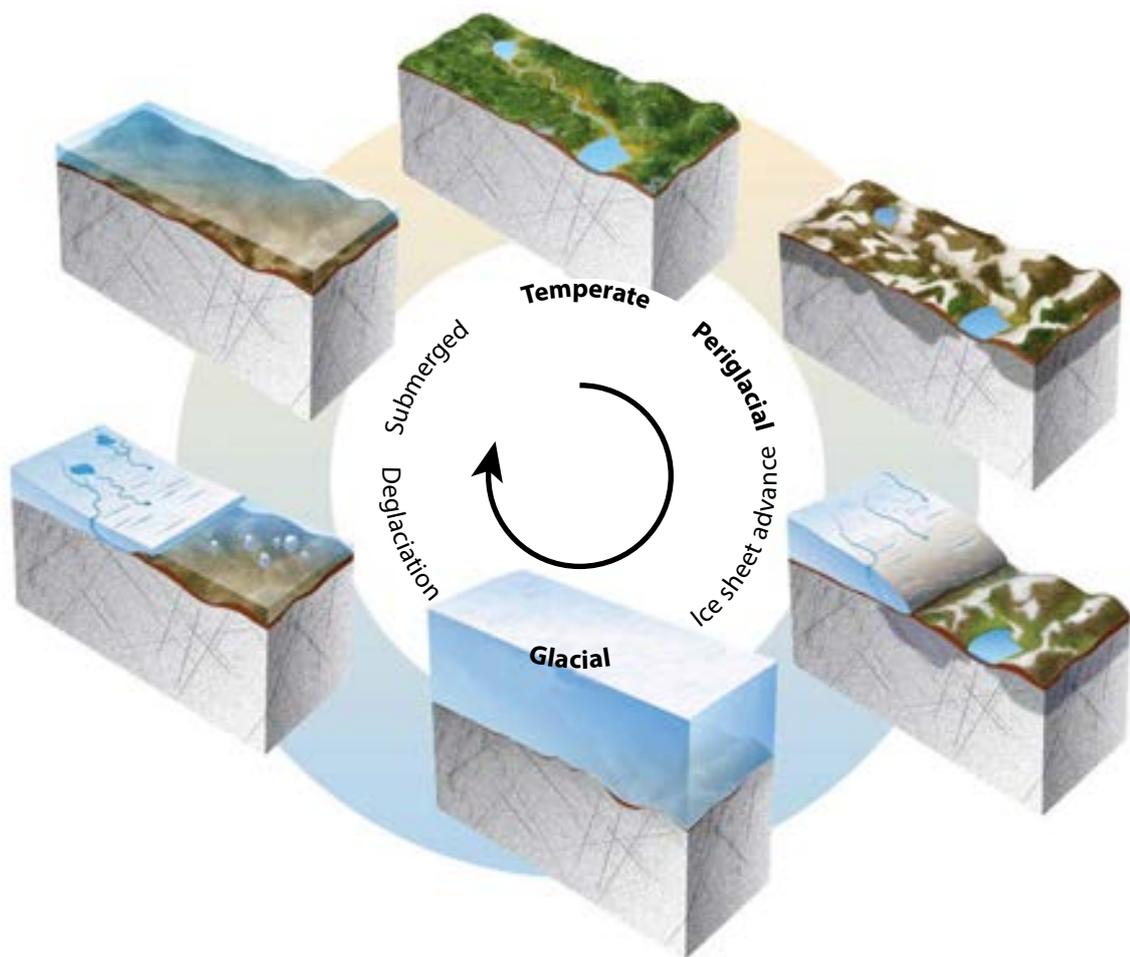


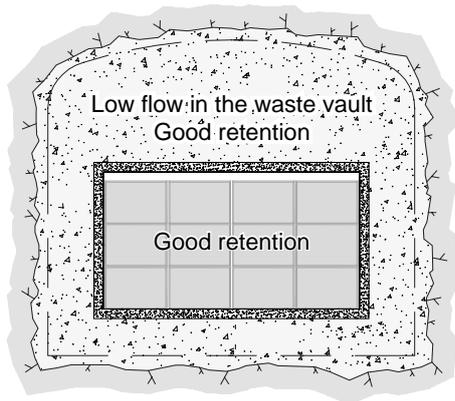
Figure 6-4. Simplified glacial cycle consisting of the three climate domains (in bold) that are considered relevant for assessments of the post-closure safety for a site by the Swedish Baltic Sea coast, such as Laxemar and Forsmark. The illustration also shows the advance and retreat of an ice sheet followed by a period of submerged conditions.

Retardation in the waste is achieved through maintaining the safety function *good retention* by means of favourable conditions with respect to for example high pH, low redox potential, low concentration of complexing agents, large available surface area for sorption and low corrosion rate of the metallic waste. The most important safety functions in the bentonite barrier in BHA are *low flow in the waste vaults* and *good retention*. These safety functions are maintained by means of low hydraulic conductivity, low diffusivity, high available surface area for sorption and high capacity for colloid filtration. The most important safety functions in the concrete barrier in BHK are *low flow in the waste vaults* and *good retention*. These safety functions are maintained by means of low diffusivity, large available surface for sorption, high pH and low redox potential.

Retardation in the surrounding bedrock is achieved by selecting a site that fulfils the safety function *low flow* in the bedrock.

The combined effect of retardation in the repository distributes the release of radioactivity from the repository over time. Thus it is ensured that the annual releases of radioactivity from the repository only constitute a very small fraction of the total inventory at closure. In addition to this effect, retention entails that a significant fraction of the initial radionuclide inventory decays before reaching the surface. This is especially true for radionuclides with short half-lives in relation to the time they are retained in the repository (or the bedrock).

Waste vault for reactor internals



Waste vault for legacy waste

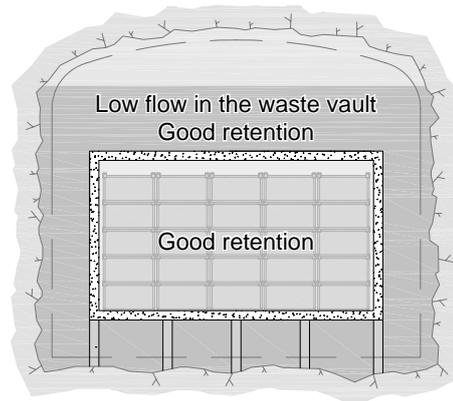


Figure 6-5. Proposed safety functions for the waste and engineered barriers in BHK and BHA. The waste vaults are backfilled with concrete (BHK) and bentonite (BHA).

Compilation of data

The natural system at the example site is largely described by the data obtained during previous site investigation for the Spent Fuel Repository in Laxemar. Since SFL would only take up around one percent of the rock volume investigated for the Spent Fuel Repository in Laxemar, data from new modelling of groundwater flow and groundwater chemistry (Joyce et al. 2019, Gimeno et al. 2019) have been used. Data from improved modelling of the surface groundwater system (Johansson and Sassner 2019) have also been used. The repository conditions are described based on the description of the natural system, as well as information from previous safety assessments and technology development at SKB. Data to describe the groundwater flow in and out of the waste vaults are taken from new modelling (Abarca et al. 2019). Data from modelling of the evolution of chemical and hydraulic properties of concrete in the waste vault for reactor internals (Idiart and Shafei 2019, Idiart et al. 2019a, b, Idiart and Laviña 2019) have also been used.

The possible effects of heat-generating waste have been studied (von Schenck 2018). Furthermore, the extent of gas formation in the waste and potential effects on groundwater flow in the repository and surrounding bedrock have been investigated (Silva et al. 2019a, b).

Reference evolution of the repository and its environs

In order to understand the overall evolution of the proposed repository concept at a relevant example site under representative external conditions, a reference evolution of the repository and its environs has been defined. It is based on the initial state and external reference conditions selected for the evaluation. The reference evolution consists of three variants based on the three variants of external reference conditions (see Table 6-1). FEPs that have been identified as having a potential impact on the post-closure safety for SFL have been included in the description.

The reference evolution is based on a combination of data from studies of expected conditions for the proposed repository concept, under the assumption that the repository will be located in the repository volume chosen for the evaluation in SFL Safety Evaluation, and data from studies of conditions in and around the Spent Fuel Repository or SFR. The description of the reference evolution is therefore not complete, but is considered sufficient for using the conclusions of the safety evaluation as a basis for the further development of SFL.

Selection of evaluation cases

In order to understand the proposed repository concept's performance and evaluate under what conditions it has the potential to meet the regulatory requirements, a set of evaluation cases was defined. The evaluation cases have been defined on the basis of the general evolution of the repository and its environs as described in the reference evolution. The set consists of a base case and cases that evaluate the significance of different properties, as well as the effects of uncertainties in the properties, of the

system analysed (Figure 6-6). The base case assumes that the current external (climate) conditions prevail at the site throughout the assessment period (one million years). Based on current knowledge of the properties of the waste and available barrier materials as described by the reference evolution, realistic, and in some respects simplified, assumptions are made regarding the conditions in the repository and its surroundings.

The term evaluation case was chosen, instead of the term scenario, in order to clarify that the selection was made from another perspective than that described in SSM's General Advice for SSMFS 2008:21 and SSMFS 2008:37. In this context, it is important to note that the evaluation cases included were chosen to serve as a basis for the continued development of SFL. The evaluation is not intended to provide a comprehensive risk analysis for SFL.

As a qualitative measure of the potential to fulfil the regulatory requirements and for comparison of how different assumptions affect the conditions for regulatory compliance, the calculated dose to a representative individual in the most exposed group is used. In a complete safety assessment, the main scenario should result in an annual dose below the dose (14 μSv) that corresponds to the risk criterion¹¹. A higher dose may be acceptable for less probable scenarios, provided that the combined risk for the main scenario and less probable scenarios is lower than the risk criterion. In the safety evaluation, the dose (14 μSv) corresponding to the risk criterion (hereinafter referred to as the dose criterion) is used instead as a reference value to evaluate the proposed concept's protective capability. Evaluation cases that result in an annual dose exceeding the dose criterion indicate that the conditions assumed to prevail in the repository and its surroundings represent a situation that is less favourable for achieving the repository's intended function. In contrast, an annual dose lower than the dose criterion indicates that the assumed conditions in the repository and its environs are more favourable for achieving the repository's intended function.

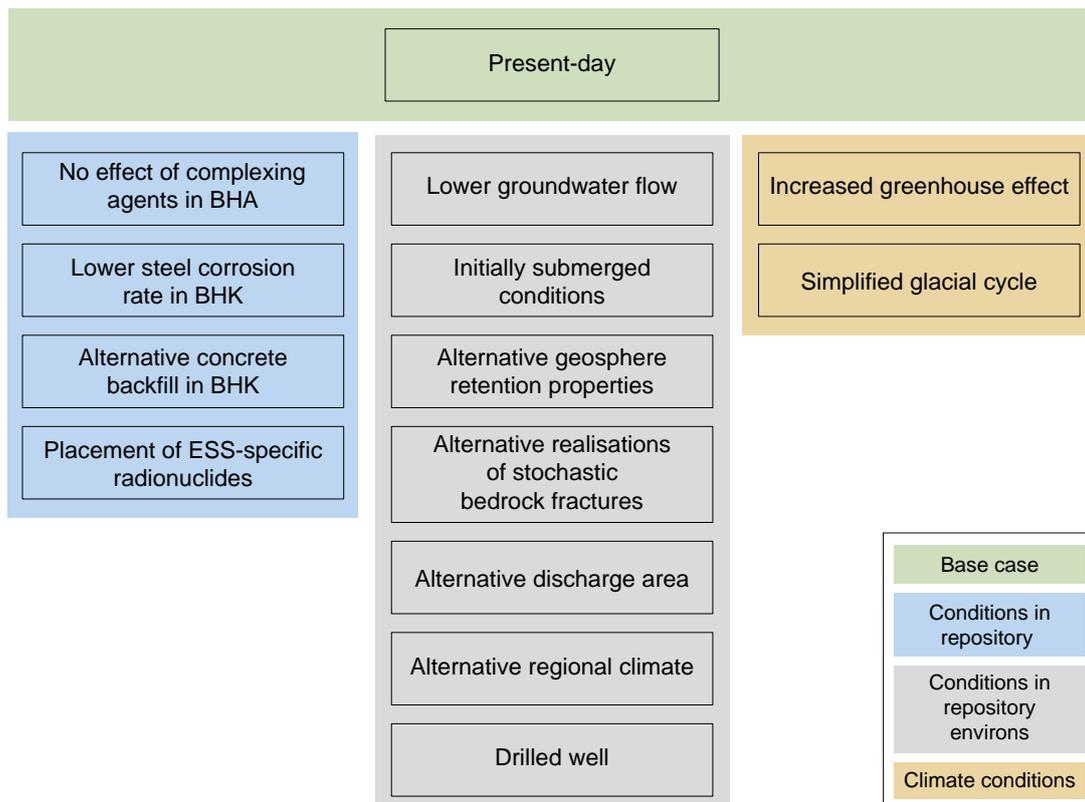


Figure 6-6. Evaluation cases included in SFL Safety Evaluation.

¹¹ SSM (SSMFS 2008:37) stipulates that the annual risk of harmful effects may not exceed 10^{-6} for a representative individual in the group exposed to the highest risk from the repository (the most exposed group). This gives an annual effective dose of 14 μSv , where the conversion to dose uses a risk factor of 7.3 percent per Sv (as specified in the regulations). The dose criterion 14 μSv corresponds to approximately one percent of the effective dose from natural background radiation in Sweden.

Base case

As a basis for the assessment, the evaluation case *present-day conditions* was chosen, defined on the basis of the description of the initial state and the base variant of the reference evolution (Table 6-1). The assumptions made in this base case were:

- Both the ground surface above the repository and the area where release from the repository is assumed to reach the surface systems are currently located more than ten metres above the Baltic Sea level. Therefore, the conditions in the repository's surroundings, i.e. the surrounding bedrock and the surface systems, are assumed to be constant throughout the assessment period. The effects of submerged conditions and shoreline displacement are instead assessed in the evaluation cases *simplified glacial cycle* and *initial submerged conditions*. The effects of landscape development in conjunction with shoreline displacement are evaluated further in the evaluation case *alternative discharge area*.
- For a significant fraction of the metallic waste in the waste vault for reactor internals, the induced radioactivity is assumed to be released through corrosion, where a constant corrosion rate is assumed. For material thinner than 1 mm and surface contamination, however, the radionuclide inventory is assumed to be available for outward transport and sorption directly after closure. The significance of these assumptions is evaluated further in the evaluation case *lower steel corrosion rate in BHK*.
- The concrete backfill in the waste vault for reactor internals is assumed to have a composition that resembles the concrete used in the construction of the existing SFR. Due to leaching, the properties of the concrete evolve over time, which leads to increased hydraulic conductivity and a decrease of pH. These changes are assumed to influence the advective and diffusive transport in the repository, as well as the sorption of radionuclides on cement in the waste and backfill.
- The calculated release from the waste vaults is distributed on different transport paths from the interface the waste vaults and the bedrock to the interface between the bedrock and soil layers. In the five realisations of groundwater flow that have been carried out (Joyce et al. 2019), a majority of the transport paths from the waste vaults end up in one of the three discharge areas. As all these areas represent similar conditions today, one of them (biosphere object 206, see Figure 6-1) was selected for the base case, and the total release of radioactivity at the rock surface is assumed to be released in this area. The area is assumed to be mireland that is drained and cultivated.

Some aspects of the reference evolution base variant have been excluded in the base case, despite the fact that they could affect the evolution of the barriers and the performance of the repository. In contrast to the concrete backfill in the waste vault for reactor internals, the bentonite backfill in the waste vault for legacy waste is assumed to have constant properties throughout the assessment period. Processes that could lead to a change of the properties and function of the bentonite during the period up to and after closure are therefore not expected to have any significant impact on the repository's capability to retard releases of radionuclides. In particular, this applies to the chemical interaction between concrete in the waste domain and the bentonite backfill, which may lead to transformation of montmorillonite in the bentonite clay to other minerals. The transformation takes place from the inside out, starting at the interface between concrete and bentonite. The extent of this process and how it affects the barrier safety functions need to be quantified in future safety assessments for SFL. Loss of bentonite by colloid release is also excluded in the base case. The extent of such release over an extended period of temperate climate conditions, where groundwater with sufficiently low salinity could penetrate into the bentonite backfill, needs to be evaluated further in future safety assessments.

As in SKB's previous safety assessments for the extended SFR and the Spent Fuel Repository, a number of simplifications are made in the calculation of radionuclide transport and potential annual dose to an individual in the most exposed group. They likely entail an overestimation of the calculated maximum radiological consequences. The evaluation does for example not consider

- solubility limitations for radionuclides
- the transport resistance of waste containers
- the possibility for dissolved radionuclides to precipitate in the solid state
- decreasing levels of complexing agents in the legacy waste over time
- that the release may be distributed over more than one geographically limited area.

Since the uncertainty in the inventory of legacy waste is considerable, additional simplifications are made for this waste vault. Here, it is assumed that the entire radionuclide inventory is available for outward transport at closure, despite the fact that some of the radioactivity in the waste is likely to be contained in solid materials.

Climate evolution

The sensitivity of the calculated dose to future climate evolution has been analysed in the evaluation cases *increased greenhouse effect* and *simplified glacial cycle*. The evolution in these cases is based on the corresponding variants of the reference evolution (Table 6-1). As in the base case, important aspects of the reference evolution variants are included in these evaluation cases, however, some aspects have been excluded even though they could affect the evolution of the barriers and the performance of the repository. Similarly to the base case, any effects from the loss of bentonite by colloid release, due to inflow of glacial meltwater with low salinity, have been excluded. The load from an ice sheet may have a mechanical impact on the surrounding rock and possibly also the waste vaults. The outer part of the concrete backfill is assumed to already be chemically leached when the ice sheet develops in the evaluation case *simplified glacial cycle*. Therefore, the possible effect of mechanical impact has not been included in the evaluation case. In a complete safety assessment, however, the combined effect of chemical leaching of concrete and increased rock stresses must be evaluated.

Conditions in the repository and its surroundings

In order to assess the potential for the proposed repository concept to satisfy the regulatory requirements, given better conditions than those assumed in the base case, a set of evaluation cases was defined (Figure 6-6). Furthermore, evaluation cases were included where the significance of inherent uncertainties of the system has been analysed.

6.1.4 Results

The dose to a representative individual in the most exposed group has been calculated using models that calculate releases of radionuclides from the repository, transport from the repository to the surface systems, uptake and exchange in the biosphere, and the doses to humans (Figure 6-7). The models are intended to cover the main FEPs for transport, accumulation and exposure to radionuclides. The models used are, to the extent applicable, equivalent with the models used by SKB in the most recent safety assessments for the extended SFR and the Spent Fuel Repository. New models that calculate transport in and out of the waste vaults for reactor internals and legacy waste have been developed (Wessely and Shahkarami 2019). These models are based on the same principles as corresponding models for the extended SFR. For transport in the bedrock, a new implementation of the conceptual model from previous safety assessments was used. For transport and accumulation in the surface systems and exposure to radionuclides, a model with the same level of detail and where most of the assumptions were based on the most recent evaluation for the extended SFR (SR-PSU) was used. The model used in SR-PSU is considered to provide a good description of the surface systems (see for example SSM 2019). The model used in SFL Safety Evaluation is, therefore, considered to provide adequate representation of the dose consequences of possible evolutions of the conditions at a facility site by the Swedish coast.

The proposed repository concept's function is briefly described below, based on the results from the base case. Results and conclusions from the evaluation are discussed and summarised in Section 6.1.5.

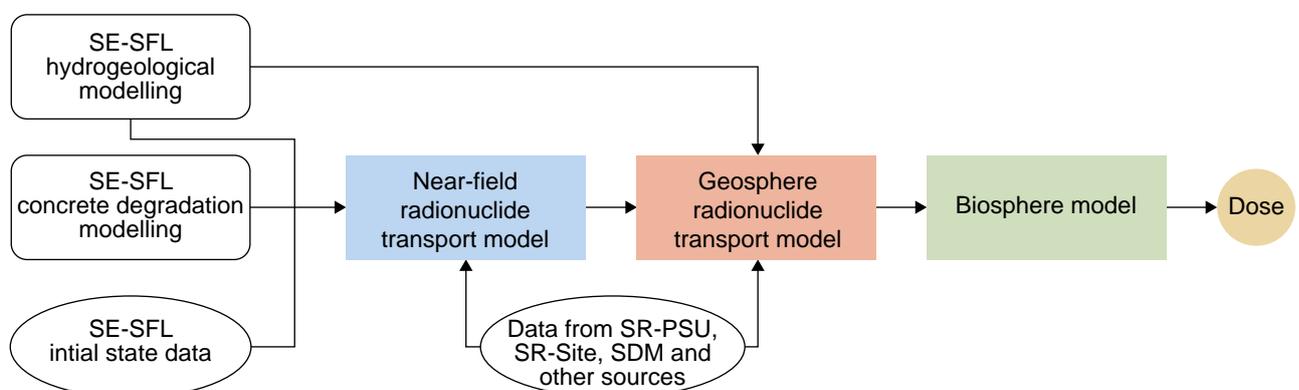


Figure 6-7. Schematic illustration of the model chain used in SFL Safety Evaluation for calculation of radionuclide transport and dose (Shahkarami 2019).

Retarding function of the proposed repository concept

This section provides a summary of the results from the base case with a focus on the proposed concept's function. The calculated annual dose, given the assumptions made in the base case, is presented in the next section. The account of the results is based on the radionuclides that give the greatest contribution to the annual dose. The contributing radionuclides largely coincide with the radionuclides that have the greatest initial inventory in each waste vault and a half-life longer than a few thousand years, i.e. technetium-99, carbon-14 and chlorine-36 from BHA, and molybdenum-93, carbon-14 and nickel-59 from BHK.

Figure 6-8 illustrates how the repository and the surrounding bedrock contribute to the retardation of radionuclides in the repository, the surrounding bedrock and soil layers. For each radionuclide, the figure shows the accumulated radioactivity (summed over the assessment period of one million years) that is transferred along the pathway from the waste in BHA (left panel) and BHK (right panel) to the biosphere in relation to the initial radioactivity. The following factors increase the residence time in the repository and thereby contribute to radionuclides decaying before they reach a recipient:

- The low advective flow through the waste.
- Most of the radionuclide inventory in BHK are released slowly from the metallic waste through corrosion (radionuclides from the thinnest metal parts and surface contamination, however, are assumed to be available for transport at closure).
- The low hydraulic conductivity of the bentonite, which results in diffusion-dominated transport in and out of BHA.
- Sorption on cement in the waste and in the backfill (BHK) and on bentonite in the backfill (BHA) (for certain radionuclides however, sorption on cement and bentonite in BHA is reduced, since they are assumed to form non-sorbing complexes with decay products from cellulose).

The analysis of the results from the base case shows that the engineered barriers in BHA and BHK provide efficient retardation of radionuclides in the waste vaults, given the properties of the example site. The retardation in the repository is particularly efficient for radionuclides with shorter half-life such as carbon-14 and molybdenum-93 (with a half-life of 5 730 years and 4 000 years, respectively), see Figure 6-8. For non-sorbing carbon-14, around 95 percent of the radioactivity decays in the waste vaults. However, retardation depends on different mechanisms in the two waste vaults, in BHA it is mainly due to slow diffusion-dominated outward transport, whereas the slow release through corrosion is the most important reason in BHK. Sorption on cement, and a shorter half-life, entails that a larger fraction of the inventory of molybdenum-93 (more than 99 percent) decays in the waste vaults.

The accumulated radioactivity that is transferred from the waste vaults to the surrounding bedrock is greatest for the long-lived radionuclide chlorine-36 (half-life 311 000 years). Only about 40 percent of the inventory decays in the waste vaults (Figure 6-8). The outward transport is, however, slow and the annual outflow of chlorine-36 from each waste vault is more than five orders of magnitude lower than the initial inventory.

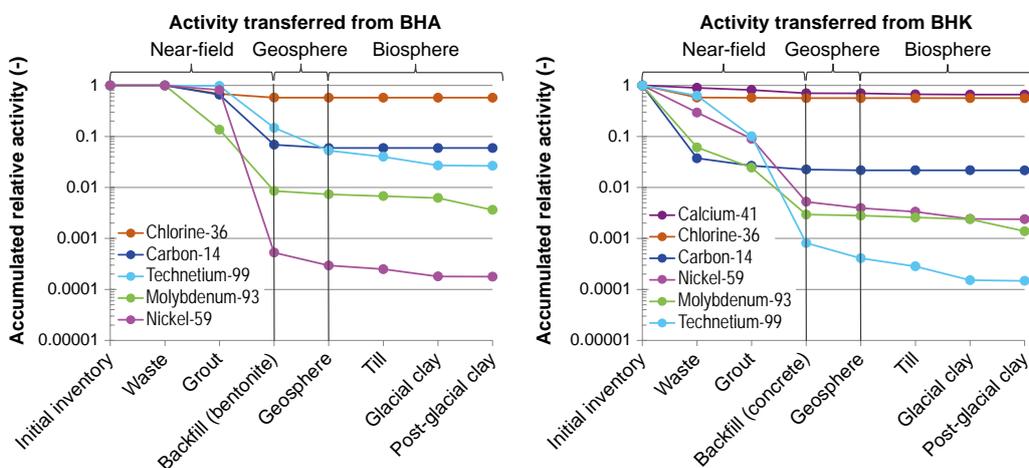


Figure 6-8. Radioactivity transferred along the pathway from the waste in BHA (left panel) and BHK (right panel) to the biosphere. The accumulated radioactivity (summed over the assessment period of one million years) that is transferred is shown in relation to the initial activity of each radionuclide.

Sorption on cement and bentonite (BHA), and a shorter half-life, entails that a larger fraction of the inventory of technetium-99 than chlorine-36 decays in the waste vaults. The retardation in BHK is greater than in BHA since sorption on cement and bentonite of technetium-99 in BHA is significantly impeded by the presence of complexing agents. Without the presence of complexing agents, the release of technetium-99 from BHA would decrease by more than five orders of magnitude.

Nickel-59 has a much longer half-life (76000 years) than carbon-14 and molybdenum-93, but, in contrast to these two radionuclides, nickel-59 sorbs strongly on cement and bentonite. This, together with the slow diffusion-dominated transport, entails that less than a thousandth of the inventory of nickel-59 is transported out of BHA. In BHK, the outward transport is delayed by both slow release through corrosion and sorption on cement, which leads to decay of nickel-59 in the waste vault. The outward transport through the concrete barrier in BHK is greater than through the bentonite barrier in BHA. Here, however, both slow release through corrosion and sorption on cement contribute to the decay of nickel-59 in the waste vault and entail that less than one percent of the inventory of nickel-59 is transported out of BHK.

Retention and decay in the bedrock have a marginal effect on the release to the biosphere for radionuclides that do not sorb in the bedrock, for example chlorine-36, carbon-14 and molybdenum-93 (Figure 6-8). Even for long-lived, weakly sorbing radionuclides such as technetium-99, the effect of the passage through the bedrock is limited.

Calculated dose to a representative individual in the most exposed group

Since there are relevant differences between both the waste and the proposed repository concept (Section 6.1.2) for the waste vaults BHA and BHK, the calculated annual dose is presented separately for the two waste vaults. In this way, the conditions under which each waste vault has the potential to meet the regulatory requirements may be evaluated.

The calculated annual dose in the base case exceeds the dose criterion by a factor of 4 for BHK and a factor of 6 for BHA (Figure 6-9). For a set of radionuclides, the annual dose is within one order of magnitude from the dose criterion: carbon-14 and molybdenum-93 in both waste vaults, and technetium-99, chlorine-36 and radionuclides in the decay chain of uranium in BHA. Here, the discharge area is assumed to be a mire that is drained and cultivated, as this is the land use that gives the highest dose. This is due to the fact that radionuclides can accumulate in peat over a long period of time before humans are exposed and that the dominating radionuclides contribute mainly through ingestion of food. However, water from wells also contributes to the dose, mainly from technetium-99 during the first part of the assessment period and from decay products in the uranium decay chain towards the end of the assessment period.

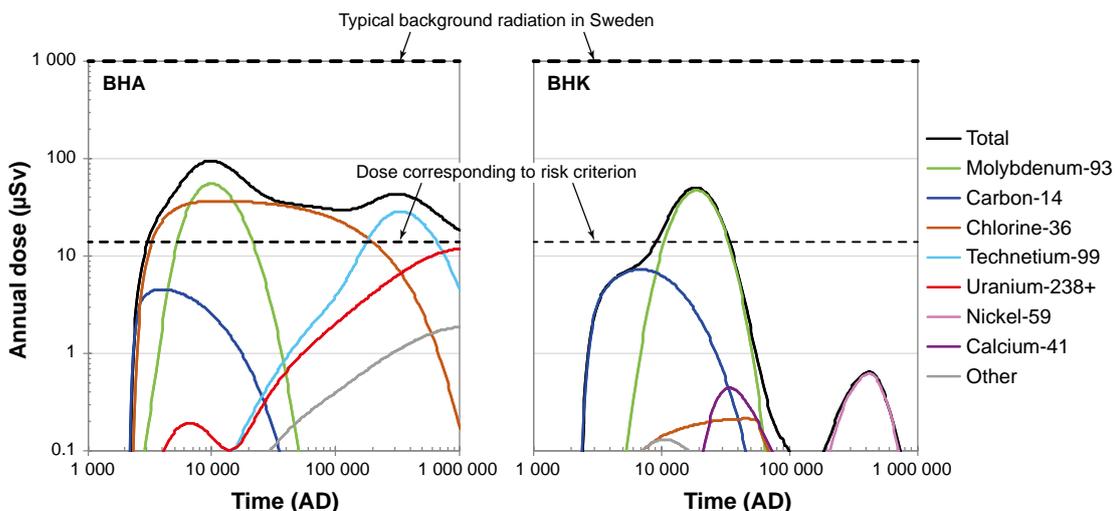


Figure 6-9. Annual dose from BHA (left panel) and BHK (right panel) in the base case (see the text). The total annual dose for all radionuclides (black line) and the contributions from the radionuclides that contribute the most to the total dose are shown. The dose corresponding to the regulatory risk criterion (14 µSv) and dose by natural background radiation in Sweden (approximately 1000 µSv) are illustrated with dotted black lines.

The variation in annual dose for a given release spans more than three orders of magnitude for nuclides with the highest geosphere releases. The differences are explained by differences in half-life, sorption in different soil layers, plant uptake and radiotoxicity. For example, molybdenum-93 provides high doses compared with nickel-59 given the same releases from the bedrock.

6.1.5 Discussion and conclusions

This section discusses the results of the safety evaluation for SFL based on the evaluation goal (see step 10 in Figure 6-3):

- Evaluation of the conditions in the repository surroundings, the waste and barriers under which the repository concept has the potential to meet the requirements on post-closure safety.
- A basis for SKB to prioritise areas where the level of knowledge needs to be improved in order to carry out a complete safety assessment for SFL that meets SSM's requirements.

Furthermore, conclusions drawn from the safety evaluation are summarised.

Given the assumptions made in the *present-day conditions evaluation case* (the base case), the calculated annual dose exceeds the dose corresponding to the risk criterion for BHA and BHK individually (Figure 6-9). The total calculated dose for SFL is about ten times as high as the dose criterion. In this context it is important to remember that the calculated dose is dependent on the assumptions made in the safety evaluation. The inventory of radionuclides and other materials in BHA is associated with large uncertainties (Section 6.1.2). In the present evaluation, a changed inventory (higher or lower) would entail a changed annual dose that is proportional to the inventory change. Some aspects of the reference evolution's base variant have been excluded from the base case, despite the fact that they could affect the performance of the barriers and the repository (Section 6.1.2). This especially concerns processes that may lead to changes of the properties and function of the bentonite during the period up to and after closure. Furthermore, as in SKB's previous safety assessments for the extended SFR and the Spent Fuel Repository, a number of simplified assumptions are made in the calculation, which are likely to entail an overestimation of the calculated maximum radiological consequence (Section 6.1.3).

The evaluation of climate evolution in the *increased greenhouse effect* and *simplified glacial cycle* evaluation cases provided a similar or lower annual dose than the dose calculated for *present-day conditions*. The results cannot be compared directly with the regulatory risk criterion, since none of the evaluation cases included in the safety evaluation was chosen to correspond to a main scenario in a full safety assessment. As described under the heading Selection of evaluation cases in Section 6.1.3, however, the calculated annual dose is used as a qualitative measure of the potential to meet regulatory requirements (on the protective capability), and for comparison of how different assumptions affect the conditions for regulatory compliance. The dose criterion, i.e. the dose corresponding to the risk criterion, is used in this regard as a reference value to evaluate the proposed concept's protective capability. Evaluation cases that result in an annual dose exceeding the dose criterion indicate that the conditions assumed to prevail in the repository and its environs represent a situation that is less favourable for achieving the repository's intended function. Correspondingly, cases that result in an annual dose lower than the dose criterion indicate that the conditions assumed to prevail in the repository and its surroundings are more favourable for achieving the repository's intended function.

Potential to meet regulatory criteria on post-closure safety in BHA

• Characterisation of waste

During the first 100 000 years after closure, the calculated annual dose from BHA is dominated by radionuclides with a high initial inventory and a half-life longer than a few thousand years (molybdenum-93, chlorine-36 and carbon-14). The inventory of the legacy waste is associated with great uncertainties, see further Section 6.1.2. The results of improved characterisation of the legacy waste cannot be predicted, but as an example an inventory of these radionuclides that is ten times lower in combination with an unaltered or lower inventory for other radionuclides, would provide an estimated annual dose lower than the dose criterion, given the assumptions made in the base case.

After 100 000 years after closure, technetium-99 and radionuclides in the decay chain of uranium-238 contribute to an annual dose that exceeds the dose criterion. These results are also affected by the uncertainty in the inventory. Furthermore, the entire inventory in BHA was assumed

to be available for transport after closure, despite the fact that parts of the waste is induced metal. Other parts of the waste, primarily radionuclides in the decay chain of uranium-238, are likely in the solid state. More realistic assumptions could lead to lower releases of these radionuclides, provided that their decay products are also retained by sorption on materials in the waste vault. For the period after 100 000 years, the assumptions that have been made concerning the occurrence and durability of complexing agents in BHA also play a significant role, since they greatly impair the sorption of e.g. technetium and radionuclides in the decay chain of uranium. This is illustrated in the evaluation case *no effect of complexing agents in BHA*, where these radionuclides do not contribute significantly to the calculated dose.

- **Groundwater flow**

The fraction of radionuclides that are retained and decay in the repository and in the bedrock would increase with a *lower groundwater flow rate* than that of the base case. The effect is exemplified in the lower groundwater flow rate evaluation case, where a factor of 100 lower flow than those assumed in the base case entails a significant reduction of the releases, as is demonstrated for molybdenum-93 and technetium-99 in Figure 6-10. The largest reduction of releases occurs in the bedrock, where typical transport times of the order of magnitude 10 000 to 100 000 years are sufficient to meet the dose criterion given the assumptions made in the safety evaluation. For a repository position with 10 times lower groundwater flow than in the base case, with typical transport times in the bedrock of the order of magnitude 1 000 years, no major reductions of the releases from the waste vault or the bedrock are expected. This is due to the fact that the release from bentonite backfill to the groundwater in the bedrock is governed by diffusion, and that the transport time is significantly shorter than the half-life of the dose-dominating radionuclides.

Existing data from preliminary investigations and site investigations indicate that more favourable hydrogeological conditions than those assumed in the base case may be found at greater depths in Laxemar or at other locations in Sweden. A significant lowering of the groundwater flow rates is also expected if the repository is located at a site that is covered by the sea. This is exemplified in the safety evaluation in the *initially submerged conditions* evaluation case, where the groundwater flow rates are assumed to be a factor of 100 lower than in the base case during a submerged period. The result is significantly reduced releases from the waste vault and from the bedrock, but the effect only lasts until the submerged period is over. The length of the initially submerged period thus needs to be significantly longer than the half-life of the radionuclides that contribute to the calculated dose and in BHA, this means the entire assessment period (Figure 6-10).

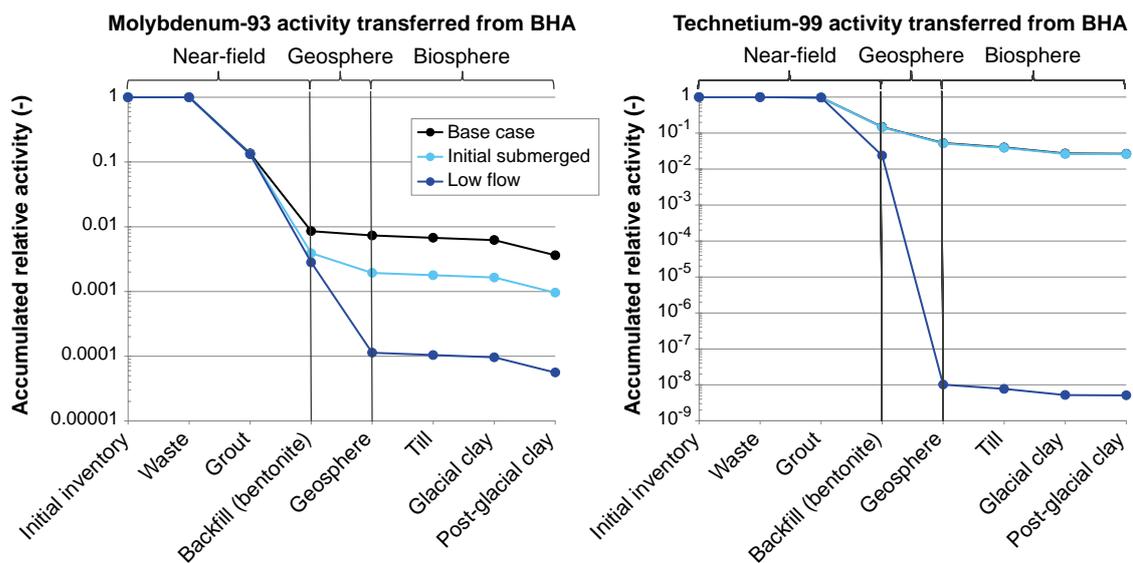


Figure 6-10. Radioactivity from molybdenum-93 (left panel) and technetium-99 (right panel) that is transferred along the pathway from the waste in BHA to the biosphere. The accumulated radioactivity (summed over the assessment period of one million years) that is transferred is shown in relation to the initial activity of each radionuclide. The results are shown for the base case (black line), submerged conditions during the first 10 000 years after closure (light blue line) and a factor of 100 lower groundwater flow rates (dark blue line). Note that the black line is overlain by the light blue curve the right panel.

- **The barrier system's ability to withstand FEPs that may affect barrier performance after closure.**

Processes that affect the evolution of the bentonite barrier over time are the most relevant for the impact on barrier performance after closure. As described in the reference evolution for BHA, montmorillonite transformation and bentonite colloid release are two processes that need to be taken into account. A study of montmorillonite transformation, occurring as a result of the interaction between bentonite and concrete, has been initiated in the safety evaluation (Section 11.3.4). Given the results of this study, the impact on the bentonite barrier safety functions *low flow in the waste vault* and *good retention* need to be evaluated. In order to quantify the effects of this process on the calculated dose in future assessments for SFL, this process needs to be adequately conceptualised in the radionuclide transport calculations.

Bentonite colloid release is discussed in the reference evolution, but the process has not been quantified for the example site in the safety evaluation, as the geochemical conditions at repository depth have been assumed to be such that colloid release is prevented. In future assessments for SFL, the process needs to be quantified based on studies performed for the Spent Fuel Repository (Section 11.3.2), and, to the extent applicable, be conceptualised in the radionuclide transport calculations.

The bentonite barrier's mechanical evolution, especially during the initial period of resaturation, is discussed in the reference evolution. To substantiate the discussion of barrier system robustness, the impact of resaturation on the bentonite barrier needs to be analysed in a future assessment.

Properties in the surrounding bedrock must be such that the integrity of the engineered barriers is not lost and the limitation of the outflow of radionuclides is maintained. A detailed analysis may provide a basis for optimisation of bentonite barrier properties and the geometry of the reference design for BHA.

Potential to meet regulatory criteria on post-closure safety in BHK

- **Corrosion of metallic waste**

The slow release of radionuclides through corrosion of the metallic waste is important in order to limit the release of radionuclides from the waste vault. The corrosion rate assumed in the base case (10 nm/year) is taken from SR-PSU. This rate is higher than recently reported corrosion rates of stainless steel in an alkaline oxygen-free environment. In PSAR for the extended SFR, a range has been defined to represent the uncertainty in the corrosion rate of stainless steel in an alkaline oxygen-free environment, where the corrosion rate assumed in the base case for the safety evaluation is ten times higher than the lowest value in the range. The sensitivity to this uncertainty was tested in the *lower corrosion rate in BHK* evaluation case. A reduction of the assumed corrosion rate to a tenth (1 nm/year) provides reduced releases and doses by about a factor of four and thereby a calculated dose just below the dose criterion. That the effect is not larger is due to the fact that the induced activity in the metallic parts thinner than 1 mm is assumed to be immediately available for outward transport after closure. A revision of the assumptions made with respect to corrosion rate and the fraction of the induced activity immediately available for outward transport at closure needs to be carried out in a future assessment.

- **Concrete backfill**

The quality of backfill concrete is a design choice, and with denser concrete than what is assumed in the base case, BHK has the potential to meet the risk criterion given other assumptions in the base case. The evaluation case *alternative concrete backfill in BHK* assumes transport properties that correspond to those measured for concrete containing additives of limestone and dolomite (Lagerblad et al. 2017). This concrete has been used for manufacturing mockup caisson structures corresponding to the caissons in 2BMA that has been carried out in the Äspö HRL (Mårtensson and Vogt 2019). Furthermore, concrete degradation of this type of concrete has been modelled and analysed by Idiart et al. (2019b). The results of the evaluation case show that the release of radionuclides from the waste vault decrease by more than two orders of magnitude for the dose-dominating radionuclide molybdenum-93 (see left-hand panel in Figure 6-11). The release of carbon-14 is also affected. The calculated maximum dose is more than one order of magnitude lower than in the base case and thus well below the dose criterion.

- **Groundwater flow**

As for BHA, the fraction of radionuclides that are retained and decay in the repository and in the bedrock increases with a *lower groundwater flow rate* than the rate assumed in the base case. The effect is exemplified in the evaluation case lower groundwater flow rate, where a factor of 10 lower flows than those assumed in the base case (i.e. with a total flow through BHK of $10^{-1} \text{ m}^3/\text{yr}$) provides a significant reduction of the releases, as is demonstrated for molybdenum-93 and carbon-14 in Figure 6-11. Since advection dominates the transport through BHK in the base case, the reduced flow results in a more substantial reduction of the transport of radionuclides from the waste vault to the bedrock than in BHA, where the bentonite barrier efficiently limits advective transport. The calculated maximum annual dose is less than a tenth of the base case and thus well below the dose criterion. Transport times in the bedrock are also increased by a factor of 10, but this does not lead to a sufficient increase of the residence time to affect the fraction of the dominant radionuclides that decay in the bedrock to any appreciable extent.

In the same way as for BHA, the effect of selecting a site that is covered by the sea at closure is exemplified in the *initially submerged conditions* evaluation case, where the groundwater flow rates are assumed to be a factor of 100 lower than in the base case during the submerged period. The result is significantly reduced releases from the waste vault and from the bedrock, but the effect only lasts until the submerged period is over. The length of the initially submerged period thus needs to be significantly longer than the half-life of the radionuclides that contribute to the calculated dose. For BHK, whose radionuclide inventory contains significantly less long-lived radionuclides than BHA, a period of about 50000 years would be required.

- **The barrier system's ability to withstand FEPs that may affect barrier performance after closure.**

The safety functions of the repository components in BHK are *good retention* in the waste domain, and *low flow through the waste vault* and *good retention* in the concrete barrier. Good retention is maintained in the waste domain by high pH, which leads to slow corrosion and thereby slow gas formation, as well as a large available sorption area on cement in the grout. In the concrete barrier, the low hydraulic conductivity of the concrete entails a low flow through the repository which, together with a large available sorption area on cement and low diffusivity in the concrete, yields good retention.

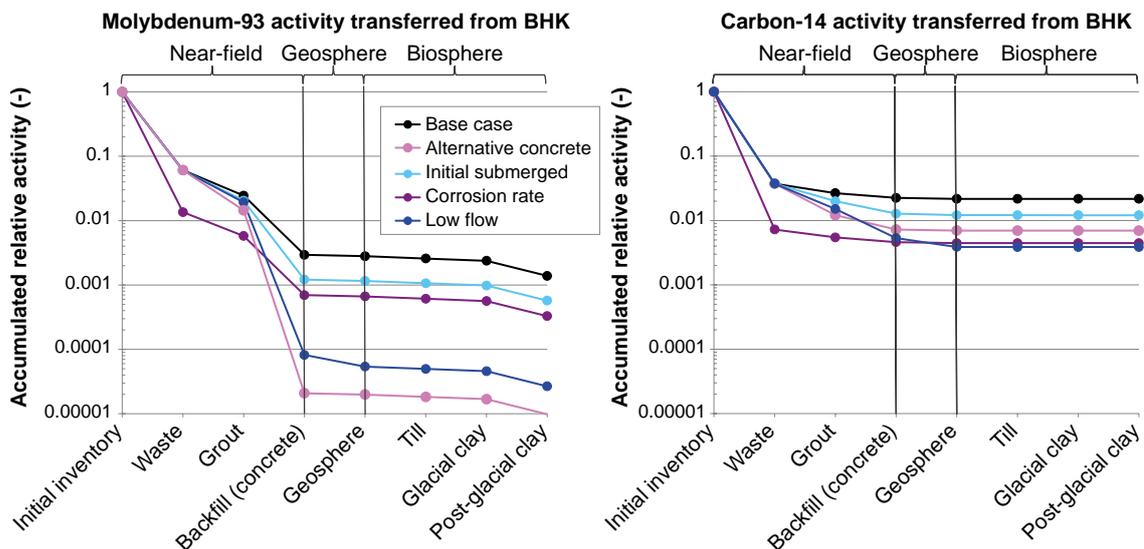


Figure 6-11. Radioactivity from molybdenum-93 (left panel) and carbon-14 (right panel) that is transferred along the pathway from the waste in BHK to the biosphere. The accumulated radioactivity (summed over the assessment period of one million years) that is transferred is shown in relation to the initial activity of each radionuclide. The results are shown for the base case (black line), submerged conditions during the first 10000 years after closure (light blue line) and a factor of 10 lower groundwater flow rates (dark blue line). Furthermore, the results are shown for the evaluation case with 10 times lower corrosion rate than in the base case (yellow line) and the case with (denser) alternative concrete backfill in BHK (magenta line).

Processes that affect the evolution of the concrete barrier over time are the most relevant for the impact on barrier performance after closure. Cementitious materials which come into contact with groundwater are affected by the chemical composition of the water, as well as by the magnitude and direction of the water flow. Dissolution or precipitation of minerals alter the concrete pore structure, which affects the hydraulic and mechanical material properties, and these processes also affect mass transport properties of the concrete matrix and the pH of the pore water. In the preceding RD&D period, SKB has worked to deepen the process understanding of degradation of concrete under repository conditions, both by means of modelling and through experimental studies (Sections 10.1.1 and 10.1.6). The modelling performed was based on the design of the waste vault BHK in SFL and included evolution of chemical and hydraulic properties of concrete in a perspective of up to one million years under the influence of hydro-chemo-mechanical processes (Idiart and Shafei 2019, Idiart and Laviña 2019, Idiart et al. 2019a). Furthermore, studies of the influence of alternative concrete compositions (Idiart et al. 2019b) have been carried out. The main conclusions are that:

- The primary degradation process is leaching of calcium, which leads to a gradual loss of portlandite and C-S-H gel.
- The degradation rate in the backfill in BHK is very low, mainly due to the low water flows in the concrete backfill.
- The degradation rate in the backfill in BHK is lower for the concrete containing large quantities of finely ground limestone that has been developed within the framework of the extension of SFR than for the concrete used in the base case of the safety evaluation. The concrete's initial porosity and transport properties have a larger impact on the extent of the degradation over time than the chemical composition of the concrete in the cases studied.

These studies also demonstrated that mechanical loads are expected to cause only very limited damage in connection with the interface between the bedrock and backfill in BHK. The only exception is glaciation loads, which may cause more severe damage in a larger part of the backfill.

All in all, these studies show that changes in the properties of the concrete in BHK after repository closure will take place very slowly. The barrier system's ability to withstand the FEPs included in the safety evaluation that may affect barrier performance after closure is therefore expected to be good during the periods covered by the assessment of post-closure safety. In the safety evaluation, the effects of possible earthquakes on the barrier system have not been evaluated. This needs to be done in future assessments for SFL and the results may contribute to optimisation of the repository design.

The significance of conditions in the repository surroundings

Most of the results presented in the safety evaluation are based on evaluation cases where the release of radioactivity from the bedrock is assumed to occur in a specific discharge area (biosphere object 206, see Figure 6-1). The *alternative discharge area* evaluation case demonstrates that the calculated annual dose may vary by several orders of magnitude depending on the properties of the discharge area, and that the differences between discharge areas may vary between radionuclides. If groundwater flow from the bedrock increases, the dose for molybdenum-93 and nickel-59 increases, while the opposite is true for carbon-14. There are also systematic differences between discharge areas in Laxemar and Forsmark that depend on lower variations in the topographic relief (and lower groundwater flow from the bedrock) in Forsmark.

6.1.6 Areas where the level of knowledge needs to be improved

On the basis of the evaluation, a need for an improved level of knowledge to support future in-depth assessments of post-closure safety has been identified. This includes needs for research and development aimed at an improved level of knowledge with respect to features, events and processes (FEP) that may affect post-closure safety. Since there are many similarities between SFL and SFR and the Spent Fuel Repository (Section 6.1.1), the needs for an improved level of knowledge for SFL often coincide with needs identified for SFR and/or the Spent Fuel Repository. During the RD&D period, there are plans for research and development identified in the safety evaluation. The efforts primarily concern the following areas:

Long-lived low- and intermediate-level waste

- Better understanding of the inventory. This applies to both predicted waste quantities and the content of radionuclides in the waste, see Section 7.2.1.
- Better understanding of gas-producing processes in the legacy waste and whether gas production may affect the bentonite barrier's capability to retard the transport of radionuclides, see Sections 7.1 and 10.1.2.

Cement barrier evolution over time in BHK

- Continued studies of interactions between groundwater and concrete under repository conditions, see Section 10.1.1.

Bentonite barrier evolution over time in BHA

- Evolution of the bentonite material after installation until saturation, see Section 11.1.
- Bentonite material properties in the saturated state, see Section 11.2.
- Evolution of the bentonite material after water saturation, see Section 11.3.

Needs for development of tools and methods used in the evaluation of post-closure safety have also been identified:

- Given the uncertainties associated with the composition of the legacy waste, the tools for handling of uncertainties in the safety assessments may need to be developed.
- Given an updated description of bentonite barrier evolution over time, developments of the tools for calculating groundwater flow and radionuclide transport in the waste vaults and the surrounding bedrock may be needed.
- Effects of earthquakes have not been taken into account in the safety evaluation. In future assessments for SFL, the barrier system's ability to withstand earthquake related processes needs to be evaluated. The results may contribute to optimisation of the repository design.
- In order to select the best available techniques and for optimisation of the repository design for SFL, it would be advantageous to have increased the functionality of the tools for modelling of groundwater flow and radionuclide transport in the waste vaults and the surrounding bedrock. In this context, methods that allow flexibility in geometries and material properties are of particular interest.

Plans to meet these needs are dependent on the results of planned research and development and will be developed in detail during the RD&D period.

6.2 Siting of SFL

SKB has previously established the basic prerequisites for siting of final repositories for radioactive waste:

- The safety during operation and after closure and the impact on the environment must meet the requirements in the Nuclear Activities Act and the Environmental Code.
- The local political and public opinion support needs to be broad and stable.

These prerequisites also constitute the point of departure for SKB's work with the siting of SFL. Results from studies, investigations and experience from siting of SKB's other final repositories, especially the Spent Fuel Repository, are an important basis for that work.

6.2.1 Requirements in laws and regulations with respect to siting

A basic prerequisite for the siting of SFL is the requirements of the Environmental Code (SFS 1998:808), the Nuclear Activities Act (SFS 1984:3) and the Swedish Radiation Safety Authority's Regulations. These are of a general nature, and thus much depends on how they are interpreted and applied in practice.

The Environmental Code and its requirements on consultations with regulators and other affected parties provide a framework for the process up until the application for permissibility and a licence for construction and operation of SFL. Within this framework, the siting process is governed to a substantial degree by what SKB states in its programme and the comments from regulatory authorities, municipalities and other reviewing bodies.

The requirements regarding siting in the Environmental Code, the Nuclear Activities Act and the Radiation Protection Act as well as the additional guidelines given in SSM's regulations and general advice are summarised below.

The Environmental Code

In the second chapter of the Environmental Code (SFS 1998:808), there is a general requirement for siting of activities, known as the siting principle, which states the following: *Section 6; In the case of activities and measures for whose purposes land or water areas are used, sites shall always be chosen in such a way as to make it possible to achieve their purpose with a minimum of damage or detriment to human health and the environment.*

Chapter 2 in the Environmental Code also contains the rule of reasonableness in Section 7, which states the following: *The rules of consideration laid down in Sections 2 to 6 shall be applicable where compliance cannot be deemed unreasonable. Particular importance shall be attached in this connection to the benefits of protective measures and other precautions in relation to their cost.*

The design of the law can be described as a far-reaching general requirement balanced by a rule that provides an opportunity for assessment of reasonableness on a case-by-case basis. The selected site shall both be suitable in itself (this is a prerequisite for considering the site at all) and cause a minimum of damage or detriment to human health and the environment. The site must thus be suitable for both the purpose of the activities and for the interest of protecting human health and the environment in a general sense in accordance with the Environmental Code's opening paragraph in Chapter 1 Section 1: *The purpose of this Code is to promote sustainable development which will assure a healthy and sound environment for present and future generations. Such development will be based on recognition of the fact that nature is worthy of protection and that our right to modify and exploit nature carries with it a responsibility for wise management of natural resources.*

The Environmental Code also includes Chapter 6 concerning environmental assessments. Chapter 6 Section 35 states that an environmental impact statement in the specific environmental assessment shall present "a description of possible alternative sites and alternative designs for the activity or measure", which includes the issue of alternative sites.

The Environmental Code shall be applied in such a way as to ensure that (Chapter 1 Section 1):

1. Human health and the environment are protected against damage and detriment, whether caused by pollutants or other impacts.
2. Valuable natural and cultural environments are protected and preserved.
3. Biological diversity is preserved.
4. The use of land, water and the physical environment in general is such as to secure a long term good management in ecological, social, cultural and economic terms.
5. Reuse and recycling, as well as other management of materials, raw materials and energy are encouraged with a view to establishing and maintaining natural cycles.

To favour sustainable management of land and water areas, the competent authorities have defined areas of national interest for different purposes. There are provisions regarding the balance between incompatible national interests in Chapter 3 Section 10. There are areas of national interest for final disposal of nuclear waste in Forsmark in Östhammar Municipality and in Simpevarp/Laxemar in Oskarshamn Municipality. The siting of a final repository for nuclear waste is also restricted by the provisions in Chapter 4 Section 3 and 4, which state that new nuclear facilities may not be constructed in certain specified areas.

The Nuclear Activities Act

The Nuclear Activities Act (SFS 1984:3) contains no specific siting provisions but refers to Chapter 2 in the Environmental Code. Siting is, however, treated in the Swedish Radiation Safety Authority's General Advice related to Sections 2 and 3 in SSMFS 2008:21. In the advice, the Swedish Radiation Safety Authority states that: *The repository site and repository depth should be chosen so that the geological formation provides adequately stable and favourable conditions to ensure that the repository barriers perform as intended over a sufficient period of time. The conditions intended primarily concern temperature-related, hydrological, mechanical (for example, rock mechanics and seismology) and chemical (geochemistry, including groundwater chemistry) factors. Furthermore, the repository site should be located at a secure distance from natural resources exploited today or which may be exploited in the future.*

The Radiation Protection Act

The Radiation Protection Act, like the Nuclear Activities Act, contains no specific siting provisions, but the Swedish Radiation Safety Authority has with the support of this Act formulated the following: *Siting, design, construction and operation of the repository and appurtenant system components should be carried out so as to prevent, limit and delay releases from both engineered and geological barriers as far as is reasonably possible. When striking balances between different measures, an overall assessment should be made of their impact on the protective capability of the repository.* (SSMFS 2008:37, General Advice to Sections 4, 8 and 9).

In the same regulations, the Authority stipulates requirements related to human health and the environment, which are of importance for the general evaluations on which siting must be based: *Human health and the environment shall be protected from detrimental effects of ionising radiation during the period of time when the various stages of the final management of spent nuclear fuel and nuclear waste are being implemented as well as in the future. The final management may not cause impacts on human health and the environment outside Sweden's borders that are more severe than those accepted inside Sweden.* (SSMFS 2008:37, Section 3).

With respect to releases of radioactive substances from nuclear facilities in operation, SSMFS 2008:23 applies. For facilities situated within the same geographical area, the following applies: *The effective dose to any individual in the reference group by a yearly discharge of radioactive substances to water and air from all facilities situated within the same geographical area shall not exceed 0.1 millisievert (mSv). The effective dose, by which is meant the dose from external exposure and the committed dose from internal exposure, shall be integrated over a period of 50 years* (SSMFS 2008:23, Section 5).

Requirements on barrier function and post-closure safety

SSM's regulations define the requirement on post-closure safety, known as the risk criterion, which a final repository must fulfil: *A repository for spent nuclear fuel or nuclear waste shall be designed so that the annual risk of harmful effects after closure does not exceed 10^{-6} for a representative individual in the group exposed to the greatest risk.* (SSMFS 2008:37, Section 5).

It can be noted that this risk criterion has been formulated so that it applies for each repository separately in the event that several repositories are located at the same site (see for example SSIFS 1998:1, SSI's comments on the regulations).

Finally, directives on how safety should be achieved by means of barriers and their function are provided. SSMFS 2008:21, Section 2 and 3 states that: *Safety after the closure of a repository shall be maintained through a system of passive barriers. The function of each barrier shall be to, in one or several ways, contribute to the containment and prevention or retention of dispersion of radioactive substances, either directly or indirectly by protecting other barriers in the barrier system.*

In the General Advice to Sections 2 and 3 in the same regulations, SSM states: *The barriers or barrier functions that are necessary in a repository depend on the radioactive inventory of the repository and other substances that affect the safety performance of the barriers and the design and location of the repository.*

6.2.2 Siting factors for SFL

A future application for a licence to construct, own and operate SFL must contain documentation to show that the selected site meets the requirements presented above. The application must also include systematic comparisons with other potential alternative sites with respect to factors that reflect the requirements on siting. The selected site shall offer good prospects for meeting requirements on post-closure safety. Assessing this requires evidence in the form of geoscientific data and other information. Furthermore, the site shall offer good prospects for constructing and operating a final repository so that its establishment has a limited impact on human health and the environment. Robustness and efficiency in execution shall be pursued, and the costs of establishment and operation shall be reasonable. There shall also be good prospects for achieving societal acceptance for a final repository at the selected site as well as at the alternative site. An extensive basis for evaluation and comparison of sites for final disposal has been developed within SKB's programme since it was initiated in the 1970s. A systematic approach based on a division into siting factors has also been formulated. This basis and the systematic approach with siting factors, which SKB now has access to and experience of, can be used to a large extent for the siting of SFL.

SKB has, both in the application for the Spent Fuel Repository (SKB 2010d) and for the extension of SFR (SKB 2013b), chosen to use the following main groups of factors for evaluation and comparison of siting alternatives:

- Post-closure safety.
- Technology for execution.
- Human health and the environment.
- Societal aspects.

This division into main groups is well established with the regulatory authorities and the municipalities, which means that there is every reason to apply it for the siting of SFL. The following is an overall description of factors included in these main groups and how they are applied to SFL.

Post-closure safety

The evaluation of the proposed repository concept for SFL (Section 6.1) shows that the requirements that need to be imposed on the rock as a barrier are similar to the requirements imposed on the Spent Fuel Repository, since both repositories must maintain their protective capability for a long time. Some similarities in requirements that are of particular importance when it comes to the siting of SFL are for example low seismicity, a repository depth sufficient to avoid any impact of future permafrost and low permeability in the repository rock. An essential difference is that the design-basis waste volumes and the chosen repository concept entail that SFL will be a small repository with a footprint of about 200×200 metres, which is approximately one hundredth of the Spent Fuel Repository. Another important difference is that the waste in SFL only generates limited amounts of heat, so the thermal properties of the rock are less important. The data developed with regard to the Swedish bedrock and its relevance for final disposal of spent nuclear fuel are thus applicable for the siting of SFL, while considering the fact that it is a geometrically much smaller final repository.

Based on the evaluation of the proposed repository concept for SFL and experience of safety assessments previously conducted by SKB for final repositories in crystalline bedrock, the following geoscientific factors are considered to be significant for final disposal of waste in SFL:

• **Permeability in the bedrock**

Hydrogeological conditions control groundwater flow in the repository volume. In crystalline bedrock, the frequency and permeability of the water-conducting fractures are the key factors. These parameters affect the inflow of substances that could affect processes in the engineered barriers and the waste, as well as the conditions for transport of solutes in the waste vaults and transport from the repository to the surface, and thereby the time it takes for radioactive substances to be transported via the groundwater. For radionuclides with shorter half-lives that sorb in the bedrock, low permeability together with a low hydraulic gradient may also mean that they will decay partially before they reach the surface. The safety evaluation shows that permeability is of great importance, especially for the outflow of radionuclides from the waste vaults, which means that low permeability is strongly recommended in the part of the bedrock where the repository will be constructed.

- **Hydraulic gradient**

A low hydraulic gradient contributes to low groundwater flow in a repository. The crucial factor for the hydraulic gradient is the terrain relief. The stronger undulations in the terrain, the greater the potential for local circulation cells due to elevated local hydraulic gradients that dominate the flatter regional gradient. For large parts of the country, for example in the mostly low-lying Swedish coastal areas, the differences in hydraulic gradient between different sites are not so large that they affect the groundwater flow more than marginally in comparison. Low hydraulic gradient is considered a preference in siting.

- **Groundwater composition**

The chemical composition of the groundwater may have an effect on processes in the waste, the repository barriers and in the bedrock. The groundwater composition is a factor that needs to be taken into account in the siting process.

- **Reducing conditions**

Reducing conditions in the repository are important for maintaining good sorption in the barriers. This is facilitated if reducing conditions prevail in the bedrock at repository depth. The large amount of iron that will be present in the repository also contributes to reducing conditions. Reducing conditions in the bedrock are regarded as a requirement for the siting. Experience shows that such conditions can be expected at all sites and at the depths considered for SFL.

- **Seismicity**

Possible consequences of earthquakes need to be taken into account in site adaptation and design of a final repository for radioactive waste. Among other things, major deformation zones through the repository should be avoided. Differences in earthquake occurrence in different parts of the country should not entail any large differences between different sites from a risk perspective.

- **Ore potential**

In a safety assessment, the probability and consequences of future inadvertent intrusion into the repository are assessed, as well as other inadvertent events caused by humans that may affect repository post-closure safety. The factors that should be considered for the disposal of radioactive waste are the risk of intrusion into the repository or its vicinity due to prospecting or mining of ore minerals (natural resources) and the risks associated with drilling wells. The lack of ore potential is a requirement in siting.

Since ore potential is often linked to specific geological environments/rock types, a qualified assessment can often be made based on previously known ore potential in the region, in combination with general knowledge of the geological environment at the site, possibly supplemented with specific information from for example geophysical data or drill cores.

- **Risk of drilling wells**

The risk that someone in the future will drill a hole in or near the repository for extraction of groundwater or energy (energy well) cannot be excluded. The safety assessment therefore analyses the consequences if people in the future would drill such a well and use drinking water from the well. The risk of inadvertent well drilling through a repository decreases with repository depth. Since SFL is planned to be constructed at a relatively large depth, the risk is generally low. With respect to siting, a low risk of drilling wells through the repository or so close that it may entail a risk of contaminated well water is desirable.

- **Climate and climate-related processes**

The climate may affect the repository's geological and hydrological surroundings and thereby its barriers in the long term. Among other things, there is a risk of pore water freezing during cold periods with permafrost, which may affect the performance of the engineered barriers and the groundwater flow pattern. Variations in sea level and crustal height also affect the shoreline position, which may be of importance in a coastal siting. Furthermore, the shear stresses that arise in conjunction with the passage of an inland ice front may affect the performance of the engineered barriers.

The consequences of sea level changes, isostatic uplift and development of permafrost and ice sheet in conjunction with future periods of cold climate will be studied in a safety assessment for the selected site. If SFL is constructed at a depth sufficient to avoid adverse effects of freezing, climate variation impact on the siting is expected to be small. The assumed future climate variations will primarily affect the technical design and repository depth.

Technology for execution

The individual siting factors of significance for the technical requirements for establishing and operating SFL are:

- **Construction and operation of hard rock facilities**

Rock conditions at the site shall offer good prospects for constructing and operating the repository's underground openings in a safe and efficient manner. For example, rock with few fractures and high strength is advantageous. Sufficient volumes of suitable rock must be available to, with a good margin, host a repository that meets the capacity requirements after site adaptation.

- **Space for facilities above and below ground**

Above ground there is a need for offices, workshops, terminal building, roadways etc. During the construction period, additional surfaces for handling and temporary storage of muck will be needed. How and where the muck is handled depends on the local conditions. SFL requires no extensive facilities above ground and the need for space is thereby less than for the other SKB repositories. Since the waste vaults in SFL are smaller in extent than both SFR and the Spent Fuel Repository, there is reason to believe that it will be easier to find sufficiently large rock volumes, without any major deformation zones and with suitable properties in other respects.

- **Transportation system**

The transport of waste to SFL will need to be carried out from the nuclear power plants, SKB's facilities, ESS and the Studsvik site. With the exception of ESS, these locations have infrastructure adapted for handling and sea transport of radioactive waste. This includes harbours and other facilities, machines, physical protection and access to nuclear expertise. In the event of siting of SFL at another coastal site, access to a suitable harbour is required and for an inland site a system for land transport also needs to be established. Technically and with respect to safety, such sites are fully possible from a transport point of view, but they require additional investments and the acceptance from different community interests needs to be secured, in particular as regards transport by road. Therefore, it is clear that from the point of view of transport, sites in the vicinity of either the nuclear power plants or the Studsvik site are particularly suitable.

- **Opportunities for coordination with existing or planned activities**

Colocation with existing nuclear activities provides opportunities for coordination. Final disposal of long-lived radioactive waste is a part of SKB's management system and a siting of the repository to one of the sites where SKB's other facilities are located, i.e. Forsmark and Simpevarp, provides especially good potential synergies and opportunities for pooling of personnel in the facilities. In relation to these sites, all other alternative sites entail disadvantages with respect to resource requirements and efficiency in SKB's activities as a whole. The possible disadvantages of colocation with nuclear or other industry must, however, be considered in the assessment. It could concern dependencies on infrastructure or services provided by other parties, risks of mutual disturbances or accidents that may indirectly affect the facility.

- **Costs**

Given that the safety and environmental requirements are fulfilled, the costs of different alternatives are an important parameter for the decisions regarding how and where SFL will be established. Cost estimates for different alternatives need to consider the total system costs for SFL and associated uncertainties in the assessments.

Human health and the environment

The siting of a final repository must take into account interests with respect to land use and different forms of impact on human health and the environment. Protected nature and nature worthy of protection must be considered, such as areas of national interest, nature reserves, Natura 2000 areas, key biotopes, the presence of legally protected species or the like. There are cultural environments in the landscape that are important for preserving historical qualities and develop them in a sustainable way. Particularly important cultural environments are those that are under some form of protection, for example areas of national interest for cultural heritage preservation, listed buildings, landscape protection and ancient remains.

In the application for a licence to construct and operate the facility, the impact on the surroundings in the short and long term must be described in an environmental impact statement. The statement must show that the siting complies with environmental objectives, environmental quality standards and sustainable management of land, material, raw materials and energy.

In general, areas with existing industry are advantageous with respect to human health and the environment since no undisturbed land will be utilised and the immediate surrounding is already affected by for example noise and air releases.

Societal aspects

In order to successfully achieve final disposal of radioactive waste, confidence and acceptance from society at the site and location in question are required, as well as access to the relevant land area. The municipal council in the concerned municipality must give its consent.

6.2.3 Existing siting data

Since the siting factors for SFL are largely the same as for the Spent Fuel Repository, the extensive body of data that has been developed since the late 1970s is useful in assessments of where there is potential for the siting of SFL.

The material consists of data and knowledge from the following categories of investigations and studies:

- The study site investigations that were initiated at the end of the 1970s in the form of measurements in boreholes hundreds of metres deep. The programme was a pioneering effort that during the 1970s and 1980s built up a comprehensive geoscientific database of the conditions at depth in crystalline bedrock at different locations in the country.
- Investigations prior to the siting and construction of hard rock facilities for SKB and other nuclear licensees; Central Interim Storage Facility for Spent Nuclear Fuel (Clab), SFR, the Äspö Hard Rock Laboratory and the interim storage facilities BFA at OKG's site and AM in Studsvik.
- Feasibility studies of the conditions in various respects for siting in a specific municipality. In the 1990s, SKB conducted eight feasibility studies, all in voluntary collaboration with the concerned municipalities.
- General siting studies, reviews of primarily geological conditions at the national or regional level.
- Specific studies, studies of specific questions, for example the possibility of siting in basic rock types, especially gabbro, and analyses of the advantages and disadvantages of siting in the northern or southern part of the country or by the coast or inland.
- Site investigations, studies and geoscientific investigations including borehole investigations at specific sites. Complete and comprehensive site investigations at sites that were selected based on the feasibility studies from the 1990s were carried out by SKB in Östhammar Municipality (Forsmark) and Oskarshamn Municipality (Laxemar/Simpevarp) during the period 2002–2009. In addition, there are results from the investigations that have been carried out for the extension of SFR.

The following is a brief summary of this extensive material with the overall conclusions drawn by SKB regarding the siting potential for the Spent Fuel Repository and to what extent the corresponding conclusions are valid for the siting of SFL.

National and regional general siting studies

SKB has conducted county-specific general siting studies for all counties except Gotland. The studies (SKB 2010d, Section 4.3) focused on the post-closure safety and thereby on conditions in the bedrock, but also included general surveys of environmental factors, existing industry and transport conditions. The main conclusion was that all studied counties contained bedrock that could warrant further studies with respect to the siting of the final repository, see Figure 6-12. At the same time, large areas were identified as potentially unsuitable.

A compilation of the advantages and disadvantages of gabbro as the repository rock type (Ahlbom et al. 1992) showed no significant differences compared with granitic rock as regards conditions of significance for the construction of a final repository or for safety in the long term.

Questions relating to advantages and disadvantages of siting the Spent Fuel Repository in the northern and southern parts of Sweden and by the coast or inland, were analysed in particular (Leijon 1998). The conclusion was that these factors are not of any significant importance. Assessments of the suitability of the site must instead be based on concrete studies of local conditions.

Questions relating to more general differences between coastal and inland alternatives have also been raised in later stages, for example by the former Swedish Nuclear Power Inspectorate (SKI) and the Swedish Radiation Protection Authority (SSI) in conjunction with the selection of sites for site investigations. More specifically, the questions have concerned whether long flow paths (and long circulation times) for groundwater from inland locations may offer advantages from a safety point of view. Furthermore, questions have been raised regarding whether low salinity of groundwater (non-saline groundwater) in inland locations, especially at sites situated above the highest coastline, may offer advantages in the form of eliminated risks of salinities high enough to affect the engineered barriers negatively. This has led to extensive follow-up studies. The studies have been summarised in SKB (2010d, Section 8.2.8):

SKB's overall conclusion is that no systematic difference can be demonstrated between coastal and inland locations with regard to the occurrence of favourable flow conditions. The supplementary analyses presented [...] have not altered this conclusion. The main reason is that investigations and analyses have shown that local conditions, mainly the permeability of the bedrock, are crucial in determining whether a site is suitable for a final repository with respect to groundwater flow. The site investigations in Laxemar and Forsmark have confirmed this contention. Notwithstanding this, the groundwater flow from a repository location may include regional components characterized by long and slow flow paths. However, it is not deemed possible by means of reasonable efforts to verify such conditions with sufficient reliability that they could be credited with any safety-enhancing function for a final repository.

Regarding groundwater salinity, SKB's view is that the salinities found in near-coastal locations, including Laxemar and Forsmark, are not high enough to risk impairing the function of the engineered barriers. The question is rather if the salinity in other geographic locations may be too low with respect to the potential for buffer erosion.

SKB believes that the conclusion regarding long flow paths presented above is valid also when considering the potential of different areas for siting of SFL.

SKB believes that based on the data in the general siting studies, the following conclusions can be drawn regarding the siting potential for SFL:

- There is potentially suitable bedrock in all the counties studied.
- Inland locations have no verifiable general advantages in relation to coastal locations.

Feasibility studies

During the period 1992–2000, SKB was involved in more or less far-reaching discussions of feasibility studies with around twenty municipalities in different parts of the country, see Figure 6-13. In eight cases – Storuman, Malå, Östhammar, Nyköping, Oskarshamn, Tierp, Älvkarleby and Hultsfred – this led to a feasibility study being conducted. In other cases the discussions were concluded, either because SKB found that a feasibility study was not justified, or because the municipality in question chose to decline.

The purpose of the feasibility studies was to determine whether there were prospects for further siting studies for a final repository in the municipality, while the municipality and its inhabitants were given an opportunity to, without commitments, form an idea of the final repository project and a possible further participation. A principal task was to identify areas with bedrock that could be suitable for a final repository. Geological studies were therefore a main component. The studies were based on existing data, and no drilling was carried out. Technical, environmental and societal conditions were also analysed. Within the framework of the feasibility studies, SKB also had an extensive, active dialogue with both the general public and the municipalities and county administrative boards.

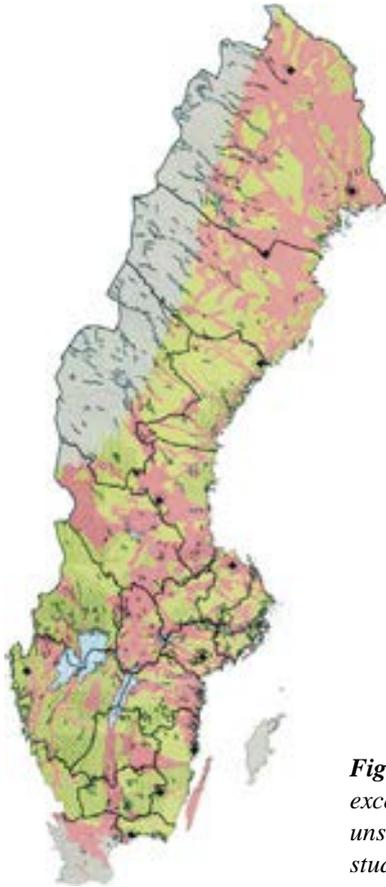


Figure 6-12. SKB's county-specific general siting studies included all counties except Gotland. For each county, a rough subdivision with respect to potentially unsuitable (red) and potentially suitable (green) areas was made for further studies concerning a final repository for spent nuclear fuel (SKB 2000).



Figure 6-13. Municipalities where SKB has carried out or discussed a feasibility study for the siting of the final repository for spent nuclear fuel (SKB 2010d, Figure 4-1).

The feasibility studies (SKB 2010d, Section 4.2) were carried out according to the programme and with the siting factors presented in SKB's supplement to the RD&D Programme 92, which entailed that the following questions were treated in particular:

- What are the general conditions for the siting of a final repository in the municipality?
- Where may there be suitable sites for a Spent Fuel Repository with respect to geoscientific and societal conditions?
- How can transportation be arranged?
- What are the most important environmental and safety issues?
- What could be the consequences, positive or negative, for the environment, economy, tourism, and other industry within the municipality and region?

The procedure was mainly the following:

- The general conditions in the municipality with respect to the specified questions were analysed.
- Areas without sufficiently good prospects for satisfying the requirements on bedrock were ruled out.
- A preliminary priority ranking was made for the remaining areas, based on an overall assessment where also technical and environmental siting conditions were considered. Areas were chosen for geological field checks.
- The results are reported in a preliminary final report, which together with other data from the study was referred to the municipality for comment.
- Geological field checks and other supplementary measures were performed.
- All results were compiled, whereby comments from the referral procedure were considered. Siting alternatives were evaluated and prioritised. The complete feasibility study was published in a final report.

SKB believes that on the basis of the extensive data from the general siting studies in eight municipalities, the following conclusions can be drawn regarding the siting potential for SFL:

- Areas with potentially suitable bedrock were identified in seven of eight municipalities. Without access to information from deep boreholes, there were not enough data for ranking the areas with regard to conditions of importance for post-closure safety.

Although there is now more knowledge of the shallow bedrock than when the feasibility studies were carried out, a corresponding study for SFL in these municipalities would most likely result in the same potentially suitable areas, possibly with some additional areas given SFL's small size, but with the same conclusion that data from deep boreholes is required to rank the areas.

- Interests with respect to land use limited the possibilities of siting to various degrees in the municipalities. For example, in the feasibility study of Östhammar Municipality, nine areas with potentially suitable bedrock were reduced to four when considering areas protected by law, the size of possible investigation areas and the proximity to potential large-scale mining.

In the siting of SFL, interests with respect to land use may limit the areas that are eligible. It could concern protected cultural environments or natural environments. There are often restrictions in coastal areas, for example the coast between Arkösund and Forsmark, where nuclear activities are only allowed where such activities already exist, or where there are certain types of heavy industry.

- Experience from the siting process for the Final Repository for Spent Nuclear Fuel showed difficulties in achieving societal support. Even though many municipalities were preliminarily interested, few gave SKB permission to conduct a feasibility study. Of the eight municipalities where a feasibility study was carried out, two municipalities declined further investigations already after the completed feasibility study. The other six municipalities participated in the process up to the choice of areas for site investigations. Of the four municipalities where SKB proposed site investigations or other continued activities, two declined.

Stable societal support is required in order to carry out site investigations for SFL. The conclusion from siting of the Spent Fuel Repository is that it may take time to achieve acceptance and that it can never be taken for granted for any site.

Study-site investigations

The sites that were the subject of more extensive investigations in the KBS project came to be called study-sites (Figure 6-14). The choice of areas for investigations was based on the extensive exploration and general assessments that had been ongoing since the middle of the 1970s. A total of 85 cored boreholes were drilled with a total length of more than 45 kilometres. The maximum depth that was investigated was about 700 metres. The boreholes were investigated by means of different measurement methods. Special care was taken to determine the location and properties of deformation zones, the permeability of the rock and the chemical composition of the groundwater at great depths. The measurement methodology, as well as the methods for evaluation of data and modelling of groundwater movements etc, was developed progressively. The results of the study-site investigations (SKB 2010d, Section 3.3) showed that it is possible to find many locations in Sweden where the geological conditions are probably suitable for constructing a Spent Fuel Repository.

The siting data for the Spent Fuel Repository includes comparisons with other areas investigated by SKB. These areas are shown in Figure 6-15 (here referred to as reference areas) together with Forsmark and Laxemar that constituted the selection pool for the Spent Fuel Repository.

With respect to SFL, where preferences and requirements on the bedrock are similar to those for the Spent Fuel Repository, the comparative study of the Spent Fuel Repository is relevant. The study is presented in Winberg (2010) and in the report "Site selection – siting of the final repository for spent nuclear fuel" (SKB 2010d). The comparative study considers different geoscientific parameters and conditions. Figure 6-16 shows the cumulative distribution of the permeability (hydraulic conductivity) for different study sites, the Äspö HRL and Laxemar and the tectonic lens in Forsmark.



Figure 6-14. Sites in the country where investigations were carried out during the period from the middle of the 1970s to 1990 (SKB 2010d, Figure 3-2).

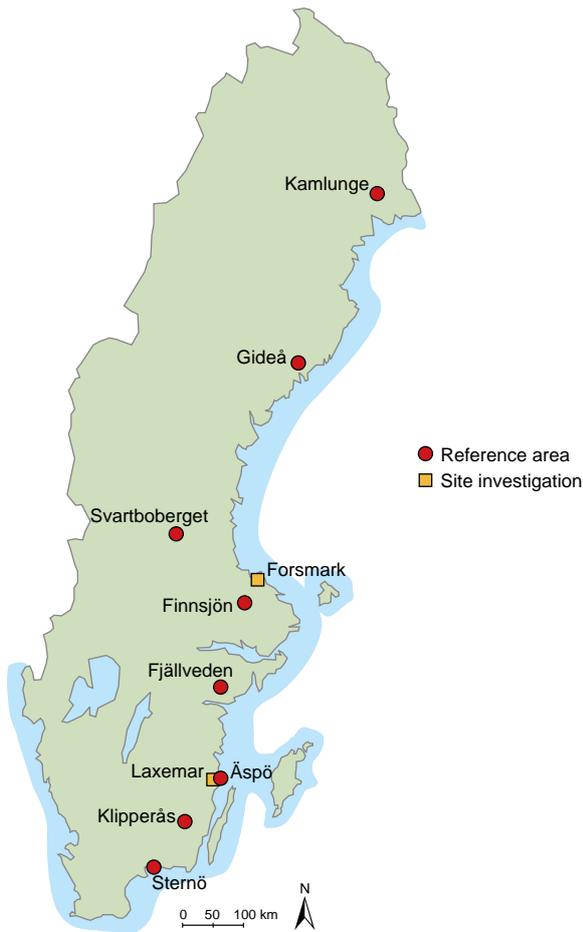


Figure 6-15. Locations for reference areas and sites for site investigations for the Spent Fuel Repository (from SKB 2010d).

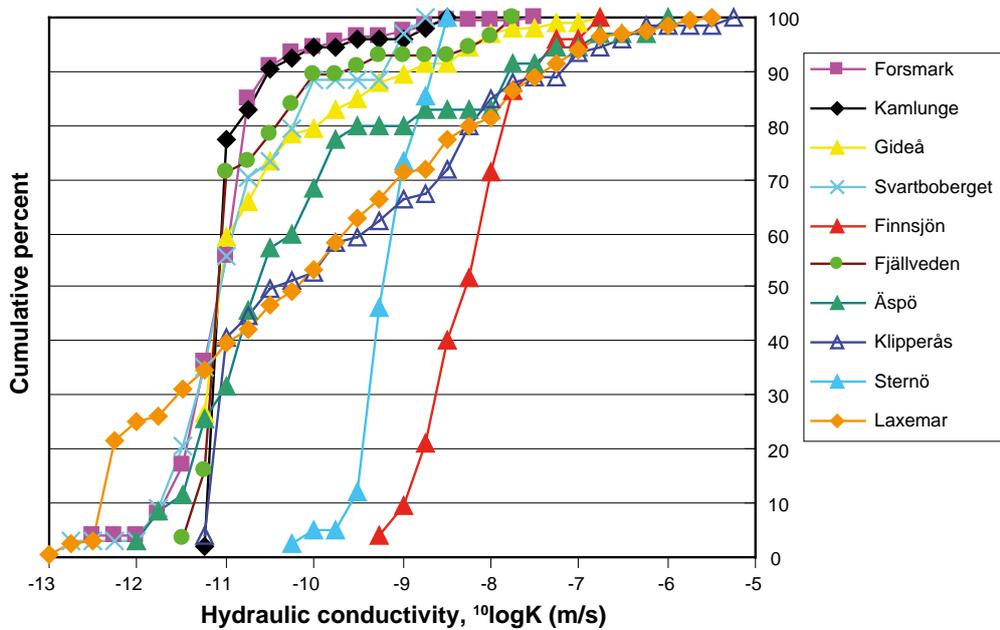


Figure 6-16. Cumulative distribution for measured hydraulic conductivity (K) on a scale of 20–25 metres, for the lens in Forsmark, Laxemar, Äspö and reference areas. The data represent the rock mass between the deterministic deformation zones in the depth range 400 to 700 metres. The cumulative distribution represents the fraction (in percent) of the measurements that are below a given conductivity value. Note that the scale for hydraulic conductivity is logarithmic.

The distribution for Sternö and Finnsjön is displaced towards higher conductivity values, probably because the lower measurement limits in these early measurements were higher compared with other later investigated areas (Winberg 2010). They are therefore not fully comparable to the distributions for other areas.

One conclusion when comparing remaining areas is that the tectonic lens in Forsmark exhibits a distribution with generally low conductivities, and above all a very small fraction of measurement sections with high conductivity. The distribution of Kamlunge is in essential respects close to identical to that of Forsmark, and Gideå, Svartboberget and Fjällveden are also closely connected to the same trend. Klipperås and Laxemar exhibit a completely different type of distribution, with conductivities evenly distributed over a wide range. Äspö, finally, connects to Klipperås and Laxemar with respect to conductivity values larger than about 10^{-9} m/s, but not in other respects.

The comparative study above (Figure 6-16) demonstrates that even if the rock in the volume in Forsmark where the Spent Fuel Repository will be located has low permeability, there are probably also other sites where that particular siting factor is met. The corresponding conclusion should apply for SFL.

Another difference that needs to be taken into account when data from the study sites are used in comparisons with SFL is the footprint required by SFL. With an extension of a hundredth of the Spent Fuel Repository, comparisons between study sites and SFL must consider the possibility that conditions may vary between different rock blocks within the studied rock volume. The required footprint for SFL is not dependent on the thermal conductivity of the bedrock as is the case with the Spent Fuel Repository.

SKB believes that on the basis of the geoscientific data from the study sites, the following conclusion can be drawn regarding the siting potential for SFL:

- There is potentially suitable bedrock for SFL in several of the investigated type areas and probably also at many other sites in Sweden with similar rock conditions.

Site investigations

During the period 2002–2009, SKB conducted site investigations in Forsmark, Östhammar Municipality and Laxemar/Simpevarp, Oskarshamn Municipality (SKB 2010d, Sections 5.3 and 5.4). The overall goal was to prepare applications with a basis for licensing under the Nuclear Activities Act for the Spent Fuel Repository and under the Environmental Code for the KBS-3 system.

The work during the site investigation phase was carried out in project form. The most important activities were to:

- Carry out an extensive investigation programme with investigations on the ground surface and borehole investigations down to a depth of about 1 000 metres.
- Prepare descriptions of the investigated sites, as a basis for site-adapted repository solutions, safety assessments and environmental studies and consequence assessments.
- Design facilities, systems and infrastructure for final repositories at the investigated sites, to a level of detail that provides a basis for the facility descriptions and safety assessments that is to be included in the application.
- Prepare safety analysis reports for the safety of the final repository at the investigated sites during operation of the facility and after closure as well as for transportation.
- Carry out studies as a basis for assessing the impact on the environment, human health and society of planned facilities and activities.
- Carry out required consultations and other communication with the concerned parties and the general public.
- Draw up a programme for the construction phase.
- Prepare the environmental impact statement that is to be included in the application.

In the final stage of the site investigation phase, an overall evaluation (SKB 2010d, Chapter 7) of all data was made in order to be able to:

- Select a site for the Spent Fuel Repository and justify this choice.
- Prepare the licence application.

In the siting of SFL, experience from the site investigations for the Spent Fuel Repository is of great value. A site investigation for SFL can further develop the knowledge that has been obtained and is well documented. This is particularly true for the investigated sites but other sites may also be considered. An important difference is that SFL, as mentioned, is a considerably smaller repository. It is therefore probable that a site investigation for SFL can be carried out in slightly shorter time and require less resources than the Spent Fuel Repository in order to identify a rock volume sufficiently large for SFL with favourable properties.

SKB believes that on the basis of the data from the site investigations in Forsmark and Laxemar/Simpevarp, the following conclusions can be drawn regarding the siting potential for SFL:

- There is a large body of data in the form of surface investigations, borehole investigations and modelling that can be used to assess the potential for siting of SFL in Forsmark and Laxemar/Simpevarp with respect to the siting factor post-closure safety.
- For these sites, there are also data for assessing other siting factors with respect to technology for execution, environment and human health and societal aspects.
- The experience from the site investigations, the safety assessment and the extensive documentation required for licensing under the Nuclear Activities Act for the Spent Fuel Repository and under the Environmental Code for the KBS-3 system can be used for siting of SFL, regardless of which site is chosen.

6.2.4 Principal siting alternatives

Experience shows that there may be potentially suitable bedrock in many places and that stable societal support is required for siting, which makes SKB believe that the siting process for SFL can start from an assessment of the reasonableness of several principal alternatives, which may all be expected to meet the requirements on post-closure safety.

The principal alternatives according to SKB are:

- Colocation with existing or planned final repositories for nuclear waste (SFR and the Spent Fuel Repository).
- Siting at nuclear facilities.
- Siting at other industrial facilities/industrial areas.
- Siting in areas where there are no other industrial activities.

Table 6-2 gives a brief description of the potential advantages and disadvantages of these alternatives. The description is not complete, but includes the most significant differences between the alternatives with respect to siting.

Figure 6-17 schematically shows the constituent parts in the chain of management of the principal alternatives described above. An establishment in Forsmark or Simpevarp requires sea transport from other nuclear facilities and short transports by land within or in the vicinity of the nuclear industrial area. This applies even if another nuclear facility would be relevant. Colocation with another final repository in Forsmark will probably not require a special operations area for SFL. An alternative is to establish a separate operations area for SFL in Forsmark, which will also be the case if SFL is to be located in Simpevarp or at another nuclear facility. Transport down to the repository is carried out by means of a ramp or shafts. The alternative of an existing industrial area will also require transfer in a harbour and transport by public road or railroad to a new established nuclear operations area for SFL. Areas where there are no industrial activities also require establishment of the industrial area with associated infrastructure, including construction of roads. For this alternative new land needs to be used for roads and industry purposes.

Table 6-2. Potential advantages and disadvantages of different alternatives for siting SFL.

	Potential advantages	Potential disadvantages
Colocation with existing or planned final repositories for nuclear waste	Infrastructure is in place in the form of transportation system and handling of nuclear waste. No new land areas need to be used. The environmental impact is small, including a possible reduction of rock excavation. The alternative permits sharing of personnel and may reduce the time required for construction. The costs are probably lower than for other alternatives, among other things because the site investigation may build on existing knowledge of the area. Local societal support is more probable than for alternatives where there is no experience of nuclear activities. Forsmark is designated as an area of national interest for final disposal of nuclear waste.	Possible mutual effects on and from the other repository may be negative and possibly so great that siting is ruled out. Colocation means that the repositories are dependent on each other, which, among other things, may entail delays. Cumulative effects need to be considered. Colocation may pose legal challenges, as altered activities at the site may mean that the conditions for existing activities are taken into review again in connection with licensing of the new activities.
Siting at nuclear facilities	Infrastructure is in place and can be shared. In an already industrialised area, the environmental impact may be limited. The alternative should provide opportunities for pooling of personnel/competence. With establishment in or in the vicinity of a nuclear industrial area, the costs should be reasonable, but probably somewhat higher than for colocation. With infrastructure in place, construction time will be reduced. Local societal support is more probable than in areas where there is no experience of nuclear activities. Forsmark and Laxemar/Simpevarp are designated as areas of national interest for final disposal of nuclear waste.	New land within or close to the industrial area may need to be used.
Siting at other industrial facilities/industrial areas	Fundamental infrastructure is available (roads, power, water supply and sewerage, etc). The environmental impact may be limited (already industrialised area). The alternative entails limited but probably somewhat higher costs than for colocation with another repository or siting at a nuclear facility. Some pooling of personnel may be expected. The possibility of local societal support is difficult to assess, but probably possible.	A new nuclear operations area needs to be established with associated installations and protection. New land within or close to the industrial area may need to be used. A transportation system for radioactive waste may need to be established. The time required for establishment is probably greater than for siting at a nuclear facility, mainly for alternatives where the industrial area is located far from a harbour, which means that a transportation system for radioactive waste needs to be established.
Siting in areas where there are no other industrial activities	An opportunity for better utilisation of favourable rock conditions. It is difficult to see potential advantages with respect to technical, environmental and societal factors. An establishment could possibly be regarded as positive in the sense that it creates work opportunities.	A new nuclear operations area needs to be established with associated installations and protection. Infrastructure needs to be established for the new industrial area (new roads, power supply, water supply and sewerage, etc) with subsequent environmental consequences. A new area needs to be used. The environmental impact of construction may be greater since it is now carried out in an "undisturbed" environment compared with the establishment within existing industrial areas. The establishment in a new area leads to increased costs and more time required compared with other alternatives. No pooling of personnel can be expected. A new transportation system for radioactive waste needs to be established that includes roads/railroad and transfer stations for inland sites or harbours for coastal sites. The possibility of obtaining societal support for such an establishment is more uncertain than for other alternatives.

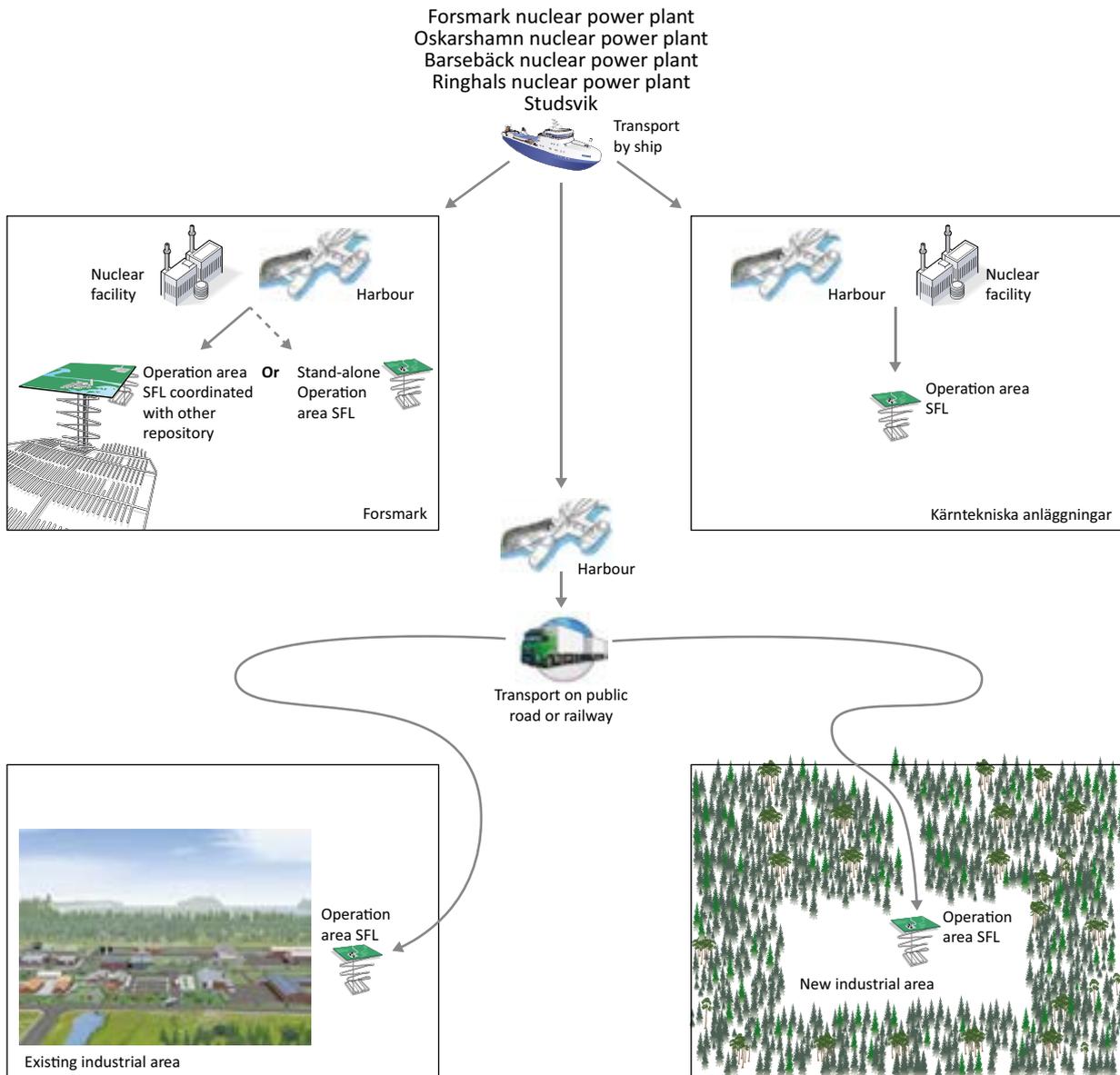


Figure 6-17. Means of transportation and facilities for handling of waste in SFL for different principal alternatives. The images show possible locations for SFL's operations area above ground. The underground part (the repository) may be placed below or laterally displaced in relation to the operations area. In the case of railroad transport, a transfer station for final transport by road will probably be required. The waste from ESS is transported by public road or railroad either to a harbour for further transport by ship, or directly to SFL.

6.2.5 SKB's conclusions and proposed siting process

Overall conclusions

The siting factors for SFL presented above are essentially similar to those applied in the siting process for the Spent Fuel Repository. Both repositories manage long-lived waste, which must be disposed of for a very long time so that it does not endanger human health and the environment. The repositories also need to be located so that inadvertent intrusion is impeded.

Even though the siting factors related to post-closure safety are essentially the same, the site investigations for SFL will presumably include smaller areas given SFL's small footprint. The need for information from borehole investigations in order to evaluate the siting factors related to post-closure safety is, however, similar since SFL needs to be located at great depths.

This similarity between siting factors for SFL and the Spent Fuel Repository also applies to other groups of siting factors related to human health and the environment, technology for execution and societal aspects.

There are differences that may be significant when the suitability of an area for siting is evaluated from an environmental, technical and societal perspective. A small SFL facility should entail less impact on the environment compared with a spent fuel repository at the same site. Since thermal properties and conditions are of no substantial importance for SFL, this factor is not significant for siting, which may entail that even bedrock with low heat conduction could be of interest. The need for personnel and other efforts during the operating period differs for a spent fuel repository that is built out continuously, while SFL is built completely during the construction phase. The estimated operating time for SFL is about ten years compared with about 40 years for the Spent Fuel Repository.

In summary, experience and data from the study site investigations, general siting studies, feasibility studies and the site investigations from the siting of the Spent Fuel Repository lead to the following fundamental conclusions:

- That there are most likely many areas with bedrock that may be considered potentially suitable for SFL and that ranking these with respect to essential conditions of importance for post-closure safety cannot be done without information from site-specific geoscientific investigations including measurements in deep boreholes.
- That stable societal support is required to carry out a site investigation and for obtaining a licence for construction of a final repository.

SKB believes that eligible sites must be considered to have good geological conditions to meet the requirements on post-closure safety based on available geological information. In such an assessment, the extensive documentation produced for the siting of the Spent Fuel Repository can be used. Provided that such conditions are available, and since it requires extensive site investigations in order to determine whether a site meets the requirements, SKB believes that the siting process should primarily be based on factors that are possible to assess in an early stage of the siting process. Hence, sites with good geological conditions should initially be evaluated on the basis of initial assessments of the impact on human health and the environment, available infrastructure, use of land and the possibility of obtaining societal support.

Siting process for SFL

The RD&D Programme 2016 described the goal of conducting an open and transparent process in consultation with SSM and the concerned municipalities, where the conditions for different actors are clarified early on and where the different steps in the process have been agreed upon and communicated. SKB wants to emphasise that the voluntary participation of the concerned municipalities in all steps in the siting process is a fundamental principle.

Based on the fundamental conditions, prerequisites and principal alternatives presented above, SKB's conclusion is that the efforts to identify potentially suitable areas for SFL, with a focus on post-closure safety combined with the possibility to obtain the required licences, should primarily include areas where there are nuclear activities, secondarily areas where there are established industrial areas. Special emphasis should also be placed on the fact that Forsmark and Laxemar/Simpevarp have been designated as areas of national interest for final disposal and thereby have a special status when it comes to sustainable management of land. Other areas where there are no industrial activities may be considered, but then only if special reasons to include them emerge in the siting process, or if the initial work shows that the selection pool needs to be expanded for other reasons.

Areas where there is extensive data from previous investigations at great depths that may be used in a preliminary assessment of the prospects for the area to satisfy the safety-related siting factors should be included in the selection pool. SKB therefore believes that Forsmark and Simpevarp should be included in the selection pool for the siting of SFL. Here, there is also data for assessing the conditions for compliance with other siting factors.

SKB considers the above account of background data and fundamental conditions and points of departure to be the first step in the stepwise siting process for SFL. The referral procedure for the RD&D Programme 2019 provides those concerned with an opportunity to offer comments on SKB's presented points of departure, the proposed process and the proposed siting factors. SKB intends to continue the stepwise siting process mainly according to the steps described below with any adjustments that may be required in the RD&D process.

1. Initial feasibility study phase

In the initial phase of the siting process, SKB needs to demonstrate that areas in the selection pool have good prospects of satisfying the safety requirements. A comparative study will therefore be conducted with other areas where there are relevant data from great depths. If the comparative study shows that there are areas with obvious advantages with respect to post-closure safety, these may be included in the selection pool provided that it is feasible and reasonable when other non-safety-related siting factors are considered.

The comparative study will initially pertain to safety-related siting factors. For areas where there is potential for compliance with these, other siting factors regarding technology, society and the environment will also be evaluated. The comparative study will also consider areas where there are no data from deep boreholes, but that have been mentioned by different stakeholders as potentially suitable for a final repository. Examples of such areas are Studsvik and certain areas in and near Hultsfred Municipality. Consultations in the siting process may yield more areas that could be included in the basis for comparison.

In this phase, a systematic review is made of the Forsmark and Simpevarp areas that constitute the selection pool and the sites included in the basis for comparison with respect to the siting factors safety, technology, the environment and society. The purpose is to as far as the data permits provide answers to the following questions:

- Can the sites in the selection pool be considered to have good prospects for fulfilling all siting requirements? If this is the case, which supplementary investigations and studies are needed to select one of the sites and an alternative and to prepare a licence application?
- Is there anything in the basis for comparison to give SKB reason to try expanding the selection pool with a new municipality/site to be included in the continued siting work? If this is the case, which supplementary investigations and activities are needed to include this site in the selection pool?

The work according to the above is carried out within the framework of the Environmental Impact Assessment (EIA) consultations with the concerned municipalities and regulatory authorities. The phase is concluded with an evaluation and assessment prior to the selection of areas for site investigations, which is presented and discussed in a consultation on scope. This phase will be initiated during the RD&D period.

2. Site investigation phase and site selection

Investigations, studies and consultations to provide data for selecting a site. After completed site investigations and safety evaluation, a site is selected considering all siting factors.

3. Licence application

For the main alternative, all documentation required for an application is produced. For the alternative site/sites, documentation is produced to a level of detail that makes it possible to review on equal grounds as SKB's main alternative. According to current planning, SKB will submit the applications around 2030.

7 Low- and intermediate-level waste

Low- and intermediate-level waste consists of operational 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 SFR, while long-lived waste will be disposed of in SFL. In accordance with the current operational -permit, only operational waste may be disposed of in SFR, but in the extended SFR facility it will also be possible to dispose of short-lived decommissioning waste.

Short-lived waste may also be disposed of in near-surface repositories at the nuclear facilities, provided that the activity level in the waste is sufficiently low, which has been described in Section 3.2. However, this chapter only 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.

7.1 Process understanding

SKB's work for improved process understanding with respect to low- and intermediate-level waste is focused on two main areas: impact on sorption and gas build-up/swelling.

In SFR, sorption of radionuclides on cement minerals is one of the most important processes to retard the release of radionuclides from the repository. The occurrence of groups of substances that may affect the sorption of radionuclides is minimised by requirements in the acceptance criteria for the waste. There are, however, 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 handled in SKB's post-closure safety assessment.

Gas production in the repository environment is mainly dependent on the degradation of the material by for example corrosion or microbial activity. If gas production is so extensive that the produced gas cannot be transported out in a controlled manner, the barriers in the final repository may be damaged. It is therefore important to understand the degradation processes leading to gas production in order to assign realistic parameters in the post-closure safety assessment.

Like gas production, swelling waste may also affect the integrity of concrete barriers and thereby affect the outward transport of radionuclides. It is therefore important to quantify the magnitude of swelling pressures that may arise.

7.1.1 Impact on sorption from cellulose degradation products

There is organic material in different waste types in SFR. A large portion of the organic material consists of cellulose which, under alkaline conditions, may be degraded to isosaccharinic acid (ISA) (Glaus et al. 1999). The degradation mechanism is well studied (Glaus and Van Loon 2008) and SKB has used the rate derived in Glaus and Van Loon (2008) to estimate the concentration of ISA in SFR (Keith-Roach et al. 2014).

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 SKB's post-closure safety assessment, sorption of plutonium is assumed to be reduced by a factor (sorption reduction factor) that is dependent on the ISA concentration. Previously, there has been limited knowledge of how the presence of ISA in different concentrations affects the ability of plutonium to sorb to cement minerals. There has therefore been some uncertainty as to which sorption reduction factor should be attributed to plutonium in the presence of ISA. In SR-PSU, SKB therefore chose a pessimistic sorption reduction factor for plutonium(III)/plutonium(IV).

Current situation

The study initiated by SKB in 2014 has been concluded and reported in the form of a PhD thesis (Tasi 2018). The study followed a four stage methodology where:

1. The redox chemistry of plutonium under alkaline conditions (pH 12.5) was studied in the absence of ISA and calcium(II).
2. The solubility of plutonium dioxide was studied in the presence of ISA but in the absence of calcium(II).
3. The solubility of plutonium dioxide was studied in the presence of ISA and calcium(II). The main purpose of this study was to provide a complete chemical and thermodynamic model for the calcium(II)-plutonium(III/IV)-OH-ISA system.
4. The sorption of plutonium and plutonium-ISA on cement was studied in a final sub-study.

A model for how the sorption of plutonium on cement is affected by ISA has been developed. Sorption experiments with the plutonium-ISA-cement system show that the sorption of plutonium is affected by ISA, and sorption reduction factors have been possible to derive for different concentrations of ISA. If plutonium and ISA are allowed to form complexes before the sorption on cement occurs, the sorption of plutonium is affected to a greater extent than if the sorption of plutonium on cement takes place before ISA is added to the system. This is important to know since most of the cellulose in the repository can be found in waste where the concentration of plutonium is low. The formed degradation products (ISA) of cellulose must then diffuse to waste packages where the concentration of plutonium is higher. This increases the probability that the sorption of plutonium will take place before the plutonium-ISA complexes can form.

The results of the study have also led to the possibility of applying a less pessimistic sorption reduction factor for plutonium in future safety assessments.

Programme

SKB is currently not planning to continue studying the plutonium-ISA system but monitors the level of knowledge and will adjust both sorption factors and sorption reduction factors if new knowledge emerges showing that such an adjustment is needed.

7.1.2 Impact on sorption from filter aid degradation products

Current situation

An example of a frequently occurring filter aid in the operational waste from the nuclear power plants is UP2. This filter aid consists of polyacrylonitrile (PAN). Under alkaline to hyperalkaline conditions, the polymer structure breaks down and forms highly soluble compounds, which may affect sorption and/or the solubility of radionuclides (Duro et al. 2012). In the most recently completed study, the impact on nickel(II) and europium(III) under repository-like conditions was investigated (Tasdigh 2015). The study showed that degradation products of the filter aid affected the sorption of europium(III), and the experiments with nickel(II) indicated that the solubility of nickel(II) increased in the presence of degradation products. Previous studies have shown that the degradation products of PAN do not affect the sorption of europium(III) to any significant extent (Dario et al. 2004).

Programme

Further studies of filter aids are planned within the framework of an EU-funded project named Cori (Cement Organics Radionuclide Interaction). SKB is participating in the preparations prior to the project application. Since the degradation of PAN is expected to be slow, degradation experiments have already been initiated. In Tasdigh (2015), an impact on pH was noted during the degradation of PAN. Such a potential impact will be further investigated to, if possible, determine if the degradation products may affect the pH in waste packages containing PAN. Cori is expected to start in 2019 and continue for four years. SKB plans to study whether degradation products of PAN affects the sorption of nickel(II), europium(III) and actinides(IV). As mentioned above, the sorption of europium(III) in the presence of degradation products of PAN has already been studied. Europium(III) is included in the planned work as a reference.

7.1.3 Sorption impact of cement additive degradation products

In both the extended SFR facility and the future SFL, large amounts of concrete will be used as an engineering material. In order to obtain a high-quality concrete with suitable properties, concrete additives such as superplasticisers need to be used. Such substances consist of, for example, organic polymers. They may degrade under the conditions -prevalent in the repository after closure (high pH, low Eh), and form highly soluble organic compounds. The effect of these organic compounds on the sorption and/or the solubility of radionuclides is dependent on the type of compound. As the composition and structure of superplasticisers differ, no general requirements can be imposed on superplasticiser content.

Before the start of construction of the extended SFR facility, it is therefore important that SKB has acquired an understanding of the effect of these organic polymers so that SKB can stipulate requirements for which additives the contractors are allowed to use in the concrete in order to avoid any impact on the safety functions which apply for the repository after closure.

Current situation

The studies initiated by SKB in 2016 have been concluded. The results showed that the structure of the studied superplasticiser included a fraction of organophosphate. The rate constant for the degradation and release of phosphate was determined. The studied superplasticiser was found to increase the solubility of both nickel(II) and europium(III) through the interaction of these elements with the split off phosphate. Superplasticisers containing organophosphate esters may therefore be assumed to increase the solubility of nickel(II) and europium(III).

The content of concrete additives is not generally known since the companies that manufacture them do not openly report the composition. This limits the possibility for SKB to study the content of possible additives in advance. SKB's study showed that the superplasticisers were hydrolysed relatively quickly in the experimental environment. This may be taken as an indication that this could occur in SFR and SFL as well, if the environment is equivalent. The solubility of nickel(II) and nickel(III) could then be affected if they are present in sufficiently high concentrations of degradation products in the pore water.

Studies completed on behalf of the British Nuclear Decommissioning Authority (NDA 2015) show, however, that superplasticisers do not affect the sorption of radionuclides. NDA's experimental set-up differs from the set-up that is traditionally used when the impact on sorption is studied. In the NDA study, radionuclides were cast together with concrete containing different types of superplasticisers. The results of these tests showed that radionuclide mobility in the concrete was not affected by superplasticisers in the concrete. 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 were conducted under more repository-like conditions than traditional experimental set-ups and therefore provide valuable information on how the real system behaves.

Programme

SKB is not presently planning to continue studying the impact of superplasticisers on the sorption of radionuclides. SKB will, however, continue to monitor the level of knowledge. In good time before the start of construction, it is important that SKB gets the opportunity to assess the superplasticiser that will be used by the contractor, so as to decide on its suitability.

7.1.4 Gas build-up from corrosion of aluminium and zinc

Anaerobic corrosion of aluminium and zinc in the waste deposited in SFR may in a short time generate large quantities of hydrogen gas. Provided that the gas cannot be transported out of the concrete structures enclosing the waste in a controlled manner, a damaging over-pressure for the concrete structures may arise. It is therefore important to understand the corrosion process and to quantify the corrosion rate for aluminium and zinc in the 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 groundwater. A series of experiments with different exposure times (2, 4, 12, 52 and 104 weeks) have been carried out.

Preliminary results from the experiments show that corrosion is initially rapid for both aluminium and zinc, around 50 µm/year for zinc and 1 mm/year for aluminium, but that the corrosion rate then decreases. The corrosion rate for zinc decreases to a very low level, approximately 1 µm/year after 104 weeks, while the estimate for aluminium is approximately 100 µm/year after 104 weeks. In the case of aluminium, a void is also formed around the metal sample in conjunction with the initial corrosion, which is preliminarily attributed to hydrogen evolution.

Programme

After completing the study together with KTH, SKB is not planning any further studies of anaerobic corrosion of aluminium and zinc. The completed experiments and the studies that are published (for example Fujiwara et al. 2017) provide adequate information to describe the corrosion of aluminium and zinc in the post-closure safety assessment. SKB will, however, continue to monitor the level of knowledge.

7.1.5 Microbial gas production

Another process that may potentially cause -damaging gas build-up in a final repository is methanogenesis by microbial degradation of organic materials, such as cellulose. It is therefore important to understand which conditions to strive for in SFR in order to limit gas production due to microbial degradation.

Current situation

In the RD&D Programme 2016, SKB reported that studies of microbial gas production would be conducted within the EU project Mind. The research within Mind has, however, mainly focused on other microbial processes than gas production. The issue is addressed to some extent in Small et al. (2017), which has studied gas production (mostly methane) in decomposition of low-level waste containing cellulose. The results show a doubling of the gas production rate when the pH in the surrounding water drops below 9.

SKB has also carried out studies of microbial production of methane. This work studied the presence of methanogens in groundwater from the Forsmark area. The results show that there are methanogens in the groundwater and that the species that occur may form methane from degradation of organic materials, representative for low- and intermediate-level waste. The results also show that these methanogens are only active up to a pH of just above 10 (Taborowski and Pedersen 2018).

Programme

SKB is not presently planning to continue studying microbial gas production in low- and intermediate-level waste. SKB will, however, continue to monitor the level of knowledge.

7.1.6 Swelling of ion exchange resins

Swelling waste may affect the integrity of the concrete barriers and thus affect the transport of radionuclides from SFR. It is therefore important to know if the waste may swell and if so, how large the swelling pressure of the waste may be.

Current situation

SKB has conducted experiments at the Äspö Hard Rock Laboratory where swelling of a bitumen-solidified ion exchange resin has been studied. The leaching rate of anions and cations from a bituminised waste form has also been investigated. The results showed that the bitumen matrix is not sufficiently tight to prevent the ion exchange resin from swelling.

Further experiments on unconditioned ion exchange resin have since been carried out to quantify the swelling pressure when the ion exchange resin is resaturated with water. In the experiments, the ion exchange resin expansion during water uptake was measured under different back pressures in order to establish an empirical relationship between expansion and swelling pressure. The results of these experiments are presented in Jensen et al. (2018).

The results for the unconditioned ion exchange resin are, however, not entirely representative of the actual waste form. Experiments with conditioned ion exchange resins are impeded by the long period of time for resaturation of bitumen-solidified ion exchange resins under back pressure.

Programme

SKB plans to conduct further experiments to measure swelling pressures that are representative for the conditioned waste under repository conditions, but where resaturation is fast enough to obtain reliable results in a reasonable time.

7.2 Radionuclide inventory

SKB continuously improves the methods for estimation and prognosis of the radionuclide inventory in low and intermediate level waste. As the level of knowledge is improved, 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 refer to 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 only exists as indirect measurements or as analyses of individual samples, it is necessary to apply -computational models for estimating the radionuclide inventory of difficult-to-measure nuclides in the low- and intermediate-level waste. For the calculated inventory to be reliable the computational models must, as far as possible, be based on established physics and the assumptions and parameters in the models must be possible to verify with measured values. Physically plausible models are also a prerequisite for extrapolating data for difficult-to-measure nuclides to 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 both extended measurement of difficult-to-measure nuclides and further development of the computational 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 extended measurements and collection of process data from the waste producers.

7.2.1 Reference inventory

Current situation

An updated reference inventory for waste in the extended SFR facility has been prepared as a basis for the next post-closure safety assessment which will be included in the PSAR. The inventory includes all disposed waste up to and including 31 December 2016 and forecast operational and decommissioning waste. The methodology for the inventory compilation has been updated with estimates of the annual production of difficult-to-measure nuclides and an updated uncertainty analysis for the radionuclide inventory.

The methodology changes are mainly aimed at replacing general assumptions of the activity in the forecast waste with data based on a more detailed statistical analysis of the reported activity for disposed waste.

SKB has also initiated a dialogue to coordinate the work of radiological survey and characterisation of decommissioning waste for the licensees that are about to initiate dismantling and demolition of nuclear power reactors.

In preparation for future updates of the reference inventory for SFL, SKB's records of interim-stored control rods in Clab have been updated with information on operating history and boron burnup, which makes it possible to improve the methodology for determining the activity of long-lived waste from the nuclear power reactors. For long-lived legacy waste, a considerable uncertainty remains with respect to the content of materials and radionuclides in the waste.

Programme

SKB will develop systems for information management in order to be able to receive information from the radiological survey prior to dismantling and decommissioning. The purpose is to continuously be able to update the decommissioning waste inventory as the licensees are conducting investigations, calculations or measurement programmes aimed at characterising the decommissioning waste. This work will serve as the basis for a future waste registry for decommissioning waste.

SKB will continue to monitor the results of sampling and sample analyses from segmentation of reactor internals and segmentation of reactor pressure vessels in order to be able to update the estimate of induced activity in the low- and intermediate-level waste. Furthermore, SKB will monitor AB SVAFO's plans for management and characterisation of the legacy waste.

The work of improving the model for calculation of induced activity in scrapped control rods will continue during the RD&D period. Using boron burnup data, the model will be able to provide a more detailed picture of the irradiation history of the control rods and thus provide a better description of the radionuclide inventory in the rock vault for core components in SFL. The calculations will also be used as support for radiological assessments regarding the continued handling of the control rods up until disposal in SFL.

SKB will monitor the planning of the spallation facility ESS to be able to assess the possibility of disposing waste from operation and decommissioning of the facility.

7.2.2 Method development for difficult-to-measure nuclides

Current situation

The inventory of difficult-to-measure nuclides whose activity is determined by measurement of process water could be underestimated if the regular analyses of these nuclides are not supplemented with water analyses from completed system decontaminations. SKB has therefore calculated and supplemented activity estimations for completed system decontaminations of primary reactor systems, where the information on difficult-to-measure nuclides was inadequate. In the calculations, released radioactivity has been calculated based on available measuring data, the estimated quantity of dissolved uranium in the reactor and vectors for fuel and activated corrosion products. The calculated amount has been compared with measurement data in applicable cases and the comparison shows that the calculations generally provide a pessimistic estimation of the activity of difficult-to-measure nuclides.

SKB has also supplemented the reported quantities of transuranic elements with calculated values for the nuclides that are not covered by regular analyses. For this, vectors from each reactor's calculated equilibrium core have been applied to the measured quantities of plutonium-239/240.

SKB has for some time calculated the amount of the difficult-to-measure nuclides molybdenum-93, technetium-99, iodine-129 and caesium-135 in operational waste by means of a computational model (Lundgren 2005, 2006). In 2016, the model was updated after a number of improvement measures were identified. Among other things, additional series of measured data were added in order to better assess the origin of the release of radioactivity and hence also the choice of parameter values. Furthermore, the concentration in process water, which was previously at the reactor type level, was replaced with

reactor-specific values. It is now also possible to vary the age and burnup of the failed fuel. During the work of developing the model, additional improvement needs have been identified. This work is ongoing and needs to be completed before the updated model can be implemented and fully replace the original.

To verify the computational model for difficult-to-measure nuclides, a measurement programme has been carried out in two stages. In the first stage, reactor water samples from the reactors Oskarshamn 1 and 3 were collected during 2014. The samples were analysed with ICP-MS (inductively coupled plasma mass spectrometry) to determine the concentration of the nuclides technetium-99 and iodine-129 (Åkerblom 2015). In stage two, the analyses included reactor water samples from Forsmark 1, 2 and 3, Oskarshamn 1 and 3, Ringhals 1 and 3, and pool water samples from Clab. The samples were collected in 2016 and analysed with ICP-MS to determine the amount of technetium-99 and iodine-129 (Ahlford 2019). The results of these measurements will constitute an important basis for the future development of the computational model for difficult-to-measure nuclides. Verifying measurements of other difficult-to-measure nuclides whose activity is determined with the computational model are also desirable, which is why the design of additional measurement programmes is under way.

Programme

In order to obtain a more reliable calculation model for the difficult-to-measure nuclides molybdenum-93, technetium-99, iodine-129 and caesium-135 in operational waste, the input source terms need to be reviewed and updated. This means that the underlying crud and fuel failure models need to be revised and improved. It is mainly the crud model for boiling water reactors (BWR) that needs to be updated with regard to the activation of molybdenum. Measurements of elemental molybdenum suggest that previous assumptions regarding how molybdenum is deposited and activated on the fuel need to be revised as there are now more data for quantifying the corrosion of activated molybdenum alloy materials in the core.

The goal is to be able to use the model for difficult-to-measure nuclides for calculating the quantity of all difficult-to-measure nuclides. Thus, measured values for nuclides where the activity determination methodology is already established (such as regular measurement of a number of transuranic elements) can be used to verify the model while the model can be used to supplement the estimated radioactivity for periods where there are no measurement data.

To facilitate further development of computational models for both development of source terms and activity determination of difficult-to-measure nuclides, SKB has identified a need to collect available operational and measurement data for all Swedish reactors since the start of operation in a database. SKB has initiated this work and the database will be updated and supplemented as new data is added.

Together with the licensees, SKB is planning further measurements of a number of difficult-to-measure nuclides, such as molybdenum-93. In order to improve the possibility of detecting these nuclides, integrated sampling will be applied for concentration of the samples. Gamma measurement and elemental analysis are carried out in connection with the sampling, in order to relate the amount of difficult-to-measure nuclides to relevant easy-to-measure nuclides and the metal content in the samples and to be able to say something about the uptake of the different filter fractions (anion filters, cation filters and particle filters). Sample collection and measurement is carried out in 2019 and 2020.

To estimate how measured and calculated amounts of activity in waste producer systems are distributed in the different parts of SFR, activity amounts need to be connected to produced waste packages. In order to improve the reliability of how the activity is distributed per package, SKB intends to, to a greater extent, gather and use information that improves the possibility of tracing waste packages to the system in which the waste has been generated.

7.2.3 Uncertainties in the radionuclide inventory

Current situation

SKB has further developed the methodology for estimating the uncertainty in the inventory in SFR. Through statistical analysis of measured and calculated activity for disposed waste, distributions of the uncertainty in the activity of disposed waste and distributions of activity in the forecast waste have

been defined. Based on these distributions, the uncertainty in the radionuclide inventory for disposed and forecast waste can be estimated through Monte Carlo simulation, where the uncertainties of different measurement and calculation methods are combined.

Programme

SKB intends to, to a larger extent than before, gather measurement data and operational data from the waste producers. Thus a more extensive body of data is obtained to parametrise and verify the models described in Section 7.2.2. Together with the extended verifying measurements that are planned for difficult-to-measure nuclides, this will lead to better data for uncertainty analysis. SKB also intends to, within the framework of the method development described in Section 7.2.2, improve the quantification of the uncertainty in measured and calculated activity.

7.3 Acceptance criteria for waste in SFL and the extended SFR facility

Acceptance criteria for waste are specified in the safety analysis report for each final repository. For facilities that have not yet been taken into operation, acceptance criteria will be established in conjunction with the submission of the SAR.

At present, however, there is a considerable amount of long-lived waste in the waste producers' interim storage facilities and additional long-lived waste will arise during the continued operation and decommissioning of the nuclear facilities. In order to avoid that future transportation and final disposal is impeded, it is important to clarify how the long-lived waste is to be managed and characterised and which requirements may be imposed on the waste today.

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

There is thus a need to specify preliminary acceptance criteria for both SFL and the extended SFR facility. As the details of the repository design are finalised, it will be possible to further define the set of requirements and finally formulate established acceptance criteria.

Current situation

The recently completed safety evaluation for SFL provides some guidance for future requirements on the waste. The assumptions regarding waste form and packaging used in the safety evaluation have been based on the management of the waste to date. The results of the safety evaluation may therefore be used in the continued work to provide answers to how the present and former management affects the conditions for future final disposal.

Acceptance criteria are established for operational waste in the existing SFR facility and preliminary acceptance criteria have been submitted in conjunction with the application for the extension of SFR. The decommissioning waste is in many ways similar to the operational waste already disposed of in SFR and the new repository parts have several properties in common with corresponding waste vaults in the existing facility. Thus, many of the requirements that apply to operational waste in SFR are also expected to apply to decommissioning waste in the extended facility. This applies in particular to the low-level waste in containers to BLA. Therefore, acceptance criteria for the additional rock vaults 2–5BLA have been developed in more detail.

There are, however, differences in the design of 2BMA compared with 1BMA that must be considered, and there is no equivalent to BRT in the existing SFR facility. The work of developing acceptance criteria for these repository parts will therefore be more extensive.

Programme

Requirements on the waste will be possible to derive from the most recently completed post-closure safety assessment for each facility. The basis for SFL will be the recently completed safety evaluation, while the set of requirements for the extended SFR facility will be based on the assessment that is carried out prior to the PSAR. If necessary, extended analyses can be made with the calculation models that have been developed for the repositories, for example, to define limits for the content and form of the waste.

In addition to requirements related to the post-closure safety of the repositories, requirements related to the construction, transportation and handling during operation will serve as a basis to further define acceptance criteria for the waste.

7.4 Management of reactor pressure vessels and large components

In 2017, Vattenfall and Uniper decided not to dispose of whole BWR pressure vessels in SFR, but that they will be segmented before disposal. There is a policy decision for Ringhals 2 to segment the reactor pressure vessel, while decisions for Ringhals 3 and 4 will be made closer to the decommissioning of these reactors. For more information on the nuclear power companies' planning and execution of decommissioning, see Part III.

Current situation

In 2017 and 2018, a development project was carried out, linked to the decision on segmentation of BWR pressure vessels, to ensure that the entire chain from dismantling to disposal in SFR is performed optimally. The work has identified possible waste containers for the licensees to use for waste to the rock vault for reactor pressure vessels (BRT) in the extended SFR facility.

Programme

At present, SKB is not planning any studies linked to the management of whole reactor pressure vessels or large components but will continue to monitor the development in the decommissioning projects.

7.5 Waste containers and waste transport containers

In order to carry out the decommissioning of the nuclear facilities optimally according to what is described in Part III, development work is required in the area of waste containers and waste transport containers for low- and intermediate-level waste .

7.5.1 Waste containers for waste from AB SVAFO and Studsvik Nuclear AB

Current situation

In the SFL concept study, a basis for waste containers has been formulated for safe and efficient management of the waste that is currently placed in moulds, 200-litre drums and 280-litre protective drums (Pettersson 2013).

AB SVAFO has conducted a feasibility study to analyse different methods for retrieving, characterising, treating and conditioning low- and intermediate-level waste . The purpose of the feasibility study was to consider the possibilities of designing and building a facility for handling and conditioning of the legacy waste prior to disposal in SFL. The conclusion of the completed feasibility study was that management of the legacy waste requires further study before a project for design of a management facility may be initiated. If AB SVAFO chooses to treat and condition their waste, the need for new waste containers could change.

Programme

The need for new waste containers will be revised when the acceptance criteria for long-lived waste and the management of legacy waste are concretised.

7.5.2 Waste containers for decommissioning waste

Current situation

In 2017 and 2018, the work continued to implement the use of a larger waste container for dismantling and decommissioning of nuclear facilities. SKB decided to postpone the development of the tetra-mould and instead proceed with the double-mould (Figure 7-1) and prepare manufacturing documentation. The completed work has included manufacturing documentation for the double-mould, lifting appliances for the double-mould in SFR, lifting appliances for the double-mould at the power plants and transport frames for handling the double-mould by forklift.

Programme

SKB has no programme for development of new waste containers for decommissioning waste but will continuously monitor the need for appropriate waste containers in ongoing and future decommissioning projects.

7.5.3 Waste transport containers for new waste containers

Waste transport containers for long-lived waste aim to permit safe and effective transport of long-lived waste from the waste producers to new interim storage facilities or SFL. The need for waste transport containers is determined by the choice of waste containers and the need for capacity.

Current situation

A new waste transport container, ATB 1T, for transport of steel tanks is being developed in cooperation with the American company Holtec International Power Division Inc. The new waste transport container is planned to be certified and commissioned in 2020.

Programme

The development of additional types of waste transport containers is determined by new waste and waste containers. If new waste containers are developed, the corresponding transport containers must also be developed. No development of new waste transport containers is conducted pending more concrete plans for management of the long-lived legacy-waste.

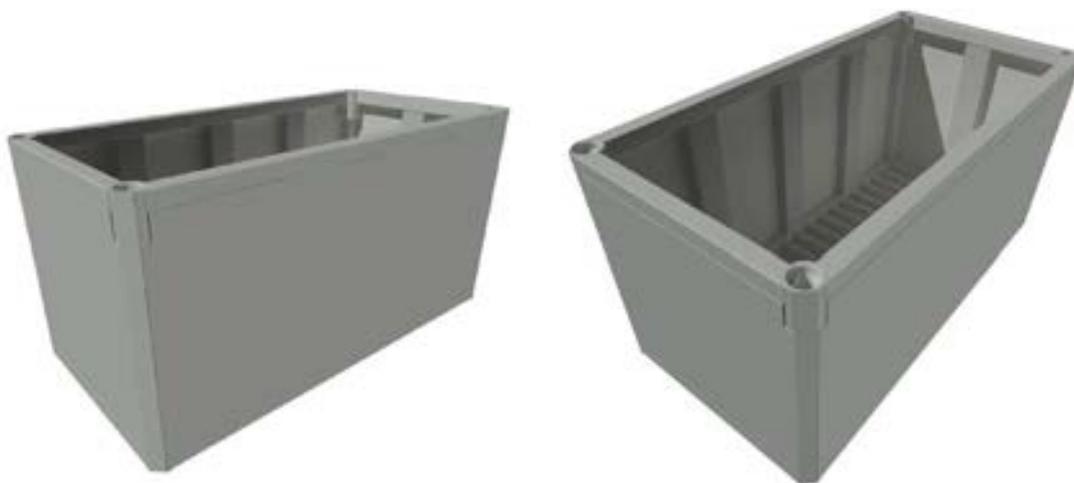


Figure 7-1. Double-mould (1.2 × 2.4 × 1.2 metres), waste container for intermediate-level decommissioning waste. The double-mould will be fitted with a cast concrete lid before the transport to the final repository.

8 Spent nuclear fuel

Spent nuclear fuel is long-lived and highly radioactive and requires radiation shielding in all handling, storage and final disposal. It comprises a small fraction of the total volume of nuclear waste to be disposed of, but contains the majority of the total radioactivity. The spent nuclear fuel must be configured in such a way that criticality cannot be achieved. Final disposal is planned to take place in the Spent Fuel Repository.

The spent fuel generates heat even after it has been removed from the reactor (decay heat). Because of the decay heat, the fuel must be cooled to avoid overheating. The decay heat is mainly dependent on the fuel's decay time, initial enrichment and burnup. The term burnup refers to the quantity of energy that has been extracted from the fuel. Burn-up is specified in megawatt-days per kilogram of uranium (MWd/kgU). Due to technical advances and changes in the operation of the reactors, fuel burnup has increased steadily since the reactors were commissioned. The reason for these changes is to achieve as efficient utilisation of the fuel as possible. This power increase is linked to development of modern nuclear fuel, for example development of new materials and structures, and the use of different additives such as gadolinium and chromium.

According to the planning premises (Section 1.1), the total amount of spent nuclear fuel will correspond to about 6 000 canisters in the Spent Fuel Repository. One canister contains about 2 tonnes of fuel. The quantity of spent nuclear fuel is specified 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 quantity of spent nuclear fuel to be deposited in the Spent Fuel Repository also includes fuel from the Ågesta reactor, spent fuel from testing programmes at Studsvik and a number of fuel assemblies with MOX fuel (mixed oxide fuel). These fuel types comprise a very small fraction of the total quantity. Approximately 20 tonnes of spent nuclear fuel from Ågesta and approximately two tonnes of spent nuclear fuel from Studsvik Nuclear AB's research activities are being interim-stored in Clab today. The facility also contains 23 tonnes of MOX fuel, which has been obtained from Germany in exchange for fuel that in an early stage of the Swedish nuclear power programme was sent to La Hague in France for reprocessing, where uranium, plutonium and waste products are separated.

New fuel types introduced at the Swedish nuclear power plants must first be approved by SKB. This verifies the compatibility of the fuel types with activities that have a bearing on fuel in the entire KBS-3 system.

SKB's programme for handling of fuel comprises several parts, from requirements for information on fuel properties before it is used in the fuel cycle, to formulating a programme for safeguards that is internationally approved. The development work that is being pursued and planned in these areas is described in this chapter. Development of 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 assessment of safety after closure 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 and thereby reduce the pessimism in future safety assessments. A desire for a better understanding and an improved systematic description of the mechanism or mechanisms behind the observed results has also been expressed by SSM both within the framework of the licensing process and the review of the RD&D Programme 2016. There are also some remaining uncertainties concerning the speciation and solubility of released radionuclides.

8.1 Fuel dissolution

This section describes the research programme for fuel dissolution in the repository environment. Fuel dissolution leads to release of radionuclides and the process is essential for the consequence analysis included in the assessment of safety after closure. In its review of the RD&D Programme 2016 and the review of SR-Site (SSM 2018), SSM supports SKB's efforts for process understanding within this area with reference to the fact that these issues have great significance for safety and require in-depth understanding in order to minimise the uncertainties in the consequence analysis. These uncertainties concern both experimental investigations and theoretical explanatory models. The purpose of the research programme described here is to address these issues.

If the integrity of the canister is breached, the fuel will come into contact with water and start to dissolve. The water that enters the canister will also interact with the material inside the canister, which affects the chemical environment and thereby the speciation of dissolved radionuclides inside the canister. Anoxic corrosion of iron forms hydrogen gas, which has proved to counteract oxidative dissolution of the fuel matrix; this is called the hydrogen effect.

In order to gain a better understanding of the mechanism behind the hydrogen effect and the influence of radiolysis on matrix dissolution in the repository environment, carefully controlled experiments are carried out on analogous materials exposed to a varying degree of radiolysis, combined with modelling.

Variations in fuel composition and operating conditions may affect the release of radionuclides from the fuel, both from the rapidly dissolved fraction in the gap inventory and from the slow dissolution of the fuel matrix. In order to study how these parameters affect fuel dissolution, leaching experiments with spent fuel are performed.

Corrosion of the metal parts, i.e. engineering materials and PWR control rods, which will be disposed of together with the fuel in the copper canister, release activation products. Release of radioactive silver from control rods and activation products in cladding material are cases that are of particular interest. Studies related to this are described below.

Current situation

In order to gain a better understanding of the processes that take place during fuel dissolution, with a particular focus on the mechanisms that cause the observed hydrogen effect, experiments are performed in a controlled laboratory environment where different aspects of the dissolution process are studied. The overlying question is how hydrogen can act as a reducing agent on an alpha-emitting fuel surface, since hydrogen is inert in normal room temperature. The reaction between hydrogen peroxide and uranium dioxide is of interest since radiolytically produced hydrogen peroxide is the oxidant with the greatest impact in the repository environment.

During the period, a PhD project (Bauhn 2018) aimed at gaining a better understanding of the hydrogen effect in systems with dominant alpha radiation was completed. The experiments conducted explored the effect of hydrogen in a solution with plutonium (Bauhn et al. 2017) and in systems with different types of uranium dioxide (Simfuel, Bauhn et al. 2018b; unirradiated MOX, Bauhn et al. 2018). The experiments showed that a surface is required to achieve the observed hydrogen effect in a system with dominant alpha radiation and that the hydrogen gas seems to prevent oxidative fuel dissolution by reacting with surface-bound hydroxyl radicals. In this reaction, water and one hydrogen atom are formed, where the hydrogen atom is reducing.

How radiation-induced dissolution of uranium dioxide is affected by the properties of the solid phase was studied in a PhD project during the period (Fidalgo 2017). One of the experiments showed that the reaction between oxygen and hydrogen on metal oxide surfaces form hydrogen peroxide, which is a more efficient oxidant than oxygen (Fidalgo and Jonsson 2016). The results of another experiment (Fidalgo et al. 2018) indicated the formation of surface-bound hydroxyl radicals from catalytic decomposition of hydrogen peroxide. The uranium dioxide redox reactivity is essential in these reactions and may be affected by additives in the material. Gadolinium-doped uranium dioxide was studied as a part of the PhD project, with results suggesting that the addition of gadolinium reduces the uranium dioxide redox reactivity (Fidalgo et al. 2019).

Kumagai et al. (2019) studied how the stoichiometry of the uranium dioxide affects its oxidative dissolution. The results show that fully reduced uranium dioxide reacted with hydrogen peroxide much faster than partially oxidised uranium dioxide. The experiments also showed that a higher concentration of hydrogen peroxide at the beginning of the experiment led to lower uranium exchange for both materials, and this is interpreted as a result of catalytic decomposition of hydrogen peroxide.

In order to study the dissolution of uranium silicide (U_3Si_2), a proposed new type of damage resistant nuclear fuel that is being tested around the world, Maier et al. (2019) compared the dissolution of uranium dioxide with U_3Si_2 . The study focused on reaction kinetics of the reaction between silicide and hydrogen peroxide and the dissolution rate under irradiation of silicide with gamma radiation. The results show that the silicide reacts slightly faster with hydrogen peroxide than uranium dioxide and that a secondary mineral (studtite) is formed in carbonate-free solutions. In experiments with gamma radiation, the formation of oxidants is the limiting factor, resulting in faster dissolution of uranium dioxide compared with silicide.

In the EU project Disco, which is coordinated by SKB, dissolution of fuel types with different additives in the uranium dioxide matrix is being studied. This is done through experiments with spent fuel, with analogous materials (uranium dioxide with different additives) and chemical modelling. The project started in 2017 and will be concluded in 2021. The so-called doped fuels that are being studied are Adopt with both chromium and aluminium and fuel doped with only chromium. The project also includes experiments with MOX fuel. Furthermore, standard fuel is studied as a reference in both oxidising and reducing environments. The analogous materials, of which some are ready-made while others take longer to produce, include additives of chromium, aluminium, gadolinium, plutonium and thorium (as an analogue to plutonium, and MOX fuel). The leaching experiments use aqueous solutions that are relevant for different types of repository concepts, and are planned so that they will produce comparable results and provide input to chemical modelling. The work in chemical modelling involves both thermodynamic modelling of the solid phase specifically with regard to oxygen potential, the dissolution process with the hydrogen effect included and chemical interactions in the immediate environment of the fuel. Progress in the project may be followed from the project website, <https://www.disco-h2020.eu>, where various documents and reports may also be downloaded.

To test the hypothesis that gradual accumulation of hydrogen due to anoxic corrosion of iron may reduce the oxidative dissolution of spent fuel, an autoclave experiment with added iron was carried out. The oxidative fuel dissolution decreased with the accumulation of hydrogen, to cease completely after approximately 220 days (Puranen et al. 2017). In order to further investigate whether iron(II) in solution may have affected the results, a new autoclave experiment was initiated with the same fuel but with magnetite instead of iron. The results are currently being analysed.

Two different high-burnup fuels, i.e. fuel with a burnup higher than about 50 MWd/kgU, have been leached using hydrogen to investigate whether they yield different results than fuel with lower burnup. A Swedish fuel with a burnup of about 65 MWd/kgU showed negligible oxidative matrix dissolution. After about one year of leaching using hydrogen, the leachant contains uranium concentrations that are equivalent to the solubility of uranium dioxide under reducing conditions and the release of caesium and strontium has ceased (Puranen et al. 2018). Previous experiments with another high-burnup fuel gave no reliable results, due to problems with sample preparation (Puranen et al. 2016).

The rapidly dissolved fraction, i.e. the IRF (Instant Release Fraction), consists of the fractions of the radionuclide inventory that are outside the fuel matrix. In order to study how much of the inventory is in the grain boundaries, two previously leached fuels have been ground to grain size with simultaneous leaching (Fidalgo et al. 2019). The results of the experiment are difficult to interpret since many grains were crushed so that not only the grain boundaries were exposed to leaching but also the internal parts of the fuel grains. Further analyses are needed in order to increase our understanding of releases from grain boundaries.

The results from the EU project First Nuclides have now been summarised in two articles with a focus on IRF from high-burnup fuels. The results show that for fuels with a high linear effect (W/cm) in the reactor, the linear effect could be correlated with the release of iodine and caesium (Lemmens et al. 2017). The results of the project have contributed to a better understanding of how the rapidly dissolved fraction of iodine, caesium and selenium correlates with the nature of the fuel specimens and the gap and grain boundary inventory of these fission products (Kienzler et al. 2017). Some insight has

also been gained into how fission gases are released during the course of the experiments. Corrosion of Zircaloy and steel was studied in the EU project Cast (<https://www.projectcast.eu/>) in which SKB participated. The focus was on the carbon-14 content in these metals, as well as the release and speciation of carbon-14. A study of corrosion of irradiated steel in a reducing alkaline environment (Cvetković et al. 2018) showed formation of organic compounds (for example different carboxylic acids) as a result of the carbon content in steel, and these organic compounds largely determine the speciation of carbon-14. Further studies of the release of silver-108m from control rods have been carried out in Studsvik. A previous autoclave experiment in hydrogen atmosphere with used control rods showed no release of silver and the experiment continued with external irradiation for four weeks. Gamma measurement at the conclusion of the experiment revealed no silver-108m in solution, i.e. the content of silver-108m was lower than the detection limit. A public report on this work is under development.

Programme

During the RD&D period, ongoing research projects at Studsvik will continue and new projects will be initiated. Planned studies include the conclusion and analysis of multi-year fuel leaching projects (several decades) in a hot cell, and the start of similar leaching experiments with a focus on doped fuel. A study of long term leaching of fuel in glass vials will continue; some of the remaining vials are planned to be opened in the RD&D period. Data from different studies that have not yet been published will be analysed and published in scientific journals.

In connection with sending damaged fuel to Studsvik to be placed in cases and transport boxes, studies of selected pieces of the damaged fuel will be initiated. The purpose is to characterise the damaged fuel with respect to oxidised uranium and radionuclide content.

An ongoing PhD project at KTH and Chalmers University of Technology will continue during the RD&D period. The goal is to learn more about the processes and mechanisms that regulate oxidative fuel dissolution, for example catalytic decomposition of hydrogen peroxide.

In the ongoing EU project Disco, matrix dissolution of spent fuel is being studied, in Studsvik and other hot cell-facilities in Europe, as well as similar experiments with analogous materials. These experiments mainly take place under reducing conditions (i.e. with hydrogen), but certain experiments are also conducted under oxidising conditions. Within the framework of the project, data from these experiments should be incorporated in updated models for fuel dissolution and investigate whether additives in so-called doped fuel affect the dissolution kinetics, for example by interfering with the hydrogen effect or the redox reactivity of the materials. Results are expected to be available towards the end of the project period, which occurs in the RD&D period.

8.2 Radionuclide speciation and solubility

If a canister is damaged and radionuclides are released from the fuel, their transport is affected by the chemical environment inside the damaged canister. It is therefore important to understand how radionuclides interact with the solid phases and dissolved species present inside a water-filled canister in a repository environment. Studies in this area are performed to establish speciation and solubility limits, which are used in the modelling of radionuclide transport out of a damaged canister.

Current situation

The PhD study conducted at FZ Jülich regarding radium-barium co-precipitation has been concluded and resulted in a PhD thesis (Weber 2017). The experiments focus on how radium is incorporated into existing barite. The results have been published in scientific articles (Weber et al. 2016, 2017) and in summary demonstrate that the nature of the solid phase is important for how quickly co-precipitation is formed. The fast reaction and complete transformation of the solid phase previously observed can be explained with the aid of these studies, as in now has been shown that the barite used in the experiments had a high concentration of liquid-filled pores and cavities (Weber et al. 2016). The uptake mechanism was investigated in more detail by Weber et al. (2017) by studying the development of microstructure and radium content in both minerals and aqueous solution during the course of the

experiment. Three different stages could be distinguished: First, the smallest pores disappeared in favour of macropores, then radium was taken up around the macropores, and finally a homogeneous radium barite had formed in the entire original mineral grain. The experiment showed a complete recrystallisation of barite to radium barite, which occurred through microstructural changes.

Reduction and precipitation of neptunium on iron was studied by Yang et al. (2017). The effects of both pure and corroded iron were studied in an anoxic environment. The results show that pre-oxidised iron, i.e. iron with a magnetite surface, was more efficient for reduction of neptunium(V) and immobilisation of neptunium in the form of neptunium(IV) on the corroded iron. After the experiments it was found that neptunium had precipitated in a 5–8 µm thick surface layer on the corroded iron. This indicates that reduction and precipitation of neptunium probably take place in the environment that is expected in a damaged copper canister.

The EU project Redupp that was concluded a few years ago has continued to result in scientific articles. Redupp focused on how the results of dissolution experiments may be affected by how the surface of the solid phase changes during the course of the experiment, for example by initial dissolution of edges, grain boundaries and other less stable areas. The most recent article describes the relationship between internal defects such as oxygen vacancies in cerium oxide and the dissolution rate: the more defects, the higher the dissolution rate (Corkhill et al. 2016).

A PhD project linked to Redupp has been concluded in the past RD&D period. The thesis (Myllykylä 2017) includes a couple of studies involving a small amount of the isotope thorium-229. Isotope exchange between thorium-229 (which is added to the aqueous solution) and thorium-232 in the thorium oxide shows how the isotope exchange continues even when the system is close to chemical equilibrium (Myllykylä et al. 2017b). In order to study more closely how the isotopic composition of the solid phase changed during the isotope exchange experiments, alpha spectroscopy was used. By means of this method, the thickness of the affected surface layer could be estimated to a maximum of 0.1 µm after 534 days in aqueous solution (Myllykylä et al. 2017).

In another EU project, Skin, in which SKB participated, sorption and uptake mechanisms for very low concentrations of radionuclides in minerals were studied. In the past RD&D period, some results from Skin have been summarised and discussed in an article (Grambow et al. 2017) regarding the measured solubility and what it represents. The results for uranium dioxide and thorium dioxide indicate that when the system is close to equilibrium, less than one monolayer is involved in the observed dissolution and precipitation process. This applies to experiments that last for about 100 days. Probably only parts of the surface of the sample, for example defects or grain boundaries, are involved in the dissolution that is observed (Grambow et al. 2017). The authors interpret this as thermodynamic equilibrium not being achieved, or not even being expected to be achieved, within a time that may be considered reasonable for a laboratory experiment.

The possible transformation of uranium dioxide and spent fuel to the mineral coffinite (USiO₄) is being studied in an ongoing project together with Amphos 21, Marcoule Institute for Separation Chemistry and Stanford University. It involves experiments and chemical modelling. The work has been initiated and the results are expected to become available during the coming RD&D period.

An ongoing international collaborative project with NEA, Thermochemical Database (TDB) Project, called NEA-TDB, has resulted in the thermodynamic database now being available online: <https://www.oecd-NEA.org/tbdbdata/>. The work with the database proceeds and updates for some elements are imminent.

Programme

Research on the possible formation of the silicate mineral coffinite and how varying silicon contents affect the dissolution of uranium dioxide is ongoing in an international project (see above). The project is planned to be concluded during the second half of 2019 and the results will be published in open literature.

Uranium speciation needs to be further studied in view of new data that suggests the possible existence of calcium-uranyl-carbonate complexes, and a project in cooperation with Stockholm University and Chalmers University of Technology is in the planning phase. Updates of thermodynamic data and

speciation are also relevant for other radionuclides, and SKB has therefore joined the next phase of the NEA-TDB project. During the next four years, the project will focus on updating the database for organic compounds, lanthanides and/or weak complexes (including those formed via hydrolysis).

The EU project Disco is also linked to issues involving speciation and solubility, since chemical modelling of fuel dissolution is included: both previous project-external results and project-internal results will be included in the models. The project will thus fill knowledge gaps and reduce uncertainties regarding radionuclide speciation and solubility in a repository environment.

8.3 Non-regular fuels and fuel integrity

Non-regular fuels refer to e.g. fuel from the Ågesta reactor, spent fuel from testing programmes at Studsvik, MOX fuel and damaged fuel.

A number of individual fuel assemblies are continuously inspected in Clab to monitor changes in the properties of the assemblies during interim storage. The integrity of the fuel is also significant for the encapsulation plant, as handling of the fuel assemblies in Clink must be done in a predetermined way. Therefore, ageing of fuel in Clab, from nuclear power plants and other industry is monitored and international research in the field is followed. As nuclear fuel in Clab is handled in pools, inspection of the fuel assemblies is relatively simple, unlike dry storage in canisters, where such inspection is not possible to perform in a simple and continuous manner. Communication is also maintained with fuel suppliers and similar facilities with experience of interim storage.

Current situation

The process of handling and emptying the Swedish nuclear power plants of damaged fuel is ongoing in a special project, which is in its final stage. Damaged fuel with failed cladding is also called leaking fuel. To re-establish the barrier function provided by the cladding during operation of the facilities where fuel is handled, the damaged fuel is treated with special methods so that it does not require further treatment prior to final disposal.

SKB has decided to use two different methods for management of leaking fuel rods. Both methods entail that the damaged fuel is placed in containers with the dimensions of PWR or BWR fuel. These containers are water-tight and thereby replace the leaking cladding. One method is developed by Westinghouse and is called Quiver. The method is developed for use in the storage pools of the nuclear power plants. Before closure of the container, its content is dried (after the fuel cladding has been punctured to ensure complete drying), and dryness is verified. The second method has been developed by Studsvik and is used for transportation of spent fuel to Clab. The Studsvik method involves transporting the leaking fuel rods to Studsvik, where they are taken to a hot cell, sawn into lengths of one metre, dried according to international protocol and thereafter encapsulated in special cases, which are gas and water-tight. They are then packed in transport boxes, which have the dimensions of PWR or BWR fuel. The Studsvik transport boxes can then be managed in the rest of the KBS-3 system.

Existing documentation also shows that there are a small number of fuel assemblies with leaking fuel rods at Clab. A preliminary study is under way concerning their management.

The long-term inspection programme for nuclear fuel is in progress after automatisisation of the fuel handling machine at Clab in order to reduce the risk of mishaps during fuel inspections. Fuel assemblies from Ågesta and Ringhals PWR fuel assemblies with sensitive zirconium alloy have been inspected. The results of these inspections have been discussed with the relevant fuel manufacturers and nuclear power plants.

For certain fuel types, weaknesses in the construction have been discovered that may cause problems when handling the fuel. Information on such constructions is documented in SKB's fuel databases, since they may affect the design of and handling in Clink.

Experiences from the operation of the nuclear power plants in Sweden and internationally are followed up through participation in international conferences such as the IAEA groups for collaboration and the OECD/NEA project Scip III (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.

Programme

When the nuclear power plants are emptied of damaged fuel, the plans for handling the known damaged, untreated fuel in Clab will be established. A detailed plan for this will be prepared during the coming RD&D period. Together with the nuclear industry, methods have been developed to manage the smaller quantities of damaged fuel that are believed to arise when the plants are emptied of existing damaged fuel.

In line with Clab's ageing programme, a renewed ageing analysis of nuclear fuel in interim storage will be carried out. One part will be a system analysis, which is planned to be initiated in 2019.

Monitoring of practices around the world is carried out to keep up to date with the latest findings regarding the integrity of nuclear fuel in a wet storage environment, but also with respect to dry handling. This includes research results in the area as well as visits to and exchange of experience with similar facilities. During the RD&D period, SKB will continue to participate in various international forums. SKB is part of the OECD/NEA project Scip III and will also participate in the next phase of the project, Scip IV, which started in June 2019. Scip IV includes more backend issues than previous Scip projects, which have focused more on the integrity of the fuel during operation at high burnups.

8.4 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. Decay heat often constitutes a limiting factor, which means that it has a considerable financial importance.

During fuel characterisation, all relevant available information in the form of fuel properties and operating 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 safeguards. There are requirements stating that all fuel assemblies must be verified prior to encapsulation, which is the prerequisite for planning.

Current situation

A project with a focus on development of measurements and calculations of gamma and neutron radiation and the decay heat of fuel bundles has been ongoing since 2015 (based in turn on previous activities and projects). The project is now in its second phase. The goal of the work is ultimately to obtain a measurement system that will satisfy the requirements for characterising fuel that is to be encapsulated in the encapsulation plant. The requirements on such a system depend on the amount of other knowledge that is available concerning the spent fuel, for example operating history, which is also being studied, see Section 8.5. Development of equipment and methods is required since the objective is that the measurement system is to be permanent, complete (capable of measuring all fuel assemblies) and robust. It should also yield unambiguous results, have low uncertainty, and a high through-put. The idea is that a combined measurement system will be used to verify the properties of the fuel, and a calorimeter will be in place to substantiate the determinations of above all (but not only) decay heat.

Gamma measurements of fuel in Clab were carried out in the autumn of 2016, within the framework of an international project (referred to as "SKB-50" but officially called Non-Destructive Assay Spent Fuel Project; Tobin et al. 2016). The results of these measurements have now been evaluated. To validate the new methodology, all PWR fuels that were included in the measurement campaign must also be measured calorimetrically. The calorimetric measurements began in January 2018. At the time of writing, a sufficient number of measurements have been carried out for determining decay heat, and evaluation of the gamma method is initiated. The efforts of measuring the remaining fuels included in the project (SKB-50) will continue when possible. The fuel measurements carried out in Clab within the framework of this international project have permitted an evaluation of the uncertainties of the constituent methods and they illustrate how a method that applies parts of the Scale/Origen code may improve the reliability of the measurements (Vaccaro et al. 2018).

Since knowledge of decay heat is central to assess the need for cooling in Clab, a review of all calculations has been conducted. The result is an improvement of the algorithms used for decay heat calculations used in the databases PlutoWeb and Dark are now improved, for example by updating existing algorithms. Furthermore, there will be a review of the algorithms used in the operations at Clab and in the transport of spent nuclear fuel.

Regarding radiation and the properties of the radiation field, measurements and calculations of emission rates, radiation doses and radiation shielding are being conducted at SKB. Radiation protection matters have an established organisation and working procedures for facility operation and transport of nuclear waste and spent nuclear fuel. Correct calculation and use of source terms and radiation doses in other applications need to be coordinated with radiation protection within a common platform of methodologies, calculation tools and competence.

The radioactivity of the fuel entails interaction between the radiation field and the surrounding materials. The effects of this interaction are important to identify and analyse in the entire handling chain, particularly for designing the capacity of radiation protection. The radiation field also affects the material properties of the barriers in the final repository, see for example Chapters 9 and 11. Therefore, monitoring of practices around the world is carried out and in relevant cases research projects concerning the ionising radiation of the fuel and how radiation field properties depend on different parameters. For this purpose, a project is under way where the radioactivity of individual fuel assemblies is used to calculate the emitted radiation and deposited energies in the canister and its near field. These calculations will be carried out based on Monte Carlo simulations at an atomic level, which are being conducted at Uppsala University. A calculation model is developed for neutron and gamma radiation from different common types of BWR and PWR fuel. The calculations are based on the emitted energies from caesium-137 and europium-154.

The impact of the calculated radiation field on the barriers is addressed in the separate chapters for the canister and the bentonite. Radiolysis of water is discussed in Section 8.1.

Programme

The fuel characterisation project will develop and refine the methods during the RD&D period. The goal for the period is to develop the submethods and detector system and verify them. During the coming years, continued calorimetric, neutron and gamma measurements are planned at Clab, as well as investigations regarding the radiation sensitivity and detector response of different detectors with associated electronics. In the next period, the industrial system will be established for use in Clink with all relevant requirements implemented.

The well-established collaboration with different international groups and organisations continues, not least the collaboration with the United States Department of Energy (DoE) through the Los Alamos National Laboratory, Oak Ridge NL, Lawrence Livermore NL and Pacific Northwest NL. In Sweden, the collaboration with Uppsala University and Lund University continues. In the large EU programme Eurad (European Joint Research Programme on Radioactive Waste Management), SKB coordinates a project called SFC (Spent Fuel Characterization and Evolution Until Disposal) during the period. SKB is carrying out an international blind test where several modelling groups predict the decay heat from a number of fuel assemblies unknown to the participants. These are then compared with calorimetric measurements of the assemblies in question. The purpose is to provide a state-of-the-art picture of the status regarding the uncertainty and accuracy of calculations of decay heat. The blind test will be carried out in collaboration with OECD/NEA, which will manage these data.

All this is carried out in cooperation with the project described in Section 8.5, as well as with the operations with respect to existing databases that document fuel data and decay heat. The project will also investigate how the degree of detail in fuel data and power history influences the calculation result.

The development of decay heat calculation algorithms in PlutoWeb and Dark that has been initiated will continue in the coming RD&D period.

SKB intends to develop the radiation research area by establishing a methodology for calculation of radiation-related parameters, and by studying which calculations to perform. The methodology will include which computer codes to use and how they are to be validated, which computing environments to use, quality assurance of input data and documentation of calculations. The methodology should

also describe which calculations should be carried out by SKB and which should be carried out by external parties. This effort will contribute to increased efficiency through coordination and traceability with respect to radiation and radiation dose calculations in various projects. Control over completed calculations may also contribute to optimising the design of radiation shields for new facilities, and to more effective operation-related radiation analyses.

The work of calculating temperatures in the fuel and the near field of the canister on the basis of the gamma and neutron radiation of the fuel will continue during the RD&D period. Parts of this work entail calculations for other fuel types and additional gamma energies, which will be used further in the calculations of the temperature evolution of the canister described in Chapter 9.

8.5 Fuel information and encapsulation optimisation

The composition of the fuel is the basis for all calculations of radioactivity, criticality, decay heat and radionuclide inventory that are required for fuel that is to be transported, interim-stored and disposed of. In order to correctly determine this, detailed information on the geometry, material, initial enrichment and operating history of the fuel is needed.

An important task for SKB and the nuclear power plants in the coming RD&D period is to ensure that the fuel information needed for management and final disposal in the KBS-3 system is compiled and preserved for the future. This is particularly important for the ongoing decommissioning of nuclear facilities. The fuel information project (MIB) is in progress during 2018–2019 with this purpose.

Current situation

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

Work is under way with quality assurance of the fuel information in SKB's databases. This is carried out partly by automated comparisons with data from the nuclear power plants, and by going through fuel documentation from the fuel suppliers and verifying it against the information in the database. The major advantage of making automated checks against the nuclear power plant data is that the comparison may be carried out without risking manual errors, which is an obvious risk if large amounts of data is entered into a database manually.

For older fuel, there is often less detailed fuel data documented. In some cases, there is no detailed operating data saved at the nuclear power plants.

The project will formulate guidelines for which uncertainties need to be associated with fuel where there is no information.

Programme

SKB's various databases should be coordinated so that all calculations and inspections within SKB are based on the same quality-assured fuel database.

Part of the project task is to propose solutions for, and facilitate, that different computer programs can be linked to this database so that calculations, to the extent they concern legacy fuel, may start from quality-assured data from the nuclear power plants instead of assumed data. This applies to calculations of radionuclide content, radiation, decay heat and criticality. The decay heat in individual fuel assemblies affects how the selection of fuel assemblies to be placed in a canister is carried out in order to minimise the number of canisters (encapsulation optimisation).

At present, a review is in progress of how encapsulation optimisation will be carried out based on the information that is and will be available at the time of encapsulation. Efforts during the RD&D period are aimed at developing tools for optimisation of canister content.

8.6 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 entails 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 calculation programs. Both methodology and calculation programs must comply with modern internationally accepted standards and requirements.

Current situation

A new methodology for burnup credit was implemented in the applications for Clink, Clab and the Spent Fuel Repository during 2013–2016. In the latest period, this methodology has been established through different activities. A central part is validation of the new version of the computer code Scale, where altered uncertainties for plutonium cross-sections have been taken into account. According to SKB's methodology for validation of calculation programs for criticality calculations, criticality experiments are selected from the OECD/NEA database where the neutron characteristics best correspond to our target system (the copper canister and Clab's storage canisters). This selection is made using the program Tsunami with the associated methodology developed by ORNL (Oak Ridge National Laboratory). The change in the plutonium cross-section affected the selection so that other experiments were selected than in the previous validation. For high burnups, there were no experiments that were sufficiently similar to the neutron characteristics of the copper canister according to the criteria recommended by ORNL. For this to be achieved, experiments with contents of both iron and plutonium are required, which are not presently included in the OECD/NEA database. SKB solved the problem by selecting a number of different experiments which together cover all the most important reactions that occur in iron and plutonium. This is considered to be sufficient to estimate the uncertainty in the Scale code.

The methodology report for the criticality analysis has been updated taking into account geometrical changes that may occur under the assumption that the integrity of the canister is lost. Efforts to ensure that the requirements stipulated in the criticality analysis can be verified during canister manufacturing have continued and resulted in a report on defects in the insert.

Programme

During the RD&D period, an update of the criticality analysis for the Spent Fuel Repository is planned, where possible geometrical changes in a canister whose integrity has been lost should be taken into account. Study of the long-term stability of the fission products used in burnup credit is also included, as the criticality analysis needs to cover periods of up to one million years. Changed facility layout and a new inventory of events for Clink require an updated criticality analysis. In addition to this, planning is in progress for the strategy for how fuel assemblies that do not meet the criticality requirements of the canister will be managed. Linked to the fuel information project (Section 8.5), a study will be carried out concerning how detailed fuel information may be used to make canister-specific calculations of criticality.

8.7 Safeguards

With respect to encapsulated fuel, the safeguards system will contain information on individual canisters' content of nuclear material, which fuel assemblies the canisters contain, when the fuel was encapsulated, transported and arrived at the Spent Fuel Repository, where the canisters are deposited, and the total amount of nuclear material in the repository. In other words, it must be possible to identify individual canisters and their content.

A general description of safeguards is provided in Section 2.4.

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

- Method for verification of declared nuclear material.
- Logistics for safeguards.
- Copper canister identification.
- Seals for safeguards.
- Unmonitored verification systems.
- Handling of deviations from normal operation within safeguards.
- Design data verification.

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

Current situation

In the past RD&D period, the principal steps with respect to safeguards were defined (Figure 8-1).

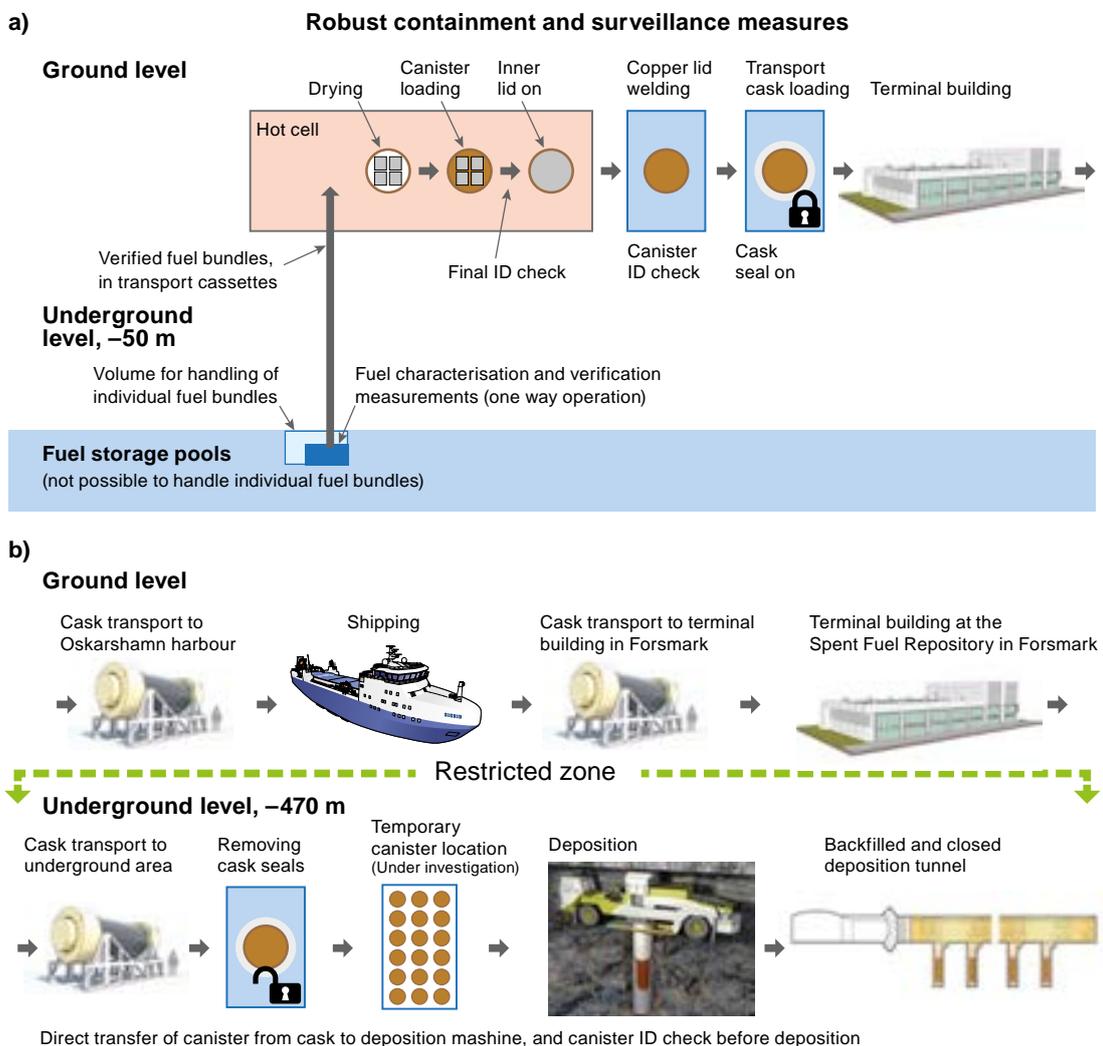


Figure 8-1. a) Graphical illustration of Clink showing planned inspections during transport of the fuel from the storage pools via the fuel monitoring station, the handling pool, drying, placement in the canister and final identity verification of the fuel before the inner lid is mounted. Thereafter, the copper lid is mounted and canister identity is verified before the canister is finally placed in a canister transport cask and the safeguards seals are applied. b) Graphical illustration showing transport and handling of encapsulated fuel from Clink to the Spent Fuel Repository with the verification of safeguards seals in conjunction with opening of the canister transport cask, KTB. The canister is transferred directly from the KTB to the deposition machine in conjunction with identity verification, and is then placed at a temporary storage site prior to the deposition campaign.

The principles are based on the following: Information on the nuclear material content of the spent fuel must be ensured. When the fuel has undergone measurements and verification in the encapsulation plant, the fuel is positioned so that its identity cannot be mistaken with other fuel, for example non-verified fuel. After handling, drying and placement of the fuel in the canister (Figure 8-1a), the identity and position of the fuel is verified by optical equipment. The identity of the canister is verified at the same time so that there can be no doubt as to which nuclear fuel has been placed in each canister. This step is critical in the handling and must therefore be carried out in such a way that no confusions can occur. When the canister has been closed, it constitutes the new smallest unit in the handling of the nuclear material.

The canister is placed in a transport cask for transport to the Spent Fuel Repository. The canister transport cask will be provided with seals and other equipment to ensure that the canister that is received in the Spent Fuel Repository has not been tampered with.

In the Spent Fuel Repository, the seals are verified by status transfer to the inspection bodies in connection with opening the canister transport cask (Figure 8-1b). In the facility's receiving station, it is possible to in conjunction with transfer to the deposition machine verify that the incoming canister identity is consistent with the consigned canister identity. After transfer, the canister is placed in a deposition position or temporarily at a storage site prior to deposition.

The final repository's openings are under surveillance or sealed to make sure that no nuclear material can be removed. The entire volume enclosing the final repository and the ground above the final repository is a restricted zone where no other activities than those reported to the inspection bodies are allowed (Figure 8-1b). The principle is that the nuclear material that has been transported down to the final repository will remain there and should not be possible to remove unnoticed in any way.

SKB has contributed to the European Commission's research and development of methods for marking and identification of copper canisters. A method that is being developed entails marking the inside of the copper canister lid, when the canister lid is welded on there is no practical possibility to manipulate the identity marking and identity reading can be carried out by ultrasound. Another method that is being developed entails recording very small and unique irregularities on the inside of the canister before it leaves Clink. Further development is needed for the methods to be reliable and possible to use in a production environment. The currently most reasonable method for safely identifying canisters is to mark the canister with an identity number both on the outside and on the inside. Reading of the inside imprint or marking will only be done in events where the seals on the canister transport cask have been damaged, and it may be suspected that the canister has been replaced during the transport to the Spent Fuel Repository.

Handling of deviations from normal operation within safeguards has been studied in a simplified analysis with respect to recording the nuclear material inventory for the events identified. Reversing of canisters from the final repository to Clink will require notification and special procedures, which are defined together with the inspection bodies. It may be appropriate to carry out a more detailed analysis of abnormal events during a future period when the facility layouts are more detailed.

Programme

A number of activities are planned during the RD&D period.

Methodology and verification of safeguards will be developed in connection with the projects mentioned above aimed at development of fuel measurement (Section 8.4) and fuel information (Section 8.5).

The nuclear material content of the fuel, which depends on the original enrichment, burnup and decay time, can be calculated either based on information from the fuel's operating history or based on measurements of gamma and neutron radiation in combination with verifying calorimetric measurements. The development of the methodology for combining fuel data with characterisation measurements for verification against declared nuclear material has been initiated and will continue over the coming years. Development of the monitoring station layout and the methodology and technology for nuclear material verification is also in progress in this area. Weighing of fuel assemblies and measurement of Cherenkov radiation are methods that SKB intends to use.

A study of the logistics of safeguards is planned in order to ensure that fuel movements in Clink will not be possible to carry out for manipulative purposes. Specifically, this applies to the handling of individual fuel assemblies prior to and in particular after the measurements carried out during fuel characterisation and nuclear material verification. Disturbances in the fuel handling or significantly deviating measurement results may entail that some fuel assemblies must be set aside or moved back in the system. When the fuel assemblies have been verified, they will be placed in a transfer canister for transport to the drying position in the encapsulation plant. The transfer canisters, which differ in design from the storage canisters, will constitute an important part of the design of Clink with respect to safeguards.

A logistics study will also be needed with regard to safeguards for finished copper canisters in Clink. The handling of the canisters from the encapsulation of the fuel must be carried out in such a way that mix-ups or other changes with the intent of removing nuclear material are not possible. The installations of the inspection bodies should be optimised and cost-efficient, while not disturbing or hindering the production in the encapsulation plant, which SKB will be working to ensure in future cooperation forums with the inspection bodies.

Development of the identification method for copper canisters will continue during the period. Available methods need to be evaluated. The methods that are deemed to be feasible and realistic must then be developed so that they can be applied in practice.

It is essential to always have continuous knowledge of the nuclear material, the copper canister's content and its position. This is ensured through the use of specially developed seals, which are proposed to be integrated in the canister transport cask construction, and other surveillance equipment, for example cameras and detectors. The proposal entails that SKB is allowed to apply the seals in Clink and break the seals in the Spent Fuel Repository without the presence of an inspector from the inspection bodies. The use of seals for canister transport will be developed in cooperation with the European Commission (Joint Research Centre Ispra), so that handling, inspection and verification of the seals may take place in a safe, efficient and reliable way. Unmonitored verification systems and methods for operator handling of seals need to be developed together with the inspection bodies.

Safeguards during abnormal events require further analysis, especially with a focus on Clink. During abnormal events, for example when the integrity of the fuel assemblies must be broken, measures may be carried out in campaigns, by entering equipment for limited actions. The equipment is then removed in order to ensure that there is normally no capacity at Clink to break the integrity of the fuel assemblies.

Design data verification will be conducted by the inspection bodies to ensure that the facilities are built according to the drawings and other design specifications presented by SKB. New technology is being developed by JRC Ispra, among others, to scan tunnels and installations with laser at high resolution, which is then compared with existing drawings or previously completed laser scanning, in order to detect changes in facility layout and installations that are not reported. Design data verification may be initiated in connection with the construction of accesses to the final repository and the construction of the encapsulation plant. SKB intends to participate in parts of this work by supporting the development of efficient methods, which affect production in the facilities to the smallest possible extent.

9 Canister for spent nuclear fuel

The canister consists of a cylindrical copper shell and a load-bearing insert of nodular cast iron (Figure 2-8). Channel tubes of structural steel have been cast into the insert to hold the spent nuclear fuel. Two different insert designs have been developed, in which the size and number of channels are adapted to fuel assemblies from pressurised water or boiling water reactors (PWR or BWR inserts). Section 4.4 gave a brief description of SKB's planned development programme for the canister.

SKB has analysed corrosion in the safety assessment SR-Site, and has thereafter on several occasions provided supplementary documentation for SSM, most recently in SKB (2019e); but in its review in the licensing process for the Spent Fuel Repository (SSM 2018), SSM has indicated several areas where further analysis is needed prior to the PSAR. SSM's review shows that the most urgent questions relate to different forms of localised corrosion in the presence of sulphide, and to creep deformation in the copper. Both questions have a particular focus on unsaturated conditions in the buffer, considering that the time for resaturation could be long.

The continued need for research on corrosion in copper-bentonite concepts for final disposal is also evident from comparisons between the final repository programmes in different countries. In a joint state-of-the-art publication (Padovani et al. 2017), research needs and possible areas of cooperation are described. The influence of pore water composition, irradiation and release of hydrogen is mentioned here, as is the use of analogues. An extensive research programme is presently being conducted in the Canadian nuclear waste programme; see for example Chen J et al. (2018a). This makes comparisons with SKB's research questions possible.

Although copper and nodular cast iron have been deemed suitable for the KBS-3 canister, further development of manufacturing methods remains to be carried out in order to determine material requirements and specifications, and to ensure that all quality requirements can be fulfilled in the canister components. It will be possible to improve the control of the copper's composition, grain size, and avoidance of defects by means of practically applicable theories and stable manufacturing processes. Also the casting and manufacturing of the nodular cast iron insert need to be developed further in order to meet SKB's material requirements, through knowledge of the structure's importance for material properties and how casting is affected by the insert design with steel profiles and by different casting methods.

SKB's friction stir welding for joining the copper shell's parts has been developed with increased process control and a gas shield for eliminating streaks of oxide particles. It cannot be excluded that welding defects need to be characterised and their height determined. In order to ensure that no welding defects or oxide streaks occur, the welding process has to be refined, and testing methods have to be developed further to be able to detect oxide streaks.

This chapter provides a more detailed description of SKB's programme in these areas.

9.1 Corrosion

9.1.1 Sulphide corrosion

In the long-term perspective, sulphide is the most important 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 a repository environment is extensive, but for future safety assessments, a better understanding is needed of the detailed mechanisms in the corrosion process.

Current situation

The formation of copper sulphide films on copper in sulphide solution and the mechanisms for this process have been studied further by means of electrochemical methods (CV, cyclic voltammetry; EIS, electrochemical impedance spectroscopy), different types of microscopy (SEM, scanning electron microscopy; FIB-SEM, focused ion beam scanning electron microscopy) and corrosion experiments.

Previously, research has focused on the question of what limits film growth, but has now progressed to questions of passivity and properties of the sulphide film and the conditions for localised corrosion (a passive film may be such a condition). The work continues to be conducted primarily at the University of Western Ontario in Canada.

The mechanisms for film growth have been studied further under applied potential in electrochemical experiments (Martino et al. 2017, 2019b, Martino 2018), whereas the corrosion experiments have investigated the effects of chloride ions (Chen J et al. 2017b), uneven growth of grains in the sulphide film and microgalvanic coupling (Chen J et al. 2017), and the speciation for the transport of the primary corrosion products in the solution (Chen J et al. 2018b). Figure 9-1 demonstrates that a passive film is only formed at high sulphide concentrations and high sulphide fluxes, higher than those expected in the repository for spent fuel.

In order to better understand if any form of localised corrosion could occur during corrosion of copper in sulphide solution, SKB commissioned three new studies where copper was exposed to a sulphide solution (Chen J et al. 2019), to sulphide gas and to a solution of sulphate-reducing bacteria (Gordon et al. 2018, Johansson L 2019, Stahlén 2019). As an initial step, different methods were also studied for removing the corrosion products without damaging the surface or creating new pits. At sufficiently high sulphide concentrations, pits were observed that could be interpreted as microgalvanic corrosion, which is a milder form of localised corrosion, and that probably appeared due to varying thickness of the film. The lowest sulphide concentration at which this was observed in the experiment was slightly higher than the highest concentrations expected in the Forsmark groundwater. In experiments where copper was exposed to sulphate-reducing bacteria, the solution was either nutrient-rich (higher concentrations of sulphate and organic carbon) or nutrient-poor (concentrations of sulphate and organic carbon that occur in the groundwater at repository depth). In some of the copper samples that were exposed to the nutrient-rich solution more pits were found than on the reference samples, which indicated that localised corrosion may have occurred during the exposure. This result is also compatible with results from some other scientific studies where activity from bacteria and the formation of a biofilm on the copper surface have been observed (Chen S et al. 2014, Dou et al. 2018). All the experiments where biofilms and/or localised corrosion from bacterial activity were observed had in common that the concentration of sulphide in the solution was higher than the highest concentrations measured in the Forsmark groundwater. The samples in a nutrient-poor solution (similar to groundwater) did not have any such pits. All samples did however have pits and scratches, which originated during sample preparation, that is to say, they resembled completely the unexposed reference samples that were also examined.

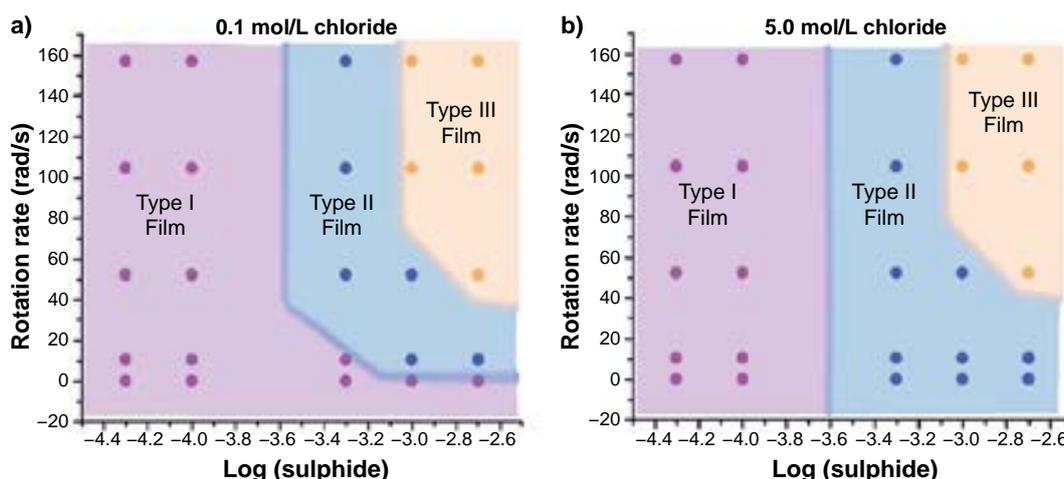


Figure 9-1. Summary of under which conditions in the electrochemical experiments a porous (Type I, II) and a compact and possibly passive (Type III) film is formed on a copper surface, as a function of the sulphide concentration and rotation rate of the electrode (a measure of the sulphide flux to the electrode), for a) 0.1 mol/L chloride, b) 5.0 mol/L chloride (Martino et al. 2017).

In a study by Huttunen-Saarivirta and co-workers, the corrosion of copper under biotic and abiotic conditions was studied, i.e. in synthetic groundwater with or without microbes extracted from the planned final repository site in Finland (Huttunen-Saarivirta et al. 2016). Measurements of both the corrosion potential for copper and the redox potential for a platinum electrode confirmed that the conditions in the long biotic experiment become anoxic and sulphidic, while in-leakage of oxygen appears to have occurred in the short biotic experiment, which is therefore not discussed further here. In agreement with other studies where copper was exposed to sulphate-reducing bacteria, above all the corrosion product dicopper sulphide (Cu_2S) was detected. When the sulphide content was measured at the end of the experiment, the concentration was approximately 2×10^{-4} mol/L, which is the lower limit at which localised corrosion attacks have been found in previous studies; but despite this, no localised corrosion attacks were observed. The results of the experiments without microbes are discussed in Section 9.1.2.

The passivation of copper in sulphide solution has also been studied by others. Mao et al. (2014) presented experimental data interpreted as degradation of a passivating layer, and these data were then used in modelling with a PDM (point defect model). Several scientific papers have been published by (partially) the same group of researchers (for instance Dong et al. 2016, Kong et al. 2018) and they interpret their results as signs of the formation of a passive film. However, Martino et al. (2017) show that the same results with a substantial current increase at higher potentials in the electrochemical measurements are obtained during exposure of copper to a chloride solution only, implying that this is not a question of the formation of a passive film. An additional scientific paper (Huttunen-Saarivirta et al. 2018) uses PDM to interpret data from electrochemical experiments, with the conclusion that a passive film is formed. The paper has, however, been commented on (Martino et al. 2019), with accompanying answers (Huttunen-Saarivirta et al. 2019b) and errata (Huttunen-Saarivirta et al. 2019a). The main arguments in the critique are that no consideration is given to the comprehensive literature showing that no passive film is formed, that an unreasonable value (12 orders of magnitude from other published data) for the concentration of cation vacancies is used, and that the use of PDM assumes a passive film (which makes it a form of circular reasoning to use it for showing that the studied film is passive).

When the canister is deposited in the Spent Fuel Repository, the surface will have a film of copper oxide, which is then expected to transform to a copper sulphide film when sulphide from the buffer (or from the groundwater via the pore water in the buffer) reaches the canister surface. Quantum chemical calculations have now been used for modelling the gradual sulphidation of the crystal surfaces (111) and (100) of dicopper oxide (Cu_2O) via a chemical substitution mechanism (Stenlid et al. 2017). The results show that surface sulphidation of Cu_2O (see Figure 9-2) is even more energetically advantageous than the actual bulk reaction. The results also strengthen the earlier experimental conclusion that the sulphidation mechanism is chemical (a pure exchange of oxygen against sulphur, which does not entail any additional oxidation of copper), and it is therefore pessimistic to add these corrosion contributions in the safety assessment (which was done in SR-Site).

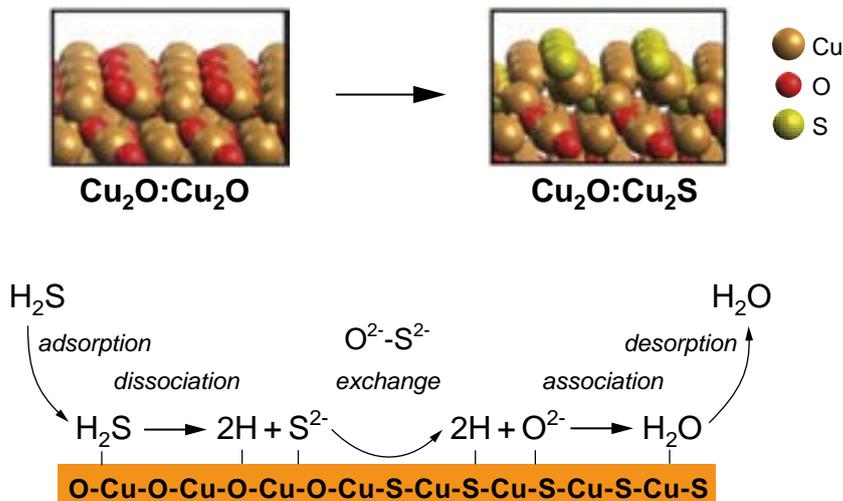


Figure 9-2. Illustration of a quantum chemical model for surface sulphidation of Cu_2O (Stenlid 2017).

Quantum chemical calculations have also been made on models of elemental copper and Cu_2S in order to gain a deeper understanding of the detailed reaction mechanism for sulphide corrosion of copper. The work so far has been focused on testing different mechanisms for the cathodic reaction and shows that as long as there is a copper surface exposed to the sulphide solution, the cathodic reaction occurs on copper rather than on Cu_2S . Preliminarily the results also show that the water molecule is a better donor of protons in the cathodic reaction than sulphide in aqueous solution, due to the considerably higher concentration of water, despite the fact that reactions with sulphide have lower activation energy (Stenlid et al. 2019). Quantum chemical calculations have also been initiated for studying the fundamental aspects of the thin copper sulphide films that are formed in the early stage of the corrosion process, with a focus on their growth mechanism and morphology (Lousada et al. 2018). In addition to this more specific modelling of sulphide on different surfaces, more fundamental development work on the quantum chemical calculation tools is carried out, for instance for developing new descriptors for the interaction between the surface and adsorbed molecules (Halldin Stenlid et al. 2019).

The analysis of the mass transport limitation in sulphide corrosion under saturated conditions in the Spent Fuel Repository in Forsmark has been refined and published (King et al. 2017). Additional data concerning the concentrations and fluxes of sulphide in the repository has been produced, see Section 12.3.2 regarding processes that affect the hydrochemical environment, and regarding bentonite Sections 11.1.1, 11.1.6 and 11.3.1. The models for sulphide transport in the near field, and corrosion on the copper, have been further developed in collaboration with Posiva, see Section 12.3.2.

A new summary of the current state of knowledge on sulphide corrosion, under both saturated and unsaturated conditions in the bentonite, is found in SKB's supplement (SKB 2019e) to the Government after the Land and Environment Court's review of SKB's application for a final repository for spent nuclear fuel in Forsmark. The conclusion is that the sulphide fluxes that could conceivably arise in the unsaturated system are one order of magnitude higher than the highest fluxes that could occur under saturated conditions. Another important conclusion is that as long as the bentonite is in place, even these fluxes are lower by a good margin than those sulphide fluxes under which local corrosion phenomena such as microgalvanic corrosion and micro cracks under applied tensile stress have been observed. (The latter has by some authors been interpreted as stress corrosion cracking, see further Section 9.1.5.)

Programme

Sulphide is the most significant corrosive agent for copper in the repository environment, since the corrosion products are so stable and reactions between the copper surface and sulphide compounds are fast. Any sulphide reaching the canister can thus be expected to react with the copper.

To strengthen the scientific basis for handling of sulphide corrosion in future safety assessments, continued knowledge acquisition is needed regarding the detailed mechanisms of the corrosion process. SKB will therefore continue the studies of copper in sulphide solution, primarily with electrochemical methods. The work essentially follows two directions: firstly, studies that explore in more detail the prerequisites for the occurrence of localised corrosion (through studies of the stability of the sulphide film, the formation of a passive layer, microgalvanic coupling etc) and secondly, studies of how other ions (chloride, sulphate, carbonate etc) affect the corrosion mechanism.

By means of experiments where the laboratory environment is made sufficiently aggressive for localised corrosion attacks to occur, mechanisms for possible growth, and mechanisms that inhibit growth are studied. The methods to remove the sulphide film will continue to be refined, so as to reduce the damage to the copper surface from the removal process, and thereafter further studies are planned of the surface topography under sulphide films from exposure to sulphide solution, and also with microbes added to the solution.

The quantum chemical part of the research programme will have continued focus on modelling of the cathodic reaction, adsorption of groundwater ions on Cu and Cu_2S , and structural and electronic properties of thin sulphide films. All efforts are aimed at achieving a deeper mechanistic understanding of different aspects of the corrosion process.

9.1.2 Corrosion under oxidising conditions

In SR-Site, localised corrosion (pitting) under oxidising conditions is treated as an uneven general corrosion (surface roughening) with a maximum additional corrosion depth, instead of the previously used pitting factors.

Current situation

Through studies of literature data, SKB reported (King and Lilja 2013, 2014) that the pore water compositions in the repository favour general corrosion and do not give rise to a passive film. Such a film is a prerequisite for localised corrosion under oxidising conditions to occur. In order to cover the uncertainty and variability in the data to an even greater extent, SKB has begun the development of probabilistic models for localised corrosion. As a first development step, the analysis has treated a situation with oxidising conditions where the buffer has been water saturated, keeping in mind that this combination will probably not occur (SKB 2019e, Section 2.6). A description of this model development is expected to be published in 2019. An important new basis for assessments is access to the database that the University of Western Ontario has compiled at the request of NWMO in Canada (summarised in Qin et al. 2017). The database contains a large number of electrode potential measurements under different conditions (temperature, pH, concentrations of chloride, sulphate and carbonate).

Data from previous field tests 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 A et al. 2019, Johansson A 2019a). Analysis of data from the tests MiniCan, LOT, ABM, the Prototype Repository and Febex shows that corrosion depths correlate with the estimated total amount of initial oxygen in bentonite and air-filled gaps in the near field; see Figure 9-3. The analysis also shows that the corrosion products in the tests that initially contained large quantities of oxygen (for example LOT and the Prototype Repository) are oxygen-containing and both monovalent and divalent, while the experiments that initially contained very little oxygen had a relatively large proportion of sulphur-containing and only monovalent corrosion products. A more detailed analysis of data and interpretations is found in Johansson A et al. (2019).

In order to be able to better characterise the observed surface of the retrieved canisters from the Prototype Repository (Taxén 2013), SKB has investigated canister copper that was only exposed to the atmospheric corrosion that occurs under normal humidity but was not exposed in any corrosion experiment (Högberg et al. 2017). Analysis of the topography of the canister surface shows an abundant occurrence of pit- and crack-like defects of around some micrometres, which are mainly deemed to have appeared during the machining (turning) of the canister surface that is carried out in connection with manufacturing. It is deemed highly probable that the topography observed on the canisters from the Prototype Repository also stems from the same step in the manufacturing process.

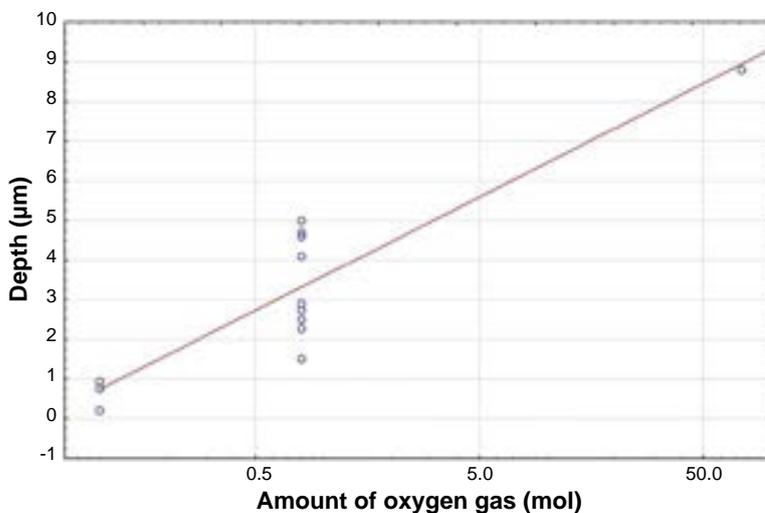


Figure 9-3. Correlation between the average corrosion depth (from mass loss) and the amount of oxygen gas in field tests (Johansson A 2019).

In a study by Huttunen-Saarivirta and co-workers from 2016, the corrosion of copper under biotic and abiotic conditions in groundwater was studied (Huttunen-Saarivirta et al. 2016). The results of the experiments with microbes were discussed in Section 9.1.1. The results of the experiments without microbes clearly show that the experiment was not oxygen-free, both the corrosion potential of the copper electrode and the redox potential measured with a platinum electrode are positive from the start and increase during the time of exposure, which implies that oxygen has leaked into the experiment. This is also confirmed by the oxygen-containing corrosion products that were found on the copper surfaces after the experiment. On one copper sample, a local corrosion attack was detected with an extent of about 50 µm and with the corrosion product having a high concentration of the elements oxygen and chlorine. This is a relatively large corrosion attack compared with the average corrosion in the experiment of about 2 µm. It is unclear whether this observation was a result of the exposure or if the attack may have been initiated already prior to the experiment.

In a subsequent study by Huttunen-Saarivirta et al. (2017), corrosion of copper was studied under oxidising conditions in bentonite, and under the influence of aerobic microorganisms. The observed corrosion agrees well with what has been observed in, for example, the field tests in the LOT series. The main corrosion products were Cu₂O and some form of atacamite, Cu₂(OH)₃Cl, but there was very little sulphur on the copper surfaces compared to the amount of oxygen. No signs of localised corrosion attacks were observed. Similar conclusions were drawn in a study with a multi-electrode, in measurements in water with groundwater composition and bentonite (Kosec et al. 2017).

The experiments at the University of Western Ontario in Canada with the purpose of measuring electrode potentials on copper in bentonite (i.e. in contact with bentonite pore water) have continued, but not unexpectedly it has been difficult to devise electrodes that are stable in the bentonite environment. A first report is found as a part of the PhD thesis of Martino (2018).

Programme

SKB continues the development of probabilistic modelling of localised corrosion and has initiated efforts for the modelling of unsaturated, oxidising conditions.

Both the Prototype Repository's inner section and remaining tests from the series LOT and ABM will provide opportunities to further evaluate corrosion under above all oxidising conditions, even though studying corrosion was not the main purpose of these tests. The retrieval of the remaining test packages follows the overall plan that was described in the RD&D Programme 2016.

9.1.3 Copper corrosion in pure, oxygen-free water

Current situation

The issue of copper corrosion in pure, oxygen-free water has attracted great attention for several years since a group of researchers at KTH presented the view that the extent of this kind of corrosion is considerably larger than established science predicts. As reported in the previous RD&D programme, SKB has therefore studied the process intensively, without finding any support for the KTH researchers' view. Since the RD&D Programme 2016, the following new material has been published:

- Results from the SKB-supported study at Uppsala University have been published in the scientific literature (Ottosson et al. 2017) with subsequent comments in the journal (Szakálos et al. 2018, Ottosson et al. 2018).
- SKB has published a summarising scientific paper, co-authored by the research group at Uppsala University, since some of their results are being published for the first time in the paper (Hedin et al. 2018). The paper describes the attempt by the Uppsala group to repeat the results of the KTH researchers, test tube experiments at Micans (Microbial Analytics Sweden AB), and the search for unknown copper compounds. It also discusses deficiencies and ambiguities in the KTH researchers' publications, along with other experiments in the area.
- A concluding measurement has been carried out on the limited number of test tubes that remained stored in a nitrogen atmosphere at 70 °C at Micans (Morsing Johansson and Pedersen 2019). No hydrogen above the background level could be detected. In Finland, similar experiments have been conducted (Ollila 2019) with results that confirm Mican's results. SKB has also carried out modelling of how canister copper emits hydrogen (Hedin 2019), which further strengthens the interpretation of the experiments with SKB's canister copper.

- A limited effort has been made to assess the possibility of formulating a kinetic model (mixed-potential model) for corrosion in pure water. A preliminary report (King and Orazem 2017) indicates, not unexpectedly, that an earlier model and the kinetic data used in (Cleveland et al. 2014, 2016) are not compatible with thermodynamic data. The model therefore does not provide a reliable estimate of the corrosion rate. Both the model and input data would have to be modified, but with the scarce availability of kinetic data, it is doubtful whether this can be achieved with reasonable efforts.
- The evaluation of kinetic data with electrochemical measurement methods has continued, and the model has been extended to include also the oxide layer that is inevitably present on the copper electrode unless it has been cathodically reduced before the start (Betova et al. 2018). The conclusion is that the possible adsorption sites on the oxide layer determine how much hydrogen can be developed, in the same way as for a copper surface without an oxide layer. Thus, the hydrogen does not originate from a reaction with the copper matrix, and the kinetic data obtained are valid only for a reaction on the surface.
- Other groups have tried to measure hydrogen from copper exposed to pure water (He et al. 2018, Senior et al. 2019), but in both cases the authors themselves note that they cannot draw the conclusion that any corrosion occurred, mainly because other sources of the measured hydrogen have not been possible to rule out.
- The quantum chemical studies of the copper/water interface that were initiated in the preceding RD&D period have continued in order to model the reactivity with more realistic models. New efforts in the area have considered the effects of both defects in the copper surface (Lousada et al. 2017a, b), solvent effects (Stenlid et al. 2016a), and the influence of an oxide film consisting of Cu₂O (Stenlid et al. 2016b, Stenlid 2017). Regardless of how the interface is modelled, the qualitative conclusion from the initial studies remains: The surface reactivity that may occur at the copper/water interface is not sufficient to explain the hydrogen evolution that has been observed in certain experiments with copper in pure oxygen-free water. The interaction between hydrogen and copper oxide and the mechanisms of hydrogen gas formation on copper oxide have been studied further with a combination of experimental and theoretical methods (Tissot et al. 2019). In order to permit better modelling of electrochemical processes on the atomic level, a method has been developed for simulating cyclic voltammograms by quantum chemical means, and the method has initially been applied to different copper surfaces in a simulated solution with varying pH (Bagger et al. 2019).

Additional comments on these studies and a new summary of the current state of knowledge are available in SKB's supplement (SKB 2019e) to the Government after the Land and Environment Court's review of SKB's application for a final repository for spent nuclear fuel in Forsmark. SKB has not been able to find any scientific support for the existence of a sustained corrosion of copper in pure, oxygen-free water above the very limited extent predicted by established thermodynamic data.

Finally, SKB has published calculations showing that even if the observed hydrogen gas formation in the KTH researchers' experiments was hypothetically interpreted as a result of corrosion, the extent of the alleged corrosion process, based on the KTH researchers' own data, would only be about one millimetre in one million years given the temperatures that will prevail in the repository environment (Hedin et al. 2017).

Programme

It is SKB's clear conclusion that there is no scientific support to claim that copper corrodes in pure, oxygen-free water in any other way than what established science states, which is undetectably small and handled in the safety assessment for the Spent Fuel Repository.

Additional final reporting is planned for the kinetic models built on "mixed-potential" and for the influence of ionic strength (experiments with a perchlorate solution). Apart from this, SKB is not planning any further studies in the area.

9.1.4 Radiation-induced corrosion

Radiation-induced corrosion of the copper canister is caused by the radiolysis products formed when water on the outside of the canister absorbs gamma radiation from the fuel in the canister. Only during the first 300 years in the Spent Fuel Repository, 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.

Current situation

After SR-Site, the effects of gamma radiation on the corrosion of copper were studied further in a PhD project at KTH (Björkbacka et al. 2016, 2017, Björkbacka 2015). The experiments were mainly conducted with copper cubes in pure water in a nitrogen atmosphere at radiation dose rates between about 0.1 and 1 kGy/h. These radiation dose rates are approximately 1 000 times higher than the maximum radiation dose rate on the outside of the canister in the Spent Fuel Repository, and the exposure times in the experiments were adapted so that the total dose was of the same order of magnitude as in the Spent Fuel Repository, i.e. 100 kGy (Björkbacka 2015).

The initial studies showed higher corrosion for irradiated samples than for un-irradiated samples, which is expected. Corrosion was detected in the form of Cu_2O and local cavities with a depth of about one micrometre and a lateral extent of a few tens of micrometres. Later studies have shown that the amount of oxidised copper due to radiolysis increases if there is initially an oxide film on the surface (Björkbacka et al. 2015). The measured corrosion effects (at a dose corresponding to that in the Spent Fuel Repository) are, however, small (μm scale) and less than the pessimistic calculations in SR-Site.

Radiolysis of moist air may give rise to ions, but it is worth noting that neither Björkbacka et al. (2017) or Ibrahim et al. (2018), the latter from the Canadian programme, reported any occurrence of ions known for inducing stress corrosion cracking of copper, for example nitrite or ammonium, in their experiments. On the other hand, nitrate was detected in both experiments.

A question that had been unanswered until recently is why a theoretical model, which has been used successfully to predict corrosion of for instance uranium under irradiation, predicted considerably less corrosion of copper than what was observed experimentally at KTH (Björkbacka 2015, Björkbacka et al. 2016). This question now seems to have received an answer, as new experiments and analyses that SKB 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 the surface chemical reactions therefore need to be described differently in the models for the two materials (Soroka and Jonsson 2019). The revised model is still under development but already predicts the experimentally observed corrosion of copper under gamma irradiation with relatively good precision (Soroka and Jonsson 2019, SKB 2019e), which is illustrated in Figure 9-4.

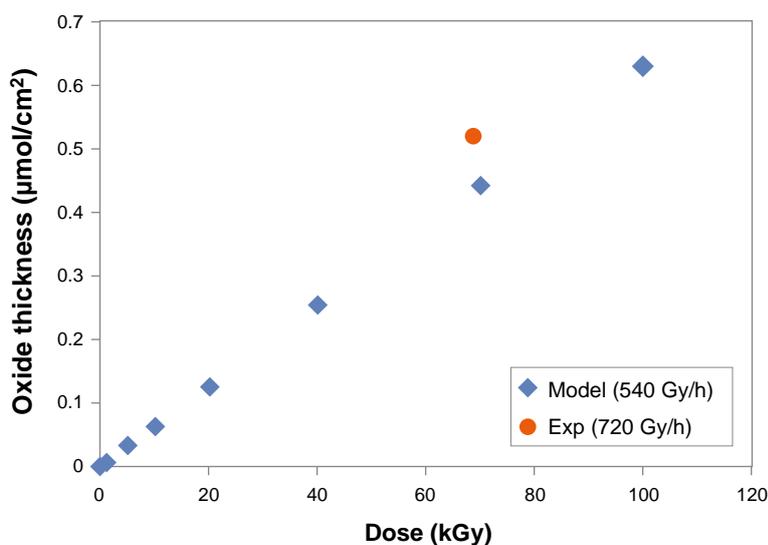


Figure 9-4. Model calculation of the oxide layer average thickness (expressed as the amount of oxide per surface area) as a function of absorbed dose, at the dose rate 540 Gy/h (Soroka and Jonsson 2019). The experimentally measured value in Björkbacka (2015) at the dose rate 720 Gy/h is also given.

Programme

In order to reduce the uncertainties in the assessment of post-closure safety regarding how radiation-induced corrosion in the Spent Fuel Repository affects the canister in the long term, the work at KTH will continue with a further study of the reaction mechanism for this corrosion process in a ground-water environment. For example, the effects of groundwater ions such as chloride and carbonate will be studied and included in the theoretical model. SKB also intends to have calculations carried out of corrosion due to radiolysis under a realistic initial dose rate and a realistic decay of caesium-137, instead of the higher and constant dose rates used in Soroka and Jonsson (2019).

9.1.5 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 handled in previous safety assessments (including SR-Site) and has then primarily concerned corrosion under oxidising conditions in the presence of nitrite, ammonium or acetate. 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 has been discussed, particularly since a Japanese research group (Taniguchi and Kawasaki 2008) presented results that indicated such a process.

Current situation

Since 2008, four other groups have carried out experiments to investigate whether stress corrosion cracking can occur in a sulphide environment. In the early experiments (Bhaskaran et al. 2013, Sipilä et al. 2014), no signs of stress corrosion cracking could be observed, and neither in Swerea KIMAB's investigation of the effect of specimen manufacturing in 2016 (Taxén et al. 2018). However, slow strain rate testing at Studsvik (Becker and Öjjerholm 2017) and at RISE KIMAB (Taxén et al. 2018, 2019) showed small superficial cracks at sulphide concentrations over 10^{-3} mol/L. Common for the latter studies was that the pH had been buffered to about 7.2 with a phosphate buffer. The follow-up study at RISE KIMAB (Taxén et al. 2019) showed that cracks with a depth of up to 30 μm were found if the sulphide concentration was sufficiently high, at both 60 and 90 °C and at different concentrations of chloride ions. The investigations also showed that the attacks were localised to grain boundaries (Figure 9-5) and that with time the crack formation stopped. A mechanistic description has now been formulated (Taxén et al. 2019), which states that tensile stresses cause cracks in the sulphide film, and that if this takes place directly over a grain boundary, which is more reactive than the actual crystal grains in copper, some corrosion will take place in the grain boundary. The corrosion attack is controlled by the transport of sulphide into the crack and as the crack becomes deeper and the corrosion products fill the crack, the local process will stop.

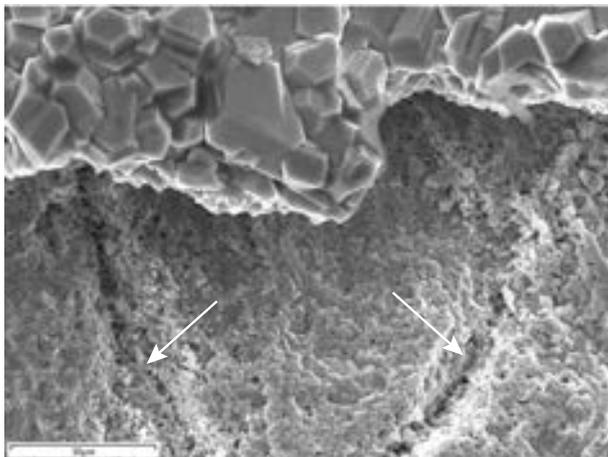


Figure 9-5. Shallow corrosion in grain boundaries of copper specimens after slow strain rate testing in a solution with 10^{-2} mol/L sulphide at 80 °C (Taxén et al. 2018). The arrows indicate superficial attacks in grain boundaries.

Already in the documentation for SKB's application in 2011, it was discussed whether the so-called Aaltonen mechanism could be a stress corrosion cracking mechanism. The mechanism entails that cracks grow as a result of an excess of vacancies, i.e. atomic "holes", being moved to the crack. SKB has had experts at VTT (Technical Research Centre of Finland) go through the published studies once again, and it was concluded that the laboratory experiments where the effects of vacancies have been studied were carried out under conditions that could never occur in the repository (Huotilainen et al. 2018).

SKB has compiled and updated the calculations of tensile stresses in the copper shell for different load cases (Hernelind 2019b) and also analysed the effect of reducing the gap between the copper shell and the nodular cast iron insert (Hernelind 2019). The conclusion is that tensile stresses cannot be ruled out, but that they can probably be reduced if the gap is reduced by minor changes in the design of the canister.

SKB has made a summary of the current state of knowledge of stress corrosion cracking in sulphide (SKB 2019e), and concludes that stress corrosion cracking in sulphide does not threaten the integrity of the canisters in a KBS-3 repository in Forsmark.

The field test MiniCan 4, which was retrieved in 2015, included bent copper samples to investigate crack initiation and precracked samples to measure potential crack growth during exposure in the Äspö HRL's groundwater environment. No crack initiation was observed in either of the two bent samples (Gordon et al. 2017, Johansson et al. 2017). Neither has any growth of the initial crack occurred in any of the precracked samples, corresponding to the results from the previously retrieved MiniCan 3. Small cracks that run from the initial crack were observed, but they are deemed to have been caused by the alternating stress used to achieve the initial experimental crack during sample preparation.

Programme

The research programme for stress corrosion cracking of copper is continuing with additional efforts, focused on sulphide environments. The purpose is above all to further verify that the observed superficial cracks are a form of intergranular corrosion. Planned experiments include slow strain rate testing with a lower strain rate and longer exposure times than previously, and a different composition of the solution so that the possible effects of pH and other ions may be studied. The experimental results will also be used to develop the mechanistic description. Whether the material properties of copper are affected by the injection of vacancies from a corrosion process will also be investigated, for example by studying the effects on the creep rate of applied potential during creep testing, as well as the significance of initiated and completed cracking of the copper shell.

With respect to stress corrosion cracking under oxidising conditions, SKB will study which concentrations of nitrite and ammonium that could arise due to the effects of gamma radiation close to the canister (on the outside, as well as in the gap between copper and iron), and likewise the acetate concentrations caused by microbial activity, in order to determine whether stress corrosion cracking may occur. For further analyses of corrosion from the inside of the canister, the overview from the Canadian programme (Wu et al. 2019) will be useful material.

9.1.6 Verification of the corrosion susceptibility of different copper materials

Current situation

Electrochemical methods for investigating the difference in corrosion susceptibility of different copper materials (welded, cold-worked etc) have been developed (Taxén et al. 2017). The results did not show any influence of cold working or welding on the corrosion susceptibility of copper. The same conclusion was drawn in the study of the significance of oxide streaks (Björck et al. 2019), where the same method was used (Section 9.4.3).

Programme

SKB is not planning any further development of methods for determining the corrosion resistance of different copper materials.

9.2 Canister material properties

The expected surrounding pressures in the final repository will deform the canister copper elastically, viscously by creep deformation, and if pressures are high also plastically. Both plastic and viscous (creep) deformation take place by sliding of atomic planes in the crystals in the balance between the stresses and the material's strain hardening and recovery. Creep deformation of a material is dependent on temperature and stress, and therefore, creep deformation will take place during certain periods of time and vary spatially in the canister. Canister copper has been creep-tested with respect to different temperatures, stresses, structures and defect occurrences. However, extrapolation will be necessary in view of the long time period in the repository and given that it is central that material properties do not change over time. In order to show that the properties do not change, processes active in the material must be possible to describe physically and chemically.

9.2.1 Creep of copper

SKB has chosen an oxygen-free copper with a low content of impurities. In order to achieve favourable creep properties (sufficiently high ductility), phosphorus is added, and the material is often called Cu-OFP (oxygen-free, phosphorus doped). 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 (down to an atomic level) to show that the material properties do not change over long periods of time. SKB has for a long time conducted development work for describing the creep processes in copper by means of models.

Current situation

The effect of cold working on the strain rate has been modelled as a balance between strain hardening, dynamic recovery and static recovery, and after cold working it is observed that most of the dislocations are found in cell walls (which are formed inside the grains in the metal), and not in the grain boundaries (Sandström 2016). The modelling work has continued in order to provide, as far as possible, physical descriptions of all input parameters, and a back stress has been introduced representing the dislocations in the cell walls (Sandström 2017).

Multiaxial stress states for slow strain rate testing have been evaluated with finite element calculations (Sui and Sandström 2017) and modelled using the developed creep model (Sui and Sandström 2016). The results verify that the fundamental models for uniaxial load may also be used for multiaxial states. Recently it has been possible to also include creep deformation during the final phase before failure (tertiary creep) in the modelling of canister copper (Sui et al. 2018). Sliding of crystal planes is what mainly contributes to this creep deformation, while cavity formation at most contributes just below one percent.

In the PhD project at KTH, the influence of the strain rate on the development of the dislocations has been studied with dislocation dynamic calculations (Delandar et al. 2016), where also the significance of sliding crystal planes has been demonstrated (Delandar et al. 2018b).

The quantum mechanical calculations of the effect of how impurity atoms interact with grain boundaries have been published (Li et al. 2017). Thermodynamic calculations have also been made for ternary systems copper-phosphorus-X (Magnusson 2017). The analysed elements (X) include substances that are known to increase, or reduce, embrittlement, and the elements present in Cu-OFP. The conclusion is that the only substances that interact with phosphorus are oxygen (by formation of phosphates) and nickel and iron, both of which (also in a mixed form) form stable phosphides.

Continued work has been performed to study the effects of phosphorus on the creep properties of copper. The appendix to Andersson-Östling et al. (2018) tested hypotheses regarding the impact of microstructure (no recrystallisation effect could be detected) and grain size (no effect of combining large and small grains). In the same study, creep-tested copper was investigated metallographically and with TOF-SIMS (Time-of-Flight Secondary Ion Mass Spectroscopy), but no enrichment of phosphorus in grain boundaries could be observed with the available detection limits. Four different Cu-OFP materials were investigated with TEM (transmission electron microscopy), also for the purpose of finding enrichment of phosphorus (Magnusson and Bergqvist 2018). Inclusions of sulphur (sulphides) could be observed, but no enrichment of phosphorus. It could also be noted that sample preparation may disturb the results (both etching with phosphoric acid and ion polishing with argon) and that crystallographic effects may be misinterpreted as particles. Slow strain rate testing has been used (Josefsson and Andersson-Östling

2019) for comparing the properties of four coppers of different compositions. The conclusion was that the addition of phosphorus is the factor of significance for obtaining increased ductility. The difference is more pronounced at 175 and 215 °C than at 75 °C, where also materials without phosphorus have a high ductility, see Figure 9-6. Slow strain rate testing at lower temperatures is thus not decisive in the way that creep testing is for the ductility at very low strain rates. The results are completely in line with previous test results for SKB's Cu-OFP (Andersson-Östling and Sandström 2009).

The size and variation of the stress and deformation in canister copper has been simulated using finite element models. SSM has had the plastic deformation in the copper calculated (Engman 2018) as a step in the review of SKB's work. SKB has presented new and updated calculations of the total plastic and creep deformation in the copper, which also show how tensile stresses may be reduced to some extent by minor design changes in the canister (Hernelind 2019a); see also Section 9.3.

Failure mechanisms in SKB's previous creep testing have been evaluated and canister copper has been found to be plastically ductile and stable during creep (Björkblad and Faleskog 2018). In Figure 9-7, dimples that extend from microscopic defects can be seen, which are typical of ductile fracture. Only in the case of extremely wide streaks of oxide particles outside the welding processing limits, the ductility is negatively affected (Björck et al. 2019).

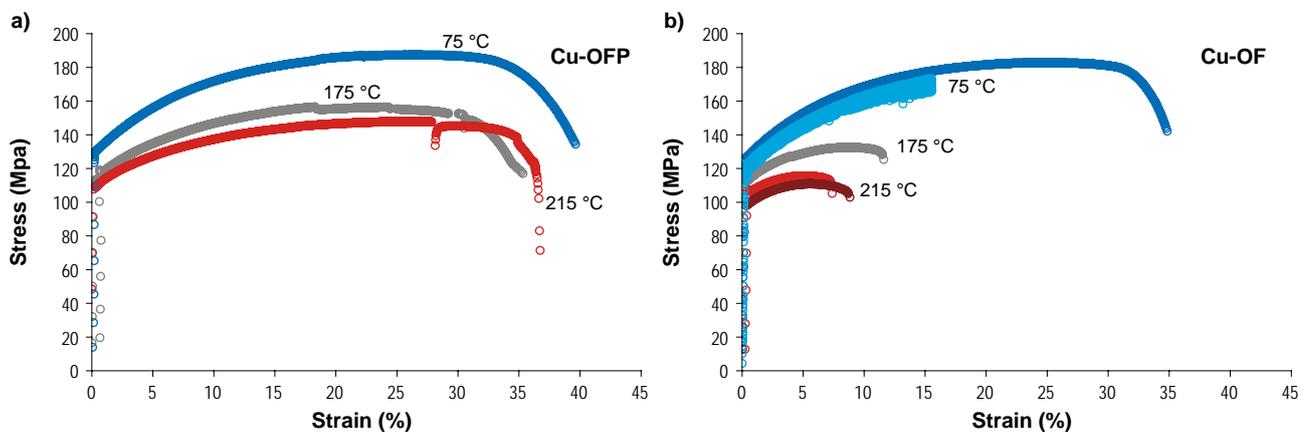


Figure 9-6. Slow strain rate testing (strain rate approx. 5×10^{-7} 1/s) of a) Cu-OFP (with phosphorus), and b) Cu-OF (without phosphorus). The graphs show engineering stress versus engineering strain, i.e. no compensation for necking has been applied (Josefsson and Andersson-Östling 2019).

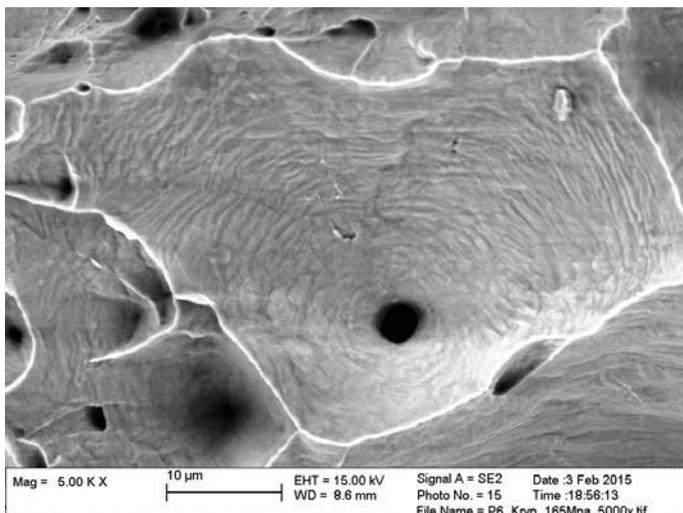


Figure 9-7. Detail from a ductile creep rupture of a cylindrical copper specimen (Björkblad and Faleskog 2018). The illustration shows the fracture surface.

Programme

To support the modelling of creep and especially the long-term extrapolation that is needed, it is central to understand the effect of phosphorus on the fracture mechanisms of the canister copper. SKB will continue in the same way as previously, with a combination of studies with different methods and scales, from the atomic scale to the specimen scale.

In the quantum mechanical calculations of interactions between atoms and dislocations and defects, the focus remains on grain boundaries, where above all studies of the diffusion of copper itself, as well as sulphur and phosphorus, will supplement the description of the mobility of defects and the ability of impurity atoms to accumulate (segregation) and form clusters. The stability and potential formation of phosphorus-containing compounds in the grain boundaries will be investigated.

Further experiments will be carried out to find phosphorus in copper and study how it is distributed, but it has proven difficult to find techniques that are useful and sufficiently sensitive. Primarily, the work will be focused on studying grain boundaries. Studies of recrystallisation at different temperatures are in progress, as well as Auger investigations of fracture surfaces, and a first publication of results is planned for 2019. The compilation of SKB's studies of creep testing since the late 1980s (Andersson-Östling and Sandström 2009) will be updated. Furthermore, creep testing is also planned to some extent, among other things in order to better estimate the fundamental effect of crystal (grain) structure, different impurity elements and phosphorous contents on the slow creep deformation's extent and fracture mechanisms.

The work with creep modelling will be supplemented with an extended description of cavity formation and how it is linked to the binding energy of phosphorus. An overview paper on creep modelling is planned.

9.2.2 Hydrogen embrittlement

Hydrogen can occur in metals both in atomic form (H) and as hydrogen molecules (H₂). If the concentration of hydrogen is sufficiently high, it may affect the metal's mechanical properties negatively. 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). However, in general copper and many other types of metals and alloys with a similar crystal structure do not experience hydrogen embrittlement.

According to the requirements on the oxygen-free copper used for the canisters, the hydrogen content may be at most 0.6 wt.ppm, and the corresponding requirement for oxygen is 5 wt.ppm. These requirements serve to prevent hydrogen embrittlement, also the type termed hydrogen sickness (where hydrogen reacts with oxygen in the form of an oxide and forms water vapour). Similar requirements on the oxygen content have been used for many years for copper in different applications and have been so successful that hydrogen embrittlement is no longer considered a technical problem in copper. The starting material for the copper canisters is sufficiently pure for hydrogen embrittlement not to pose a problem, but it must be ensured that the material is not affected in the manufacturing of the canisters or in the final repository in such a way that hydrogen embrittlement could occur and affect the canister properties negatively.

Current situation

SKB has worked for a long time with the requirement of keeping the oxygen content in the canister copper at a low level. This is achieved on the one hand by exclusively using copper with low contents of oxygen (at most 5 wt.ppm) and on the other hand by developing the welding technique for closure of the copper canister with a gas shield so that the welding of the canister lid and bottom does not lead to the build-up of high contents of oxides in the weld joints; see further in Section 9.4.3.

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 enter any large quantities, other than close to the surface, and likewise to reliably measure the content at different places in the material.

In some experiments, moderate increases of the hydrogen content have been observed when copper in water was 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. The uptake of hydrogen that was measured after copper was exposed to gamma radiation is less than the hydrogen content in SKB's canister copper (Lousada et al. 2016). Furthermore, it has been measured in very thin samples, which means that the effect on the copper canister would be even less significant. Measurements of hydrogen in copper after slow strain rate testing in sulphide solution at Studsvik (Section 9.1.5) indicated that the hydrogen content had increased from 0.5 to about 1.0 wt.ppm during the exposure (Forsström et al. 2017). The results can be questioned, however, since 1) the increase of hydrogen content was independent of the sulphide concentration, 2) also the non-exposed part of the specimen had the same content of hydrogen as the exposed part after the testing.

In some tests, the mechanical properties after hydrogen charging were also measured and an effect was observed that approximately corresponds to that which could be expected if the material had actually changed near the surface. A new survey of previously published testing of copper did not identify any mechanism that could cause crack propagation in copper during the time spans that are relevant in the repository (Leijon et al. 2018).

Extensive quantum chemical calculations 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 (Lousada and Korzhavyi 2019). The calculations show that even if the copper surface was completely covered with hydrogen, regardless of origin, the concentration inside the metal is very low.

The risk that hydrogen enters and affects the properties of the copper material at depth has also been analysed by theoretical means based on diffusion calculations, both for a case of hydrogen pores forming in the metal and for a case where this is not assumed to occur (SKB 2019e). The results show that the extent of the corrosion that may occur in the final repository is too small for the hydrogen released to have any significant effect on the canisters.

Finally, SKB in cooperation with four different laboratories commissioned a so-called Round Robin study of the sensitivity of hydrogen measurements by melting analysis. The study highlights the importance of similar sample preparation for being able to compare samples with respect to hydrogen content (Granfors 2017).

Programme

The research programme for hydrogen in copper is continuing with electrochemical experiments in order to better understand what charging conditions are required to force hydrogen into copper, what is required for pore formation to take place, and how the mechanism for transport in copper under conditions with high current density differs from conditions with the low current densities that may be the result of gamma radiolysis and sulphide corrosion in the repository environment.

The efforts with quantum chemical calculations will continue for the purpose of gaining a better understanding of how hydrogen in copper interacts with impurity atoms and defects in the material, and how this may affect the transport properties of hydrogen in copper.

During the RD&D period, SKB also intends to have methods for measuring changes and variations in the hydrogen content of copper during experimental tests evaluated further, in order to be able to better evaluate whether different experimental environments have any effect on the hydrogen content. Development of measurement methods will also be useful for formulating requirements and verifying that requirements in the manufacturing of the canisters are met.

9.2.3 Radiation effects on copper and nodular cast iron

In the documentation for the safety assessment SR-Site, it was concluded that the radiation doses that the canister materials (copper and nodular cast iron) will be subjected to are at least one order of magnitude too low to provide measurable effects on the mechanical properties of the canister. This is based primarily on previous calculations of radiation damage in the two materials as a result of gamma and neutron radiation.

Gamma- and neutron radiation may also have an embrittling effect on the material properties of the nodular cast iron, by means of the precipitation of copper particles or the formation of LBP (late-blooming phases). 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

SKB has had new calculations of radiation damage on copper and nodular cast iron carried out by a research group at KTH (Yang et al. 2019). The calculations of radiation damage have been conducted for both gamma and neutron radiation. Almost twenty years have passed since the previous calculations were made, and both software and data have evolved. The new calculations result in estimates of the radiation damage in the materials of the same order of magnitude as in the previous calculations.

In both the previous and new calculations of radiation damage, the effect of the irradiated material "self-healing" at the same time as radiation damage is created has been cautiously disregarded, which entails that the results are pessimistic regarding the extent of the radiation damage. Recently completed estimates of the efficiency of self-healing suggest that the actual extent of the radiation damage is at least a factor of 100 lower than the already low extent in the calculations (Padovani et al. 2019).

In parallel with the new radiation damage calculations, SKB has also had a research group in the UK attempt to experimentally verify small radiation damage in copper after gamma irradiation, with a dose representative of the conditions in the final repository (Padovani et al. 2019). Already in advance it was clear that the small extent of radiation damage estimated by calculations is too low to be detected. The ambition of the experiment was instead to use the sensitivity of the experimental methods to verify that the extent of damage did not exceed that which could be detected in the experiment, which is also what was found.

The feasibility study to investigate whether LBP can occur in the nodular cast iron has been completed (Korzhevyyi et al. 2018). In this study, thermodynamic calculations of phase diagrams (Calphad, CALculations of PHase Diagrams) were used in combination with quantum mechanical calculations for the analysis of interactions between dissolved atoms and vacancies in the copper lattice. The work was later extended to a systematically structured database of thermodynamic data for concentrated dissolved and precipitated phases in iron-based alloys (Delandar et al. 2018a). Similarly, there is now a systematic study with quantum mechanical calculations of interactions between dissolved atoms, impurity atoms and vacancies in iron (Gorbatov et al. 2016), which may be used in studies of the formation of clusters and precipitation of phases.

Programme

SKB is presently not planning any continued efforts with respect to direct radiation damage in canister materials. The new calculations of radiation damage will serve as a basis for the planning of experiments to investigate possible precipitation of copper particles. This will in turn be used to ensure that the requirements on the maximum copper content in nodular cast iron are appropriate. The possible impact of radiation on the steel channel tubes will also be taken into account. Precipitation of other phases in the nodular cast iron will be studied further, with calculations and possibly experiments.

9.3 Technical design

Current situation

SKB has updated the design analysis in collaboration with Posiva (Jonsson et al. 2018) in order to clarify requirements and compile and incorporate the new, verifying stress and damage tolerance analyses that have been performed as a response to the request for supplementary information by SSM. For the analyses of the mechanical load cases, the new technical design requirements (Posiva SKB 2017) have been taken into account.

The requirements on the canister's strength and permissible defect size are formulated deterministically based on technical design requirements and material standards. In order to evaluate the robustness of the canister, SKB has also performed probabilistic analyses of both the isostatic load case and the shear load case (Jonsson et al. 2018).

Even though mechanical loads vary between parts of the canister in the final repository, the requirements on canister strength have not been adapted to this fact. The size of acceptable defects varies in different parts of the insert cross-sections, but otherwise the set of requirements regarding defects and material properties has been uniform for the whole insert. SKB sees opportunities to relax standard requirements in relation to material properties and permissible defect sizes, while maintaining the integrity of the canister. Furthermore, SKB can justify that not all parts of the canister require testing.

Programme

In order to adapt the requirements to changed technical design requirements, variations in mechanical loads and the systematic variation of properties within canister components, SKB is expanding the set of requirements and carrying out an assessment of the compliance with requirements in the PSAR.

Permissible handling loads are analysed as well as the permissible size of handling defects in the copper shell. New studies of the copper shell are performed on the maximum residual deformation that the copper shell may be subjected to. These studies use both elasto-plastic material models and material models containing creep (Section 9.2.1). The different load cases are evaluated with respect to whether they are deformation controlled or force controlled for the copper shell. Furthermore, the impact of an eccentrically mounted insert is studied. Thereafter, the requirements on the copper material's ductility can be revised and a new assessment of compliance with requirements be carried out.

Preliminary results show that there is potential for revising the requirement on elongation after fracture in nodular cast iron. The set of requirements with respect to idealised defects in nodular cast iron is being studied further, and the preliminary results indicate that these requirements may be relaxed, in particular for the central parts of the insert. SKB is further investigating how this is to be inspected in serial production and with which safety (confidence) the requirements must be satisfied.

9.4 Manufacturing, inspection and testing

9.4.1 Design and processing of copper components

SKB imposes requirements on the strength and ductility of the canister copper for encapsulation, transportation and final disposal. Dimensions and material properties have been previously presented in SKB (2010b). In order to achieve the properties and dimensions postulated in the design specifications, the copper components are manufactured by casting, hot working, and machining according to standards and the industry practice of metal plants. The hot-working processes used are extrusion of tubes, pierce and draw processing of tubes and forging of lids and bottoms. The processes require sufficient force and power of the presses used in the manufacturing, and also that the metal can be processed without breaking or forming other defects.

Current situation

SKB has knowledge on tensile strength and deformation resistance, and experience of hot working. Figure 9-8a shows SKB's measurements (Nilsson 2019) of the strength of extruded tubes T77 and T85, and it also demonstrates that the yield strength R_p can be described by a fundamental grain size model according to Hall–Petch adapted to the measurements. The spread in measurement data indicates that also other hardening mechanisms may affect the yield strength. SKB's measurements of the yield strength R_p have proven to be lower than the standard value for copper > 99.9 percent from ASM Specialty Handbook (Davis 2001), and this can be assumed to be a result of the larger grain size. For the same sample material, Figure 9-8b shows how the copper's tensile strength is affected systematically by the temperature. The straight lines show the sum function of normal distributions from minus three standard deviations to plus three standard deviations, where the proportion of 0.50 corresponds to the expected value.

The deformation resistance and yield stress in hot working depends mainly on temperature and strain rate, but also to some extent on the metal grain size. Figure 9-9 shows that SKB's measurements (Nietsch 2018) of the maximum yield stress of copper in hot working can be modelled as a function of a temperature-compensated strain rate, the Zener–Hollomon parameter. The maximum yield stresses

presented in the figure have been measured on the solidification structure of copper ingots. The model can be used to interpolate the yield stress of copper and can be included in force calculations for the processing of work pieces of copper ingots. In the extrusion and forging of copper components, the copper will have a smaller grain size and the model has to be calibrated and possibly modified.

Since the temperature and strain rate affect the new formation of crystal grains in the copper, the newly formed grain size is often modelled as a function of the temperature-compensated strain rate. The models can be used to choose a suitable combination of temperature and press speed in the concluding processing steps in order for the metal structure to be sufficiently fine-grained.

Figure 9-10 shows the copper lid thickness after machining for different manufacturers. The plate thickness has been measured for 35 lids and is on average 49.96 mm, with a standard deviation of 0.16 mm. The upper or lower control limit of the manufacturing process is defined as the mean value ± 3 standard deviations for each manufacturer. Although the manufacturing processes are not completely stable, the plate thickness of the copper lids is well within the requirements (Jonsson et al. 2019).

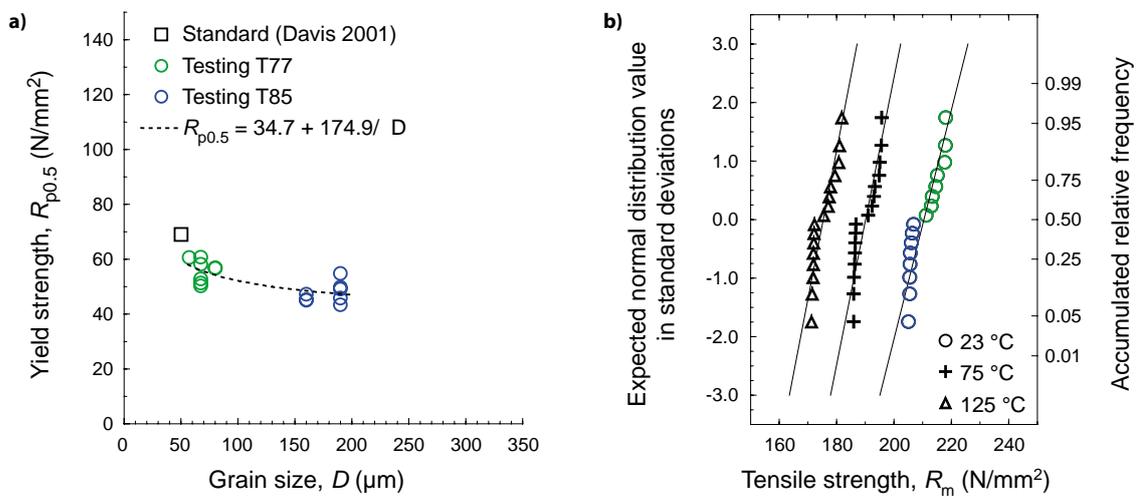


Figure 9-8. a) Yield strength as a function of grain size at room temperature. b) Measurements of tensile strength dependence on temperature.

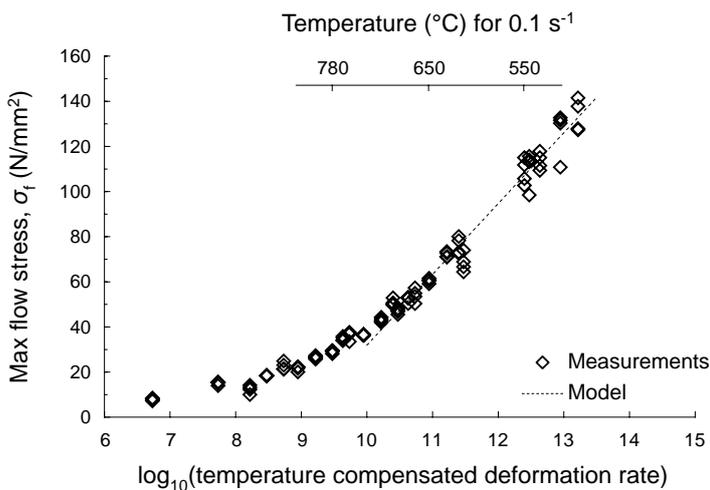


Figure 9-9. Yield stress of canister copper in hot working as a function of the temperature-compensated strain rate.

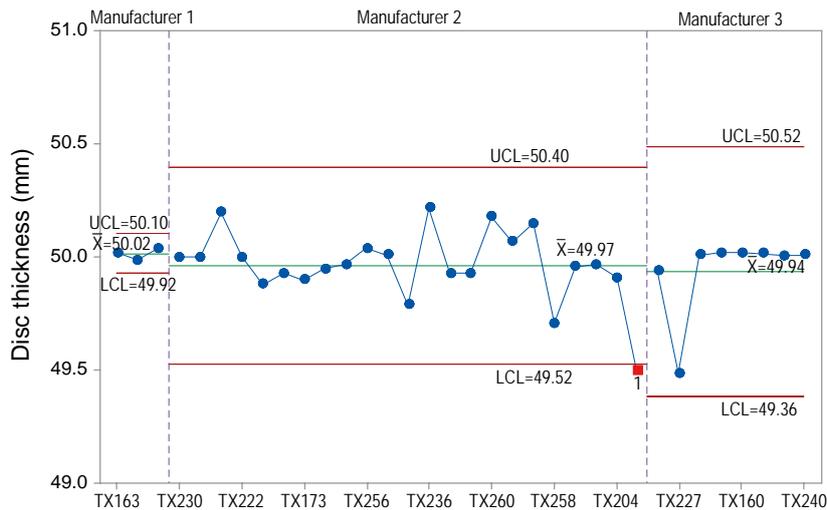


Figure 9-10. Chart for plate thickness of copper lids. Sorting is done per manufacturer and thereafter in the order of time of machining the lids. UCL denotes the upper control limit of the process, LCL, the lower control limit for each manufacturer (Jonsson et al. 2019).

Programme

SKB intends to describe how hardening mechanisms affect the strength of the canister copper based on measurements of yield strength, tensile strength, grain size and temperature, but also of other factors such as manufacturing processes. SKB will formulate requirements on both the handling of the canister in Clink and the yield strength of the copper in order to ensure canister strength during lifts and that no indentations will occur. SKB intends to develop models for yield stress and newly formed grain size in hot working based on the compression of small copper cylinders. Furthermore, SKB plans to investigate the requirement on the average grain size in copper components, which may lead to a need for formulating an additional requirement for sound attenuation in order to achieve sufficient sensitivity during ultrasonic testing.

SKB intends to study and clarify acceptance requirements regarding material composition (for example maximum oxygen content in the copper shell) and to supplement and justify the set of requirements for defects in the copper components (Jonsson and Rydén 2014). SKB plans to validate the finite element calculations underlying the calculated permissible defects through suitable practical experiments. SKB plans to study and improve the methods for quality assurance of the phosphorus content and other specifications of the chemical composition and density in the copper components, as well as methods for achieving a more even grain size.

The hot forming of lids and bottoms can be carried out as open-die forging or with a confining tool (impression-die forging). The choice of process and design is affected by the series size (Bodin 2003). SKB will develop the forging to improve the strength, reduce the spread in grain size in the finished product and minimise the risk of cold shuts. Possible and probable defects in the copper components will be analysed and classified. Acceptance criteria for defects by means of finite element simulations of extrusion and forging processes are being developed. Preliminary defect analyses and acceptance criteria will be presented in the PSAR. SKB plans to extrude copper tubes and to forge copper lids with suitable process parameters for serial production and to test and check against the requirements and acceptance criteria, to verify that the properties of the copper components are achieved and to be able to take corrective action in the manufacturing processes.

In addition, SKB is investigating the possibility to select pierce and draw processing as an additional reference method for the manufacturing of copper tubes. Pierce and draw processing results in a tube with an integral bottom. SKB and Posiva are carrying out the development of the pierce and draw processing in collaboration. It is now possible to produce canister copper that meets the manufacturing requirement for an elongation after fracture of at least 40 percent and the reference design requirement for an average grain size of at most 800 µm, but not always the manufacturing requirement for an average grain size of at most 360 µm. More work remains to develop the pierce and draw processing

and verify that an even grain size can be achieved also in the integral bottom of the tube, but also to determine what defects may occur during pierce and draw processing and formulate acceptance criteria for these defects. SKB also intends to determine how the verification of copper manufacturing by pierce and draw processing can be carried out.

9.4.2 Casting of the canister insert

In order for the canister to guarantee radiation safety during transportation and deposition and to contribute to a safe repository, SKB must show that the mechanical strength is sufficient in the entire insert and that no unacceptable defects or deviations remain in the insert.

Current situation

SKB has chosen a ferritic nodular cast iron for the canister insert, which balances strength and toughness in view of isostatic loads and the earthquake scenario in SKB's assessment of safety after closure. The material designation for SKB's nodular cast iron is EN-GJS-400-15C, from the standard SS-EN 1563. Nodular cast iron is also the type of cast iron that has the highest ageing resistance to for example blue brittleness. By casting the nodular cast iron around steel profiles, space for the fuel assemblies is ensured.

SKB's commissioned research at KTH and Frecast AB has investigated the structure and graphite nodule size distribution of SKB's test-manufactured nodular cast iron inserts, and has simulated the solidification and cooling processes. Figures 9-11 and 9-12 show different solidification processes for BWR and PWR inserts. As can be seen in Figure 9-11, PWR inserts cool much more slowly than BWR inserts. Small graphite nodules are formed during rapid solidification and coarser nodules during slow solidification, but the differences in nodule size do not completely explain the variations in mechanical properties (Tadesse and Fredriksson 2018).

Iron and steel may become harder and more brittle at storage after cooling or deformation. The relationships between brittleness and material composition are not fully clarified, but during the 20th century, the steel ageing tendency decreased through research and practical development by manufacturers. There are investigations that show that nodular cast iron in practice only becomes brittle at 400 °C (Yanagisawa and Lui 1983). Dynamic ageing may occur during tensile testing at an elevated temperature; it is identified as spikes (striations) in the tensile test curve and is called the Portevin-LeChatelier effect. Simplified, the striations can be attributed to dissolved substances interacting with the ongoing deformation of the metal's crystal grains, and both the diffusion and the deformation of the substances are temperature-activated. This effect does not result in any drastic reduction of the material's toughness (elongation after fracture).

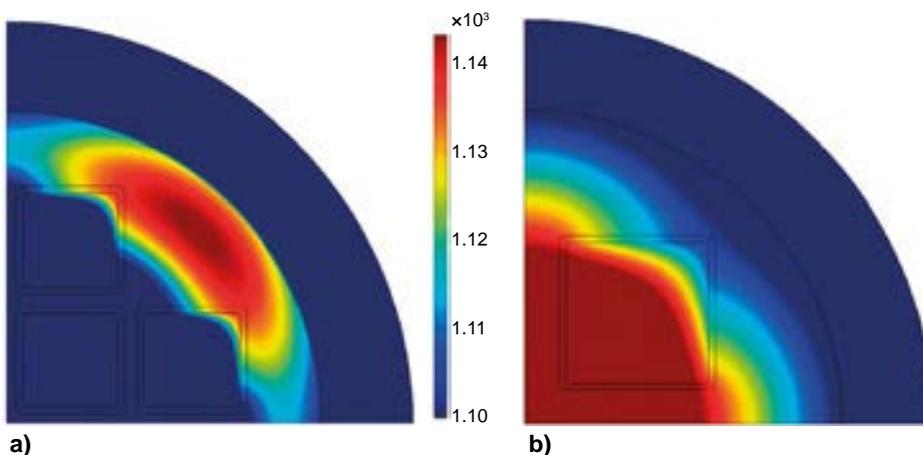


Figure 9-11. Temperature profile in the latter part of the solidification process.

a) BWR insert, $t = 1\text{ h }43\text{ min}$, b) PWR insert, $t = 8\text{ h}$. Figure from Tadesse and Fredriksson (2018) according to <http://creativecommons.org/licenses/by/4.0/>.

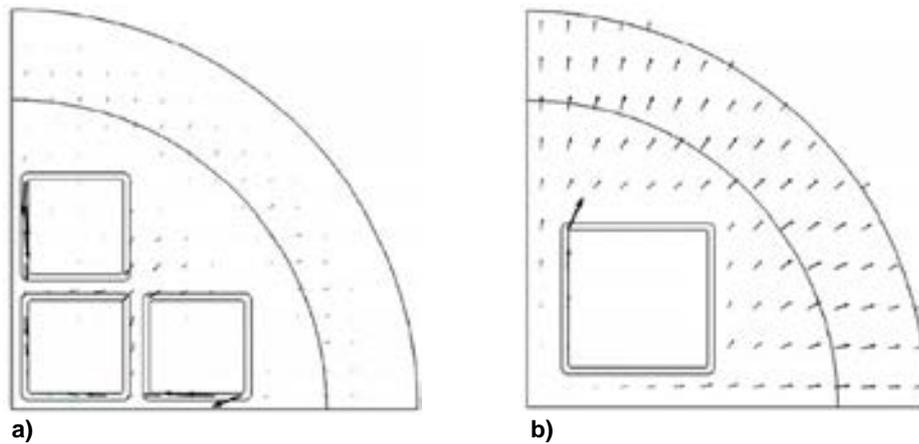


Figure 9-12. Heat flow before complete solidification. a) BWR insert, b) PWR insert. Figure from Tadesse and Fredriksson (2018) according to <http://creativecommons.org/licenses/by/4.0/>.

Static strain ageing only occurs for materials and temperatures that exhibit characteristic signs of ageing. Since no signs such as a significantly increased yield strength or striations have been observed during tensile testing of nodular cast iron, static strain ageing is not expected to occur if plasticisation occurs at temperatures below 125 °C (Sarnet and Holst 2017). After a survey of the available literature and the load cases in the final repository, SKB is of the opinion that neither static nor dynamic strain ageing will affect the mechanical properties or functions of the canister insert in the final repository. Firstly, plasticising loads are needed for the phenomena to occur, while the canister insert is designed so that plasticisation will not occur except in unlikely cases, and secondly, the temperature in the final repository is considerably lower than the temperatures at which in particular dynamic ageing occurs. Furthermore, nodular cast iron is the type of cast iron that is most durable against strain ageing (SKB 2017).

Programme

During 2018–2019, SKB has cast three BWR inserts and two PWR inserts in order to develop the casting processes and compare the quality levels of different foundries. The strength and structure of the nodular cast iron will be measured and analysed for several heights and sample positions. For these, the chemical composition, density and ultrasound properties will be measured, and fracture surfaces will be characterised for a suitable selection in relation to strength results.

Casting a steel structure into the nodular cast iron is effective but unusual. SKB will investigate how the structure interacts with the nodular cast iron melt during casting and whether the structure or material may affect the properties of the nodular cast iron.

Even though SKB believes that neither static nor dynamic strain ageing of the nodular cast iron will affect the canister insert's mechanical properties or functions in the final repository, the nodular cast iron will be subjected to tensile tests at elevated temperatures. Dynamic strain ageing and possible temperature-dependent brittleness will be evaluated for two different nodular cast iron inserts according to the experiment plan in Table 9-1.

SKB will further develop the manufacturing technique and clarify the manufacturing requirements on the steel parts of the insert. Manufacturing requirements for the nodular cast iron insert are being investigated. SKB plans to update the set of requirements for non-destructive testing of the insert and to link the defects more clearly to the eventually selected manufacturing methods.

Table 9-1. Experiment plan for investigation of the dynamic strain ageing of nodular cast iron.

Insert	Height (mm)	Specimen	Temperature (°C)	Rate (1/s)
i76	2306	2	20	2.8×10^{-4}
i76	2306	5	125	2.8×10^{-4}
i76	2306	11	300	2.8×10^{-4}
i76	2306	14	400	2.8×10^{-4}
i76	2336	2	400	2.8×10^{-4}
i76	2336	5	20	2.8×10^{-4}
i76	2336	11	125	2.8×10^{-4}
i76	2336	14	300	2.8×10^{-4}
i76	2377	2	300	2.8×10^{-4}
i76	2377	5	400	2.8×10^{-4}
i76	2377	11	20	2.8×10^{-4}
i76	2377	14	125	2.8×10^{-4}
i76	2419	2	125	2.8×10^{-4}
i76	2419	5	300	2.8×10^{-4}
i76	2419	11	400	2.8×10^{-4}
i76	2419	14	20	2.8×10^{-4}
i57	2336	2	300	7.7×10^{-5}
i57	2336	5	300	2.8×10^{-4}
i57	2336	11	300	7.7×10^{-5}
i57	2336	14	300	2.8×10^{-4}
i57	2372	2	300	2.8×10^{-4}
i57	2372	5	300	7.7×10^{-5}
i57	2372	11	300	2.8×10^{-4}
i57	2372	14	300	7.7×10^{-5}
i57	2372	2	300	2.8×10^{-4}

9.4.3 Welding

Friction stir welding (FSW) was chosen in 2005 as the reference method for sealing of the canister, after an evaluation against the alternative technique of electron beam welding. In 2014, Posiva also selected FSW as the reference method for sealing of the canister. Since then, SKB and Posiva have worked together to develop and verify FSW.

FSW joins the material in the solid state, which differentiates the process from classical fusion welding. By letting a probe rotate, the material is heated by friction. The increasing temperature makes the material softer so that, if the temperature is sufficiently high, it starts to move with the probe without the material melting. When the probe is then moved along the joint line, a homogeneous joint is created. FSW creates a joint with material properties comparable to the parent metal (SKB 2010b).

Current situation

After the RD&D Programme 2016, development has focused on both industrialisation and automation of the welding process, as well as investigation of oxide streaks in the weld metal.

In this period, the work on oxide streaks has had the purpose of understanding the effect of oxide particles on mechanical properties, mainly toughness (ductility) but also corrosion. This has been carried out by systematically varying the oxide thickness on joint surfaces prior to welding. After that, tensile testing and creep testing have been carried out on welded material. The corrosion properties have been investigated by means of electrochemical studies of weld material and by calculations of the chemical dissolution of oxide bands. These results have been used to define a preliminary process window for the oxygen content in the gas shield during welding, < 100 at.ppm oxygen (Björck et al. 2019).

As regards the automation and the so-called cascade controller (which alters the welding tool's rotational rate), it keeps the tool temperature around the whole joint line within ± 5 °C. The controller has been developed so that it will work even if the tool temperature signal vanishes, a so-called fail-safe function that has been tested and verified at two separate occasions, where the controller controls only by heat input when there are no temperature signals.

The so-called depth controller, which alters the welding tool's pressure against the canister, has since the RD&D Programme 2016 been verified during several full circumferential welds as an additional step in automation. The results (Cederqvist et al. 2018) show that with a probe length of 51 mm and a centred height position, the joint line hooking can be limited to below 2 mm even though the shoulder depth has been varied in the range 1.2–3.4 mm.

In order to investigate tolerances for lids and tubes and to simplify the joint line geometry, a new straight joint line design has been test-welded (instead of the current design with a kink). Both non-destructive and destructive testing show the same results as with the current design, i.e. no discontinuities except a joint line hooking less than 2 mm.

Programme

SKB and Posiva will continue to verify the welding procedure and prepare it for qualification. Qualification takes place by pre-production testing. The focus will be on the new joint line design and manufacturing tolerances between lid and tube, and on defining process windows for all parameters in preparation for the planned qualification procedure.

9.4.4 Inspection and testing

The canister and canister components will be inspected to verify that the manufacturing processes work as they should (controlling inspection), that the properties assumed in the design specifications are achieved and to ensure that no errors or deviations remain (acceptance inspection). Controlling inspections can with advantage be carried out by direct measurement during the manufacturing process.

Current situation

The canister and canister components can be inspected and tested mechanically and physically and be analysed chemically, using standard methods. The mechanical properties can be tested in test volumes connected to the canister components by means of cast-on specimens and sample rings from excess lengths of canister components. Even if the measured properties in principle apply to the investigated test volume, the properties of the canister and canister components can be determined by technical reasoning.

In order to detect defects in the weld, copper components and nodular cast iron inserts, physical testing methods based on ultrasound, magnetism and radiography have proved appropriate. SKB has customised such standard methods based on the geometry, dimensions and material of the canister, and the size, nature and orientation of defects. With physical testing methods such as ultrasonic testing and radiography, possible defects can be detected inside the canister weld or canister components. Surface-breaking defects can be detected with a magnetic testing method (eddy current).

Finite element calculations of the shear load case indicate that an elongation after fracture of 3 percent is a reasonable minimum requirement to impose on the nodular cast iron insert (Jonsson et al. 2018).

In the forging of copper lids, cold shuts may be formed under unfavourable tool conditions (Kopp et al. 1988). Small contact lengths between the die and material and small tool radii increase the risk of cold shuts forming, as shown schematically in Figure 9-13a. The size and orientation of the defect entail that a copper lid with cold shuts cannot be accepted. SKB has customised eddy-current testing that successfully detects such cold shuts in copper lids (van den Bos 2018). Figure 9-13b shows how the indication of the eddy-current testing corresponds to cold shuts.

SKB has developed a combined set of requirements, including acceptance criteria for the canister and its components. As a consequence of ultrasonic inspection of the insert's central parts having been questioned in SSM's review report (SSM 2018), a study has been initiated with the aim of applying radiographic inspection techniques to these areas. Preliminary results of this study indicate possibilities for inspection using radiography in these areas, especially considering that the acceptance criteria could probably be revised.

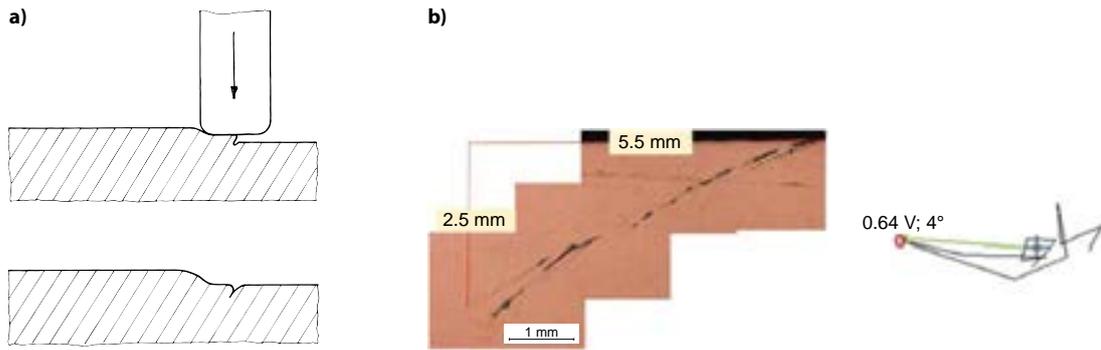


Figure 9-13. a) Formation of cold shuts due to unfavourable tool radius and feeding. b) Indication in eddy-current testing at 1 kHz and the corresponding cold shut (3 mm) (van den Bos 2018).

Programme

Testing of the mechanical properties of copper components and nodular cast iron inserts takes place in test volumes in connection to the canister components. SKB plans to study the feasibility of testing methods in the test volume and what extent of testing is needed. For example, tensile testing is feasible and decisive for each manufactured copper tube, to check that the manufacturing process has functioned as intended. On the other hand, SKB needs to investigate how tensile testing measurements are related to the canister copper's strength.

SKB plans to conduct studies and development regarding the inspection programme for the canister. As a starting point for these studies, which to some extent are conducted in cooperation with accredited inspection and qualification bodies, SKB intends to use the existing regulation SSMFS 2008:13, which applies to mechanical equipment in nuclear power plants, with compensatory measures. Other guidelines and standards (IAEA) may also be used to provide principles and strategies within subareas where SSMFS 2008:13 is not considered to be fully applicable for the canister in a KBS-3 system. The purpose of an inspection programme is normally to provide a system and a strategy for how fulfilment of requirements and quality assurance of, for example, a product is to be obtained. Within the system it must then also be studied in general how the inspection programme activities should be reported and documented, since this is a part of quality assurance.

SKB intends to propose detailed acceptance criteria for non-destructive testing. This will be carried out by means of further systematic analyses of possible and likely defects in the canister components and the seal weld, which are compiled in defect descriptions. These will then be linked to the calculations made regarding the handling of the canister in manufacturing and deposition, and the canister post-closure safety. In parallel with this work, further development is pursued of the technique for non-destructive testing for detection of defects linked to these more detailed acceptance criteria. Development work is focused on the preparation of supplementary ultrasonic technology for inspection of the insert's volume, and on the evaluation of alternative techniques for surface inspection of the insert. In addition, work has been initiated with a focus on development of eddy-current techniques for surface inspection of the copper components.

The development of non-destructive testing will also be more specifically focused on techniques for sizing linked to the defects that are expected to occur in the canister components and welds and the acceptance criteria that are formulated. In addition to the inspection techniques developed for final inspection, a review is planned of the inspection currently carried out of starting material such as copper tubes. Based on this review, customisation of standard methods and possibly technology development will be initiated.

To substantiate the developed techniques for non-destructive testing, SKB is conducting further studies of possible variations of the geometry, material and defects in the canister components and welds and how they may affect the canister's testability. Based on these studies, requirements on for example sound attenuation and surface finish may be defined. To substantiate the inspection techniques even further, work continues on ultrasound simulation of both the inspection technique and its response linked to artificial and real defects.

With respect to other inspections of the canister, for example regarding dimensions, a review will be made of tolerances. Based on this review, development of techniques for the verification of compliance with requirements may be initiated.

In the encapsulation plant, the weld quality will be inspected by ultrasonic and radiographic inspection. The methodology for inspection and testing of the canister lid weld and requirements on the weld has to be determined to a level of detail that makes them possible to use for the design of Clink. It cannot be excluded that defects will have to be characterised and determined in height, and thus techniques for non-destructive testing remains to be developed. Since oxides cannot be detected with these methods, the friction stir welding process has to be qualified, and SKB will therefore formulate criteria for the maximum permissible oxygen content in the gas shield.

10 Cementitious materials

SFR contains large amounts of cementitious materials in waste matrices, engineered barriers and other structures. Also, the concept for SFL that has been evaluated in the recent SFL safety evaluation also includes large amounts of cementitious materials. Finally, also the Spent Fuel Repository will contain cementitious materials in plugs, grouting and rock support.

Chapter 4 briefly presents SKB's planned development programme for cementitious materials, linked to the design of repository structures, material development and production methods for construction and closure of the three repositories. Furthermore, it provides a general description of the programme for research on processes of importance for the safety after closure of the systems. This chapter provides a more detailed description of SKB's programme in these areas. The research programme aimed at gaining a better understanding of processes is presented as a whole for the three repositories (Section 10.1), while the programme for design, material and production methods is described separately for SFR (Section 10.2), SFL (Section 10.3) and the Spent Fuel Repository (Section 10.4).

10.1 Cementitious materials – evolution after closure

This section describes the scientific research SKB plans to conduct in order to gain a better understanding of how the function of cementitious materials in the repository environment changes during the time periods covered by the safety assessments.

10.1.1 Groundwater impact

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 concrete's pore structure, which affects the hydraulic and mechanical properties of the material. Mass transport properties in the concrete matrix are also affected.

Current situation

In the RD&D period, SKB has worked to deepen the process understanding of degradation of concrete under repository conditions, both by means of modelling and through experimental studies.

The completed modelling started from the design of the rock vault for core components, BHK, in SFL and included studies of the evolution of chemical and hydraulic properties of concrete in a perspective of up to one million years under the influence of hydro-chemo-mechanical processes (Idiart and Shafei 2019, Idiart and Laviña 2019). Furthermore, studies of the effects of alternative concrete compositions (Idiart et al. 2019b) have been carried out; see also Section 10.1.6.

The following principal conclusions can be drawn from the completed studies:

- The primary degradation process is leaching of calcium, which leads to a gradual loss of portlandite and C-S-H gel.
- The degradation rate in the backfill in BHK is very low, mainly due to the low water flows in the concrete backfill.
- The concrete's initial porosity and transport properties have a larger impact on the extent of the degradation over time than the chemical composition of the concrete in the cases studied here.

These studies have also demonstrated that mechanical loads are expected to cause only very limited damage in connection with the interface between the bedrock and backfill in BHK. The only exception is glaciation loads, which may cause more severe damage in a larger part of the backfill.

All in all, these studies show that the properties of the concrete in BHK in SFL will only be affected at an extremely slow rate during the period after closure and the repository's capability of limiting the release of radionuclides can therefore be expected to be good during the periods covered by the assessment of repository safety after closure.

Applied to 2BMA, the results indicate that the concrete containing large quantities of finely ground limestone that has been developed within the framework of SR-PSU will have a slower degradation rate than the concrete used in the construction of the existing SFR facility, which is the basis for all current analyses.

Programme

Based on the results obtained in the studies that were performed within the framework of the SFL Safety Evaluation, SKB intends to continue to develop its programme for studies of interactions between groundwater and concrete under repository conditions. The programme is currently not finally established, as all results from ongoing studies are not yet available. Of the possible development areas identified, however, the following are worth mentioning:

- Evolution of the porosity of materials as a function of precipitation and dissolution reactions and the connection 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.

10.1.2 Modelling of gas transport

The gas that is formed by degradation of materials in the final repositories during the time after closure may build up an internal pressure if a well-functioning system for gas transport is not available. The pressure may affect the structural integrity of the concrete structures. In order to avoid detrimental loads on the barriers, gas transport out of the waste domain must be facilitated.

Current situation

With simplified models for metal corrosion and gas transport, SKB has previously made an analysis of how gas production may affect the structural integrity of the concrete barriers in 2BMA (Eriksson et al. 2015). According to the analysis, the corrosion of aluminium and zinc in the waste has the greatest impact on the barriers. The calculated limit values for how much aluminium and zinc the waste can be permitted to contain are close to the estimated quantities of these metals in the inventory (SKB 2013a, 2019b).

Recently, SKB has also completed a study concerning transport of gas in a multiphase flow through the barriers in BHK, where the main source of gas formation is corrosion of steel in the absence of oxygen (Silva et al. 2019a), but the further transport of gas through the bedrock has also been studied (Silva et al. 2019b). The main conclusions of these studies are:

- Gas formation does not significantly affect groundwater flows in the near field or the hydraulic properties of the concrete (Silva et al. 2019a).
- The presence of a deformation zone with high permeability comprises a preferential transport route for the gas formed in the repository and reduces the maximum gas pressure in the repository (Silva et al. 2019a).
- The model's description of gas flow distribution in the bedrock depends on whether it is described with a homogeneous or a heterogeneous model (Silva et al. 2019b).
- The transport properties of the bedrock affect the maximum gas pressure that can arise in the rock vault (Silva et al. 2019b).

SKB has also commissioned a study of the influence of the backfill on groundwater flows and radionuclide transport starting from the design of 2BMA (von Schenck et al. 2018). The study investigated the release of radionuclides as a function of the properties of the backfill around the concrete caissons, but the effects of gas formation processes or gas flows were not taken into account.

Programme

In the coming RD&D period, SKB needs to gain a deeper understanding of two-phase flows of hydrogen and water through both the rock and the barrier materials (concrete and bentonite). Initially, as a continuation of the work carried out by von Schenck et al. (2018), the gas and water flow in the near field in SFR will be modelled with a two-phase flow model for porous media. In such a model, water flow is also affected by the gas flow. One of the most important parameters of two-phase flow modelling is the so-called relative permeabilities of the two phases. There are several models that describe these as a function of the degree of saturation. SKB plans to study the differences and similarities between the different models.

In the longer term, models where the two phases (gas and water) are described separately may also be required. In particular, this applies if the rock is described as a discrete fracture network instead of a porous medium. Such models, too, need to be investigated in terms of their dependence on numerical conditions.

When the ability to simulate and calculate the gas transport capacity of both the rock and the barriers has improved and can reflect experimental results, the methods may be used to assess how engineered gas release systems should be designed to expel the gas formed while affecting the flow-limiting ability of the barrier as little as possible; see also Section 10.2.3.

10.1.3 Impact of organic waste degradation

Organic materials that degrade in a cement matrix may affect the properties of the cementitious materials. This can be reflected in a change in the pore water composition but could also affect the ability of the concrete to limit the release of radionuclides.

Current situation

The project Concrete and Clay, which has been pursued by SKB in 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- and intermediate-level waste, and how they are dispersed in a concrete matrix.

In 2014 and 2016, samples were retrieved comprising steel containers with groundwater from repository depth, organic material representative of low- and intermediate-level waste, and a small quantity of crushed cement paste (Wold 2014, Dvinskikh et al. 2016). Analyses showed that the degradation of organic waste was very limited, and only in a few cases could very low amounts of degradation products be identified.

Programme

SKB has initiated the retrieval and analysis of additional samples from the Concrete and Clay project comprising organic material embedded in concrete cylinders. The focus of these studies is the extent to which degradation products from the materials are spread in a cement matrix. Additional retrieval and analyses of similar experiments in this project will be carried out in the coming five-year period.

10.1.4 Impact of corrosion of metallic waste

As metallic material corrodes in the presence of cement, the corrosion products can react with the cement minerals and thereby alter the properties of the cementitious materials. In addition, the mechanical pressure that occurs if voluminous corrosion products congregate on or around the metal surfaces may lead to crack formation in the matrix.

Current situation

Studies of the corrosion of metallic materials in a cementitious material and of how the corrosion products interact with the cement matrix are also being carried out in the Concrete and Clay project.

The retrieval and analyses that were carried out in 2015 (Kalinowski 2015) have been followed up with a quantitative long-term study of the corrosion of aluminium and zinc in a cementitious environment. It will continue even into the coming RD&D period before the final results are reported.

The study includes both embedded samples and samples placed in cement-buffered water in closed containers. Preliminary results indicate a high initial corrosion rate, which then declines sharply as a layer of corrosion products is formed around the sample.

Programme

In the RD&D period, SKB intends to retrieve and analyse an additional specimen of concrete with metallic material similar to that studied by Kalinowski (2015) as a part of the Concrete and Clay project. This analysis will provide additional information about the corrosion rates of embedded material and how corrosion products spread in the cement matrix over long periods of time, which may lead to a better understanding of this process. The time for future retrievals and analyses has not yet been finally determined.

10.1.5 Impact of bentonite on cementitious materials

In the existing silo in SFR and the planned rock vault for legacy waste in SFL, BHA, contact between cement and bentonite will occur. When these materials become water-saturated some time after closure, the chemical interactions may lead to changes in the cement material's composition, properties and structure. This process may also include degradation products from the waste.

Current situation

In a number of previous studies, SKB has investigated the interactions between bentonite and cementitious materials and their impact on safety after closure for the silo in SFR; see for example Höglund (2001), Gaucher et al. (2005) and Cronstrand (2007, 2016), of which Gaucher et al. (2005) is the most extensive study.

In 2014, SKB also deposited a large number of samples within the Concrete and Clay project in the Äspö HRL for studies of interactions between different types of bentonite and cement paste consisting of pure Portland cement as well as of low-pH paste (Mårtensson 2015). A major fraction of the cement samples, which consisted of small cylindrical pellets, also contained powder of one of several metals and metal salts, representative of low- and intermediate-level waste, but also samples without such materials were used (Mårtensson 2015).

In 2018, one of a total of five packages was retrieved and analysed (Figure 10-1). The studies could not detect any release of the tracers embedded in the small cement pellets and thereby spread in the bentonite, other than for the element caesium, for which a gradient approximately 10 mm wide could be observed in the bentonite (Kalinowski 2018).

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 contents 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.

Since 2015, SKB has participated in the international research project Cebama (www.cebama.eu), in which interactions between different types of cementitious materials and adjoining materials such as clay or bedrock material of various types are being studied. The project, in which around twenty research institutes and about ten waste organisations have been active, is in its final phase and compilation of the various research initiatives and final reporting is in progress.



Figure 10-1. Retrieval of a package of 30 bentonite blocks (left-hand picture). Each block also contains four small material samples, which is why the blocks are sorted into four parts and vacuum-packaged before analysis (right-hand picture).

Programme

SKB is planning further retrieval of samples within the Concrete and Clay project for continued studies of interactions between cement and bentonite in the presence of metals and metal salts, representative of low- and intermediate-level waste. The focus of the analysis will be studies of ion transport and mineral transformations in the interfaces between cement and bentonite, and dispersion of degradation products from the materials that are mixed into the small cement pellets in the bentonite. The next retrieval will only be carried out around 2021 and will probably include two experiment packages. The two last packages will be retrieved and analysed as late as possible, but the exact time has not yet been determined.

10.1.6 Impact of additives

When manufacturing cementitious materials, different types of additives may be used, in addition to cement, water and aggregate. These materials – for example silica, finely ground limestone or fly ash – can be added either in the cement production or when mixing the cementitious material. The purpose of these additives can be to reduce the environmental impact of the material or to control the properties of fresh and hardened material, but there may also be financial reasons. Regardless of the reason for using additives, they will not only affect the composition and properties of the fresh and hardened material, but may also affect how its properties change after closure.

Current situation

SKB has over the years commissioned a number of studies on the long-term chemical transformation of concrete in repositories for low- and intermediate-level waste; see for example Lagerblad and Trägårdh (1994) and Höglund (2001, 2014). These studies have been based on a cement composition equivalent to construction cement from Degerhamn. This is a pure so-called Portland cement without additives, corresponding to the cement type CEM I, which was used in the construction of the concrete structures in the existing repository parts in SFR.

Within the programme aimed at developing concrete for the caissons in 2BMA, a type of concrete containing relatively large quantities of finely ground limestone has been developed (see Section 10.2.1). As this additive can be expected to affect the long-term chemical and physical evolution of the material, a number of studies have been initiated and carried out during the RD&D period. These studies have included modelling of the expected chemical and hydraulic evolution of this concrete, but also of concrete with other additives (Idiart et al. 2019b). In addition, an experimental study of the properties of the concrete for 2BMA is in progress, with the purpose of creating a foundation for a well-described initial state of the concrete structures made with this type of concrete.

Idiart et al. (2019b) 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 that were investigated in this study, the concrete developed for 2BMA exhibited the best properties, which was attributable to its very low initial porosity rather than to its chemical composition.

Preliminary results from the experimental study also show that the properties of the concrete for 2BMA correspond well to those assumed in the assessment of repository safety after closure.

Programme

In this RD&D period, SKB plans to complete the modelling studies to evaluate the properties after closure of cementitious materials with altered binder composition. These studies will provide a better understanding of how the use of future materials may affect the safety of the repositories after closure. SKB also intends to complete and report on the experimental study of the properties of the concrete for 2BMA.

10.1.7 Freezing

When the pore water in a cementitious material freezes, it will expand and the material will be exposed to an internal pressure. If the material is sufficiently water-saturated and a sufficiently large fraction of the pore water freezes, the internal pressure may become so large that cracks form or that the material completely falls apart. The temperature at which this occurs is mainly dependent on the pore structure and the degree of water saturation.

Current situation

During the years 2007–2013, SKB commissioned a number of studies regarding freezing of the pore water in concrete and its impact on the properties of the concrete at temperatures that can be expected to prevail in conjunction with permafrost reaching repository depth in SFR (Emborg et al. 2007, Thorsell 2013, Tang and Bager 2013, Pålbrink and Rydman 2013).

Emborg et al. (2007) and Thorsell (2013) demonstrate that a sufficiently large fraction of the pore water has frozen at a temperature of between -3 and -5 °C for the concrete to be severely damaged. These results could not, however, be confirmed by Tang and Bager (2013) or Pålbrink and Rydman (2013). These studies showed instead that if concrete freezes in a volume that does not permit volume expansion corresponding to an environment that can be expected to prevail in conjunction with permafrost, it will not break in the manner described by Emborg et al. (2007) and Thorsell (2013).

All the above cited studies were conducted on very young concrete, which was not exposed to the mineral and structure transformation that can be expected during the long periods of time that are relevant in the final repository context. A more likely scenario is that the concrete, as a result of these processes, has a different pore structure at the first freezing than exhibited by the fresh concrete. This means that the concrete's freezing properties, at the time of the first permafrost, will probably not correspond to the properties exhibited by the investigated materials in the studies referenced above.

SKB has therefore commissioned two studies of the freezing properties of concrete that was leached with the method developed by Babaahmadi (2015) and thereby obtained a pore structure corresponding to that of aged concrete (Karlsson 2017, Fridh 2017).

Karlsson (2017) showed that the porosity increased by almost 25 percent in the leached concrete compared with the non-leached material. This resulted in a change of the material structure, clearly measurable by ultrasound, which was further affected by the subsequent freezing. Despite this increase, the specimens retained their shape after freezing to -10 °C, and no structural collapse could be observed.

Fridh (2017) carried out a more comprehensive study in which there was no evidence that leached concrete will fall apart completely when it is exposed to temperatures down to -3 °C.

SKB has also carried out a supplementary study where the effects on SFR due to freezing of concrete have been studied; see Näslund et al. (2017) and the references therein. The study, which is described in the supplement (Brandefelt et al. 2013), shows that the coming 50 000 years will be dominated by warm climate. However, it cannot be ruled out that the temperature in the repository may be as low as $-3\text{ }^{\circ}\text{C}$ during a potential cold period around about 54 000 years.

Programme

At present, SKB has no programme for studies of the freezing properties of concrete. This is justified since the completed studies have shown that even highly leached concrete will retain its integrity at the temperatures that can be expected to occur in SFR in conjunction with permafrost, and because dose calculations have shown that even the earliest permafrost will not affect the time for reaching the maximum dose or the level of this dose. As regards SFL, the great repository depth entails that the probability of a permafrost affecting the properties of the concrete barriers will be significantly lower than for SFR. SKB is following the research in the area and does not rule out future efforts if such needs should be identified.

10.1.8 Internal and external loads

During construction and operation and in conjunction with closure and backfilling, the concrete structures will be exposed to loads that could lead to formation of cracks. Apart from this, the concrete structures may also be subjected to both internal and external loads during the period after closure as a consequence of for example gas production and swelling waste, or as a consequence of rock breakout and of load from the backfill material, all of which could lead to crack formation.

Current situation

SKB has previously studied the strength of the concrete structures in the repository. The outer silo walls have proved to be resilient to both internal and external loads (von Schenck and Bultmark 2014).

Recently completed studies (Idiart et al. 2019a) within the safety evaluation for SFL have also demonstrated that mechanical loads are only expected to cause very limited damage in connection with the interface between the rock and concrete backfill in BHK. The only exception is glaciation loads, which might cause more severe damage in a larger part of the backfill.

During the RD&D period, SKB has decided to adjust the reference design for 2BMA, which entails an increase of the thickness of the outer walls and base slab of the caissons, and the introduction of inner walls. The adjustment results in a structure that can handle the external loads without the need for grouting of the waste. This solution also creates room for swelling of the waste without affecting the concrete structure. In order to handle the gas pressure, SKB has studied different designs of systems for gas transport; see also Section 10.2.3.

SKB has decided to carry out certain adjustments in the reference design of 1BMA as well. This entails that the concrete structure's outer walls and lid will be reinforced by an external concrete structure. As a result of the planned reinforcements of the concrete structure in 1BMA, grouting of the waste is no longer required in order to handle the external loads that will affect the concrete structure during backfilling and resaturation (Elfving 2017). For this reason, SKB no longer plans to carry out grouting of the waste in 1BMA, but the possibility remains if the need should arise again.

Programme

SKB will follow up previous calculations and investigations of the concrete structures in the existing SFR with a study of the base slabs in the waste vaults. In some cases, the base slabs may be locally subjected to loads exceeding the specified design-basis imposed loads in the construction drawings. SKB will study the effects this may have and whether the possible fracturing of any base slab entails that the initial state assumed in the safety assessment is no longer achieved. The design-basis load cases unilateral water pressure, internal swelling pressure and gas pressure may also require updating.

10.2 Design of concrete structures and materials in SFR

This section presents SKB's programme for the design of concrete structures, material development and production methods for SFR.

10.2.1 Waste vault for intermediate-level waste, 2BMA

The waste vault for intermediate-level waste, 2BMA, will consist of a more than 250 metres long rock vault in which a number of free standing caissons will be built of unreinforced concrete (Figure 10-2). During the RD&D period, SKB has decided to adjust the reference design of 2BMA, entailing an increase of the thickness of the outer walls and base slab of the caissons and the introduction of inner walls. This means that the waste will be deposited in shafts corresponding to those found in the silo today.

The introduction of inner walls and the increase of the thickness of the outer walls, lid and base slab of the caissons result in a structure that can handle all loads during all repository phases. This means that grouting of the waste is no longer required in order to achieve structural stability over time. Grouting of the waste in 2BMA is therefore no longer planned. This solution also creates room for swelling of the waste without affecting the concrete structure. In order to handle the gas pressure, SKB has studied different designs of systems for gas transport; see also Section 10.2.3.

Current situation

Since the beginning of 2015, SKB has conducted a programme for development of materials and technology for construction of the caissons in 2BMA. This work started from the set of requirements, the reference design and the method for construction that were in effect at the time of formulation of the programme (SKB 2015b). The starting point in the design of the concrete recipe was to use aggregates made of excavated rock spoil from the extension of SFR as far as possible, and that a minimum of chemical additives would be used.

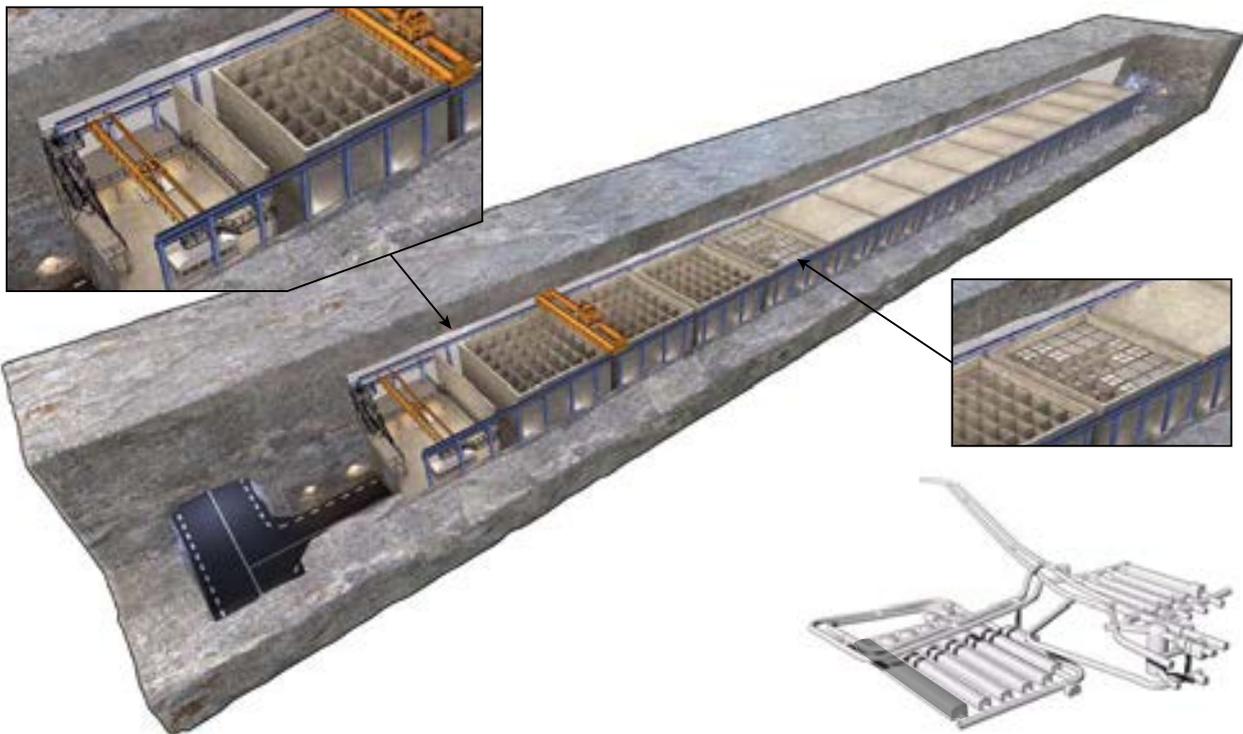


Figure 10-2. 2BMA – waste vault for intermediate-level waste in the extended part of SFR.

The work was carried out in a stepwise process, which includes the following areas:

- Studies of the rock material that will arise during excavation of the new rock vaults with respect to using it as aggregates in concrete and identification of suitable quarries to use in the development work (Lagerblad et al. 2016).
- Development of concrete in a concrete laboratory and upscaling to production plant scale for concrete (Lagerblad et al. 2017).
- Casting of large concrete structures in the shape of corner sections whose design and dimensions correspond to those planned for the caissons in 2BMA. The work is carried out in a representative rock vault environment in the Äspö HRL (Mårtensson and Vogt 2019).
- Casting of a caisson in quarter scale according to the adapted reference design (Elfving et al. 2017) in a representative environment in the Äspö HRL, including manufacturing of concrete in a mobile concrete station placed in connection to the tunnel entrance. The work was carried out in the end of 2018. Follow-up and analysis of the long-term properties of the structure is currently in progress. The final results are expected in 2020.

In the completed studies, a concrete has been developed that meet al. property requirements on both fresh and hardened concrete. The concrete uses to 100 percent aggregate made of crushed rock similar to that which can be expected to arise during excavation of the extension of the SFR facility. In addition, a low cement content is used, which limits the thermal output during the hydration in combination with a relatively low content of superplasticisers and retarders. Commercial very finely ground limestone filler is also added to achieve a sufficient volume of paste, so that the fresh concrete will have sufficient workability. For a more detailed description of the development work, see Lagerblad et al. (2017).

Material development was followed by casting of two large components corresponding to corner sections of caissons in the Äspö HRL, according to the reference design applicable at the time (Figure 10-3). During this work, one section was constructed according to the reference method, which entailed constructing a base slab and an L-shaped wall in one uninterrupted sequence. Some time later, a second section was constructed including an L-shaped wall, which was built on the existing base slab. The joint between these was made tight with a joint seal of copper sheet. Figure 10-3 shows the structure after casting of section 1 (left) and section 2 (right).



Figure 10-3. Caisson section 1 including base slab and L-shaped wall (left), and caisson section 2 including L-shaped wall, which was built on the base slab of section 1.

The first casting, where the base slab and walls were constructed in one uninterrupted sequence, involved unforeseen technical difficulties in building the caissons according to the reference design applicable at the time. However, material studies completed after the second casting showed that the hydraulic properties of the joint between the base slab and wall corresponded to those of concrete without a joint. This indicates that the construction of the base slab and walls at two separate occasions in accordance with conventional techniques could be a potential solution, which would simplify the casting procedure considerably. For a more detailed description of this work, see Mårtensson and Vogt (2019).

The last stage in the programme for development of material and technology for construction of the caissons in 2BMA included casting of a quarter section of an unreinforced caisson in accordance with the current reference design (Elfving et al. 2017) and was carried out in the Äspö HRL in the autumn of 2018. In addition to casting and studies of the properties of a concrete structure, this work also investigated the effects of manufacturing the concrete at a mobile concrete station directly adjacent to the tunnel entrance. Issues that were studied were the effect of the need for additives, and transportation and production aspects.

The work included casting a caisson in quarter scale compared with the planned caissons in 2BMA (Elfving et al. 2017) with the outer dimensions $8.1 \times 19.3 \times 4.5$ metres ($l \times w \times h$), where the base slab and walls have a thickness of 600 mm and 680 mm, respectively. The caisson foundation consisted of an unreinforced concrete slab which was cast directly on cleared bedrock in order to achieve a stable foundation where no subsidence will take place. Before casting the base slab, the foundation was covered with reinforced plastic foil in order to reduce adhesion and thereby allow unrestricted movements of the caisson base slab due to temperature variations.

The walls were built four weeks after the base slab. Before casting the walls, the base slab was heated carefully by means of heating mats in order to expand it to a limited degree. By controlling the cooling of the base slab after the completed casting of the walls, the effect of temperature-induced wall shrinkage could be limited and the development of stresses in the joint be prevented. The risk of crack formation could thereby be minimised.

The work was carried out in the autumn of 2018 and casting included a total of about 250 m^3 of concrete, of which 100 m^3 were used for the base slab and 150 m^3 for the walls. Figure 10-4 shows the outside and inside of the completed caisson.

A follow-up of the structure's properties was initiated immediately after the completed casting and studies of material properties have been conducted on samples produced in conjunction with the casting process. The follow-up is expected to proceed during much of 2019 and the final results of the project will therefore be reported in 2020.



Figure 10-4. Cast caisson, outside and inside.

Programme

SKB's programme for further work on the development of the caissons and the production method for 2BMA over the coming years includes the following activities:

- Further development of the design and technology for construction.
- Analysis, calculations and determining the dimensions of design details (such as the casting joint and the impact of external load).
- Further development of the production method for construction.
- Verifying tests of design and production technology.
- Preparation of operational and maintenance programmes.

10.2.2 Waste vault for reactor pressure vessels, BRT

The waste vault for reactor pressure vessels, BRT, will consist of a more than 250 metres long rock vault in which a compartmentalised concrete structure will be built of reinforced concrete (Figure 10-5).

The waste containers are placed in the compartments and are grouted with high strength concrete in conjunction with closure. Thus, a network of load-bearing walls is formed, which can take up the external loads.

Current situation

In the RD&D period, SKB has decided to adjust the reference design of BRT in order to adapt it to the decision to segment the reactor pressure vessels, after which the segments are packed in containers before final disposal.

This means that the programme related to disposal of whole reactor pressure vessels, which was presented in the previous RD&D programme, has not been implemented. Instead, work is now in progress focused on the detailed design of the concrete structure in order to adapt it to the different types of waste packages that the reactor owners intend to deposit in BRT.

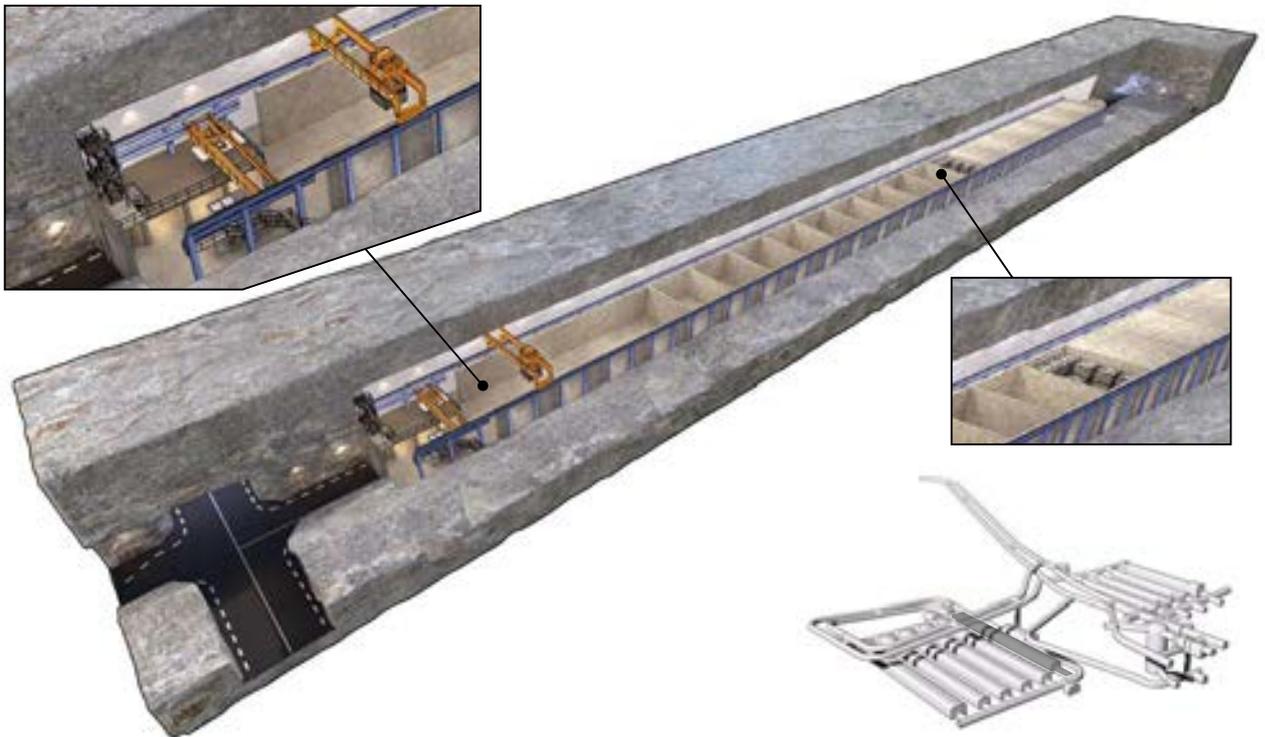


Figure 10-5. BRT – waste vault for reactor pressure vessels in the extended part of SFR.

Programme

SKB intends to continue the detailed development of the design of the concrete structure in BRT.

The results from the programme for development of material and technology for construction, which is being conducted for the caissons in 2BMA, are considered applicable to BRT as well, which is why no separate programme is planned for BRT within this area.

10.2.3 System for gas transport

During the long periods of time considered in a safety assessment, the waste, waste containers and any reinforcement will degrade with expected gas formation as a result. To facilitate the transport of gas from the different waste vaults, a system for gas transport is required, even though some transport may take place through the concrete pore system.

Current situation

According to the closure plan for the silo, a thick concrete lid provided with a number of gas outlets filled with a porous material will be cast on top of the silo (Luterkort et al. 2014).

In conjunction with the design and construction of the silo, SKB carried out a number of analyses of gas formation and gas transport in the silo, and of the long-term function of the outlets with gas-permeable material (Moreno et al. 2001, Höglund and Bengtsson 1991), where Höglund and Bengtsson (1991) show that the system function can be maintained during the first 5 000 years after closure.

As a part of the continued development of the barriers conducted by SKB for 2BMA, a system for gas transport has been developed (Elfving et al. 2017).

The gas transport system design consists of

1. horizontal (empty) gas transport channels that run in the interface between the radiation shield lids and the more watertight concrete overlay (Figure 10-6),
2. vertical gas transport channels filled with porous cementitious grout that are placed along an outer wall (and that intersect the concrete lid; see Figure 10-7).

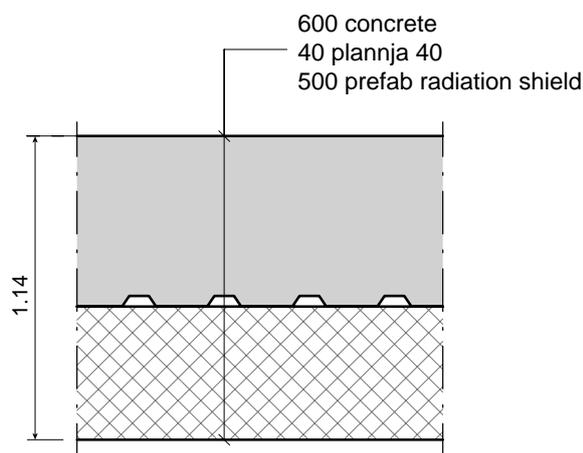


Figure 10-6. Horizontal gas transport channels that run in the interface between the radiation shield lids and the concrete overlay. These channels will be left empty. Dimensions in metres.

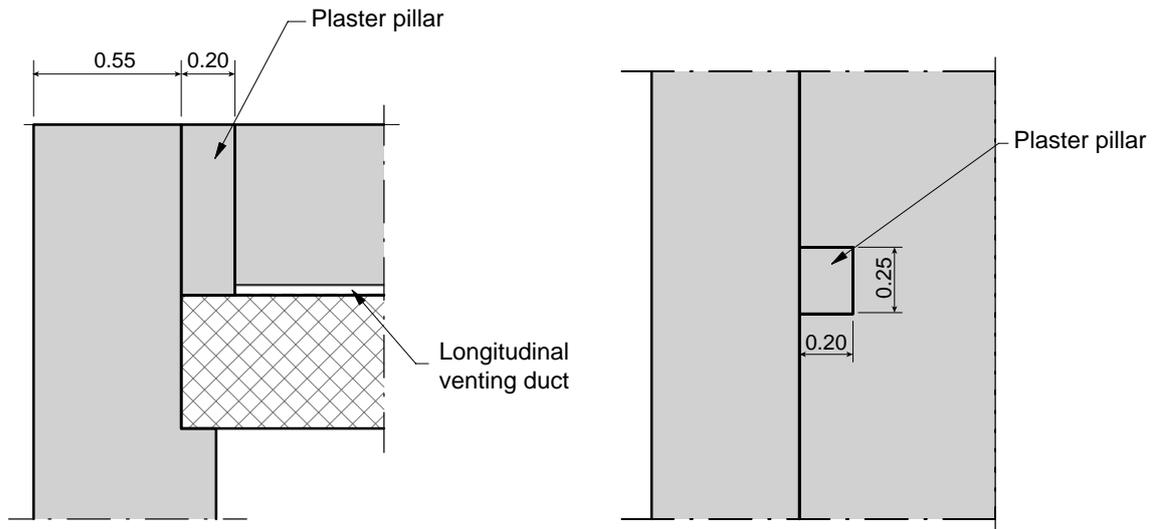


Figure 10-7. Vertical gas transport channel filled with porous cementitious grout; cross-section to the left and view from above to the right. The outer wall is seen to the left in both figures and the concrete lid to the right. In the cross-section (left), the radiation shield lid can also be seen (shown as a grid). Dimensions in metres.

The gas transport system is designed in such a way that the concrete structure's structural integrity will not be affected.

For 1BMA, studies are under way concerning the need for and design of a possible gas transport system.

Programme

SKB intends to continue the studies concerning the design of the gas transport system for 2BMA and its long-term properties, as well as the analysis of the need for and design of a corresponding system for 1BMA. If a need is considered to exist, the design work will start from the ongoing work for 2BMA.

10.3 Design of concrete structures and materials for SFL

10.3.1 Concrete structures

In the proposed design of the rock vault for core components, BHK, and the rock vault for legacy waste, BHA, the waste is deposited in concrete structures, which serve as radiation shields during the operating period and ensure that there are escape routes (Figure 10-8). In both BHA and BHK, the primary barrier is the backfill material installed around the concrete structure in conjunction with closure: concrete for BHK (Section 10.3.3) and bentonite for BHA (Section 11.4.3).

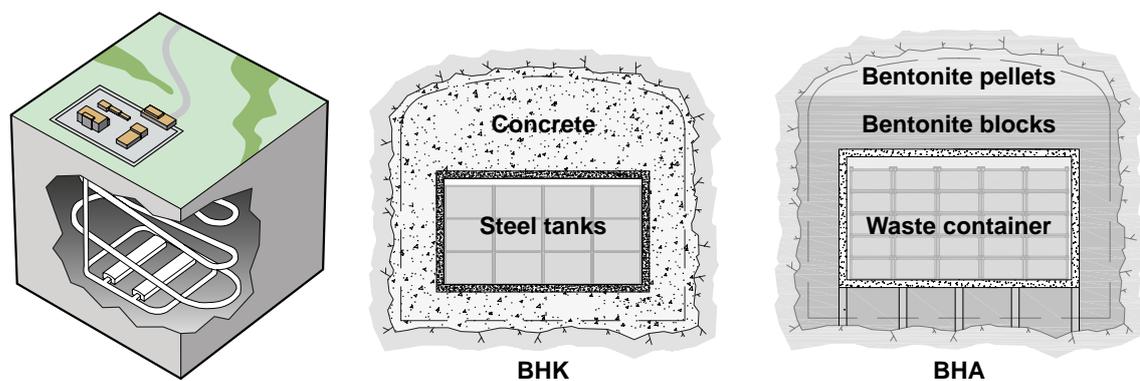


Figure 10-8. Proposed design of SFL.

Current situation

At present, SKB is not pursuing the development of concrete for the concrete structures in BHK and BHA. The concrete that is currently being developed for the caissons in 2BMA is deemed to be adequate also for these waste vaults. The results achieved from the work so far verify this assessment (Section 10.2.1).

Programme

SKB sees no need for carrying out any experimental development work concerning concrete for the repository structures in BHA and BHK in the RD&D period. Neither does SKB see a need for development of technology for construction of the repository structure in BHK, since it is deemed to be possible to design and construct using conventional methods.

However, SKB has identified a need to initiate work linked to the design, dimensioning and construction technology of the repository structure in BHA. The work must start from the design illustrated in Figure 10-8, entailing that the repository structure will be built on a bridge-like construction, which will be surrounded by bentonite on all sides. The following general issues requiring targeted development work have been identified:

- Material and method for the construction and dimensioning of pillars.
- Dimensioning and technology for construction of the load-bearing structure.
- Technology for construction of the repository structure on the load-bearing structure.

In addition, there are questions linked to the requirements on the rock floor and requirements on free work height under the load-bearing structure, which are related to the installation of the bentonite.

10.3.2 Grouting of waste containers

In conjunction with closure of the two rock vaults in SFL, the waste will be embedded in cementitious grout according to the current plans. The purpose of the grouting is to stabilise the waste and create a support for the concrete structure in conjunction with backfilling and resaturation of the repository.

Current situation

During 2012–2015, SKB carried out a number of studies and development work on grout for 1BMA and 2BMA (Lagerlund et al. 2014, Lagerlund 2014, 2015a, b), which will serve as a basis for the development of a grout for BHK and BHA.

Programme

Given the very long time period until closure of SFL, SKB sees no immediate need to initiate development work for a grout for SFL within the RD&D period.

10.3.3 Foundation for the concrete structure and backfilling of BHK with concrete

In the proposed design of BHK, the safety after closure of the waste vault is based on the waste being surrounded on all sides by a thick layer of concrete (Figure 10-8). In order to ensure the safety after closure for the whole period of time covered by the safety assessment, materials and methods for installation of materials must be chosen carefully. This includes both constructing the foundation for the concrete structure and backfilling the rock vault with concrete in conjunction with closure.

Current situation

During the RD&D period, SKB has conducted an internal study concerning the choice of materials and methods for constructing the foundation for concrete structures and backfilling BHK. A number of different methods have been compared and aspects related to the environment, safety after closure and technical feasibility have been analysed and evaluated (Mårtensson 2017a, b).

Programme

SKB is not presently planning to conduct a separate experimental development programme linked to constructing foundations for concrete structures and backfilling BHK with concrete within the RD&D period. This is justified by the long time spans that remain until the start of construction and closure of SFL, as well as the need to prioritise development work linked to SFR.

The work linked to constructing the foundation for concrete structures and backfilling the waste vault with concrete will instead be focused on compiling ongoing studies and modelling studies (Sections 10.1.1 and 10.1.6) into a cohesive basis in preparation for the design of the experimental development programme.

10.4 Design of concrete structures and materials for the Spent Fuel Repository

10.4.1 Plugs for deposition tunnels

Current situation

The purpose of the plugs is to keep the backfill in the deposition tunnels in place and to minimise leakage from the tunnels during the operating period, i.e. the time from closing the tunnels until the closure of the adjoining main tunnel. The development and verification of plugs for closure of the deposition tunnels in the Spent Fuel Repository have been carried out by means of analytical and numerical calculations, laboratory experiments, scale tests, and a full-scale test of the plug system (the Domplu experiment) at a depth of 450 metres in the Äspö HRL (Grahm et al. 2015). A graphical illustration of the test and a picture of the installed plug are shown in Figure 10-9.

The full-scale test has shown that it is possible to construct the plug system in an appropriate manner with an unreinforced concrete dome of the low-pH concrete mix design B200, which meets performance requirements.



Figure 10-9. Schematic cross-section and picture of the installed Domplu plug.

The full-scale test was taken into operation in 2014 and the plug was at that time pressurised with water to 4 MPa in order to simulate the groundwater pressure at repository depth. The test has been monitored with different instruments since installation. In the summer of 2017, the operational phase of the test was concluded with a test period where a gas tightness test and a pressurisation test were particularly significant. After the completed test period, Domplu was dismantled for final evaluation (Enzell and Malm 2019).

The gas test was carried out to verify if the plug design can meet the requirement for stopping convection of air during the operating period. The completed gas test shows that Domplu is gas-tight (Åkesson 2018).

The pressurisation test was carried out to verify that the concrete structure can withstand the design pressure of 9 MPa. Both the completed analyses and the pressurisation test show that the concrete dome is well designed for this load.

Core samples taken in conjunction with dismantling show that the concrete structure is homogeneous and that it has had good contact with the rock.

At the end of September 2014, the measured leakage through the plug was 44 mL/min (corresponding to 2.6 L/h). Leakage has gradually decreased since installation and at the time of dismantling, the measured leakage was about 25 mL/min (corresponding to 1.5 L/h). The decrease is due to the fact that the Domplu plug's bentonite seal has had continuous access to water. This has facilitated complete water saturation and build-up of a swelling pressure large enough to efficiently seal the tunnel.

Programme

During the RD&D period, SKB intends to complete the evaluation of the construction and dismantling of the Domplu plug and compile these analyses into a basis for further studies on material and method of construction of tunnel plugs

10.4.2 Low-pH cementitious materials for grouting and rock support

Current situation

The current technical design requirements and other requirements on the Spent Fuel Repository assume the use of low-pH cementitious materials for grouting and rock support from –200 metres, based on a study of the effect of using conventional cement (OPC) in the upper 200 metres of the Spent Fuel Repository (Sidborn et al. 2014). SKB has previously commissioned development of a low-pH cementitious material for grouting (Bodén and Sievänen 2005) and rock support (Bodén and Pettersson 2011).

The construction, operation and closure of the Spent Fuel Repository cover long periods of time, which means that the availability of components in the current mix design could change considerably and some products may be completely discontinued over time. For this reason, it is important that the recipe is designed in such a way that the properties of the materials are not completely dependent on a specific product and that the constituent components can be replaced with products with similar properties.

SKB has also carried out long-term studies of the corrosion of steel components embedded in standard concrete and in concrete of the low-pH type. The studies have been in progress for a total of nine years, and retrieval and analysis have been carried out after five years (Aghili et al. 2014) and nine years (Sederholm et al. 2018).

The studies showed negligible corrosion in both standard grout and low-pH grout as long as chlorides did not penetrate into the embedded steel samples. The studies also showed that the high density of low-pH concrete entailed that the time up until the corrosion process began was longer than for the somewhat less dense standard grout. This is explained by the lower diffusion rate of chloride in low-pH concrete compared with standard grout. However, when chlorides did reach the embedded steel samples, the corrosion rate in low-pH concrete was considerably higher than in standard grout and severe attacks could be observed.

No corrosion could be identified on grouted rock bolts after nine years of exposure (Sederholm et al. 2018). The study also showed that no leaching occurred in any of the bolt grouts and that the adhesion between the grout and rock bolts was good.

Programme

SKB has, after completing the development work described under “Current situation”, not yet designed a programme for further development work with respect to low-pH cementitious materials for rock grouting and rock support. During the RD&D period, the focus will instead be on compiling the work completed so far and thus create a basis for the planning of possible future studies. In addition, SKB is planning to update the study reported in Sidborn et al. (2014), i.e. an analysis of the extent of the pH plume in the Spent Fuel Repository from tunnel sections where standard cementitious materials are used. The analysis will be based on more realistic assumptions and calculation preconditions. With new input data and boundary conditions, results based on actual conditions may be obtained, which provides a better foundation for formulating requirements.

11 Clay barriers and closure

The main purpose of the clay barriers, i.e. the buffer and backfill in the Spent Fuel Repository, the silo clay barrier in SFR, and the backfill in the rock vault for legacy waste (BHA) in SFL (Figure 11-1), is to restrict the water flow around the canister and the waste packages. This is achieved by low hydraulic conductivity and a swelling capacity that makes the installed barrier homogenise, fill cavities and seal against the rock and other repository components.

The closure consists of plugs, material installed in boreholes and the material installed in all underground openings outside the waste vaults (the rock vaults in SFR and SFL and the deposition tunnels in the Spent Fuel Repository) to seal them.

The closure in the repositories will maintain the principle of multiple barriers by preventing the formation of conductive water paths between the repository area and the ground surface and by preventing the backfill from expanding out of the deposition tunnels and rock vaults. In the upper part of the ramp and shafts, the closure should considerably impede inadvertent intrusion into the repository.

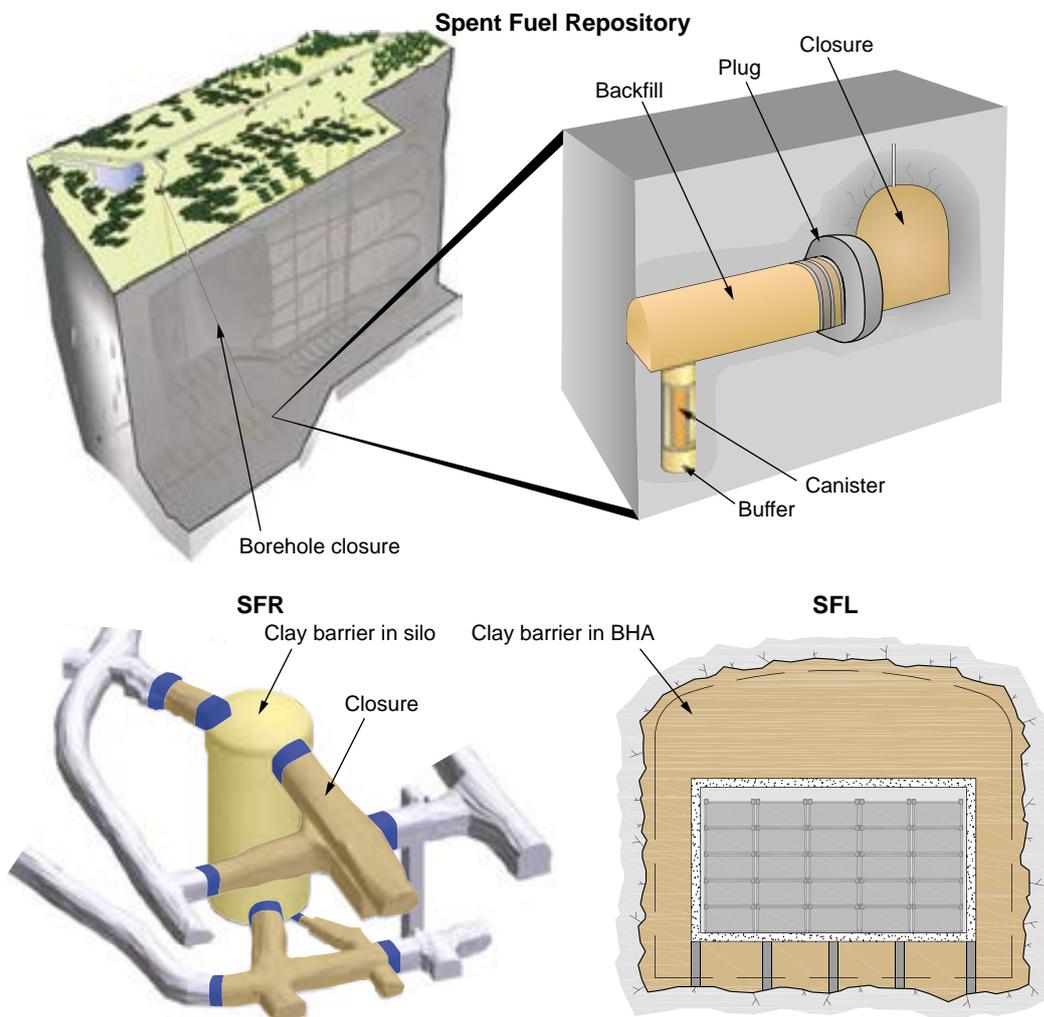


Figure 11-1. Top: The clay barriers and closure in the Spent Fuel Repository: buffer, backfill, closure and borehole seal. Bottom left: the silo clay barrier and closure in SFR. Bottom right: clay barrier in the rock vault for legacy waste (BHA) in SFL.

11.1 Bentonite evolution after installation until saturation

In both the Spent Fuel Repository and the rock vault for legacy waste in SFL (BHA), 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 be developed as the bentonite absorbs water from the surrounding rock.

In order to more extensively understand and describe the evolution of the bentonite material until saturation, efforts are required linked to the chemical evolution during the unsaturated period, piping/erosion, swelling, homogenisation of blocks, pellets and cavities, water vapour circulation and microbial sulphide formation under unsaturated conditions.

11.1.1 Gas phase composition during the unsaturated period

In the rock in Forsmark, the saturation time for the buffer around the canister is expected to vary between a few tens of years to several thousand 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 an unsaturated state for a relatively long time. A possible issue with respect to canister corrosion is the chemical composition of the gas in the unsaturated bentonite. The oxygen content (O_2) and the hydrogen sulphide (H_2S) are of particular significance. The questions that must be answered are:

- Will oxygen be consumed by the repository components? If yes, what is the reaction rate?
- Will hydrogen sulphide be generated from minerals in the buffer?
- Can hydrogen sulphide be generated microbially in an unsaturated buffer?
- Can other gases form?

These questions are of particular interest for the Spent Fuel Repository since the gas composition will primarily affect the copper canister. It is not impossible, however, that the composition of the gas phase during the unsaturated period may also be of interest for BHA.

Current situation

Two experiments have been designed and carried out to investigate the evolution of the gas composition in the buffer in an unsaturated KBS-3 repository (Birgersson and Goudarzi 2018). One of the experiments included a central heater in the form of a copper tube and Ibeco RWC bentonite blocks and pellets, configured as a scale model of an isolated unsaturated buffer in a KBS-3 repository (10 cm copper tubes, and bentonite blocks with a diameter of 30 cm). The second experiment was conducted under isothermal and isolated conditions (room temperature or 50 °C) and only included bentonite pellets. The evolution of the oxygen concentration in the experiments was measured periodically by means of an in situ system. Before the experiments were concluded, gas was sampled for analysis. Even though the bentonite used was chosen due to its relatively high sulphur content, no sulphide gas was detected in any of the samples. This result is a strong indication that such gas is not expected in an unsaturated KBS-3 repository. High oxygen consumption was observed in the experiment containing a copper tube – after about 50–60 days, the oxygen concentration was approximately one percent in this experiment. The copper heater surface was investigated, and the results are presented in Johansson A (2019b). However, the experiment without copper components showed no noticeable oxygen consumption at room temperature. From this difference, it can be concluded that the oxygen is primarily consumed as a result of aerobic copper corrosion.

Oxygen consumption during the saturation phase has also been studied in the Full-scale Emplacement Experiment in the underground laboratory in Mont Terri (Switzerland) (Giroud et al. 2018). The test showed that the oxygen was consumed very quickly, in the fastest case after a few weeks from back-filling the experiment. The interpretation of this experiment, however, is that the excavation-disturbed zone in the bedrock and the bentonite consume the oxygen, and that the reaction with metal is only responsible for a small fraction. Laboratory experiments performed in conjunction with the field experiment show that oxygen is consumed by bentonite already at ~55 percent of relative humidity. The observation that noble gases are consumed at rates that are comparable with oxygen implies that this is probably a sorption process.

Programme

Studies of the gas composition in unsaturated bentonite will continue. A new experimental set-up similar to the one used in Birgersson and Goudarzi (2018) has been built in SKB's research laboratory on Äspö. The difference is that hydrogen sulphide, hydrogen, sulphur dioxide and carbon dioxide can be measured in situ.

The experimental set-up of the new test (Figure 11-2) is based on two containers of stainless steel, which makes it possible to conduct two experiments in parallel. Both containers have a central heated canister, one with a copper heater and one with a heater of stainless steel, which means that similar experiments can be carried out with or without the presence of metallic copper.

11.1.2 Piping/erosion

A hydraulic problem during the operational phase concerns piping and related erosion effects in the buffer and the backfill. The inflow of water to the deposition holes, which 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 provide water at a more rapid rate than the swelling buffer can absorb, the arising 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 piping in the bentonite and a continuous water flow, as well as progressive erosion of bentonite particles. In such a case, the continued evolution is determined by the bentonite swelling rate, the flow rate through the buffer and the buffer erosion rate.

The same phenomena may occur in the backfill in BHA. In that repository, however, the amount of bentonite is so large that a mass loss in the early stage will hardly have any significance for barrier performance at all.

Current situation

Results from earlier studies have been published in Börgeresson et al. (2015). The goals have been to understand and develop models for critical processes that occur in an early stage after the installation of the buffer and canister, such as piping, erosion, water uptake in pellet-filled slots and early water absorption, and to create opportunities for stipulating requirements on the tightness of the end plugs in the deposition tunnels.



Figure 11-2. Left: Two stainless steel containers equipped with sensors for temperature and relative humidity and connections for online analysis of gas composition. Right: A central heated copper heater with exchangeable copper shields for easy installation and analysis.

Programme

A remaining issue after many piping and erosion experiments is that the equipment used has permitted clay to leave the test system. When small inflows (0.01–0.05 L/min) are studied in such systems, the channels are not always stable and may open and close as the experiment progresses. A channel closure leads to the pressure increasing until the clay breaks and the channel is opened again. Each time a channel opens, some clay is carried away. Altogether, this results in a higher clay concentration than the one measured at higher flow rates, when the channel is stable (10 g/L or more). However, this is likely to be an artefact caused by the test configuration since 1) there are no outlet holes so large in deposition holes or tunnels 2) the “carried-away” clay in a deposition hole must be transported for ~6 metres in clay to reach the tunnel. It is likely that it will sediment on the way and not leave the deposition hole.

The current plan is to perform a slightly larger experiment where both buffer and backfill are included. The inflow is to be small (possibly 0.01 L/min or 0.05 L/min) and the flow will go through the buffer to the backfill and either leave the system in a way that makes the clay unable to escape (porous wall around the backfill) or the void volume in the backfill should be large enough to accommodate large quantities of water (several cubic metres). Based on the inflow pressure, it is then possible to assess whether the channels form in the clay, and after the experiment the clay material remaining in the buffer can be analysed to see how much clay actually disappeared.

11.1.3 Water uptake

When the bentonite blocks and the pellet fill have been installed in a deposition hole and the deposition tunnels in the Spent Fuel Repository, or in the backfill in BHA, the bentonite will absorb water from the surrounding rock. During the saturation phase, the bentonite will develop a swelling pressure that exerts a mechanical force on the rock and the surrounding barriers. The 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 relative humidity in the barrier, which can be regarded as a negative capillary pressure in the buffer pores leading to water uptake from the rock. The hydraulic conditions in the rock surrounding the barrier determine the course of the saturation process. If the supply of water is unlimited, full water saturation will be achieved in a few years in the buffer in the Spent Fuel Repository and in about ten years in the backfill in the deposition tunnels and in BHA. As the water supply will be limited by the flow in the rock, water saturation will in reality take much longer.

The water uptake and the time for full water saturation have no direct bearing on the barrier function in any of the repositories. The process may, however, have an indirect significance for the performance of the repository, for example by:

- The maximum temperature – a saturated buffer has higher thermal conductivity.
- The swelling and homogenisation process.
- The time for when species (e.g. corrosive agents) in aqueous solution can be transported between the rock and canister.
- the time for 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.

Current situation

SFL's near-field hydrogeology has been evaluated by means of a scale model of the repository (Abarca et al. 2019). One part of this evaluation focused on transient unsaturated flow simulations to estimate the time required to reach full water saturation of the waste vaults BHA and BHK. The maximum flow to the repository is always limited by the groundwater flowing through the rock. The distribution of the degree of saturation during the water saturation process is shown in Figure 11-3. The results show that most of the water enters the repository through areas in contact with fractures in the rock. The loading area, which is completely filled with bentonite, takes the longest time to saturate. The waste is saturated faster than the bentonite due to its lower porosity.

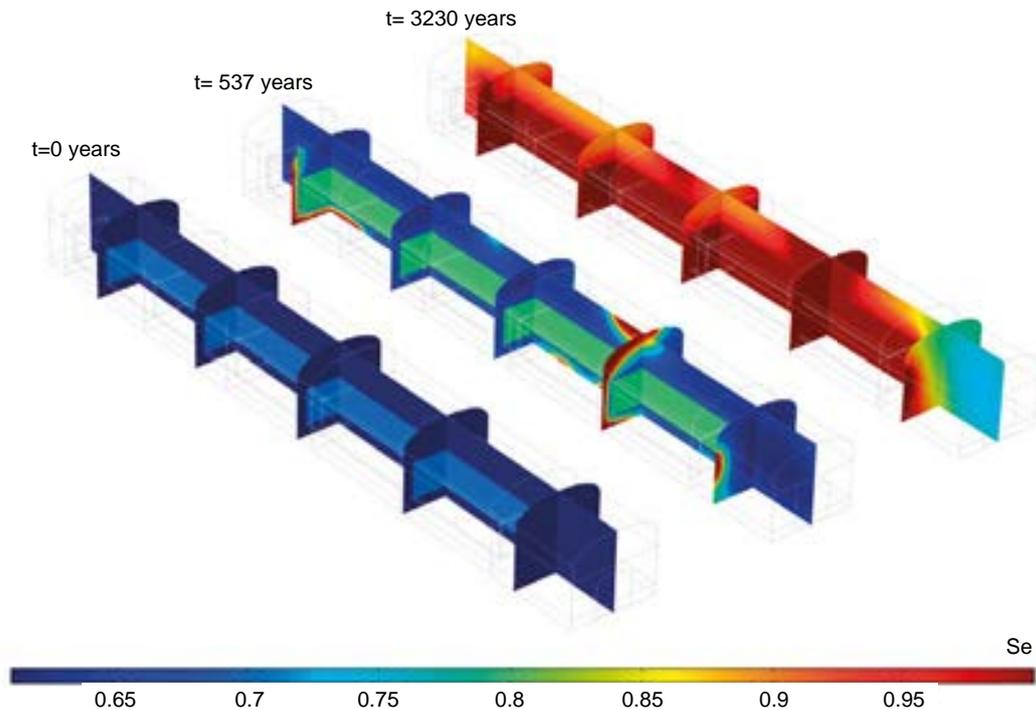


Figure 11-3. Initial water saturation in BHA and calculated distribution of the degree of saturation at 537 and 3 230 years (Abarca et al. 2019).

Most of the model development and testing of models for water uptake is carried out within the framework of the SKB Task Force on Engineered Barrier Systems (Task Force EBS). During the past period, the results of some of these studies have been reported. Phase 1 of Task Force EBS, concerning modelling of THM processes (thermo-hydro-mechanical processes) in the buffer and backfill materials for final disposal of radioactive waste, took place between 2005 and 2010. This phase included several calculation tasks with the purpose of modelling both well-defined small-scale laboratory experiments and large-scale field tests.

The modelling performed by the two Swedish modelling groups funded by SKB is described in Börgesson et al. (2016a, b).

In Malmberg and Åkesson (2018), modelling of the BRIE experiment in the Äspö HRL is described – both the water uptake experiment and the actual field test. That modelling was also performed within the framework of Task Force EBS. The main goals of the modelling were 1) to verify the hydraulic material model for bentonite clay and 2) to develop and test the conceptualisation of water transport properties of the rock in conjunction with the BRIE experiment. The purpose was to gain a better understanding of the wetting process in the planned final repository in Forsmark. The water uptake experiment consisted of individual bentonite blocks (cylindrical) in metal containers. The bentonite blocks had free access to water via filters surrounding the outer boundary, but neither the filters nor the metal containers were included in the model. The results from the modelling of the experiment agreed very well with the experimental results, showing that the parameters specified in the task definition provided a good representation of the used bentonite blocks.

Programme

The work of verifying and updating the models for water saturation of the buffer continues. Most of this work is carried out within the framework of Task Force EBS. Work will continue within the framework with modelling of laboratory and field experiments, in order to obtain better and more flexible tools for the coming safety assessments for the Spent Fuel Repository and SFL. An important issue for a repository in Forsmark is the ability to handle a very slow water uptake, either in a discrete fracture or through the entire rock matrix, and to determine how it will affect the saturation process and the water distribution in buffer and backfill. The connection between slow water uptake and the mechanical evolution of the bentonite barriers is an important issue as well.

11.1.4 Swelling, homogenisation of blocks, pellets and cavities

Water uptake after deposition will lead to swelling of the buffer and backfill in the Spent Fuel Repository, which are inhomogeneous at installation. This causes all gaps in the buffer, between the rock and buffer and between the canister and buffer, to disappear, and the buffer to be homogenised. However, some inhomogeneity will remain due to friction in the bentonite, which means that there will be a remaining density distribution and the barrier in question will have different hydraulic properties in different parts. This remaining inhomogeneity is significant for the technical design requirements and the configuration (pellets and blocks) used when the buffer is emplaced. The purpose of the programme is to assess whether the expected degree of homogenisation in the buffer is sufficient for the buffer to maintain its intended functions in the long term. Swelling and homogenisation are also important processes for compensating for local mass losses from erosion or undetected mishaps during installation.

The swelling and compression properties of the backfill are important for the function of the Spent Fuel Repository. The design with blocks and pellets in the backfill imposes requirements on the homogenisation capacity both between the blocks and pellet fill and for the healing of erosion channels. The swelling pressures and compression properties are also important for, for instance, buffer swelling and the effect on the plug. The same issues are relevant in BHA in SFL, where the large volume of bentonite may lead to further questions concerning the upscaling from laboratory experiments to a real scale.

Current situation

Developing predictive models for the mechanical behaviour of buffers, backfills and bentonite seals is a common need for waste organisations that are planning to use engineered bentonite barriers. These types of models are complicated, both conceptually and numerically. An effective way of tackling this is cooperation between different groups, where they jointly try to find better and more effective solutions. This is the background for the EU project Beacon.

The purpose of Beacon is to develop the understanding of the fundamental processes that affect the homogenisation of bentonite, and to improve the capacity of the numerical models. The project has 26 participants from nine countries and is in progress during the period 2017–2021.

In parallel with Beacon, SKB has continued with further experiments and model development regarding the mechanical properties of bentonite. Results from several test series are presented in Dueck et al. (2016). Test series including swelling have been conducted in two different scales, with the results from the tests in the larger scale (cylindrical samples with a diameter of 100 mm) permitting relatively speaking a higher resolution in the base variables determined from the samples. In another test series, the friction between confined water-saturated samples and different surfaces has been studied. Self-healing of the bentonite buffer has also been studied in two tests in medium scale (cylindrical samples with a diameter of 300 millimetres), where the loss of large volumes of bentonite was included. The self-healing of the cavity is effective, but no complete homogenisation is achieved.

Dueck et al. (2019) contains a compilation of laboratory experiments that have been conducted to study the homogenisation process in bentonite clay. The main purpose of the report is to present experimental results from experiments performed up to and including July 2016 and to provide results that may be used for modelling, model improvement or determination of mechanical parameters for the THM modelling of the bentonite buffer installation. Analysis of the experiments, conditions, experimental results, and limitations of the experiments will be carried out later in the project. Results of supplementary swelling tests with varying water supply and of tests that investigate the friction between bentonite and other surfaces are being presented in this status report. Additional results from studies of buffer homogenisation where the effect of the loss of bentonite materials has been studied are presented. Differences in density that remain after a long time have been studied in a series of tests with so-called long tubes.

One of the most important tasks for the backfill in the tunnels in the Spent Fuel Repository is to limit the upward swelling of the buffer from the deposition hole. If the buffer can swell upwards, its density will decrease and may thus change its important properties. One case that could lead to such an expansion of the buffer is when the water flow into the deposition tunnel is very low while there is a simultaneous water flow into the deposition hole. This entails that the buffer in the deposition hole will saturate quickly and begin to swell, resulting in a pressure build-up that pushes the dry tunnel backfill upwards. This upward swelling has been modelled previously, but the results have not been validated against field or laboratory tests. The field test Buffer Swelling Test was carried out at the

Äspö HRL in order to be used for validation of the models. A simulated deposition hole with a depth of about 1.5 metres was made in the tunnel floor and equipped with four hydraulic jacks in the bottom of the hole. A steel plate with a buffer block on top was placed on the hydraulic jacks. The tunnel above the simulated deposition hole was then filled with backfill blocks and bentonite pellets (Sandén et al. 2017).

After installation, the buffer blocks were pushed upwards against the backfill to simulate a swelling buffer against a dry backfill. The vertical forces on the steel plate, the displacement of the buffer block and the resulting pressure on the rock surface in five positions above the hole were measured throughout the experiment. According to the experimental results, the total displacement of the buffer block of approximately 150 mm was distributed between the different parts in the following way:

- The 70–90 mm thick pellet layer on the floor was compressed by about 30 mm.
- The 430 mm thick pellet layer in the ceiling between the upper backfill block and the rock surface was compressed by about 40 mm.
- The block section was compressed by about 80 mm through compression of slots, elastic compression of the blocks, and lateral movements.

Measurements of the uniaxial compressive strength and tensile strength of the backfill blocks show that the strength is low in comparison to other tests, mainly due to the low density and low water content of the blocks. A preliminary conclusion is that the resistance to upward swelling can be significantly improved if the backfill blocks are made with higher density and thereby have higher compressive strength. Börgesson and Hernelind (2017) describes model calculations of the mechanical interaction between buffer and backfill. The purpose of the calculations has been to understand and evaluate the effects of different factors of the interaction between the buffer and a dry backfill on the final state of the buffer after swelling and homogenisation. Sandén et al. (2017) has modelled the Buffer Swelling Test and the comparison between modelled and measured results showed very good agreement.

Leoni et al. (2018) presents a comparison between the results from two different calculation models that have been used for modelling the Buffer Swelling Test (Sandén et al. 2017). Despite the differences between the codes used and differences in the material models, geometries and input data, both techniques successfully modelled the course of events up until the failure of the blocks. One conclusion is that both the codes and the models can be used to study the processes in the backfill when the buffer swells upward against a dry backfill of blocks and pellets. In Figure 11-4, there is a comparison between the results from the models and the test with respect to normal stress as a function of displacement. The field test showed that the backfill reaches its maximum resistance to upward swelling of the buffer when the swelling pressure from the buffer is close to the uniaxial compressive strength of the individual blocks. If this can be confirmed for a broader range of compressive strengths, it should be considered in the continued development of the backfill concept.

Programme

Most of the continued work in this area until 2021 will be carried out within the framework of Beacon. In the next phase, the models will be tested in experiments on a larger scale. In addition to the part funded by the EU, SKB will also continue with laboratory studies and model development under its own auspices. The main goal is to develop verified models with predictive capability. These will be used to verify the repository designs of the buffer and backfill in the Spent Fuel Repository and the backfill in BHA in SFL.

There is a need to verify and develop the models for buffer swelling and the interaction between the buffer and backfill, so that they can later be used for calculations of different wetting cases, since there are currently no models that are verified for different wetting cases. It is appropriate to carry out a number of scale tests to

- gain a better understanding of different wetting cases and determine whether the dry case is the most unfavourable,
- compare with modelling approaches,
- scale up the results to full scale.

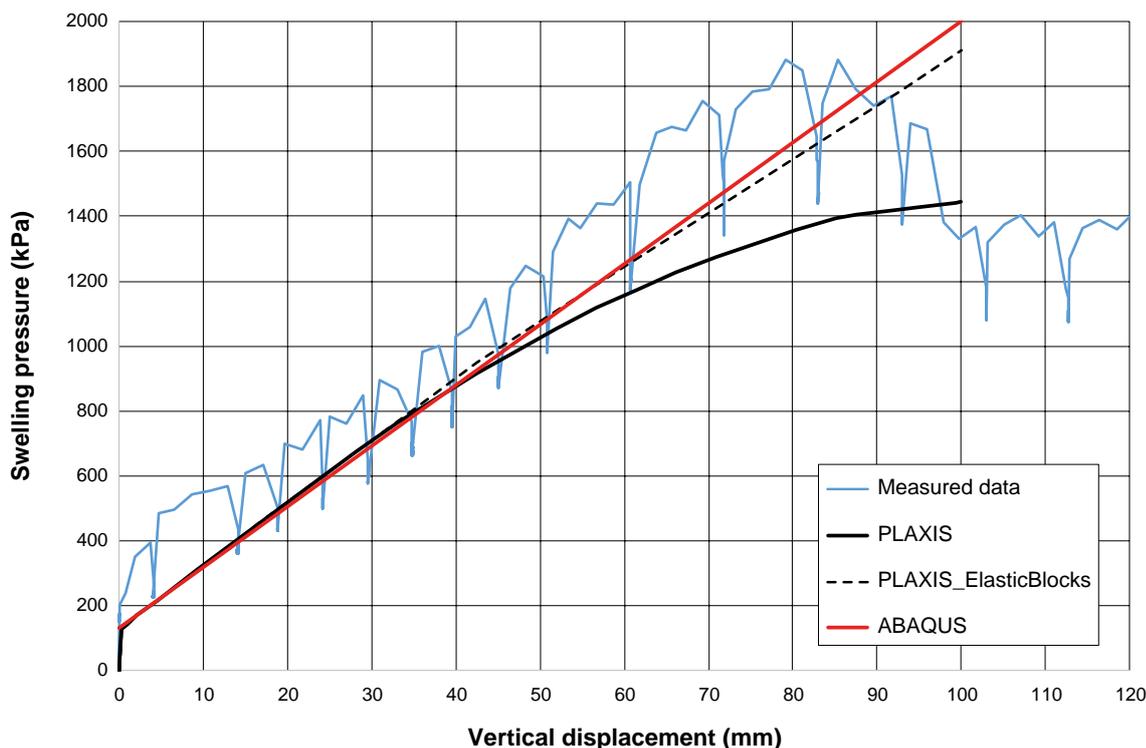


Figure 11-4. Comparison between the measured and modelled relation between vertical displacement of the plate and normal stress on the plate. The black solid and dashed lines represent the Plaxis results, and the red line corresponds to the Abaqus results. The “dips” in the measured values are a result of increasing the load in steps with a relaxation time in between. (Leoni et al. 2018).

The calculation models that can be used for these calculation cases are above all Code Bright, COMSOL and 3DEC, but further knowledge and development of these tools are required. This is in progress, partly within Beacon.

11.1.5 Water 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, thereby causing salt enrichment against the canister which, if extensive, could possibly cause corrosion (the sauna effect). SKB’s opinion is that this cannot occur to an extent that is harmful to the canister or the bentonite, as water is not expected to be transported out from the deposition hole. An experimental programme to verify this opinion has been completed.

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) presents tests carried out to study vapour transport in and sealing capacity of the inner slot between the bentonite blocks and the canister in a deposition hole. These processes were studied as a part of an investigation of so-called sauna effects, i.e. potential salt accumulation in a deposition hole as a result of vapour formation and vapour transport during the initial phases in a KBS-3 repository.

The studies specifically showed that the rate of the water uptake process was reduced considerably when water was fed through a bentonite component, compared to feeding water directly to the inner slot. Unsurprisingly, the bentonite component functions as a flow resistance, which decreases the water inflow to the inner slot (compared with direct feeding). The bentonite component through which water

was fed remained essentially water saturated throughout the test, despite being exposed to drying conditions in the inner slot. The rate at which liquid water vaporises in the inner slot was thus low in relation to the transport of water through the bentonite component.

Birgersson and Goudarzi (2017) argue that it is highly unlikely that larger quantities of salt are enriched in a KBS-3 repository during the saturation phase, and that the process can be disregarded.

Åkesson et al. (2019) describes measurements of vapour transport through bentonite. Tests have been conducted with test equipment in which specimens of bentonite were placed above vessels of water (or saline solutions). The test equipment was in turn placed in a climate chamber. This entailed that the specimens could be exposed to a vapour content gradient that was defined by the water (or saline solution) or the climate chamber and by the geometry of the test equipment. The tests were conducted with samples consisting of blocks, split blocks, references without bentonite samples, or of pellets.

The mass loss in the reference cell was, unsurprisingly, much faster than in the cells with bentonite samples. The mass loss in the cells with blocks was similar, regardless of whether the block was fractured or not, while the pellet-filled cells lost more water. The uptake of water in the bentonite was less than the loss from the cells in all cases. It is therefore clear that bentonite will not limit vapour transport to any appreciable extent, unless condensation occurs. All tests gave similar results.

Programme

SKB (2019e) draws the conclusion, regarding the sauna effect, that salt accumulation as a consequence of vaporisation of the inflow of groundwater from a fracture intersecting a deposition hole will not occur in a KBS-3 repository. The conclusions are based on laboratory investigations, model calculations and field tests. Thus, the conclusion is that the sauna effect will not be active in a KBS-3 repository in Forsmark. SKB therefore believes that further studies of vapour transport in bentonite are unwarranted. The area is monitored, however, and further efforts may be required if new questions arise.

11.1.6 Microbial sulphide formation under unsaturated conditions

In the Spent Fuel Repository in Forsmark, it is likely that a number of deposition holes and deposition tunnels will be dry, or partially dry, even in a relatively long-term perspective. An issue that will be relevant in such a case is the production and transport of canister corrodants in the gas phase. Diffusion is rapid in the gas phase, so in practice there are no transport limitations in an unsaturated repository. The key factor is then the amounts of corrosive agents available in the gas phase in tunnels and deposition holes. The amount of oxygen can easily be estimated with the available void volume. Although unlikely, it cannot be completely ruled out that hydrogen sulphide could be formed by microbial sulphate reduction in the unsaturated buffer and backfill. In the buffer this is handled pessimistically by an assumption of mass balance and a requirement on the maximum sulphur content (all sulphide in the bentonite is assumed to cause canister corrosion). In the backfill, it is more difficult to verify a requirement on the total content of sulphide in the installed material since the quantity of material is so large. Transport of sulphide from the backfill to the canister will be controlled by the concentration of sulphide, in the form of hydrogen sulphide, present in the gaseous phase in the unsaturated tunnels. It is therefore important to determine the concentrations of hydrogen sulphide in the gas phase in equilibrium with bentonite. Under saturated conditions, the diffusion resistance in the backfill and in the buffer will greatly restrict the transport of sulphide.

A prerequisite for microbial activity is the occurrence of nutrients for the microbes. Nutrients can either be supplied with groundwater or be present as impurities in the buffer and backfill materials at installation. During the unsaturated period, the supply of nutrients from groundwater will be very limited since the amount of water needed to saturate the barriers is relatively small. In order to limit the amount of nutrients in the installed materials, SKB imposes the requirement that the materials must not contain more than one percent by weight of organic carbon. This requirement is independent of the form of the organic carbon.

Current situation

During the latest research period, a series of experiments were conducted (Svensson et al. 2019b) to investigate the activity of sulphate-reducing bacteria (SRB) in bentonite as a function of the availability of water (liquid and different relative humidities). The focus was on experiments with the commercially available *Pseudodesulfovibrio aespoeensis* (previously called *Desulfovibrio aespoeensis*), which was originally isolated from the Äspö HRL, but experiments were also performed with an enrichment from natural groundwater from a borehole in the Äspö HRL. The experiments were performed in small glass tubes placed in larger plastic tubes (Figure 11-5), which were then stored in an oxygen-free environment. A demonstration experiment in a measurement cylinder was also carried out (Figure 11-6) to determine whether the natural occurrence of SRB in bentonite could be activated. When gypsum, lactate (a source of both carbon and energy), nutrients and liquid water were added, sulphate reduction was observed regardless of whether bentonite had been added or not. There were strong indications that the bentonite reacted with sulphide since the amount of formed sulphide was lower or below the detection limit in the presence of bentonite. The conclusion of this study is that both an energy source (lactate in this case) and liquid water is required for sulphate reduction to take place. Not even with 100 percent humidity could sulphide be observed during the test period.

Programme

Further experiments will be carried out, both with *Pseudodesulfovibrio aespoeensis* monoculture and with naturally occurring microbes isolated from boreholes in the Äspö HRL. In order to reduce the uncertainties, one or more alternative methods will be tested for measuring any formed sulphide in the bentonite solid phase. Experiments will be carried out where the carbon and energy sources are in the gas phase (carbon dioxide and hydrogen).

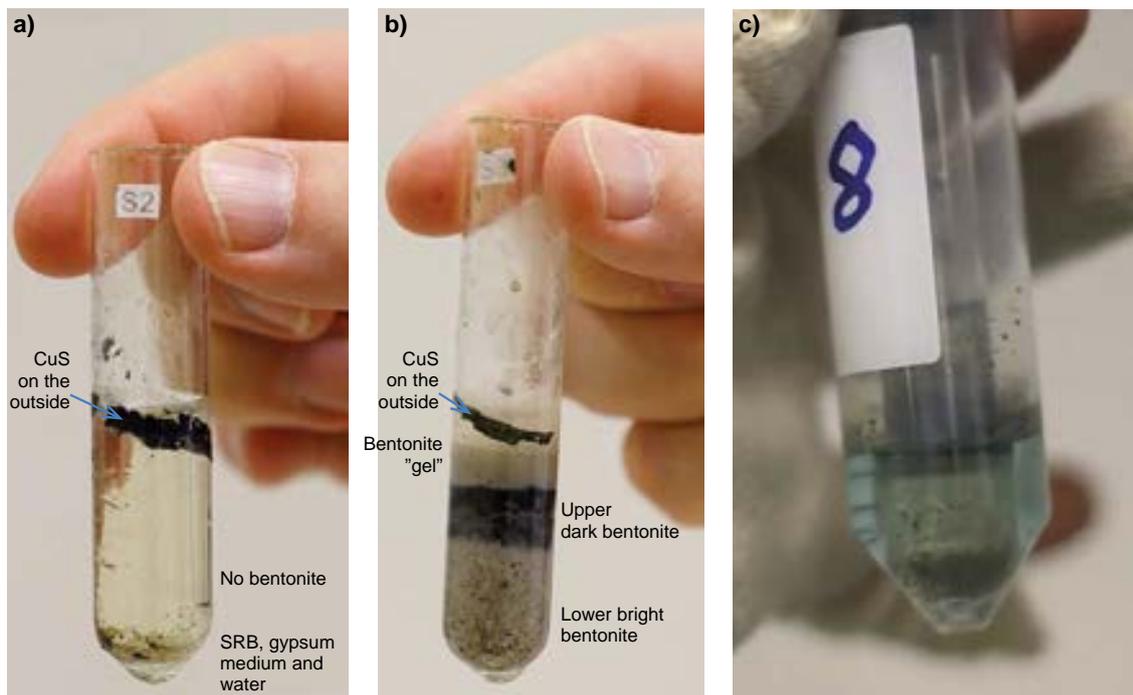


Figure 11-5. Picture a) and b) show two glass tubes with newly formed copper sulphide from microbial transformation of sulphate. Picture c) shows how one of these glass tubes has been placed in a plastic tube with liquid to provide the relative humidity in the system. The liquid also captured any formed hydrogen sulphide in the stable phase copper sulphide. There is no bentonite in the left glass tube, while the glass tube in the middle contains bentonite. Liquid water was available for the microbes in both cases. These tests were performed in an oxygen-free atmosphere.

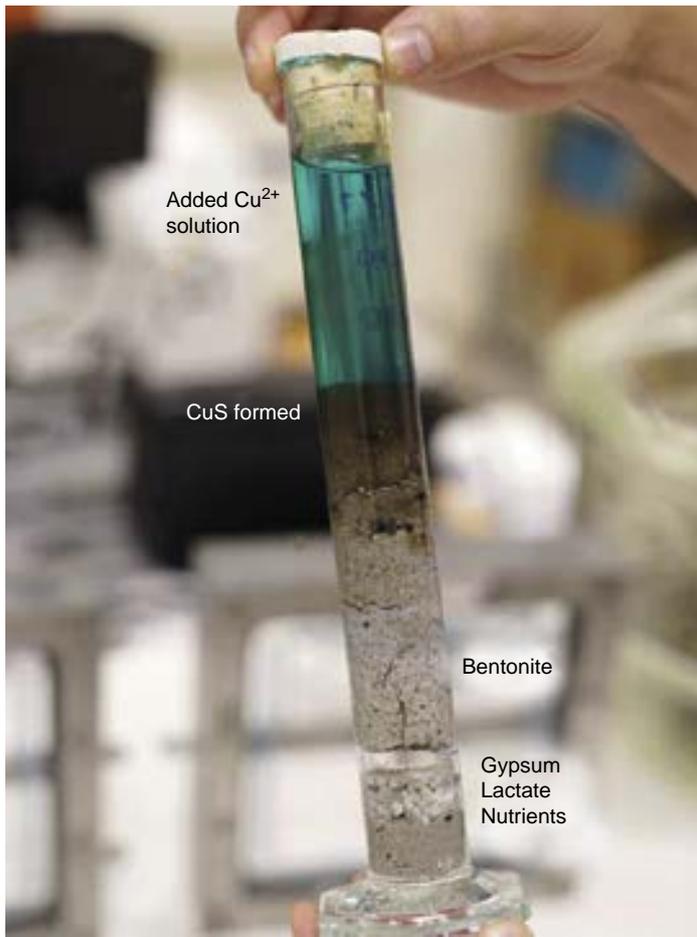


Figure 11-6. A demonstration experiment where naturally occurring sulphate-reducing bacteria in Wyoming bentonite were successfully activated by the addition of liquid water, gypsum and nutrient solution. Copper sulphate was later added as a reagent, and copper sulphide precipitated.

11.2 Bentonite material properties in the saturated state

The main function of the clay barriers is to restrict the water flow around the canister in the Spent Fuel Repository and around the waste packages in the silo in SFR and in BHA in SFL, as well as to limit the advective transport in the deposition tunnels in the Spent Fuel Repository. This is achieved by means of low hydraulic conductivity, so that diffusion is the dominant transport mechanism, and by means of a swelling pressure that makes the buffer self-sealing. The buffer should also hold the canister in place in the deposition hole, dampen the shear movements of the rock and retain its properties during the period being assessed. In addition, the buffer should limit microbial activity on the canister surface, and filter colloidal particles. The buffer must not significantly degrade the function of the other barriers.

The most important properties are hydraulic conductivity, swelling pressure and shear strength. These properties can be related to the density of a given bentonite. The relation is unique for each bentonite type and the properties vary with the composition of a given type of bentonite (the montmorillonite content, for example). Detailed material investigation is therefore important since changes in the montmorillonite content will have a direct impact on the bentonite's swelling-pressure curve.

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 primarily significant for the buffer in the Spent Fuel Repository.

11.2.1 Material composition

Current situation

A research laboratory has been set up at the Äspö HRL for analysis of and experiments with bentonite materials. The main purpose is to provide the infrastructure for research, development and quality control of bentonite, and to stimulate internal competence development. During the past period, the focus has been on developing and testing the methods and characterising reference material (Svensson D et al. 2017a). The methods that have mainly been used 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, analysis of carbon and sulphur, and thermal conductivity.

Extensive efforts have been made to quantify uncertainties and validate the methods (Svensson et al. 2019a), in part to investigate the smallest detectable size of impurities in the bentonite and the repeatability achieved by each method.

In the activities connected to the experiment Alternative Buffer Materials (ABM), a large number of bentonites are being analysed by several different international teams, which will serve as a type of benchmark test for each laboratory. This is SKB's main forum for the material composition and long-term stability of bentonite, and the analyses will be conducted in the research laboratory.

Programme

There is still some need for further improvement and testing of the methods and characterisation of reference material. This will be carried out to a limited extent in activities in conjunction with the ABM.

11.2.2 Swelling pressure and hydraulic conductivity

Current situation

In the research laboratory built at Äspö HRL (Section 11.2.1), nine measuring cells have been installed to measure swelling pressure and hydraulic conductivity. The way in which a change in the montmorillonite content of the bentonite affects the swelling pressure and hydraulic conductivity has been investigated in the past period, among other things. This was done by mixing a Wyoming bentonite (MX-80) with five and ten percent by weight of finely ground sand in order to simulate different bentonite qualities. Regarding swelling pressure, it is clear that a decrease of the montmorillonite content by five percent significantly affects the swelling pressure negatively. It is therefore important that the methods used for measurement of the material composition (Section 11.2.1) are designed so that they can detect small changes in the montmorillonite content of the bentonite that will be used as buffer. The hydraulic conductivity increases slightly when the montmorillonite content decreases.

Swelling pressure and hydraulic conductivity have been measured for several different bentonites. Most bentonites behave in a similar manner but bentonite from Bulgaria, for instance, deviates in that it has a much higher swelling pressure than the other bentonites at a given density.

Programme

Measurements of swelling pressure and hydraulic conductivity will continue in order to characterise more bentonites, both as raw material and after heating in long-term field tests (the experiments ABM and LOT, Long term test of buffer material). Analyses of the materials from the field tests will provide insight into whether there are any processes that alter the bentonite material's properties during the heated period in the rock. The results can also be used to determine which mineralogical parameters are most significant for the hydromechanical properties.

It is of the utmost importance to have a large body of data regarding the relation between swelling pressure and hydraulic conductivity as a function of bentonite type and density before a material is selected for use in the Spent Fuel Repository.

11.2.3 Shear strength

Depending on the mechanical properties of the buffer, shear movements in the rock may damage the canister and potentially cause loss of its containment capability. Bentonite shear strength is therefore an important parameter for the dimensioning of the canister and for the evaluation of the shear case in the assessment of safety after closure. 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.

As the ability of the bentonite to protect the canister is the critical factor, the issue is only significant for the buffer in the Spent Fuel Repository.

Current situation

A standardised testing method for shear strength, expressed as the uniaxial compressive strength, has been developed and is presented in Svensson D et al. (2017a). In that type of test, the sample is compressed axially at a constant rate. The samples that are used for the test are relatively small. The diameter is normally 20 mm and the height, 40 mm. The samples are first compressed and are then saturated in a separate unit. The test is carried out with a strain rate of 0.8 percent of the sample height per minute with continuous measurement of the load and deformation of the sample. The compressive strength is determined as the maximum stress applied to the sample before failure.

Modelling and laboratory tests have shown that the stresses on the canister will be reduced to an acceptable level if the shear strength of the buffer, measured with a uniaxial compressive strength test at a strain rate of at least 0.8 percent, is restricted to a maximum value of 4 000 kPa.

Programme

Since the shear strength, or rather the uniaxial compressive strength, is a control parameter of the material in the bentonite buffer of the Spent Fuel Repository, the method for uniaxial compression tests on clay samples will be established internally at SKB and tested on suitable clay materials. The work entails method development with laboratory analyses of bentonite at the Äspö HRL. Establishment of the method at Äspö HRL involves, among other things, supplementing software and documentation of logging systems and developing methodology for the conversion from sodium bentonite to calcium bentonite in larger batches than previously.

The uniaxial compression tests will be carried out using press equipment (Tritech) with a capacity of 50 kN. This equipment can be programmed so that a constant strain rate is obtained. During a compression test, the change in position is measured by means of a deformation sensor and the axial load by means of a load cell.

Brittleness of bentonite samples exposed to increased temperatures has been observed in different studies. The brittleness mainly manifests as reduced strain at failure. In these contexts, the effect of increased temperature and dehydration has mainly concerned water-saturated conditions and equilibrium with deionised water. How the combined effect of increased temperature, dehydration and increased salinity affects the stress/strain/strength will be investigated.

11.3 Bentonite evolution after water saturation

11.3.1 Sulphide formation and sulphide transport

Sulphide dissolved in the pore water in the bentonite can act as a corrodant for the copper canister. In order to assess the diffusive transport of sulphide in the bentonite to the canister, it is important to understand the concentrations of sulphide in the pore water. Microbial processes may give rise to the formation of sulphide under certain conditions. The dry density or swelling pressure of the bentonite has a great impact on the microbial activity. Sulphide is in principle 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.

A safety function for the buffer was defined in SR-Site, which involves the limitation of microbial activity. The safety function indicator was defined as *high swelling pressure* due to the uncertainties associated with the mechanism. In Posiva SKB (2017), the question was discussed further, and it was established that the conclusions regarding swelling pressure/dry density and potential other factors that limit microbial activity are somewhat incomplete. There are, however, results indicating a clear threshold where microbial sulphate reduction ceases in MX-80 bentonite (Bengtsson et al. 2015). The function indicator for limiting microbial activity was defined as a swelling pressure > 2 MPa and applies to the entire buffer volume.

Current situation

In order to determine which sulphide concentrations are relevant in the bentonite/sulphide system, experiments were performed with two different bentonite clays, Ibeco BF (Milos, Greece) and MX-80 (Wyoming, USA) and in some cases sulphide solutions (Svensson D et al. 2017b). The main purpose of the study was to determine the equilibrium concentration of sulphide in water/bentonite mixtures. During the determination of dissolved sulphide in the aqueous solutions, concentrations lower than the reporting limit of the method were obtained. When sodium sulphide solution was added, it was discovered that the bentonite decreased the amount of sulphide in solution to a concentration that was lower than expected. The mechanism for this could be absorption, transformation or some type of reaction but, based on these data, it is not possible to draw further conclusions. The observation that the bentonite reduced the amount of sulphide in solution when sodium sulphide was added substantiated the observation that no sulphide could be detected in solution equilibrated solely with bentonite.

Bengtsson et al. (2017a) investigated three different bentonite clays, Wyoming MX-80, Asha and Calcigel, under saturated densities in a range from 1 500 to 2 000 kg/m³, for microbial sulphide-producing activity as a function of total density at full water saturation. The results for the three clays indicate that in a range of saturated density, the sulphide-producing activity measured on the copper discs decreases from high to very low or below the detectable limit. Accumulation of copper sulphide with radioactive sulphur (Cu_x³⁵S) as a function of the bentonite's saturated density for copper discs installed in test cells with bacterial additives from the experiments is shown in Figure 11-7. For each bentonite type, there was a relatively small range of saturated density in which the sulphide-producing activity decreased from a high level to a very low level or to the absence of Cu_x³⁵S. The analysed density where the formation of Cu_x³⁵S had decreased to a very low value was similar for Calcigel and MX-80 at ~ 1 870 kg/m³ (Figure 11-7). The corresponding value for Asha was lower, ~ 1 830 kg/m³. Consequently, at least one more factor that affects the microbial activity of compacted bentonites, other than density, is needed in order to fully explain the difference in the values of Asha compared with MX-80 and Calcigel.

To further study the effects on bacterial sulphide production in different bentonites, the series described above continued with two additional materials and three densities for each material (Bengtsson et al. 2017b). This study included an iron-rich bentonite clay (Rokle) and a bentonite clay with a low content of iron (Gaomiaozi, GMZ). The methodology of using radioactive sulphur-35 (³⁵S) as a tracer showed very low sulphide production in Rokle in all densities, with the exception of a local production in the lowest density. The situation was the opposite for GMZ. Sulphide production could be detected in all tested densities. The results are presented in Figure 11-7 together with the results from Bengtsson et al. (2017a).

A weakness in the studies reported in Bengtsson et al. (2017a, b) is that they have been carried out by the same personnel in the same laboratory and that there were no independent verifications of the results. In order to remedy this, Haynes et al. (2019) repeated a small part of the tests from Bengtsson et al. (2017a). Two saturated densities of Calcigel were studied: 1 750 kg/m³ as a positive control where sulphide production was expected, and 1 900 kg/m³ where no activity was expected. Accumulation of copper sulphide on the copper discs was identified by exposing a phosphor imaging plate to the corroded copper surface and thereby analyse the radioactive sulphur-35 that had been introduced to the bentonite after the saturation stage. For Calcigel with a density of 1 750 kg/m³, the plate showed that sulphur-35 accumulation is responsible for 99.96 percent of the intensity (Figure 11-8). The density of 1 900 kg/m³ gave a negative result (0 percent), which is lower than the background signal for the phosphor imaging plate (0.04 percent) (Figure 11-8). The conclusion is that no sulphide was formed in the higher density.

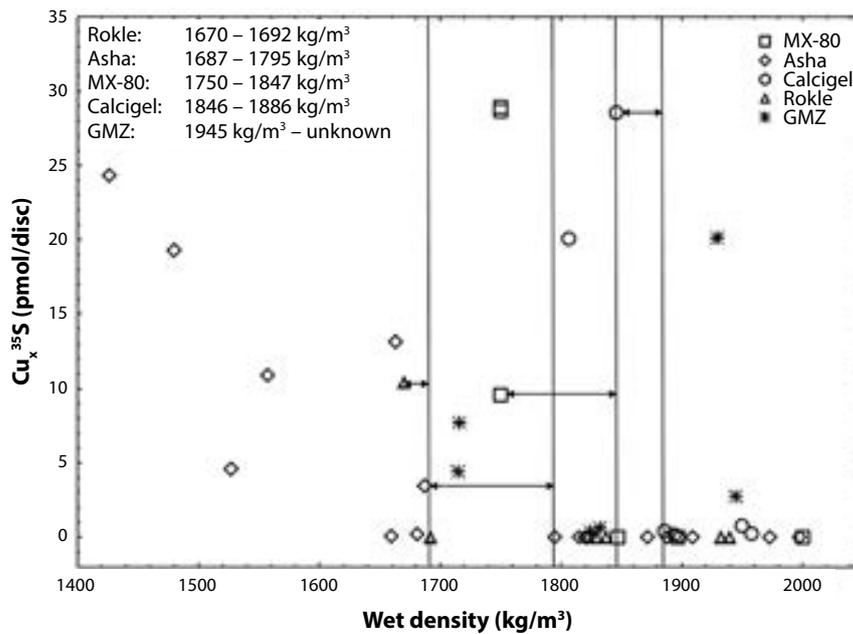


Figure II-7. Accumulated amount of Cu_x³⁵S on copper discs in picomole (pmol) as a function of saturated density. The respective ranges where the sulphide production shifts from high to low are indicated by arrows. The ranges are defined in the top left of the figure (Bengtsson et al. 2017b).

	Background	1 750 kg/m ³	1 900 kg/m ³
			
Beta radiation count (cps)	9	120	9
Fluorescence intensity	0.04 %	99.96 %	0 %

Figure II-8. Pictures of the copper discs after they have been removed from the pressure cell, together with the measured dose rate from beta radiation and the intensity from the phosphor plate (Haynes et al. 2019).

Programme

It is evident that there is a clear limit for when microbial sulphide production in bentonite can occur. For a given bentonite material, the limit can be expressed as a density in the saturated state. However, there is currently no understanding of why the limit differs between different bentonite materials. The swelling pressure or the montmorillonite content is not the whole explanation. Empirically, the sulphide production appears to be inhibited by a higher content of iron in the clay. There is, however, no good theory as to why the iron would be significant, and considering the limited body of data, it could be a random event.

The limit for microbial sulphide production in saturated bentonite is critical for the technical design of the buffer and to some extent the backfill, and for the assessment of safety after closure of the Spent Fuel Repository. There is currently not enough data for determining the definite limit, and the understanding of the bacteria-inhibiting mechanism is insufficient. The experiment methodology described in Bengtsson et al. (2017a) is both costly and relatively time-consuming. It is also rather inflexible since it requires that the laboratory can handle radioactive isotopes. SKB is planning a programme to improve the understanding and collect additional data. The measurement method also needs to be developed. One possibility is to analyse sulphide on copper plates by SEM-EDX (scanning electron

microscope, equipped with an energy-dispersive X-ray spectrometer), which would eliminate the need for radioactive sulphur and thereby make the method considerably more flexible. Over the next few years, the programme will focus on developing more experimental data to determine, in more detail and with better statistics, the limit in density where sulphide production ceases. With improved data, it may be possible to determine which physical, chemical and mineralogical properties control the activity of sulphide-producing microbes. This will be crucial for the choice of materials, technical design requirements and assumptions in the safety assessment.

11.3.2 Colloid release/erosion

The water uptake and the resulting swelling of the bentonite buffer in the Spent Fuel Repository are restricted by the walls of the deposition hole, and thereby a swelling pressure develops in the bentonite. The same applies to the backfill in BHA. If fractures intersect the deposition hole or the disposal vault, there are no rigid swelling restrictions at the intersection surfaces. The swelling then continues into the fractures until equilibrium or steady-state conditions are reached. This free swelling may lead to separation of individual montmorillonite layers (dispersion) and some of the buffer could be transported away with the groundwater. The maximum free swelling of bentonite is strongly dependent on the charge and concentration of the ions in the interlayer spaces. At too low concentrations of solutes in the groundwater, the interlayer distance between the individual montmorillonite layers may increase so much that the clay/water system takes on a sol character, i.e. single or small groups of montmorillonite layers behave like colloidal particles.

The conditions of significance for potential colloid formation in a repository are the local salt concentration in the pore water and the ratio between mono- and divalent ions in the montmorillonite in the contact between bentonite and ground water. The controlling processes are ion exchange of the original montmorillonite counterions with ions that stem from the extra minerals and from the surrounding groundwater.

In the post-closure safety assessment, SR-Site, colloid release and erosion could not be disregarded and the calculated mass loss was so great in a few deposition holes that advective conditions could not be ruled out.

Current situation

The modelling to describe erosion of the bentonite clay in SR-Site has been revised based on experiments performed in narrow slots (Neretnieks et al. 2017). The numerical accuracy for describing the processes in the expanding narrow transition zone between clay and water was not sufficiently high in the previous model. A new method for solving equations based on splitting the area into two zones was developed and was found to achieve the necessary accuracy. However, this model considerably underestimated the erosion in the experiments. More thorough analysis of the experiments showed that a previously unknown phenomenon highly contributed to increasing the erosion rate. When they have been released from the clay, the eroded clay particles will very soon form flocs, which are rapidly transported with the flowing water in the slot. Including this unexpected process in the model resulted in good agreement between the model and experiment. A comparison of the results from the new model and the model used in SR-Site is illustrated in Figure 11-9.

Flocculation is also found to have another unexpected effect, namely that flocs sediment in sloping fractures. The resulting erosion can be more severe than erosion caused by flowing water. Erosion loss in fractures with an aperture larger than 0.1 mm seems to be determined by how quickly flocs are released from the clay.

BELBaR was an EU-funded collaborative project based on the idea of improving the post-closure safety evaluations of geological final repository concepts, where a technical barrier system with bentonite as a component is combined with a natural barrier of crystalline rock. BELBaR included national organisations for management of radioactive waste from a number of countries, research institutes, universities and consulting companies that are working with the post-closure safety of repositories for radioactive waste. A summary of the conclusions from BELBaR is found in Shelton et al. (2018). In BELBaR, a considerable number of experiments were performed in which bentonite clay was allowed to penetrate into an artificial fracture containing either stagnant or slowly flowing

water. The clay intrusion into the fractures and the release of clay particles to the water in the fractures were studied. An important purpose of these experiments was to gather data that could be used to further develop quantitative models for calculating the loss of bentonite from deposition holes and tunnels. Furthermore, a number of experiments were performed aimed at studying the underlying mechanisms for bentonite expansion and release of clay colloids. Previously derived conceptual, mathematical and numerical models were refined and modified based on experimental results. The large amount of information produced during the four-year project has been published in a number of reports and publications in journals.

Neretnieks and Moreno (2018) presents an evaluation of how well the current model for mass loss corresponds to experimental data from BELBaR. The main purpose of the report was to collect and compile available data, mainly from BELBaR but also from other sources, in order to be able to use them for model testing and to verify the current model (Neretnieks et al. 2017) against the experimental data from BELBaR. The model used to simulate the loss of bentonite from the deposition hole, tunnel or repository vault has two main model components. The first one is the dynamic clay expansion model, which describes how compacted bentonite absorbs more water, generates a strong swelling pressure and penetrates empty spaces – the fractures. This model is valid for a large number of ion concentrations in the pore water. For monovalent ions, the swelling can be extensive. This also applies for calcium-dominated clays up to the point where the swelling pressure decreases towards zero, which, according to the model, occurs at volume fractions within the range 10–20 percent. The second model component describes the loss at the rim of the clay. This may occur in a fracture with stagnant water by means of sedimentation, and in a fracture with flowing water, by means of colloid erosion. Flocculation of released colloids seems to take place in both cases. Erosion by flocculation at the clay/water interface can only take place in water with low ion concentrations. Both model components need to be resolved at the same time since they affect each other.

The formation and properties of very loose clay flocs have been experimentally shown to be caused by the presence of positively charged edges on the clay layers (Hedström et al. 2016). Under conditions where a sol is formed, it has been found that flocculation of colloids is an important process, which can greatly affect erosion both in stagnant water in sloping fractures and in horizontal fractures with flow (Schatz et al. 2013, Schatz and Akhanoba 2017). Flocculation has been incorporated into the updated model for bentonite expansion and erosion (Neretnieks et al. 2017). Flocculation has been studied further in Neretnieks and Moreno (2019). Rheological properties of flocs and sediments affect how fast and how deeply into the fracture network montmorillonite can flow. The flocs of montmorillonite particles age, making the flocs stronger. They form a gel-like structure that is strong enough to stop further flow into the fractures.

The model for expansion of bentonite in a fracture presented in Neretnieks et al. (2017) predicts that bentonite may expand up to a hundred metres in a fracture under a 100 000-year period (Figure 11-10) also for ion concentrations in groundwater where no colloids are formed.

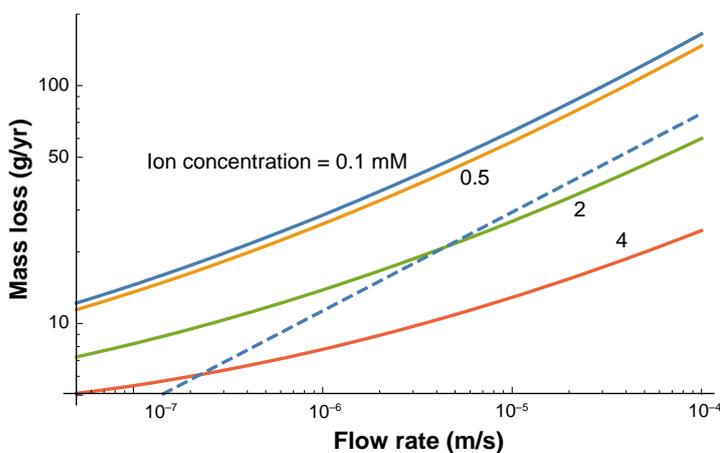


Figure 11-9. Mass loss as a function of the flow rate for different ion concentration in mM for a fracture with the aperture 0.1 mm. The dashed line shows the results from the model used in SR-Site.

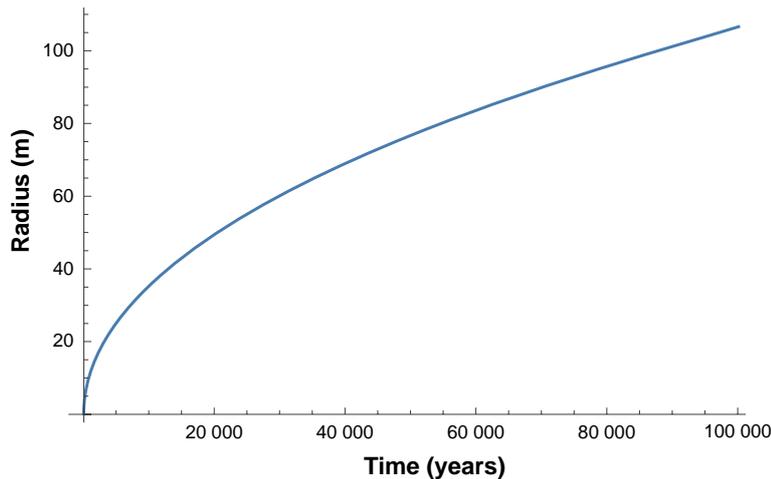


Figure 11-10. Radius to the rim of the clay as a function of time for an ion concentration $> 4 \text{ mM}$ (Neretnieks et al. 2017).

This expansion is not in full agreement with the observations made in field and laboratory tests. Börjesson et al. (2018) has studied swelling of bentonite in a fracture in a groundwater with a sodium chloride concentration above the critical coagulation concentration (CCC). The CCC concept works for pure sodium systems. The processes involved when bentonite swells until it becomes a non-swelling gel, which will be the result when the groundwater is above the CCC, have been analysed. Since problems with colloid erosion of bentonite are greatest for homoionic sodium montmorillonite, limit values have been determined for such systems. The CCC for sodium bentonite is approximately 20 mM of sodium chloride. At that chloride concentration, the clay will stop swelling at a clay concentration of approximately 60 g/L. In order to estimate how a bentonite clay swells into a fracture, knowledge is needed of

- swelling pressure and swelling properties of bentonite in clay concentrations higher than 60 g/L,
- the resistance to swelling caused by shear resistance or friction in the bentonite and between the bentonite and fracture surfaces,
- the shear strength of the non-swelling gel at 60 g/L.

A theory has been developed regarding bentonite penetration into planar horizontal fractures. The theory provides the final state at equilibrium when the swelling is finished and the swelling pressure has been balanced by friction against the fracture walls. From the theory it can be concluded that bentonite expansion in fractures under conditions above the CCC is limited to distances that are much smaller than the diameter of the deposition hole.

The erosion behaviour of compacted bentonite in artificial fractures has been studied experimentally under different physical-chemical conditions. The results are published in Alonso et al. (2019). Experiments were performed in both planar and uneven fractures with an experimental set-up specifically designed to evaluate

- bentonite expansion in horizontal fractures,
- sedimentation in sloping fractures, and
- erosion as a function of water flow.

The experiment series focused on questions where there were no data from previous experiments, or areas where the previous data are considered difficult to interpret.

Experiments performed with different bentonites in narrow fractures (width $< 1 \text{ mm}$) show that the clay expansion into the fracture ceases after some time or expansion distance. An example of the results is illustrated in Figure 11-11.

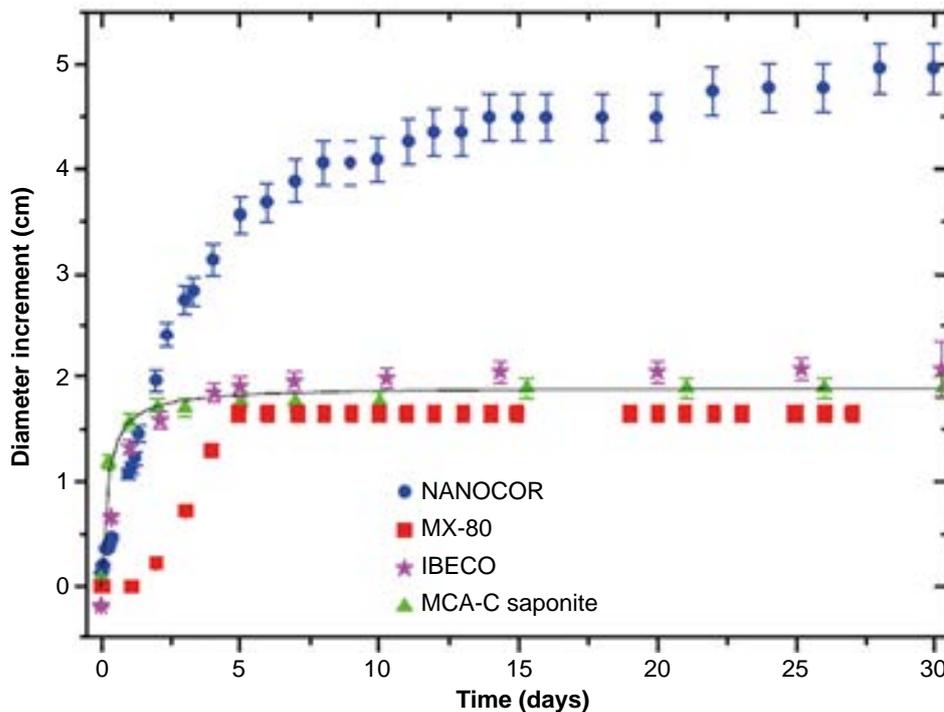


Figure 11-11. Expansion (diameter increase, in cm) of different clays, Nanocor®, MX-80, Ibeco and MCA-C, compacted to 1.4 g/cm^3 , within a 0.2 mm planar fracture positioned horizontally and filled with sodium chloride 10^{-3} M (Alonso et al. 2019).

This supports the conclusions in Börjesson et al. (2018), which were presented in the preceding paragraph. Studies performed with natural clays and calcium-rich clays in planar and uneven fractures showed that under the experimental conditions that prevailed, no sedimentation of clay occurred in vertical fractures with apertures smaller than 0.4 mm . The effect that accessory minerals and highly soluble salts may have on the erosion process was analysed in detail. The results show that coarse mineral fractions can efficiently limit clay expansion into narrow fractures. In the rim of the expanding fraction, however, both accessory minerals and clay were identified, and the clay particles themselves actually form a coherent filter. The studies also confirmed that the water chemistry is more significant for erosion than the flow rate. However, the question of which physico-chemical parameters limit the expansion and sedimentation of bentonite in the fractures is still not fully answered.

Programme

The major remaining issue with respect to colloid formation and erosion is the significance of flocculation. It is evident that clay colloids form a new phase, a “secondary gel”, when they are exposed to a force, either through a flow in the fracture or from gravitation in a vertical or sloping fracture. Regarding the flow in a horizontal fracture, it seems as if the release of particles from the clay rim to the flowing water is the limiting factor and flocculation is of secondary importance. The situation for sloping fractures is different: the formation of a “secondary gel” controls the mass loss. There is currently no conceptual model that can describe this process. The quantitative model presented in Neretnieks et al. (2017) is based entirely on empirical data. It is also challenging to conduct sedimentation tests under stagnant conditions, since small quantities of salt in the clay may greatly affect the result.

A new experimental programme has been initiated in SKB’s research laboratory at the Äspö HRL. The objective is to verify the results presented in Alonso et al. (2019), and to collect additional data. Figure 11-12 shows preliminary results from a sedimentation experiment with a “non-saline” pure montmorillonite.

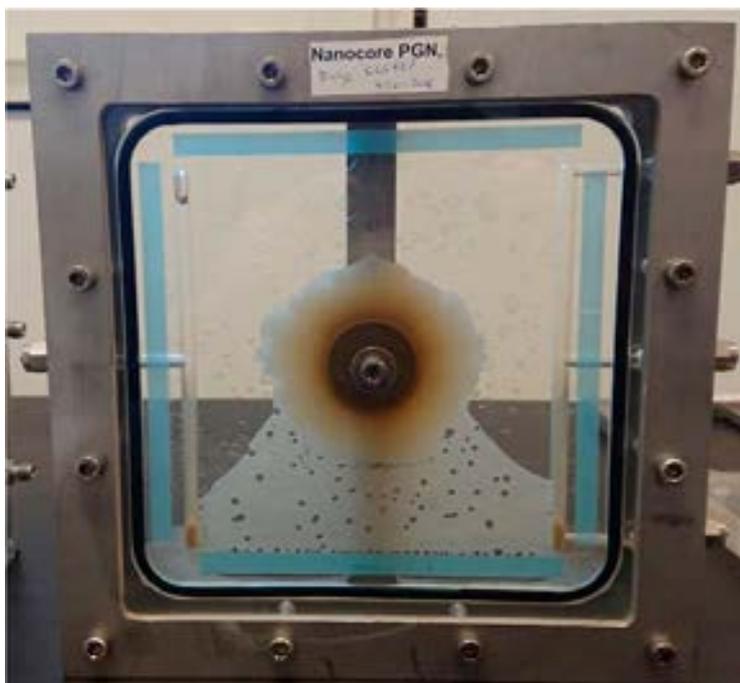


Figure 11-12. Sedimentation experiment with a montmorillonite washed in water with 1 mM of sodium chloride in a 0.2 mm artificial fracture. In the middle, there is a disc of compacted clay, with the expanding paste and the non-swelling gel forming a circle around it. The sedimenting flocculation (secondary gel) can be seen below the circle. A sol phase is also visible in the channels on the sides.

In addition to the experiments carried out by SKB under its own auspices, experiments and model development by external suppliers will continue. At present, the focus will be on sedimentation/flocculation.

11.3.3 Self-healing of bentonite

In the event of mass loss of bentonite from a barrier, for example due to erosion, it is important to understand how the barrier self-heals. The programme for this is described in Section 11.1.4.

11.3.4 Mineral stability

Bentonite has been selected as a barrier material, since it is expected to be stable in the long term in the environments prevailing 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 alteration of the mineral structure. The factors that primarily determine the stability are temperature, availability of potassium, and pH. Redox conditions may be important as well. Potassium concentrations in Swedish groundwaters are generally low, but there may be relatively large quantities of potassium in the rock. An elevated temperature is expected during a relatively brief period in the buffer in the Spent Fuel Repository, while the backfill and 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 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 is negligible.

Important field tests for studying the mineral stability of bentonite clay are LOT and ABM in the Äspö HRL. LOT studies the ageing of MX-80 bentonite with a copper heater at 90 °C and 130 °C. ABM investigates more than ten different bentonites from different parts of the world together with a steel heater at 130 °C (the exception is test package 5, which was exposed to higher temperatures). In both types of test, compacted bentonite rings of 30 cm diameter are used, stacked around the heater, which is a total of four metres high in LOT and three metres high in ABM.

Current situation

The most important results from ABM's test package 2 are presented in Hadi et al. (2017) and can be summarised as follows:

- The identified processes agree with those reported in previous tests at the Äspö HRL and the Swiss hard rock laboratory, Grimsel. Furthermore, the study facilitated a more detailed understanding regarding iron/bentonite interaction in a repository-like environment. Corrosion of the heater gave rise to an iron front, which extended ~5–20 mm into the adjacent bentonite, with the exception of the iron-rich bentonite material (Rokle) where no iron increase was observed. The main fraction of this iron accumulation in the clay was found to be iron(III) oxides. Some iron(II) was also found, which had diffused further into the clay but whose nature has not yet been identified.
- Cation exchange reactions between sodium, calcium, magnesium and potassium ions occurred both horizontally and vertically in the experiment. The cation exchange capacity (CEC) showed almost no change as a function of the distance to iron and the heat source.
- Accumulation of magnesium and of calcium sulphate close to the heater.
- No indications of smectite transformation in any of the samples. This is supported by the nearly constant CEC values and the constant aluminium/silicon ratio close to the heater. This result is in line with other in situ experiments and indicates the stability of smectite under tough repository-like conditions.

Studies are under way regarding the interaction between the bentonite barrier and cement in the waste form in BHA in SFL. Two sets of one-dimensional reactive transport models for bentonite transformation in BHA have been developed and implemented by means of ICP, an interface between the software Comsol Multiphysics and Phreeqc. The first set of models only include the bentonite system and treats the concrete as a boundary condition. The second set handles the complete interaction between the concrete and bentonite barriers. The purpose is to assess the mass of montmorillonite transformation in the bentonite barrier as a result of the interaction with the cement surfaces of the waste container during a period of 100 000 years.

With the current assumptions, the models predict a relatively extensive transformation of the bentonite barrier.

Programme

Work is still in progress on ABM's test package 2, and it is now partly being carried out in parallel with the analyses of ABM's test package 5 (Figure 11-13). Package 5 was exposed to very high temperatures (approx. 200 °C), after which the blocks dried and fractured to a large extent. The chemically more affected samples appear to stem from test package 2, where a new magnesium-rich trioctahedral smectite could be identified in the contact surface with the steel heater, although in a very small content and only in one type of bentonite. It is also clear that different bentonites interacted differently with the steel heater. It cannot currently be determined whether these differences between the clays would also appear in an experiment with a copper heater.

The work with test package 2 and 5 from ABM continues. In particular, swelling-pressure data and hydraulic conductivity data will be collected, as well as X-ray diffraction data for the identification of possible transformation phases. An important experience from the analysis of test package 2 was that the bentonite clays can have different chemical stability and can interact in different ways with the metallic heater. The fact that only Wyoming bentonite (MX-80) has been tested in the field with copper is a problem. Therefore, new field tests of the ABM type are planned, but with copper heaters instead of steel heaters, so that there is a flexibility in the choice of buffer in the Spent Fuel Repository in the future.



Figure II-13. Sample of Calcigel bentonite from ABM's test package 5, which was retrieved in 2017.

Two test packages from LOT will be retrieved next. Standard analyses will be carried out, similar to those carried out for previous packages, but a large number of samples will also be investigated by X-ray diffraction (XRD) and X-ray fluorescence spectroscopy (XRF) in order to find magnesium-rich phases in contact with the heater. Previously, small increases of magnesium have been observed (for example in LOT and the Prototype Repository) and the current goal is to try to find the reason for these observations. Another focus area is to study the redox chemistry of iron in the bentonite. The bentonite is dominated by iron(III) at installation, but at the time of retrieval of the outer section of the Prototype Repository (Olsson et al. 2013), an increase in the iron(II)/iron(III) ratio in the buffer was observed. This increase could not be fully explained, so more information will now be collected in order to improve the understanding of the process.

Diocahedral smectite has been identified in the Kiruna mine, down to depths of at least 1 200 metres beneath the surface. The smectite is relatively pure and swells with water as expected. The Kiruna mine is one of the largest iron ore deposits in Europe. Significant zones of clay have appeared close to the magnetite/apatite deposits in up to about 50 metres thick soft transition zones. There are indications that the smectite in Kiruna might be extremely old and therefore potentially very interesting as a natural analogue provided that it can be dated.

There are obvious challenges regarding the interaction between cement and bentonite in BHA. There are technical solutions, such as a reduction of the amount of grout around the waste or a more massive bentonite barrier, but these solutions might not be practically achievable. The uncertainties that could be most significant for the process are the transport properties in the concrete in the waste container and the montmorillonite dissolution kinetics. Precipitation kinetics of the secondary phases could also affect the system, and this mechanism has been completely neglected in the study. Further studies will be necessary, both experimental studies and modelling. Satoh et al. (2013) shows that the reactive surface in compacted bentonite is considerably smaller than the one measured in dispersed solutions. This effect has not been addressed in the ongoing study. Preliminary calculations show that the effect may be critical for the durability of the bentonite barrier in BHA. The next steps in the programme are therefore:

1. Modelling of the interaction between bentonite and concrete in BHA considering the smaller reactive surface.
2. Further experiments to verify that the surface reduction during compaction is a correct approach to the process.

The results will be of great importance for the design of the barriers in BHA.

11.4 Barrier design

The documentation presented in the application for the Spent Fuel Repository described conceptual designs of buffer, backfill including dome plug, and closure. Since then, technology development has been conducted to perform basic design of the buffer, backfill and plugs in deposition tunnels. Based on updated technical design requirements, the designs of the buffer and backfill need to be revised.

A feasibility study concerning the SFL facility design has been carried out and indicates a need for targeted development activities regarding technical solutions for the design and construction of repository structures in the rock vaults.

11.4.1 Buffer in the Spent Fuel Repository

Current situation

Based on the updated technical design requirements (Posiva SKB 2017), a new methodology for adapting the design of the buffer to different bentonite materials has been developed. The methodology has been used for MX-80 by Luterkort et al. (2017, see Section 7.2). It is also presented here how the expansion that takes place before the backfill is installed above the deposition holes (see further Section 11.6.2) and the upward swelling of the buffer after the installation of the backfill have been taken into account.

Programme

During the RD&D period, the feasibility of installing the buffer in the form of segmented blocks will be investigated. This will include a full-scale installation experiment and modelling of the early thermal, hydraulic and mechanical processes that may lead to expansion or shrinkage of the buffer. The design and manufacturing method for buffer pellets will also be studied.

11.4.2 Backfill in the Spent Fuel Repository

Current situation

A laboratory for routine characterisation of bentonite materials has been built and put into operation at the Äspö HRL. Two of the central requirements for the backfill, swelling pressure and hydraulic conductivity, can be determined by means of the methods that are now being used in the laboratory. By being capable of making these determinations under its own auspices, SKB has had the opportunity to build a body of data during the RD&D period, containing relevant materials and densities, for describing the relation between density and swelling pressure and the relation between density and hydraulic conductivity. The backfill design regarding installed density has been compared with these relations and has fulfilled the requirements.

The models for how the backfill counteracts the upward swelling of the buffer have been developed further based on results from tests in laboratory scale and full scale (Börgesson and Hernelind 2017). The models have been used to show that the requirements on the buffer's swelling pressure and hydraulic conductivity are met even after the buffer has compressed the backfill and a state of equilibrium has been reached.

Specifications of the backfill components and the installed backfill as a whole have been developed, and a programme has been proposed for verifying the agreement with production specifications.

Programme

The backfill design will be updated based on changes of the tunnel profile, which is a consequence of a policy decision on a new rock excavation method. The models that describe the ability of the backfill to restrict the upward swelling of the buffer will also be further developed and used in the design update. How inflowing water is to be handled will also be studied as a basis for the updated design of the backfill. This work will also consider logistics studies providing data on how long the installation of buffer and backfill will take in the final repository.

11.4.3 Design of clay barriers in SFL

Current situation

During 2015–2019, an assessment of the proposed concept with respect to the safety after closure has been carried out. The safety evaluation was performed to give SKB a basis for assessing whether the proposed concept (Section 2.1.2) has the potential to meet the requirements on safety after closure.

The results of the safety evaluation indicate that the repository concept has the potential to meet the requirements on safety after closure, given that the future site exhibits sufficiently favourable properties, in particular regarding solute transport in the rock. The sensitivity cases for different barrier solutions analysed in the safety evaluation form a basis for the continued work with developing the engineered barriers.

Programme

During the RD&D period, the development of the design of waste vaults continues, with a focus on repository structures for the handling and storage of the waste and on methods for backfilling with bentonite. The development of structures for safe handling and storage of the waste must in particular consider the feasibility of an efficient and robust process for backfilling the waste vaults. The work includes further evaluation concerning which initial state can be achieved given the practical limitations in the waste vaults, and how the initial state can be verified at closure.

11.5 Manufacturing, inspection and testing of buffer and backfill components

In order to ensure that buffer and backfill can be installed in the Spent Fuel Repository in such a way that the technical design requirements are fulfilled, good control and oversight of the entire chain are needed, from mining of bentonite to finished installation of buffer and backfill. Work is under way for designing the whole process from the choice of materials to the finished installation. The different steps in the process and the quality inspections required to ensure the quality of the installed barriers are described based on this work. This section gives more detailed descriptions of the following areas:

- Material supply and quality assurance of bentonite materials.
- Manufacturing of buffer components.
- Manufacturing of backfill components.

11.5.1 Material supply and quality assurance of bentonite materials

Current situation

By adapting the density and water content of the buffer and backfill components, the technical design requirements can be met for many different materials. The methodology called adaptive design by SKB has been evaluated and developed further within the framework of recently completed technology development projects. A central part of this work is the methods with which the materials are characterised (Table 11-1). The choices of methods in the table are those that are currently in effect, but they may change based on future development work. The different characterisation levels are also selected to be used at different places in the manufacturing process in the Spent Fuel Repository.

The following is a brief description of how materials can be evaluated and how their design can be adapted. Materials that are deemed to fulfil the long-term function are purchased in small volumes, sampled and characterised in stages in accordance with characterisation level 1 and 2 in Table 11-1. The best materials, for which a buffer and/or backfill design can be produced, are purchased in larger volumes, after which the quality is confirmed (level 1) and a test for full-scale production is carried out (level 3). If necessary, supplementary analyses are also carried out (level 2).

The latest technology development project, in which seven different materials have been evaluated, indicates that SKB will be able to establish a list of potential suppliers and materials with designs for both buffer and backfill. This is positive from both a quality and procurement perspective. With several available materials, the choice of material can be made based on an integrated view starting from its long-term function, production of components and cost.

Table 11-1. Methods for characterisation of bentonite.

Sampling	
	Sampling, labelling and archiving of bentonite clay for material characterisation at the Äspö HRL
Characterisation level 1	
1a	Determination of water content in bentonite Determination of granule size distribution of bentonite
1b	Chemical analysis of bentonite clay by means of X-ray fluorescence spectroscopy, XRF Determination of swelling pressure and hydraulic conductivity – simplified method Determination of cation exchange capacity (CEC) in bentonite clay with Cu-triethylenetetramine/UV-Vis Determination of carbon and sulphur in bentonite
Characterisation level 2	
	Determination of swelling pressure and hydraulic conductivity Extraction of exchangeable cations, EC Phase analysis of bentonite clay by means of X-ray diffraction, XRD Determination of particle size Determination of the compaction curve for bentonite Uniaxial compression tests on bentonite Determination of thermal properties of bentonite
Characterisation level 3	
	Pellets, dimensions and strength Block weight, dimensions and visual inspection

The goal of SKB's buffer and backfill facilities will be to ensure that quality-assured components are produced and delivered in a cost-effective way. As a part of the development work, a strategy for the process has been formulated. The process has been divided into four main parts:

- Supplier and material evaluation.
 - Choice of material.
 - Purchase.
- Quality control for delivery of materials on an industrial scale.
- Quality control during the manufacturing process.
 - Sub-process Hargshamn (bentonite reception facility).
 - Sub-process Forsmark (production plant).
- Quality control of the components.

The strategy also includes a proposal for how the process will be controlled and organised as a part of SKB's management system.

A proposal for the scope of sampling and analysis has been formulated for the different parts of the process, and it will be optimised as more material statistics are produced. The starting point is that a control is carried out in conjunction with the delivery (samples on level 1 and 2) and that the material is controlled in conjunction with the manufacturing of blocks and pellets, where also dimensions and strength are controlled (level 1 and 3). In the sub-processes where the material is processed from coarser granules to production material, primarily process-governing parameters such as water content and granule size distribution (level 1a) are controlled.

During the latest RD&D periods, SKB has built a laboratory on Äspö, which can carry out the majority of the analyses in accordance with Table 11-1. A few analyses are still being sent to external laboratories. The laboratory permits testing and development of sampling and analysis methodology prior to the construction of the Spent Fuel Repository.

Programme

In order to have a robust and reliable system for quality assurance prior to integration tests and commissioning tests in the Spent Fuel Repository in Forsmark, the steps in the preliminary quality plans need to be tested for large volumes of material in conjunction with future full-scale tests, so as to build up an understanding of the homogeneity of the materials on different scales and thereby be able to optimise sampling plans and quality plans. In this work, a good understanding of the material suppliers' quality systems will be important. Plans for how the implementation in the final repository will be carried out also need to be developed in more detail.

A strategy needs to be formulated regarding the amount of processing required for the procured bentonite materials. This affects the design of the facility in the receiving harbour. If inhomogeneous materials are procured, the facility and quality system must be adapted in order to be able to homogenise materials on a large scale.

11.5.2 Manufacturing of buffer components

Current situation

Buffer blocks have historically been manufactured in relatively large quantities, and during the past period, they have been manufactured for the experiments carried out on full-scale blocks. Studies have been conducted to investigate how the absence of lubricant during production affects the blocks, and completed tests (Eriksson 2017) indicate that this will work. These tests have also studied the homogeneity in terms of density. Without lubricant, the homogeneity in density becomes slightly lower than when lubricant is used, but the differences in density within the block is relatively small compared with the differences within the deposition hole after installation. The manufacturing method currently in use is deemed to be possible to use in a future production plant in order to produce buffer blocks with sufficient capacity and quality.

However, the pressing technology that is used today is fairly labour intensive, and the process needs to be made more efficient to increase material yield and automate the production. The press die for manufacturing of buffer blocks is therefore being assessed to reduce the need for processing of the buffer blocks and to simplify the production.

In parallel with the work on the press die, work is also under way for studying the possibility of dividing the buffer blocks into smaller parts, so-called segmented blocks, in order to be able to use smaller and less expensive presses. This would facilitate production and handling of the blocks during production, but at the same time make the installation more difficult.

The manufacturing process has been updated, and details on the materials' behaviour and the requirements in different process steps are being described for use in a detailed design of the production equipment.

Programme

The production process will continue to be optimised in order to be able to produce blocks in an efficient manner. The process will also continue to need to be developed, and requirements and prerequisites of the different process steps need to be refined prior to the detailed design of the production system.

Large series of buffer blocks will be produced to obtain better statistics on the extent of variations in density and dimensions that can be expected in production. Buffer blocks will also be produced with new bentonite materials in order to ensure that the manufacturing functions with other materials.

Further development of the segmented blocks will be carried out. Among other things, full-scale blocks for a deposition hole will be manufactured. The segmented blocks will then be installed in a deposition hole in the Äspö HRL in order to gain experience with the installation process. With respect to pellet manufacturing, new methods for measuring quality will be developed.

11.5.3 Manufacturing of backfill components

Current situation

The manufacturing of the backfill blocks is deemed to be an industrially mature process, with machines and technology that are being used in the industry. The manufacturing greatly resembles similar industrial production of, for instance, refractory brick. The difference is that the size of the backfill blocks is greater than those that are normally produced. This means that, among other things, de-airing of the blocks takes a longer time. In a future production, vacuum is assumed to be used in the press, which facilitates de-airing and reduces the risk that trapped air under high pressure causes fractures in the blocks. This has not been possible to test, since there is no equipment available.

Backfill blocks have previously been manufactured for full-scale tests, and the results show that it is important to continuously check the material and maintain an even quality in order to fulfil the requirements on density and dimensions.

Programme

The production process will continue to be optimised in order to produce blocks in an efficient manner. The process will also need to be developed in more detail, and requirements and prerequisites of the different process steps need to be refined prior to the detailed design of the production system. Larger series of backfill blocks need to be manufactured in order to increase the current statistics. This may also facilitate refining the requirements on the material, such as granule size distribution and water ratio.

Like buffer pellets, backfill pellets will also need to be quality tested to verify that they meet the stipulated requirements. Methods for quality testing backfill pellets will therefore also be developed.

11.6 Disposal and installation of buffer and backfill

The strategy for equipment and technical systems is to use available standard equipment wherever possible. If there is no standard equipment, the possibility of making a modified standard is studied, and the last alternative is to develop a separate special machine. For transportation and installation of buffer and backfill components in the Spent Fuel Repository, some special machines are needed.

The strategy for equipment for transportation and installation of buffer and backfill components is based on a modular approach using the same equipment for several applications (Figure 11-14). Using modules has many advantages, such as:

- Faster error handling.
- Easier to maintain service expertise.
- Simpler stocking of spare parts.
- Lower development costs.

Figure 11-14 shows that transportation equipment, buffer handling equipment and backfill equipment are all based on a universal chassis, which is a key component in this strategy. Studies are being conducted to verify that it is possible to construct a universal chassis that meets SKB's requirements and can be used for planned applications.

The equipment will first be tested separately and then integrated with other equipment and systems. Full-scale integration tests, in which deposition of the canister, installation of buffer, backfill and plugs are tested, will be conducted at the Äspö HRL or another suitable site. The purpose is to verify that SKB's sequence for deposition and backfill works as intended before a detailed design of the special equipment needed in the Spent Fuel Repository is created.

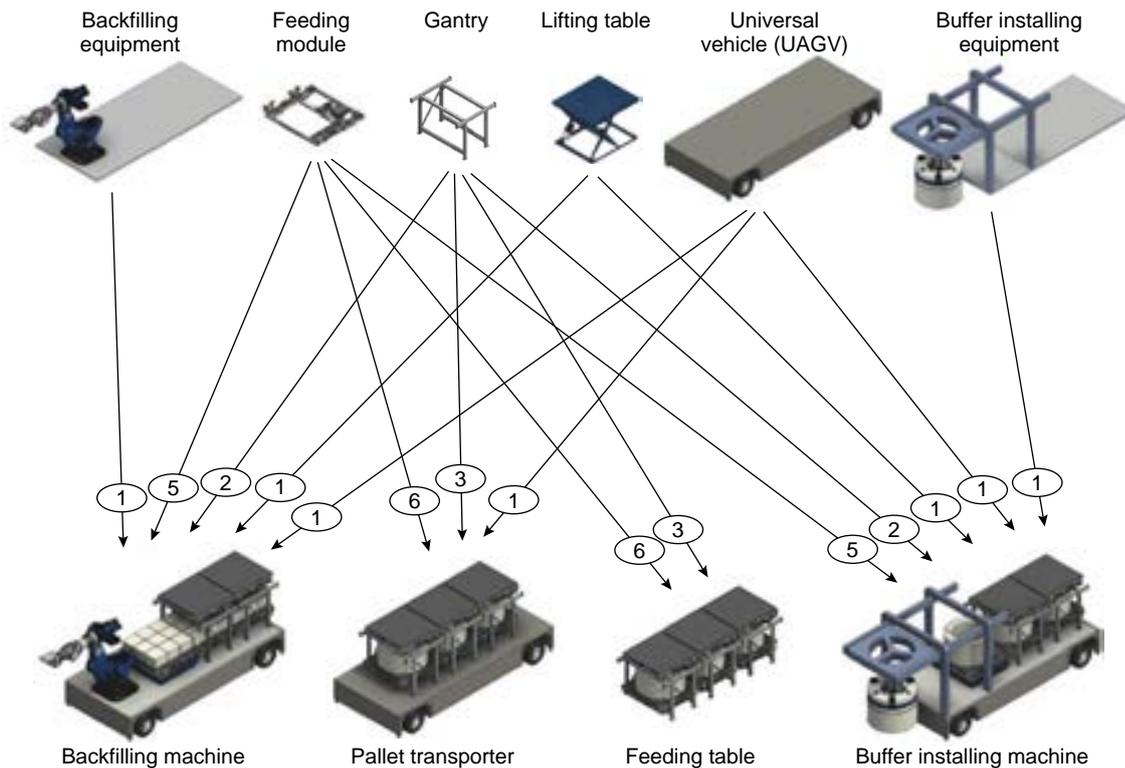


Figure 11-14. Strategy for modularisation.

11.6.1 Deposition

In the Spent Fuel Repository, equipment is required in order to be able to carry out processes such as the deposition of canisters. Most of these processes include individual steps that must be carried out in a controlled and safe manner and with high repeatability, which requires some automated functions.

Current situation

To control and monitor automated functions and processes in industrial applications, some kind of supervisory system is used in most cases. The supervisory system gives an operator the opportunity to get an overview of what is happening in the process. In order for these processes to be visualised in the system, the equipment or machine must have an interface that can transfer adequate information to the supervisory system. A demonstration system that provides the means for controlling and monitoring different prototype machines and equipment, independent of the supplier, has been developed and designed to manage all the functions of a prototype of the deposition machine.

In 2015, testing of this demonstration system was completed using the prototype machine for deposition, Magne, showing that the stipulated requirements on reliability, availability and repeatability in automated operation can be fulfilled. The results of the testing campaigns show that the technology is applicable and also possible to implement on a deposition machine operating in a fully automated mode without a driver to assist.

Programme

Based on the completed work, the degree of automation in the Spent Fuel Repository will be determined for deposition and backfilling, followed by further development of the supervisory systems required to control the processes.

During the RD&D period, the deposition machine's mechanical components such as grapple and hoist will be developed further. A safety hatch for deposition holes will also be developed.

The need for further technology development for retrieval of canisters, buffer and part of the backfill will be clarified in the RD&D period.

11.6.2 Installation of buffer

Current situation

During the past RD&D period, SKB tested two installation methods in full scale in deposition holes in the Äspö HRL (Luterkort et al. 2017). One of the methods was planned to be used in relatively dry deposition holes, and the second, in relatively wet deposition holes. The method for dry deposition holes worked well and the buffer could stay in the deposition hole for three months with a canister equipped with a heater, without adversely affecting buffer density. The method for wet deposition holes, in which a developed buffer protection was installed in the upper part of the buffer, was less successful. There was fracturing in the buffer blocks at the top of the canister, and the thermal gradient led to water condensing in the fractures. It could be concluded that this installation method was not robust. But the method for dry deposition holes, where buffer blocks and canister are installed in succession and then stands without support until the backfill is installed, could be shown to work well for a large majority of the planned deposition holes in Forsmark. The sequence for deposition is carried out in such a way that the buffer in wet deposition holes is supported by the backfill after a relatively short time. Most of the wet holes can thereby be used as well.

There is a prototype for a buffer lifting tool, which has been tested for some time. Prototype equipment for installation of pellets in deposition holes has also been developed and tested at the Äspö HRL.

Programme

During the RD&D period, the knowledge of early thermal, hydraulic and mechanical processes in the buffer for different water inflows will have to be developed further. The knowledge needs to be sufficiently developed to assess how the processes are affected by different designs of the buffer components, for example the effect of segmented blocks. The size of heaving and possible shrinkage in different water inflow situations must be possible to determine so that it can be taken into account when the buffer is designed. The boundary between dry and wet deposition holes must also be determined based on the knowledge of the early THM processes. To achieve this, tests on different scales and modelling will be carried out.

To verify that the installation of the buffer works as intended and yields results within acceptable intervals, a new prototype equipment will be developed and constructed. Tests will be carried out in full scale both above ground and in underground conditions as a part of the integration tests at the Äspö HRL.

11.6.3 Installation of backfill

Current situation

Equipment for the installation of backfill has been developed and tested, and the tests were concluded with an installation test at the Äspö HRL (Figure 11-15). A total of 12 metres of backfill was installed in a tunnel at a depth of 450 metres with the same dimensions as a future deposition tunnel. The installation was carried out using prototype equipment and some temporary solutions including manual handling that will not be used in the Spent Fuel Repository, which means that no conclusions regarding capacity can be drawn. The assessment is nevertheless that the proposed installation concept is suitable for backfill. The test showed that it is possible to install blocks and pellets in such a way that the installed density will be sufficient for the backfill to meet its requirements (Arvidsson et al. 2015). New equipment for automatic installation of the pellet bed has been constructed as a prototype and has been tested with good results.

An important issue for backfill installation is the knowledge about the water distribution in the rock and information on the size of the water inflow to the deposition tunnel, which is obtained from the detailed characterisation in conjunction with the excavation of the tunnels. In cooperation with Posiva, work has been carried out to develop methods for water management during the installation phase. The work has resulted in a number of applicable methods, where the choice of method is dependent on the water inflow's size and distribution in the deposition tunnel (Sandén et al. 2018a).



Figure 11-15. Block installation at a depth of 450 metres in the Äspö HRL.

Programme

To verify that the installation of the backfill works as intended and yields results within acceptable intervals, prototype equipment for the backfill installation will be constructed during the RD&D period. Tests will be carried out in full scale both above ground and in underground conditions as a part of the integration tests at the Äspö HRL.

Regarding water management during backfill installation, work to develop and test suitable methods will continue. Potential methods will also have to be tested in full scale. Cooperation with Posiva is ongoing and will continue with respect to these issues. If the tunnel profile changes as a result of a new excavation method, the backfill design needs to be adjusted.

11.7 Borehole sealing

There are a large number of investigation boreholes adjacent to the area where the surface facility for the Spent Fuel Repository is proposed to be constructed and in the rock geometry where the underground facility is proposed to be built. A number of these boreholes must be sealed before the start of construction. Remaining investigation boreholes that are open today need to be sealed in later stages of the construction process, since they can be utilised for monitoring during the construction phase. There are also some boreholes for the extension of SFR that must be sealed prior to the construction phase.

The purpose of sealing the boreholes is to restore the hydraulic properties of the bedrock so that the boreholes will not constitute flow paths for groundwater and thereby contribute to an increase of radionuclide transport to the ground surface (from a canister if it is damaged).

Current situation

The SR-Site evaluation of the reference design for the borehole seal showed that the impact of poorly sealed boreholes was very limited and that the requirements were possibly too strict, since even open boreholes without seals appear to have a very limited impact on the groundwater flow in the Spent Fuel Repository.

Also subsequent sensitivity analyses (Luterkort et al. 2012) show that a hydraulic conductivity in the sealing material of up to 10^{-6} m/s does not result in increased radionuclide transport to the ground surface to any significant extent. This has led to SKB revising the requirement on borehole seal tightness. In a technology development project, SKB has developed a method for borehole sealing to meet the new requirements. The project has carried out verifying laboratory tests of the seal's constituent materials (sand, bentonite, concrete, and copper expanders) and has conducted a full-scale installation test in borehole KAS13 on Äspö. The laboratory tests and the installation test have been documented in a technical report (Sandén et al. 2018b).

The proposed method can be summarised as follows:

- Classification of investigation boreholes into three classes, borehole class 1–3, based on hydraulic contact with the repository, borehole depth and distance to the repository.
- Simplified installation method for borehole sealing of investigation boreholes that are not in hydraulic contact with the repository. These boreholes belong to borehole class 1 and 2, where class 1 includes shallow boreholes and class 2 deep boreholes. The boreholes in class 1 are sealed with bentonite pellets (Figure 11-16a). The boreholes in class 2 are sealed with sand or concrete in the part of the borehole that is in the rock. The upper part of the borehole, which is in soil, is sealed with bentonite pellets (Figure 11-16b).
- Borehole sealing according to the sandwich method for borehole class 3. Boreholes in this class are in hydraulic contact with the repository and therefore require the tightest seal. These holes are mainly sealed with sand and with strategically placed tight sections consisting of bentonite, copper and concrete (Figure 11-16c).

The completed full-scale test included installation of two different kinds of concrete (quartz-based concrete and construction concrete), 106 metres of sand, about ten metres of bentonite, and seven copper expanders.

Programme

Based on the results from the development of the method for borehole sealing, the following analyses have been made regarding remaining questions:

- Installation has, at present, been carried out in cored boreholes with a diameter of 76 mm. In order to be sure that the method also works for coarser percussion-drilled boreholes, tests should be carried out in this type of borehole.
- Concrete is an installation component in the borehole seal. The long-term properties of the concrete need to be evaluated.
- The design currently includes a top seal of *in situ* cast concrete in borehole classes 1 and 2. The need for a top seal has to be assessed further.
- In borehole classes 2 and 3, the installation material consists mostly of sand. The sand is uniform with a grain size distribution of 0–2 mm. The rock volume in Forsmark contains fractures with apertures larger than 2 mm, which entails that some of the sand could disappear into these with a risk of altering the sand fill density in the borehole over time. The extent of this needs to be studied further.
- Installation of the borehole seal's tight sections in borehole class 3 is carried out with a drilling rig. In deep boreholes, the transport of the drill string down to the intended level is time-consuming. Therefore there is a need to evaluate the possibilities of optimising the installation of tight sections.

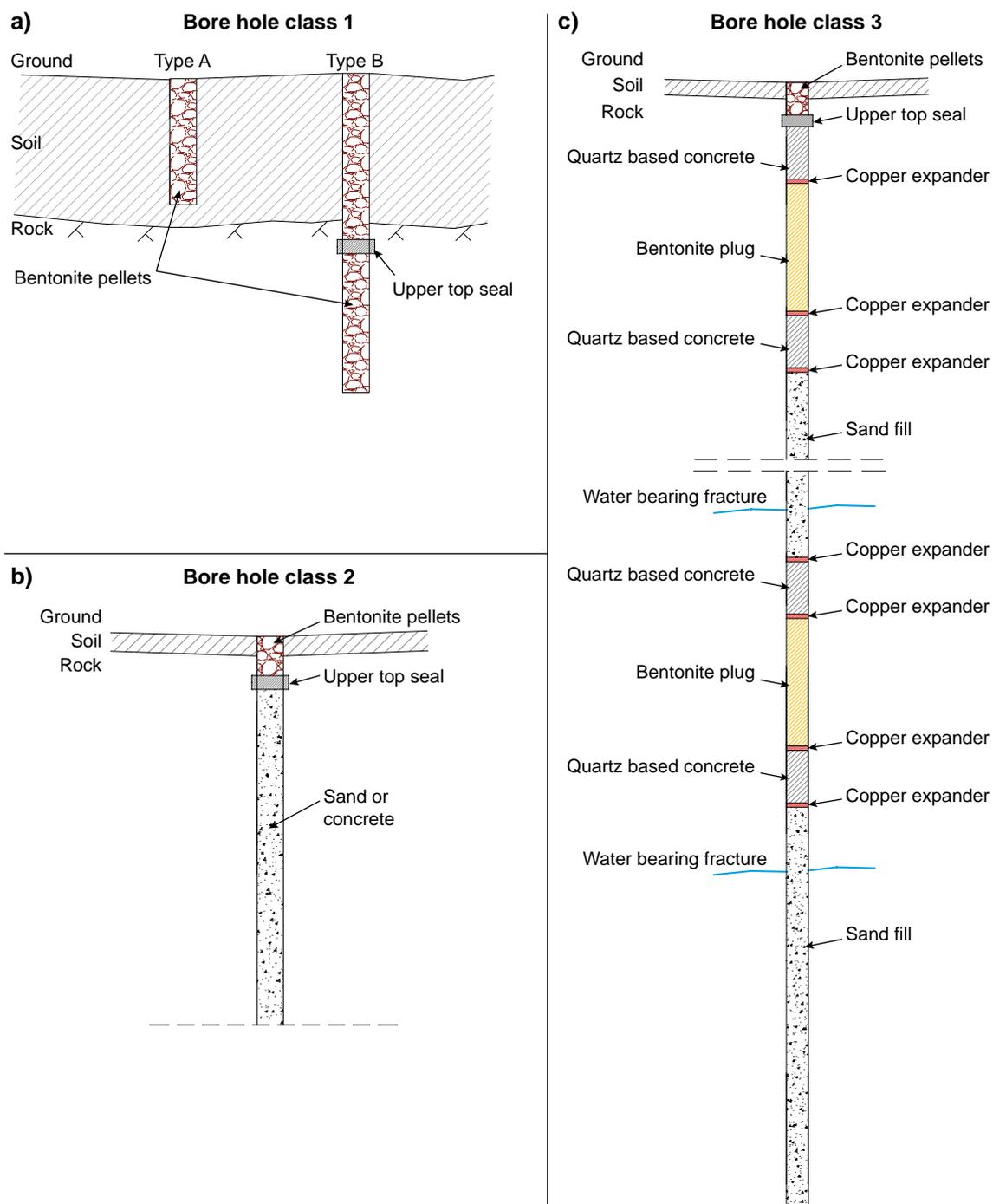


Figure 11-16. a) Design of borehole seal for boreholes in borehole class 1. b) Design of borehole seal for boreholes in borehole class 2. c) Design of borehole seal for boreholes in borehole class 3.

11.8 Closure

11.8.1 Closure of the Spent Fuel Repository

Current situation

A simplified design of the closure has been proposed based on completed sensitivity analyses. These are described in Luterkort et al. (2012). The size and function of closure components may have an impact on repository design, which means that continued efforts are needed in this area.

Programme

An overall closure plan will be drawn up to yield more details with respect to the closure sequence and the function and size of the closure plugs.

11.8.2 Closure of the Final repository for short-lived radioactive waste

SR-PSU and subsequent analyses have resulted in updated requirements on the closure components for SFR. There is thus a need for further development of the design and further studies concerning installation of the closure. Detailed design of the extension of SFR and improved knowledge of rock properties can also affect the design of the closure components. Based on the revision, the closure plan will be updated prior to the PSAR.

Current situation

In conjunction with the submission of the application for the extension of SFR, SKB formulated a closure plan for the extended facility (Luterkort et al. 2014). Closure is described on a conceptual level and the knowledge concerning materials, design and installation will be improved prior to the PSAR.

Separate studies of individual closure components, for example modelling of concrete plugs, have been carried out during the period to develop the understanding of property evolution over time and to define the requirements in more detail. Further feasibility studies have been conducted concerning the practical conditions for the execution of closure in parts of the existing facility. The studies are meant to provide a basis for a comprehensive update of the closure concept at a later stage.

Programme

The set of requirements linked to closure of the facility will continue to be developed during the RD&D period, both property requirements for different backfill and closure components and the practical conditions for installation in the repository.

SKB is planning for continued technology development of concrete plugs to achieve a robust design that meets the relevant set of requirements and the practical conditions in the repository. When the final layout of the extension has been determined, a more precise calculation of the dimensions of each concrete plug may be required.

Based on a developed set of requirements, development of the concept for the earth dam plug and the transition material can continue. Calculations, parameter studies and modelling are primarily envisaged for studying the property evolution over time of the earth dam plug and its different material components.

The developed set of requirements are also a basis for a more detailed analysis of how bentonite in the tight sections is to be designed and installed. Possible solutions for achieving sufficiently high density in the hydraulically tight sections with bentonite pellets or granulated bentonite need to be identified and evaluated.

12 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 during which the waste must be isolated so that it will not constitute a risk for human health and the environment. The bedrock will also constitute a barrier that as far as possible prevents or retards the transport of radionuclides from a repository to the surface system. 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 in the final repositories must also be designed in such a way that long-term stable conditions are not jeopardised. A large part of the technology development and research issues related to the bedrock and the underground openings in the final repositories are common for the three different repositories.

For the assessment of post-closure safety, there are a number of issues that require further research. These issues need to be resolved prior to the SAR with respect to both the Spent Fuel Repository and the extension of SFR, in which the state of knowledge in the PSAR and at the time of the decision to start construction are important milestones. The recently completed safety evaluation of SFL was based on current knowledge of the barrier function of the bedrock, and on previous knowledge from the work on the Spent Fuel Repository and SFR. The safety evaluation of SFL has pointed to a few areas where further knowledge is needed (see also Chapter 6).

Underground openings consist of the rock cavities and constructions in the rock that are required for the final repositories' underground parts. In the design of the underground openings, the following must be taken into consideration:

- The spatial geometry and location of the openings.
- The rock surrounding the openings, which is affected by construction.
- Structures and foreign materials used for sealing and rock support, and for conducting operations in the facility, which remain in or on the rock surrounding the openings after deposition, backfilling and closure.

The most important factor with respect to the barrier function of the rock in the Spent Fuel Repository is the placement of deposition areas and deposition holes considering the thermal, hydrogeological, mechanical and chemical properties of the bedrock.

12.1 Characterisation and modelling of rock properties

12.1.1 Mechanical properties of rock mass

Coupled rock mass models assume that the properties of the rock can be described on different scales. The descriptions must correctly represent a crystalline fractured rock taking into account its inherent deformation zones and fractures on different scales. Empirical methods for characterisation of the mechanical properties of rock mass are very effective for the design and construction of the repositories, but the assessment of post-closure safety requires a deeper and more fundamental understanding and characterisation of the coupled thermal, hydraulic and mechanical processes (THM processes) that affect the behaviour of the rock mass.

The numerical tools for modelling a synthetic rock mass have been continuously developed and in recent years, they have undergone a very rapid development. Synthetic Rock Mass (SRM) refers to models of rock in which the fracture network is explicitly represented together with the intervening, so-called intact, blocks.

Current situation

For characterisation of the mechanical (M) and hydromechanical (HM) properties of the rock mass, discontinuum methods have been developed based on the discrete element method (DEM) in the software UDEC (Itasca 2014b) and 3DEC (Itasca 2013). These tools have been used by SKB as complements to the traditional, empirical methodology for calculation of the mechanical properties of the rock mass (see for example Olofsson and Fredriksson 2005, Glamheden et al. 2007).

Particle-based numerical tools of the type Particle Flow Codes (see for example Itasca 2014a) have previously proven capable of reproducing fundamental and more subtle aspects of fracturing and fracture propagation in rock (Potyondy and Cundall 2004). This type of software has recently been developed to include discrete fracture networks and thereby permit more complex representations of the rock mass (Mas Ivars et al. 2011). The software development has made it possible to model failure in intact rock and to include the effect of isolated (not block defining) fractures on rock strength and stiffness. The modelling of SRM is, however, still limited by numerical computation capacity, which leads to limitations in terms of model size (scale), degree of detail and complexity. Despite this, SRM models have decisively contributed to the development of DFN-based (Discrete Fracture Network) analytical equations, which can describe the properties of the rock mass. A strategy in which analytical and numerical models are used in combination has been formulated, and Darcel et al. (2015) recently applied this strategy on behalf of Posiva with the aim of defining an analytical method to predict effective elastic properties on different scales (up to 100 metres). In the past three years, the strategy has been further developed with respect to elastic properties (Davy et al. 2018a, Darcel et al. 2018).

The HM properties of individual fractures are of crucial importance for the rock's barrier function, since they determine how the local fracture transmissivities vary with the stress and stiffness of the rock mass and how the 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 improve the knowledge of how to scale up a fracture's shear properties. Conclusions and experience from the Post project with respect to fracture shear strength, based on laboratory tests, numerical simulations and shear tests on fractures in situ, are summarised in Siren et al. (2017). Following the recommendations from the Post project, a second phase has been initiated, in which NWMO and SKB are collaborating. This new project phase is focused on laboratory tests and numerical and analytical modelling. Recently, Stigsson and Mas Ivars (2019) presented an improved procedure for estimating JCR (Joint Roughness Coefficient) based on a mapped fracture trace. The effect of large-scale undulation on the shear mechanical behaviour of fractures has been studied, supported by SKB (Lönngqvist and Hökmark 2015). The effects of fracture geometry on hydromechanical properties of crystalline bedrock were studied in a PhD project (Thörn 2015).

In conjunction with high normal stresses, the effect of dilatation (volume increase) decreases, but there are still uncertainties regarding the extent to which this affects transmissivity. An increase of the normal stress from 2 to 4 MPa appeared to suppress the increase of transmissivity very effectively in hydromechanical shear tests performed by Olsson (1998) and Esaki et al. (1999). On the other hand, increases in transmissivity of between one and two orders of magnitude were observed in tests performed under high normal stresses on artificially created granite fractures. It is not clear whether the behaviour of the artificial fractures can be considered to be representative of natural fractures. An experiment performed with a normal stress of 20 MPa, and repeated shear in the opposite direction, gave insignificant transmissivity effects for shear displacements up to 10 millimetres. The current understanding of the THM aspects in Forsmark and Laxemar (Hökmark et al. 2010) assumes that high normal stresses suppress transmissivity effects.

The remaining uncertainties associated with the HM properties of the fractures thus mainly concern the scale effect and changes in transmissivity as a function of shear load and normal load.

Programme

Efforts are planned for continued development of a “state-of-the-art“ methodology for calculating the hydromechanical properties of rock mass. The methodology should include the following:

- Establishment of effective HM properties on different scales based on DFN in the initial state.
- Rules for applying these properties onto DFN models, for example the transmissivity of fractures, have to be analysed and developed so that a relevant initial state can be obtained. Studies of how to conceptualise this in large-scale models (for example DarcyTools, 3DEC) are required.
- Comparison of in situ and/or laboratory tests and modelling is used to establish constitutive relations for describing how the effective HM properties vary with the stress field and the pore water pressure.

In order to improve the hydromechanical understanding of fractures, efforts to clarify the relationship between aperture and transmissivity are planned. The hydraulic aperture, and thereby the transmissivity, is a function of mechanical aperture, its standard deviation, and the fracture contact area. A survey of previous studies is planned (literature review) linked to flow tests in conjunction with shear in crystalline, hard rock and replica material. Furthermore, 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, coupled HM modelling of flow changes caused by shearing will also be carried out.

The following mechanical properties and fracture parameters must be determined or estimated in order to assess the impact of fractures on constructability and on the safety of the repositories after closure:

- The effect of normal stress on normal stiffness, shear stiffness, friction angle, cohesion and dilatation angle.
- The shear displacements where dilatation starts and ceases.
- The effects of fracture minerals and fracture width (mineral filling) on the mechanical and hydraulic properties of fractures.
- Scale effects on shear strength, normal stiffness, shear stiffness and dilatation angle, as a function of fracture roughness, undulation, contact surface and mineral filling.

12.1.2 Induced rock mass deformation caused by thermal, seismic or glacial load

SSM is of the opinion that certain conceptual issues from the review of the RD&D Programme 2013 were not sufficiently addressed in the RD&D programme 2016. SSM has particularly emphasised issues relating to fracture formation, fracture propagation and coalescence of existing fractures in the vicinity of the deposition holes. The processes and scenarios for which SSM particularly requested further analyses were the reactivation of deformation zones and fracture opening/closure due to large scale, thermally induced stresses in the final repository's near-field or near the surface (Min et al. 2005, 2013, 2015, Rutqvist and Tsang 2008), and the effect of an ice sheet on fracture opening/closure, fracture propagation and short-circuiting of the fracture network between deposition holes close to each other (Min et al. 2005, 2015, Backers and Stephansson 2012). Regarding technical modelling aspects, SSM emphasised that when it comes to rock mechanics and coupled processes, SKB's models should attempt to quantify the effect of realistic fracture geometries on the state of the rock and processes relevant to safety.

Current situation

Regarding long-term strength and stability, a coordinated study dealing with the behaviour of microfractures, subcritical fracturing and creep was presented in the RD&D Programme 2010 (Damjanac and Fairhurst 2010). The investigative work was based on interpretation of results from short-term creep tests on rock samples, numerical model analyses of the effect on rock strength of reduced fracture toughness due to stress corrosion, evidence from plate tectonic processes and observations of rock stresses in quarries. The article concludes that there is a stress threshold (i.e. a deviatoric stress that can be sustained indefinitely) for crystalline rock types (40–60 percent of the uniaxial compressive strength). Furthermore, it was concluded that extrapolation of an exponential model to the results of short creep tests provides a realistic, time-independent strength corresponding to a driving-stress ratio of about 0.45. This implies that a linear extrapolation to an ultimate zero strength is unwarranted (Potyondy 2007).

Since the RD&D Programme 2007, SKB has investigated long-term processes for rock strength evolution. There is, however, still a need to study the dynamic processes in fractures and faults, which are triggered in very short periods of time when exceeding the strength of the materials, for example during earthquakes. This need was also mentioned by SSM in the review of the RD&D Programme 2016. In this context, temperature effects should also be taken into account. Furthermore, there is a need to conduct studies to clarify whether time-dependent fracture growth affects stability and permeability in the Spent Fuel Repository's near-field during the different repository phases. In such studies, stress corrosion cracking at fracture edges (i.e. subcritical fracture growth affected by groundwater chemical

conditions) has to be taken into account in all load cases (tension, shearing, tearing) since fracture growth not only occurs under tension, but also under shearing or tearing at high confining pressures (Backers 2005, Backers and Stephansson 2012).

Programme

The following development is planned with respect to modelling capacity and capability:

- Fracture propagation. Development of analytical and numerical methodology for studying how fractures coalesce on different scales. This requires a programme with a limited number of laboratory tests.
- Long-term rock strength. Supplementary literature review and data collection are planned.
- Transmissivity changes in the rock mass due to stress changes caused by rock excavation or by seismic, thermal or glacial load, including the effect of realistic fracture geometries.
- Spalling and its effect on transmissivity in deposition holes and tunnels as a result of stress changes caused by rock excavation and by seismic, thermal and glacial load.

12.1.3 Rock stress

The rock stress model for Forsmark contains large uncertainties, and efforts to try to reduce these uncertainties are necessary. In its review of the RD&D Programme 2010, SSM pointed out that SKB should follow up previous initiatives that appear to have yielded promising results for Laxemar and Forsmark (Mas Ivars and Hakami 2005, Hakami 2006, Hakami and Min 2009). Moreover, SSM considered that SKB should study how the rock stress models for the near-field of the Spent Fuel Repository are related to the large-scale stress models that are used to calculate the stresses during a glacial cycle (Lund et al. 2009). An updated rock stress model in 3D, based on the current structural geology model (Stephens and Simeonov 2015), has been developed (Hakala et al. 2019). The main purpose of the model was to improve the methodology for estimating the spatial variability of the shape and orientation of the stress ellipsoid in high resolution (30–100 metres blocks).

Current situation

Two projects concerning rock stress measurements have been carried out over the past years. The SLITS-project (SLIm borehole Thermal Spalling) has developed a method for determination of the rock stress orientation (Hakami 2011) in boreholes. The method is based on inducing thermal stresses in a borehole until spalling occurs and, assuming correlation between the stress field and the induced fractures orientation, the maximum horizontal stress can be calculated. The second project (Hakala et al. 2013) has developed an LVDT (Linear Variable Differential Transformer) cell for rock stress measurement in tunnels. The instrument is installed in several pilot boreholes with a diameter of 120 millimetres in a tunnel section and is then overcored with a drill bit with a diameter of 200 millimetres. Inverse modelling of the in situ stress field is based on strain data collected from the LVDT sensors, which measure in four different directions, combined with a detailed laser-scanned model of the tunnel section where the measurement has been performed, and a modulus of elasticity measured on solid drill cores from the pilot boreholes. Verification measurements with an LVDT cell in the Äspö HRL at the 450-metre level, where the stress field is well-characterised, show that the results of the measurements with the LVDT cell agree with results from previous studies, which involved overcoring measurements, and with the previously developed stress model (Christiansson and Janson 2003).

Programme

The new updated rock stress model for Forsmark in 3D (Hakala et al. 2019) is one of the components in 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 for spalling and the stability within different repository volumes. The updated methodology will also provide boundary conditions for modelling of induced fracture displacements (due to seismic, thermal and glacial load) and coupled hydromechanical processes. The plan is to update the model continuously in agreement 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 detailed plans for this are presented in the detailed site investigation programme that will be submitted to SSM as a part of the application to construct the repository.

12.1.4 Modelling of discrete fracture networks

Discrete fracture networks, DFNs, are used in modelling of rock mechanics, groundwater flow and transport of solutes. The explicit description of individual fractures offers a great advantage in that entities used in the modelling can be measured directly in the field (for example fracture intensity and orientation). The description and parameterisation of individual fractures are often important input to the analysis of various issues that play an important role for the safety of the repositories after closure (e.g. buffer erosion and radionuclide transport from a damaged canister), and for questions that have to be answered during the construction and operational phases. In addition to modelling of explicit networks, DFN modelling includes different methods to up-scale DFN models to continuum models, which are often used on a larger scale. Both the explicit and the up-scaled models constitute input data for modelling of rock mechanics, hydrogeology, hydrogeochemistry and transport of solutes.

Current situation

Tools and methodologies for conditioned DFN models have been developed in the previous RD&D period. The purpose of these methodologies is to increase the determinism in the stochastic DFN models, primarily in the vicinity of tunnels and deposition holes where different types of data can be measured. An increased determinism is expected to lead to a lower uncertainty regarding the entities calculated in groundwater flow models based on underlying conditioned DFN models, and which are propagated to different assessments of post-closure safety. Furthermore, a larger degree of determinism means that the models can also be used to test different criteria for rejecting/accepting deposition hole positions. The methodologies for conditioning, implemented in two different tools, has been tested using data from synthetic sites described in a simplified manner and is presented in Bym and Hermanson (2018) and Appleyard et al. (2018). The methodology used in Appleyard et al. (2018) has also been successfully tested on data from Onkalo in Finland in Baxter et al. (2018).

One way of increasing the confidence in the modelling technique is to explore other DFN tools with which alternative fracture conceptualisations can be tested. MoFrac is a simulation tool for DFN modelling that was originally developed by Srivastava (2002). The code has a rule-based fracture generator, which is mainly based on the geostatistical relationships that can be derived from site investigations above and below ground. The generated fractures in the network are strictly conditioned to measured structures such as lineaments, fracture traces and borehole intercepts, and the code supports the generation of both undulating fracture surfaces and truncation.

In a project headed by CEMI (Canada) in cooperation with NWMO, MIRARCO (Canada) and SKB, the source codes, which were originally produced by Srivastava (2002), are being updated and extended, with the aim of permitting broader application areas mainly within mining and the nuclear waste industry. MoFrac has been tested on both synthetic data and site data (Junkin et al. 2017, 2018). The work with further development of MoFrac will continue during the current RD&D period.

Furthermore, an alternative method, UFM (nearly Universal Fracture Model), for generation of DFN models has been developed in a joint project between Itasca, Université de Rennes and SKB (Davy et al. 2010, 2013, 2018a, b, Lavoine et al. 2017, Le Goc et al. 2017, Darcel et al. 2018). The method focuses on the fracturing process in the generation of fracture networks. It has been shown that the fracture size distribution can be described as the result of two main processes:

- A growth process, whose rate, dl/dt , appears to be proportional to the fracture size, l , raised to the power of a constant, b , according to: $dl/dt = Cl^b$, where C is a constant of proportionality.
- A hierarchical braking process in which large existing fractures retard or impede the growth of smaller fractures.

This methodology partially simplifies the statistical analysis of fracture data through the introduction of a general, double Pareto distributed model, which appears to be able to account for most of the data from the site investigations on scales relevant for the assessments of post-closure safety. The methodology also offers a generic tool that based on simple rules controls the initiation, growth and termination

of fractures, which can be conditioned on observations in situ (see further under “Programme”). One advantage of the UFM concept in relation to generation methods based on statistical distributions is that the resulting fracture networks better mimic measured data in terms of how fractures are truncated against each other. The models will thus appear visually more realistic and confidence can thereby be considered to increase.

During the RD&D period, UFM models have also been used in hydrogeological applications, for investigations of how to reduce the fracture network from a geologically defined DFN model including all fractures to the fractures that are potentially hydraulically active, i.e. open. Also the effects of correlation relationships between fracture size and transmissivity, and between stress (rock stress) and transmissivity, have been investigated. Furthermore, the resulting models have been compared with corresponding Poissonian models, which have been traditionally used by SKB in for example the Forsmark site modelling (Follin et al. 2007). The results indicate that UFM models are generally more channelled, but have lower effective permeability, than the corresponding Poissonian models. These results are consistent with previous studies (Maillot et al. 2016) in which DFN models that only contain potential hydrogeologically active fractures were generated. In Le Goc et al. (2018), the models have been calibrated against PFL data (Posiva flow log data) and, moreover, methods have been developed for efficiently discriminating between resulting models by means of comparative analysis with the underlying data. The comparison technique is based on calculating the effective permeability and a so-called “channelling indicator” (which quantifies channelling of the flow in individual fractures and fracture networks) as a function of the spatial scale of the measurements. This comparison has proved effective for dismissing certain models (parameter combinations) in an early stage of the analysis.

Furthermore, a project is in progress concerning quantification of the uncertainties of mapped objects in boreholes and how these uncertainties affect the interpretation of data measured in boreholes and what effects this may have on discrete fracture network models (Stigsson and Munier 2013, Stigsson 2016, 2018).

Programme

During the current RD&D period, the preparation of a new modelling methodology report that describes the next generation of DFN models (i.e. conceptual assumptions and data requirements) will continue. This work also includes implementing the aforementioned developed methodologies for conditioning in SKB’s different modelling tools. Within the framework of the preparation of the methodology report, a number of conceptual questions linked to DFN modelling will be explored. These include, for instance, the relationship between intensity and size in water-conducting fractures, a possible heterogeneous distribution of fracture intensity (the effect of clusters), the hydraulic properties of deformation zones and the possibility of applying a DFN concept for describing these zones, and uncertainties in the relationship between aperture and transmissivity. Also methods based on genetic fracture generation, rather than purely statistical methods, are being studied (see below).

The effects of uncertainty in measurement data will be further studied. Previous work shows how different measurement uncertainties affect the total uncertainty in the orientation of fractures mapped in boreholes (Stigsson and Munier 2013, Stigsson 2016). An ongoing research project addresses how this total uncertainty can be estimated, and how the total uncertainty is propagated further when DFN-based models for flow and transport are produced. The developed methodology for estimating the total uncertainty in orientation can also be applied to fracture orientations mapped in tunnels and deposition holes.

Fractures are typically assumed planar when using DFN models in SKB’s hydrogeological modelling. To address this in the subsequent transport modelling, scaling factors are occasionally applied to approximate the effect of the channelling occurring due to internal aperture variability (i.e. the spatial variation in the fracture aperture of individual fractures). With the development of high performance computing (HPC), fracture networks can now be simulated with internal aperture variability. Several different aspects may be explored, for example the validity of the simplified approach with scaling factors, but at least as important is investigating how the calibration of models against measurement data is affected when internal fracture heterogeneity is included. Furthermore, possible channelling effects in fracture networks and in single fractures can be studied. A joint project with Posiva continues, in which the calculation tool DFNworks (Hyman et al. 2015) is used to investigate these issues. The results of this work will be directly incorporated into the new DFN methodology.

The work that was initiated in the RD&D Programme 2016, aiming to increase the understanding of alternative DFN conceptualisations, will continue during this RD&D period. Given the outcome of these efforts, the alternative conceptual models will be described in the modelling methodology report for DFN as equivalent concepts in relation to the traditional, statistically based method.

Development of the user interface for modelling using for example MoFrac and UFM is planned, and special functions for interaction between the tools will be developed to facilitate the exchange of fracture network models between different tools. Moreover, SKB plans to further improve the UFM methodology through the introduction of alternative initiation processes for fracturing, which so far has been assumed to be uniform in space, and through the introduction of correlations to fracture orientations and thereby indirectly to paleostress fields (i.e. how the stress field has varied over time). The possibility of defining a degree of saturation, i.e. a state where fracturing essentially ceases, will be investigated based on global energy consumption in the fracturing process. Development of a new method to continuously nest different scales with the purpose of more efficient generation of large models is also planned. Since the codes are still in a development phase, it will also be necessary to develop an efficient user interface and interactions with other DFN codes. This work has been initiated.

SKB continues to further develop the DFN models of the UFM type for hydrogeological analyses. Specifically, it will now be investigated how partially open/closed fractures and general internal fracture variability can be included in the UFM concept, and what the effects will be on the resulting connectivity structure, flow and transport. The use of graph theory to study flow and transport in this type of individual fractures and fracture networks in a simplified manner will also be studied. The functionality produced will be implemented in a new package of calculation tools, DFNLab, focused on precisely the UFM concept. The UFM concept will, together with other potential genetically based fracture generation concepts, be included as a potentially important part in the new DFN methodology.

An ongoing doctoral project continues (a part of the EU project Enigma), which, by means of field experiments at the Äspö HRL and numerical modelling, investigates whether conditioning of fracture networks (on a deposition hole scale) can be performed on data from hydraulic tests combined with tracer tests and geophysical measurements (GPR, Ground Penetrating Radar). Specifically, fresh water with added tracer is injected into a borehole under the tunnel floor while the breakthrough is studied in nearby boreholes. The injected water's difference in salinity compared with the formation water with elevated salinity permits GPR measurements from the tunnel floor to study fracture network properties. During the RD&D period, this project will be concluded and the results reported.

12.2 Seismic impact on repository safety

Under the assumptions made in SR-Site, the effect of earthquakes on the post-closure safety constitutes a significant contribution to risk for the Spent Fuel Repository (SKB 2011b). The risk contribution is strongly linked to the frequency/magnitude relation suggested for the short-term (Böðvarsson et al. 2006) and long-term (Hora and Jensen 2005, Fenton et al. 2006). The uncertainties in, above all, the long-term predictions are substantial, in particular regarding earthquake activity in conjunction with deglaciation of an ice sheet. An underestimation of the seismic activity entails an underestimation of the long-term risk, while an overestimation leads to over-dimensioning of the final repository.

The methodology for modelling of earthquake effects on a KBS-3 repository has been developed over a long time (Fälth and Hökmark 2006, Fälth et al. 2007, 2008), and after extensive tests (Fälth et al. 2014) and case studies (Fälth et al. 2016, 2017, Fälth 2018), it has reached a high level of maturity. Additional calculation cases are, however, required, in order to narrow down a pessimistic, but relevant, range of outcome, as well as a number of alternative conceptualisations to challenge the assumed pessimism in the current approach. Further development efforts that indirectly concern earthquake modelling are presented in Section 12.2.3.

In the safety evaluation of SFL, no analysis was made of the effects a possible earthquake could have on the repository's barrier system. The barrier system's resilience to these processes must be evaluated in future assessments for SFL. The results may contribute to optimisation of the repository design.

The studies that are planned in the present research programme aim at gaining a better understanding of glacially induced earthquakes and their connection to present-day seismicity, with the goal of reducing uncertainties through in-depth understanding and thereby increasing the confidence in assessments of post-closure safety.

12.2.1 Seismic monitoring

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 9 000 earthquakes (Figure 12-1) with magnitudes between approximately -2 and 5 . The last large expansion of the network, in south-western Sweden, was instrumented in 2008, and since then SNSN has relatively good coverage in the seismically most active areas of the country. Currently, there are 65 permanent stations installed and a varying number of temporary stations.

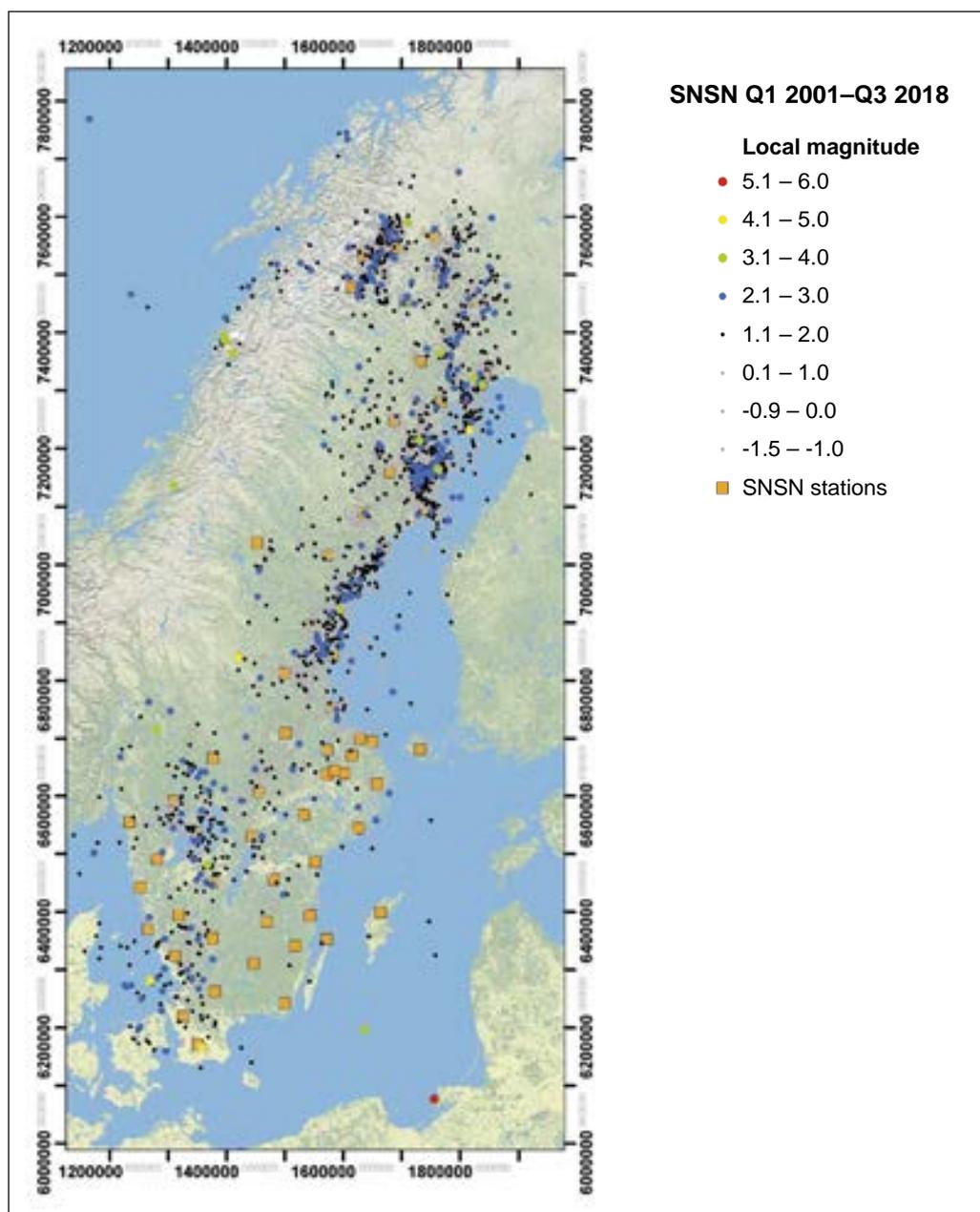


Figure 12-1. Earthquakes recorded by the Swedish National Seismic Network (SNSN 2015) during the years 2001–2018. Data from SNSN (Böðvarsson 2012).

Since 2008, SNSN has continuously gathered data in real time from all stations, which means that the quantity of data available for analysis has increased significantly compared with previous years when only data segments from detected events were collected. The new, refined seismic network has fundamentally improved the potential for interpretation of earthquake activity in Sweden, which, along with paleoseismic studies, serves as a basis for prediction of future earthquake activity. Continuous, long-term monitoring of earthquakes is critical for being able to capture patterns of frequency and magnitude, which can vary both in time and space.

A tomographic analysis (see for example Tryggvason and Linde 2006) resulting in a three-dimensional velocity model is concluded and reporting of the results is under way. The results have been used for relocalisation of earthquakes. Linked to this, a so-called “multi-event analysis“ of earthquake localisations have been performed, which considerably improves the precision of location determinations and calculated magnitudes. The relocalisations will serve as a basis for the ongoing new calculations of focal mechanisms and stress inversion, which, among other things, will be used to calculate the stress field at great depth.

Programme

The cooperation with SNSN proceeds as planned, and SKB is not planning for any specific research efforts in relation to the permanent seismic network during the RD&D period. However, there are plans for extending the permanent network with a number of temporary stations (Section 12.2.2).

12.2.2 Investigations of glacially induced faults

Current situation

Burträsk is the most seismically active area in Sweden (Figure 12-2c). Several years ago, Uppsala University supplemented the seismic stations in the Burträsk area to obtain more exact data (Lund et al. 2015, 2016). The results have been very successful, since it has been shown that the earthquakes cluster along the fault scarp that was first identified by Lagerbäck and Sundh (2008) and later described in detail by Mikko et al. (2015). Reflection seismology has previously shown a well-defined fault plane down to a depth of about three kilometres with a dip around 55° (Juhlin and Lund 2011). The earthquakes cluster in the elongation of this plane and analyses further indicate (ongoing work) that only the upper crust, down to about 15–20 kilometres, participated in the post-glacial reactivation of the deformation zone (Lund et al. 2015). This affects the calculations of the magnitude of the earthquake and is of great importance for our understanding of glacially induced earthquakes, and thereby our ability to handle future earthquakes.

Registrations by SNSN (2015) show a very distinct, north-east trending cluster of earthquakes located north-west of Iggesund (Figure 12-2b). As a clear correlation between recent earthquakes and glacially induced faults has been detected for most of the faults registered until now (Lindblom et al. 2015), this means that the cluster of earthquakes identified north-west of Iggesund is a strong indication that there may be a so far undetected, glacially induced fault in this area. The relative proximity to Forsmark makes it particularly urgent to explore these indications more closely.

Lantmäteriet’s Lidar measurements (Lysell 2013, Lantmäteriet 2015) have been remodelled for the county of Uppland at a higher resolution (1 metre). Based on this, as a supplement to the previously reported analyses (Mikko et al. 2015), a high-resolution analysis of potential fault scarps has been carried out (Öhring et al. 2018). The results of this study substantiate previous completed studies in the investigation area (Lagerbäck and Sundh 2003, 2008, Lagerbäck et al. 2004, 2005, Mikko et al. 2015) in that it is not possible to confirm the occurrence of glacially induced faults in the county of Uppland.

Programme

A project has been initiated to intensify the investigations of the Burträsk fault (Figure 12-2c), the currently most seismically active area in Sweden. The main purpose of the project is to obtain a better understanding of the mechanisms that contributed to and triggered the large glacially induced earthquakes in northern Scandinavia. The intention is to use seismic data from the area around the Burträsk fault (Lund et al. 2016) as a basis for building a numerical model of the fault (Fälth et al. 2014) and to construct a background stress field that can be applied in the model (Fälth et al. 2014). By simulation

of glacially induced fault movements using various modelling assumptions (e.g. variations of material properties, stress model, pore pressure) and comparisons with observations of fault movements on the ground surface in the Burträsk area, the objective is to investigate which parameters mainly affect the stability and movement of the fault.

For the area north-west of Iggesund (Figure 12-2b), a project was recently initiated with the purpose of identifying the source of the seismically active band. The project includes the establishment of a dense network of temporary seismic stations to increase the precision in the localisations and thereby facilitate identification of the structure that is currently seismically active. Furthermore, structural geological studies are planned, including Lidar analyses and structure modelling, quaternary geological field studies and, if necessary, follow-up investigations using reflection seismology.

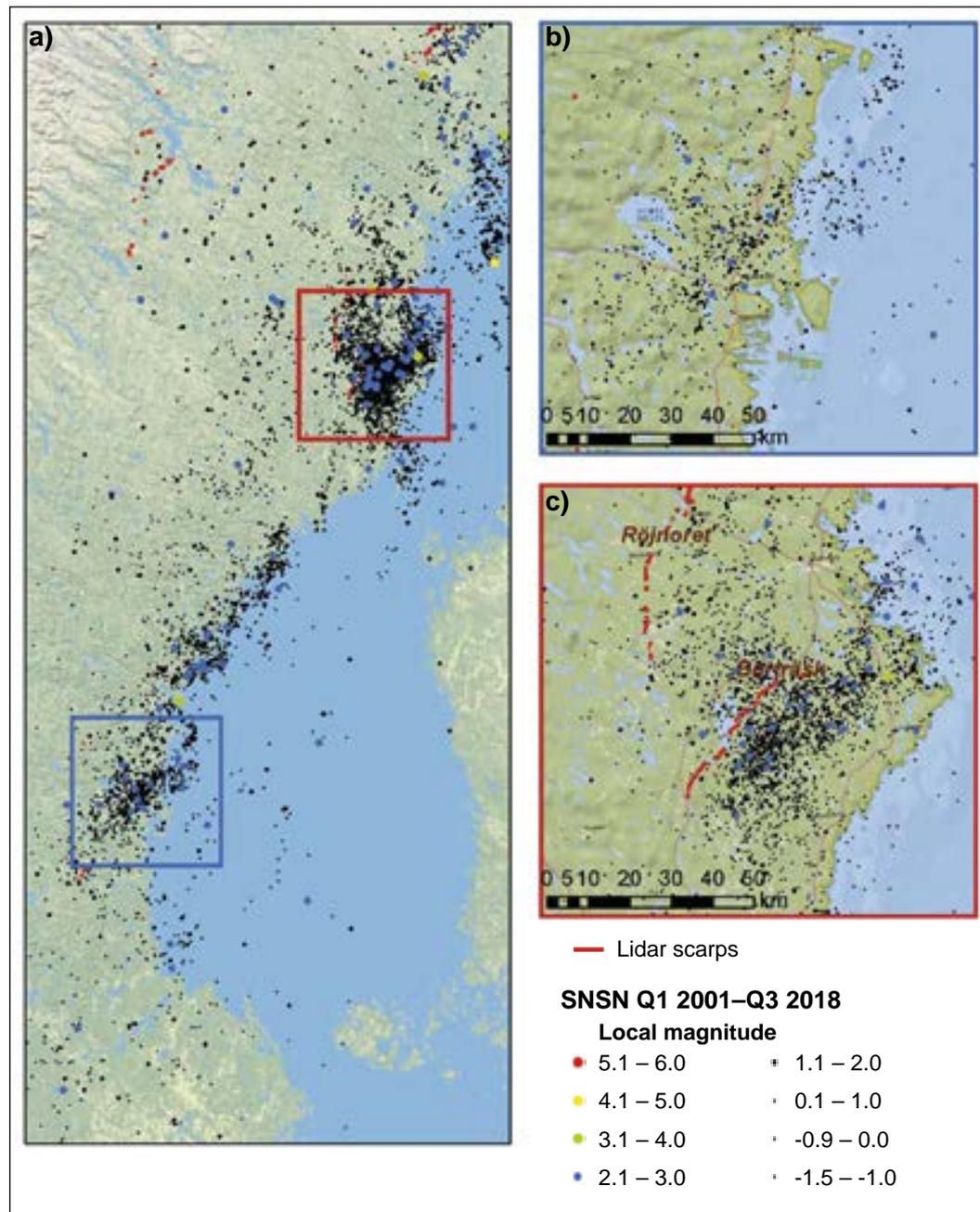


Figure 12-2. a) Cluster of earthquakes recorded by SNSN (2015) b) north-west of Iggesund and c) in connection with the Burträsk fault. Fault scarps are identified by Mikko et al. (2015) on the basis of the New National Elevation Database, NNH, (Lantmäteriet 2015) and Lagerbäck and Sundh (2008) using mainly photogrammetry and mapping.

The supplementary Lidar studies in the county of Uppland that has been carried out (Öhrling et al. 2018) cannot confirm the occurrence of glacially induced earthquakes in the investigated area. However, for a few identified lineaments supplementary field studies are required in order to definitely be able to dismiss them as glacially induced faults. Such field studies will probably be carried out during 2019. In an initial phase, the lineaments will be investigated with traditional, quaternary geological field methods and if possible strategically localised excavations. If the results of these investigations show that the lineaments are still difficult to interpret, supplementary studies with geophysical methods will also be carried out.

12.2.3 Modelling of seismic impact on the final repository

Current situation

Methodology for modelling of earthquakes and their effects on a final repository was defined in Fälvh and Hökmark (2006) and applied after further refinement (Fälvh et al. 2007, 2008) in SR-Site (Fälvh et al. 2010), partly in the form of a PhD programme (Fälvh 2018). In addition to these efforts, development of methodology and calculation tools has been carried out in cooperation with Posiva, with application to both rock stress conditions at the end of the latest glaciation (Fälvh and Hökmark 2011, 2012) and in the present day (Fälvh and Hökmark 2015). The methodology development and different applications have been summarised in a specific report (Hökmark et al. 2018), in which the modelling results from the latest years are also presented. Among other things, the following aspects have been analysed:

- The effect of very large earthquakes in the vicinity of the repository (Fälvh and Hökmark 2015).
- The effect of branching faults.
- Implementation of undulating target fractures (i.e. the fractures that host the induced the shear displacements) (Lönnqvist and Hökmark 2015).
- Implementation of undulating fault surfaces and an improved conceptualisation of fault edges (Fälvh et al. 2017, Fälvh 2018).
- Analysis of present-day and induced earthquakes (Hökmark et al. 2018).
- The effect of the fracture network (DFN) around the target fractures (Hökmark et al. 2018).
- The effect of thermal load on the shear load case (Hökmark et al. 2018).

Programme

The shear load cases that have been analysed have intentionally been based on what is judged to be pessimistic or very pessimistic conditions and assumptions. With an increased process understanding, substantiated by observations, more realistic assumptions can greatly reduce the estimation of induced shear, and thereby of long-term risk, but 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 in these conditions and properties on the calculated shear displacements. The following studies are planned:

- Determine how the stress field variability, in both orientation and magnitude, within the repository volume, affects the calculated shear displacements and, if possible, quantify these effects to size and location.
- Examine the effect of alternative glaciation models (with respect to different size, extent, resilience and dynamics) on the calculated shear displacements (Section 14.2).
- Improve the conceptualisation of fault undulation and edges.
- In-depth studies of the target fracture conceptualisation (mainly undulation).
- Modelling of the target fractures near (100–200 metres) the fault and, if possible, even target fractures that connect to the fault.
- Modelling the effect of forebulge on steep faults.
- In-depth analyses of the stability of steep, local, deformation zones in Forsmark.

- Modelling of target fractures with orientations that maximise shear displacement in different parts of the repository volume.
- Include the interaction between bentonite, the canister and rock in the calculation of shear displacement.
- Develop the handling of repeated earthquakes and their effect on above all the target fractures.

12.3 Groundwater flow, groundwater chemistry and transport of solutes

12.3.1 Development of calculation tools for groundwater flow and transport of solutes

SKB mainly uses the calculation tools ConnectFlow, DarcyTools and MIKE SHE for hydrogeological and surface-hydrological modelling, but also the general simulation tool COMSOL. For modelling of wstreams and their interaction with groundwater, Mike11 is also used, a one-dimensional tool for channel flow that is directly linked to MIKE SHE.

The calculation tools are mainly used for different purposes but are also completely or partially overlapping. ConnectFlow and DarcyTools are used primarily for questions concerning the deep groundwater system, but they also handle the near-surface groundwater system, whereas MIKE SHE is primarily used for the near-surface groundwater system and surface hydrology (linked to atmospheric processes), but the tool also handles parts of the deep system. The calculation tools are functional for their respective usages, but require constant maintenance and further development to remain up-to-date.

Current situation

SKB's capacity for hydrochemical modelling has been increased to include geochemical and transport processes in the hydrogeological calculation tools DarcyTools and ConnectFlow. Transport in this case mainly concerns the chemical components that control the salinity, density, pH and redox properties of the groundwater. In the case of DarcyTools, this extended capability has been achieved by developing an interface for communication with the modelling tool PFLOTRAN, which is designed for HPC (High Performance Computing) machines. This coupled modelling tool is called iDP (Molinero et al. 2016, Ebrahimi et al. 2018). Openly available code libraries from PHREEQC have been included for this purpose into ConnectFlow (Joyce et al. 2015). Both calculation tools have been used to study how leachate from cement affects the groundwater chemistry and they enable studies regarding if and how water with elevated alkalinity may reach deposition hole positions. ConnectFlow has been further developed so that reactive transport can be simulated in the DFN module. The calculation tool iDP has also been used to study reactive transport (Trincherro et al. 2017a, Iraola et al. 2017) and oxygen intrusion into the bedrock (Trincherro et al. 2017b, 2018b, 2019). In Trincherro et al. (2017a) and in Trincherro et al. (2019), the rock matrix has been simulated as a micro-DFN, i.e. the rock matrix has heterogeneous porosity properties and also a heterogeneous distribution of minerals where sorption can take place. Another calculation tool that has been used and has proved useful is the integration of COMSOL and PHREEQC (Nardi et al. 2014, iMaGe 2019).

Other DarcyTools developments that have taken place during the RD&D period mainly involves new calculation routines for determination of all quantitative entities that are propagated to calculations in the assessment of post-closure safety. Furthermore, routines for connectivity analysis between different CAD objects and the fracture network have been developed (objects such as tunnels and boreholes can create connectivity between previously non-connected fractures), the traceability of runs has improved as a result of new output files, the routines for fracture generation have been rewritten (but still only Poissonian networks), the possibility to read and import DFN networks from other codes has been included, preparations for geometric conditioning has been carried out, and the routines for inspection of the convergence in numerical runs have been modernised to current programming and computer standards.

In ConnectFlow, functionality in the connection to PHREEQC has been developed to directly calculate local K_d values (for sorption of radionuclides). A prerequisite for the geochemical coupling in DFN models was to first implement multicomponent advection and matrix diffusion. Furthermore, density-

driven flow and particle transport in explicit DFN models have been implemented, and the functionality for nested models has been improved. Finally, anion exclusion and surface complexation have been added as possible processes.

The safety evaluation of SFL (SKB 2019c) identified that the interface between concrete backfill and the bedrock and between bentonite backfill and the rock is crucial for the water flow through SFL and for the transport of solutes from SFL to the surrounding rock. The properties of these interfaces are therefore of great importance for the assessment of post-closure safety.

Within the framework of the SFL safety evaluation, a new method for detailed analysis of flow paths and travel times within and between different layers of regolith has been developed (Joyce et al. 2019, Johansson and Sassner 2019). Through a refined numerical description of different regolith layers in MIKE SHE, it is now possible to in greater detail study flow paths within and between different types of regoliths from the time when the water leaves the rock and until it leaves the saturated groundwater zone at the ground surface. As a result, the description of flows from the rock surface to different biosphere objects can be refined. Instead of exporting water balances from the MIKE SHE model to the dose model for entire biosphere objects, it is now possible to study only the part of the object that is actually exposed to deep groundwater discharge.

The focus of model development of surface hydrological models in the RD&D period has been on evapotranspiration processes. The potential evapotranspiration, PET, has previously been calculated by SMHI using the Penman equation according to Eriksson (1981) and Sjögren et al. (2007). Studies based on measurements in, for example, Norunda show that the original Penman equation or modifications of it is not optimal for boreal forest ecosystems, while less complex models based only on temperature are easier and more robust to calibrate for hydrological applications (Halldin 1988, Grelle et al. 1997, Oudin et al. 2005, Rohde et al. 2006). Both in the site model SDM-Site and in the SR-Site safety assessment, SKB has observed that the Penman equation as applied by SMHI overestimates the actual evapotranspiration in Forsmark. SKB has therefore carried out a study where different calculation methods for PET were evaluated and used in Forsmark for the purpose of finding the most effective method. This new site-adapted calculation method will be implemented in relevant tools prior to future updates of the site model.

The largest effort in the previous RD&D period regarding the integration between superficial and deep groundwater has taken place within the framework of the site understanding work. An updated modelling methodology for site-descriptive modelling with a focus on better interaction between both conceptual and quantitative modelling of hydrology and hydrogeology is under way. The work will be concluded in 2019.

Programme

It can be initially noted that the primary development of hydrogeological models required in future stages of SKB's programme is linked to the development of the DFN methodology described in Section 12.1.4. Other development efforts concerning hydrogeological modelling are described in this section. These efforts should primarily be regarded as continuous development efforts to maintain and develop the functionality of the calculation tools in line with the progress of scientific research.

The development of the hydrochemical calculation tools iDP (DarcyTools-PFLOTRAN) and ConnectFlow -PHREEQC will continue in order to be able to apply the tools to other problems where reactive transport has a significant impact, for example the simulation of field experiments, support for site modelling and safety assessments, transport of non-radioactive, environmentally hazardous substances, the influence of leachate from low-pH cement and the intrusion of glacial water. Development of the DFN-based reactive transport modelling in ConnectFlow is continuing. The use and development of the calculation tool Comsol-PHREEQC (iMaGe 2019) continues, and functionality for linking DarcyTools and PHREEQC will be established during the period.

Additional efforts will be made to take into account the variation in time and space of mineral and groundwater composition. The simplifications underlying the dynamic K_d concept will be further analysed in order to fully defend the methodology. Furthermore, there are plans for coupling the hydrogeochemical calculation tools (PFLOTRAN, DarcyTools and/or ConnectFlow) and the transport model (MARFA) in a more effective way. For MARFA, there are also plans to include the process diffusion into stagnant water in the fracture planes.

Further development efforts for ConnectFlow primarily involve code parallelisation and other efforts to improve code performance, specifically for reactive transport in DFN models and in nested continuum/DFN models.

As there is now functionality in DarcyTools for calculating the entities that are needed for the assessment of post-closure safety, a major effort is required to verify that DarcyTools produces results equivalent to ConnectFlow, or that any differences can be explained in an acceptable way. A partial repetition of the hydrogeological analyses that were carried out with ConnectFlow in SR-Site (Joyce et al. 2010) is therefore planned.

It must also be investigated how the methodology for conditioning produced within the development of DFN models (Section 12.1.4) is best incorporated into DarcyTools. The difficulty here is that DarcyTools up-scales the explicit DFN model to a continuum representation before groundwater flow is simulated, which means that geometric/hydraulic conditioning is difficult and time-consuming in practice in the current version of DarcyTools. As described above, however, functionality for purely geometric conditioning has already been implemented in DarcyTools. In practice, it may turn out that geometric conditioning is adequate and more practical given the current requests for fast model updates and the numerically heavy computation requirements of the combined geometric/hydraulic conditioning. Nevertheless, possible approaches for combined conditioning in DarcyTools will be studied further.

Under “Current situation”, there is a description of how micro-DFN methods have been developed to describe matrix diffusion and sorption in detail on the mineral grain scale. These models cannot be used on the spatial scales that are of interest for the safety assessment. A project has therefore been initiated concerning how these micro-DFN models can be up-scaled for use in models for radionuclide transport on larger scales.

A PhD project has been initiated aimed at understanding and quantifying clogging (build-up of geological material in fractures) under different hydrogeological and hydrogeochemical conditions (Doolaeghe et al. 2019). The purpose is to develop a descriptive model and to implement this mathematically in a DFN model in order to be able to study how clogging affects the hydraulic connectivity and transport properties of a discrete fracture network.

A related issue is clogging in the repository due to introduced materials such as bentonite. A corresponding numerical study is planned in order to increase the understanding of the extent to which eroded bentonite may affect the flow conditions in the fracture network around a repository.

Detailed models to calculate the water flow through SFL and the transport of solutes from SFL to the surrounding rock will be developed.

A PhD project has been initiated in order to develop a site-specific model for coupled modelling of hydrology and biogeochemical processes with a focus on carbon. Carbon has long been used as an analogue for how radionuclides move within and between different ecosystems. The model has been developed for the Krycklan catchment outside Umeå in Västerbotten. The purpose of the model development is, however, to make it possible to apply governing processes and equations on both Forsmark and other analogous sites, for example Two Boat Lake on Greenland. This development work provides a better coupling between flow and transport models in the surface system. It is a great advantage if flow and biogeochemical transport and processes can be handled in the same modelling tool.

Within the framework of the operational detailed site investigation programme for the surface system, a number of integrated initiatives are planned for coupled chemical and surface-hydrological modelling in order to study transport processes in discharge areas of different age and character. A number of “type ecosystems” will be identified representing different stages of landscape succession for a future climate in Forsmark. Multidisciplinary investigations will be carried out in order to obtain relevant data for model studies in future safety assessments. An important purpose of these model studies is to better describe transport of solutes from the bedrock top surface to uptake in plants.

12.3.2 Processes affecting the hydrochemical environment

Current situation

As described above, development of hydrochemical calculation tools where the transport of solutes is coupled with chemical reactions is in progress. These tools are used to simulate oxygen intrusion in the bedrock and sulfide transport in the near-field of the Repository for spent nuclear fuel. The development has been documented in a number of publications (Trincherò 2017b, 2018b, 2019). Thus, insight has thereby been gained on the effects of heterogeneity in the distribution of iron(II) minerals in the rock matrix. The comparison with analytical equations that was used in SR-Site shows that these provide pessimistic values of oxygen intrusion into the rock matrix.

In a recently finalised collaborative project with Posiva, processes that may affect the occurrence and concentrations of sulfide at repository depth were studied thoroughly. The project has resulted in a number of reports (Chukharkina et al. 2016a, b, Nilsson et al. 2017, Johansson et al. 2019, Svensson et al. 2017b, Bengtsson et al. 2017a, b, Haynes et al. 2019, King and Kolář 2019, Pękala et al. 2019), and the final reporting of certain modelling results is being concluded. Some conclusions that can be drawn based on results from the project are:

- Aluminium should not be used in borehole sections since it generates a considerably larger amount of hydrogen than stainless steel. Hydrogen is an important energy source for e.g. sulfate-reducing bacteria.
- The bentonite clay in itself constitutes a sink for sulfide.
- Different clays have different threshold densities above which microbial sulfide formation does not occur.

SKB has also been a coordinator for the EU project Mind (Microbiology In Nuclear Waste Disposal, <http://www.mind15.eu>). In the project, degradation of organic matter in low- and intermediate-level waste, and microbial processes in a repository for spent nuclear fuel have been studied. A number of publications are available via the project website. The results from Mind indicate, as supported by previous studies, among other things, that microbial activity can only occur in the presence of nitrate (NO_3^-) at a pH > 10. Several of the results confirm previous studies regarding for example threshold densities and salinity (Section 11.3.1 and the project website). In particular, Mind has created a valuable network of researchers who work on microbial issues in a repository environment.

Quantum chemical modelling has been used to deepen the mechanistic understanding of bacterial sulfate reduction, which in extension may be used to more precisely describe the sulfur chemistry in groundwater at repository depth (Wójcik-Augustyn et al. 2019).

Programme

The development of hydrochemical calculation tools continues and reporting of the results from the sulfide project will be concluded. SKB will evaluate different methods for analysing organic matter in order to find the most suitable method to characterise as many as possible of the main dissolved organic substances in the groundwater. These substances may constitute potential energy source for sulfate-reducing bacteria and thus result in production of sulfide. Other microbial processes will be studied, such as the effect of acetate supply on sulfate and uranium reduction in deep groundwater.

In accordance with SSM's request in their review of SR-Site (SSM 2018), weathering studies of the surface layers in Forsmark will be conducted. These processes may increase the salinity of infiltrating meteoric water, which could reduce the negative effect of fresh water on bentonite, especially during longer periods of temperate climate. Among other things, the effects of the gradual leaching of calcite in the soil profile will be investigated aside from other factors that define the chemical environment.

12.3.3 Transport properties and processes affecting solute transport in the bedrock

The bedrock transport properties and transport processes for both radioactive and environmentally hazardous substances constitute an important area of knowledge needed for assessments of repository post-closure safety. Transport of solutes both to the repository and from the repository must be taken

into account. The focus when it comes to transport to the repository is mainly on reactive or corroding components such as dissolved oxygen, sulphide and leachate from cement, but also the salinity of the groundwater is of interest. Regarding the transport from the repository, radionuclides that are transported with the groundwater while simultaneously interacting with the rock are the primary focus. The same methodology can be used for the transport of pollutants. Knowledge generated in the area also provides an important basis for e.g. the construction of the repositories and for interpretation of field measurements.

The most important processes that affect the transport of solutes are matrix diffusion and sorption. The need for further development and knowledge accumulation particularly applies to conceptual understanding, reduced uncertainty in transport parameters and further development of modelling tools. A fundamental understanding is important also when it comes to advective transport, dispersion, electromigration, gas transport and colloidal transport. Increased knowledge will primarily lead to less pessimistic assumptions in the assessment of post-closure safety.

In recent years, hydrogeological modelling has been developed so that geochemical processes and transport processes now can be integrated with the flow modelling (Section 12.3.1). Efforts are 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.

Current situation

When it comes to transport of solutes, efforts have been made within the fields of matrix diffusion and sorption, with regard to conceptual understanding, reduced uncertainty in transport parameters and further development of the modelling tools (for example MARFA). Efforts have also been made concerning advective transport (Soler et al. 2019), dispersion, electromigration (Puukko et al. 2018), gas transport (Silva et al. 2019b) and X-ray microtomography (Iraola et al. 2017, Voutilainen et al. 2018) in order to increase the understanding of transport processes and provide input data (for example Byegård et al. 2017) to the calculation tools. Sensitivity studies of transport parameters have also been carried out to investigate the effect of the variability of the rock, for example calcite occurrences on fracture surfaces and their effect on sorption and matrix diffusion (Drake et al. 2018, Crawford and Löfgren 2019). The effect of possible radial matrix diffusion from channels in fractures has been studied (Neretnieks 2017). The effect of diffusion into stagnant water in fractures, and thereby increased matrix diffusion, has been implemented in CHAN3D (Shahkarami 2017).

Theoretical studies of sorption mechanisms have been carried out (CROCK 2013), which make it possible to obtain sorption data for rock material of varying sizes and under different geochemical conditions. A feasibility study concerning sorption on cement and corrosion products has been carried out.

Traditionally, K_d is a constant distribution coefficient that indicates how much of a substance is sorbed and how much in solution. MARFA, which has been developed to apply a dynamic K_d concept, where K_d is a function of the mineral and groundwater composition in time and space (Trincherio et al. 2016), has been further developed in the safety evaluation of SFL (Trincherio et al. 2018a).

The SKB Task Force for modelling of groundwater flow and transport of solutes comprises an important international platform for modelling of field experiments, conceptual understanding, comparisons of results and calculation tools, demonstrations and training. In Task 8, the modelling was focused on the field test BRIE (Bentonite Rock Interaction Experiment), which was performed in the Äspö HRL. It studied the hydraulic interaction between the rock and the bentonite, where the resaturation process was a central part of the issue, in two downscaled experimental deposition holes that are 30 centimetres in diameter. This study is evaluated in two publications (Finsterle 2019, Finsterle et al. 2018). Furthermore, extensive efforts have been carried out in the modelling task Task 9, which focuses on modelling of tests that investigate rock transport properties such as matrix diffusion and sorption. The experiments that comprise the basis for the modelling exercise are REPRO (Aalto et al. 2009, Poteri et al. 2018a, b), which is being conducted in Onkalo in Finland, and LTDE-SD (Nilsson et al. 2010), which was carried out in the Äspö HRL. The evaluation of the first task can be found in Soler et al. (2019).

The state of knowledge in modelling of gas transport, both as dissolved gas in groundwater (Silva et al. 2019b) and as two-phase flow, has been improved for the benefit of future assessments of post-closure safety.

SKB has also supported a PhD project (Krall 2016) on fracture minerals and uranium occurrence in Forsmark. The stability of uranium-minerals and amorphous phases has been studied as a function of changed geochemical conditions. The work indicates that the oxidation of uranium(IV) to uranium(VI) occurred geologically early, and that these older uranium(VI) minerals are soluble under prevailing groundwater chemical conditions.

Programme

In order to reduce the uncertainty associated with transport parameters, new measurements are planned on site-specific material under well-defined conditions. The intention is to determine sorption data especially for the nuclides where data are insufficient and deemed to be of importance for the safety assessments. The goal is to be able to estimate the uncertainties related to differences between laboratory and in situ measurements, different size distributions of rock material, mineral distributions, complexation and geochemistry. The variability in transport parameters should also be possible to handle in the modelling. Sorption on cement and corrosion products will be further studied. The development of electromigration for determining transport parameters is continuing both experimentally and by means of modelling to support the experiments and their interpretation. Another goal is to investigate the effect of matrix diffusion on pore water and salinity. The theoretical studies of sorption mechanisms will continue.

There are plans for long-term tests, in situ or under corresponding conditions, of the interaction between radioactive tracers and natural calcite to study the metal exchange between calcite and groundwater (in the outermost part of calcite crystals), and how organic complexation in the groundwater affects this exchange.

Within the SKB Task Force for groundwater and transport of solutes, Task 9 will continue with the modelling described under “Current situation”, with a focus on reporting the results of the modelling of LTDE-SD and predictive modelling of the through-diffusion test TDE (Aromaa et al. 2018) within REPRO. In one of the modelling tasks, the goal is to utilise the knowledge that has been obtained in the modelling of field tests for the safety assessment. The knowledge has been obtained during the work with Task 9, but also from previous modelling tasks in, for example, Task 4 and 6.

12.3.4 Climate effects on geosphere processes

Large temperature reductions at the ground surface may result in freezing and if the ground is frozen for more than two consecutive years, permafrost arises by definition. Permafrost in regolith and bedrock leads to reduced permeability and causes volume changes, which in turn may lead to new formation, widening and/or propagation of existing fractures. How the hydrogeological system is affected during different stages of permafrost growth or thaw is site specific and requires analysis in site models.

The ice load from an ice sheet affects the hydrogeological system differently depending on the thermal basal conditions of the ice. The most important factor at depth is whether the ice sheet is cold-based or warm-based. If the ice sheet is cold-based, it represents a large mechanical load that may lead to compression of the pore system of the bedrock. If the ice sheet is warm-based, water is generated at the base of the ice. Water infiltrates into the ground and the groundwater pressure builds up in the same way as the ice load. In the case of more permeable bedrock, the latter situation does not result in great changes of the hydrogeological system (even if the hydrostatic pressure increases).

Current situation

The understanding of the effects of freezing and thawing processes on hydrology in periglacial areas is limited, and there is a general lack of data for conceptualising the hydrological periglacial system (Vaughan et al. 2013, Bring et al. 2016). Within the framework of GRASP (Greenland Analogue Surface Project), the hydrology in permafrost environments (periglacial environments) has been studied on catchment scale. The project has allowed SKB to strengthen its conceptual knowledge and process understanding of permafrost hydrology.

Field investigations, combined with conceptual and numerical modelling, have been carried out in order to increase the conceptual understanding of the hydrogeological system. The field investigations, which started in 2010, have been conducted in a small catchment area on western Greenland in the vicinity of

Kangerlussuaq. Two groundwater systems occur in areas with permafrost, one above the permafrost in the active layer and one constantly unfrozen system underneath the permafrost. The active layer thaws and freezes depending on seasonal variations in air temperature and a shallow groundwater flow occurs in this layer during the summer months (Figure 12-3). The investigations within GRASP have focused on processes in the active layer, since they have large effects on the water partitioning within the catchment. The lake in the catchment area is underlain by a talik, i.e. an unfrozen area in the permafrost. The talik allows for a hydraulic contact between the surface water system and the unfrozen groundwater system underneath the permafrost. By measuring pressure level variations in both the lake and talik, water exchange between the lake, the active layer and the deep unfrozen system underneath the permafrost has been studied. Pressure variations in the bedrock have been measured within the framework of the now concluded GAP project (Greenland Analogue Project).

Analysis of collected data has led to an increased understanding of the periglacial hydrological system, and the main hydrological flows have been identified and quantified. The resulting conceptual model, which constitutes the platform for the numerical modelling performed for the catchment area, is presented in Figure 12-3.

In the RD&D period, knowledge from the GRASP project regarding handling of active layer dynamics in the MIKE SHE model has been used in a PhD project for the purpose of investigating the effect of ground frost on hydrology. Ground frost has so far been neglected in SKB's hydrological models for Forsmark. The PhD project uses knowledge from Greenland and has implemented ground frost in a MIKE SHE model of the Krycklan catchment. The model study shows that ground frost has a great impact on runoff processes during snowmelt and that the partitioning of surface water and groundwater is affected by ground frost formation above all in peatland areas and wetlands (Jutebring Sterte et al. 2018). Since Forsmark is characterised by a relatively high proportion of wetlands, this study will be important for future updates of the hydrological site model in Forsmark.

A scientific publication was published within the framework of GRASP in 2017 (Lindborg 2017) followed by a publication in 2018 concerning soil frost processes in boreal landscapes (Jutebring Sterte et al. 2018).

Unless the mechanical load and the hydraulic pressure interact, a unilateral increase of the hydrostatic pressure, especially near the surface, could cause fracture opening, fracture shearing and fracture propagation. In the current THM assessment (analysis of thermo-hydro-mechanical conditions) of the effect of glaciation on rock mass properties, a worst-case scenario has been postulated based on a lower limit of fracture normal stiffness, an assumption leading to fractures that are very sensitive to normal stress variations (Hökmark et al. 2010). A better understanding of how the ice load affects the underlying rock mass would make it possible to bound the glacial cycle's consequences for the THM behaviour of the rock mass and thus also the consequences for rock mass transport properties in a more realistic way.

During the past 20 years, SKB has participated in the research project DECOVALEX, which was aimed at developing and validating software for modelling of THM processes in fractured rock (Chan et al. 2005). SKB has also contributed with supervision in a PhD programme at Chalmers University of Technology. The PhD project has partly focused on different aspects of permafrost growth and freezing of groundwater in the bedrock, where for example volume changes could lead to formation of new fractures and/or opening and propagation of existing fractures; see for example Lönnqvist and Hökmark (2013). Furthermore, state-of-the-art knowledge of processes and modelling capability has been reported by Selvadurai et al. (2014). There are, however, still unresolved questions and uncertainties, which need to be addressed in order to assess the possible effects of permafrost on the bedrock prior to future milestones, for example the PSAR and SAR for the Spent Fuel Repository and SFR.

The development of coupled hydrogeological and hydrogeochemical calculation tools, which is described in Section 12.3.1, makes it possible to model geochemical processes and transport processes for flow conditions that change continuously in time, for instance the changes that take place during a glacial cycle.

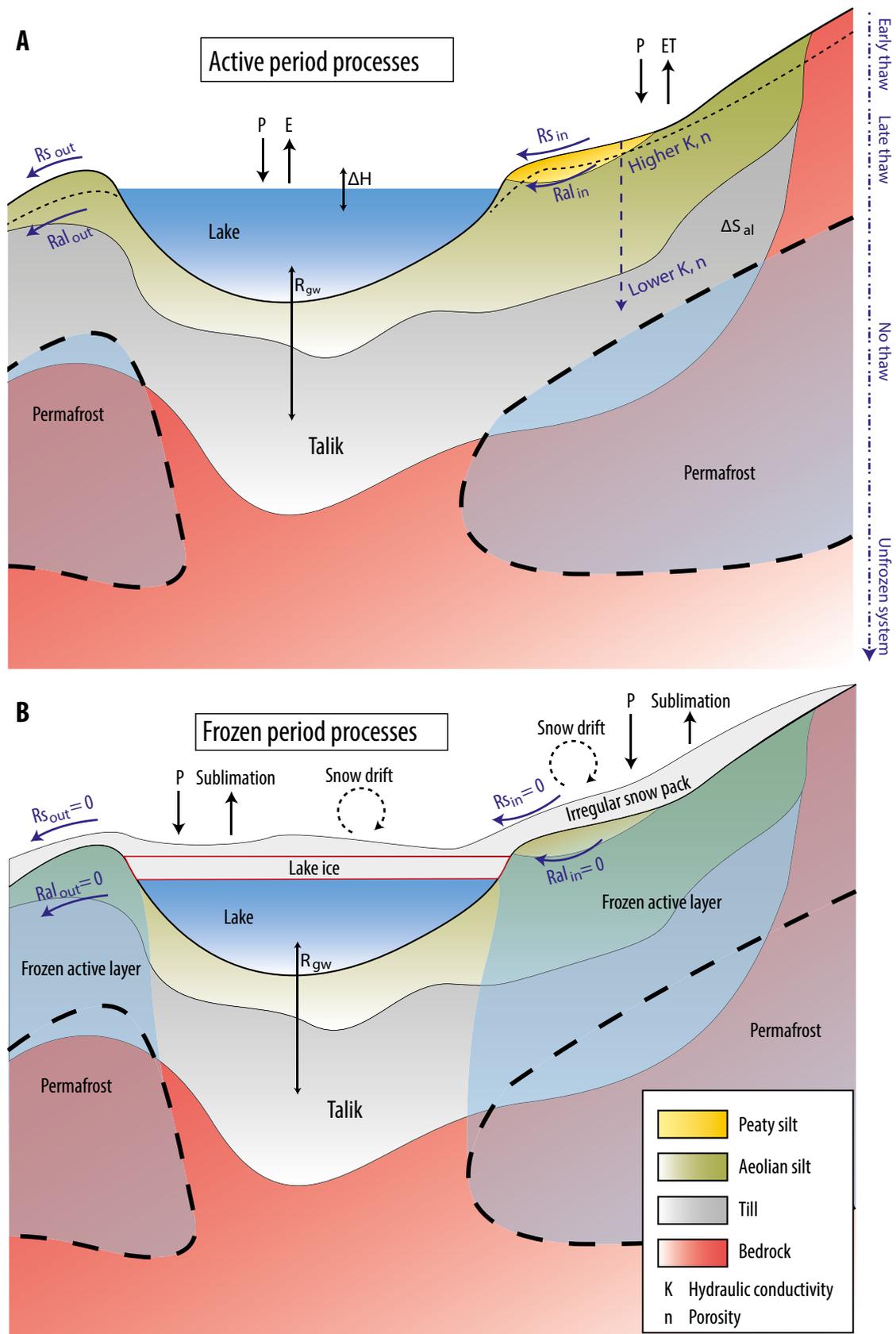


Figure 12-3. Conceptual model of the hydrology in the periglacial catchment area investigated by SKB in Greenland (Johansson et al. 2015). Two models have been developed, one model for the active period and one for the frozen period. R_s – surface runoff, R_{al} – flow in the active layer, R_{gw} – groundwater exchange between lake and talik, P – precipitation, E – evaporation, ET – evapotranspiration, ΔH – lake level variation, ΔS_{al} – reservoir change in the active layer.

Programme

The focus of the programme is to develop the understanding of spatial and temporal changes of rock and regolith properties caused by permafrost and glaciations, and the effects on groundwater flow and solute transport. The knowledge of water flow and transport through frozen ground has to be improved. In situ experiments are therefore planned at the GRASP site in combination with modelling, where specific water temperature effects on water flow and permafrost dynamics will be studied.

The value of the meteorological and hydrological time series that have been collected in Greenland within the framework of the now concluded projects GRASP and GAP increases with time. Longer time series provide a better understanding of the natural variation and reduce the uncertainty in the models that have been developed for the periglacial system. The established monitoring programme on Greenland will therefore continue, as a first step for the next two years.

The projects GRASP and GAP have contributed new and valuable knowledge to both SKB and the scientific community. Data from the two projects have, however, not been analysed together and the conceptual understanding of the coupled glacial/subglacial/periglacial system is still insufficient. Even though the two projects have contributed a substantial amount of new knowledge of processes in areas with deep, cold and continuous permafrost, several knowledge gaps remain linked to the time of freezing and thawing of permafrost. SKB is therefore planning a new project in the period 2019–2022, in which these knowledge gaps and the link between the glacial and periglacial systems will be in focus. In cooperation with other nuclear waste organisations and representatives from academia, SKB is establishing a network for site-specific hydrological modelling in periglacial environments. The network will mainly focus on three questions: the coupled glacial/periglacial system, hydrological processes in sporadic, discontinuous permafrost and biogeochemical processes in periglacial environments. Within this network, SKB plans to fund a PhD position.

Three-dimensional THM models on different scales will be established and the uncertainty range will be studied. The problem is highly site-specific and the models that are established must therefore take into account the impact of ice loads and the potential effects of thermal basal conditions of a future ice sheet in Forsmark. The development is initially of a research character; operational, site-specific models can be available at the earliest prior to the application for an operating licence.

There is a need to increase the understanding of the effects of freezing on the mechanical stability of the rock at different depths. Fracture opening and new fracturing may result in new flow paths in the rock closest to the tunnel openings, even long after closure of the repository. Studies to develop a fundamental understanding of failure mechanisms will therefore be conducted, and three-dimensional coupled THM models will be established, incorporating both DFN descriptions and mechanical properties that are relevant for the different repositories.

The coupled hydrogeological and hydrogeochemical modelling tools will be used to analyse the geochemical evolution during a glacial cycle. Temperate and glacial climate conditions and conditions with permafrost are included here in order to determine if and how present day geochemical and hydrogeological conditions may be re-established after a glaciation cycle. The new knowledge will be applied within site-specific and safety-related modelling of Forsmark.

12.4 Detailed site investigations and site-descriptive modelling

Detailed site investigations (including monitoring) and site descriptive modelling carried out in conjunction with the construction and operation of the final repositories will provide information on rock properties, state and basis for assessment of compliance. Detailed site investigations and site descriptive modelling are carried out fully integrated with rock excavation during the construction of each facility and, specifically for the Spent Fuel Repository, during the subsequent successive development extension of the deposition areas during the operational phase.

12.4.1 Methodology for detailed site investigations

Since the publication of the detailed site investigation programme for the construction and operation of the Spent Fuel Repository (SKB 2016), the work with the reference design of the facility in Forsmark has continued, along with the work of updating the technical design requirements relating to post-closure safety. As a part of the continued development of the detailed site investigation programme to formulate operational programmes, the work will continue with developing, or updating and describing, methods, instruments and modelling methodology to enable verification that stipulated requirements have been fulfilled.

Detailed site investigations with associated site descriptive modelling have several purposes. They should provide the information required for enabling progress in underground construction and contribute to improved knowledge concerning conditions of importance for the assessment of post-closure safety. During construction of the Spent Fuel Repository, and to an even greater extent during the operational phase when deposition tunnels are built, an important task is to gradually adapt the facility layout to the prevailing rock conditions so that the requirements linked to the repository's nuclear safety (technical design requirements) are fulfilled. Furthermore, collected data and updated models that describe the bedrock provide a comprehensive basis for updated assessments of post-closure safety. The detailed site investigations will also provide the information required for environmental monitoring.

Current situation

The detailed site investigation programme (SKB 2016) comprise the basis for formulating operational programmes that describe the detailed site investigations and modelling with the associated quality assurance in the different phases of the construction and operation of the Spent Fuel Repository. Harmonised technical design requirements for a repository for spent nuclear fuel have been jointly established by Posiva and SKB (Posiva SKB 2017). In its review of the application for the KBS-3 system, SSM identified remaining needs and uncertainties linked to site understanding and realisation (SSM 2018).

The work on a preliminary programme for detailed site investigations and site descriptive modelling for the preparation of deposition areas has been initiated. This programme includes a description of strategies for investigation cycles and site descriptive modelling in conjunction with the preparation of main tunnels, transport and deposition tunnels, and deposition holes with preceding pilot boreholes. Furthermore, work on the operational programme for preparation of accesses and the central area has been initiated. This programme includes detailed site investigations and site descriptive modelling in conjunction with excavation of the skip shaft and ramp, with subsequent preparation of the central area and the remaining three service shafts, the latter excavated by raise boring. As a foundation for the operational programmes, updates of discipline-specific and integrated modelling methodology has been carried out, as well as development of methods and instruments. The latter work is initially focused on the tools that will be used in the construction of accesses and the central area in the Spent Fuel Repository, and some development for deposition areas that is expected to take a long time and therefore needs to be started early.

Programme

Prior to the PSAR and detailed design, an operational programme for detailed site investigations with associated modelling prior to the start of construction of accesses and the central area will be formulated during the RD&D period. Furthermore, a preliminary operational programme for deposition areas will be prepared. The focus of both programmes are thus on the Spent Fuel Repository, but the developed methods/tools and modelling methodology can also be used within the framework of the extension of SFR, and for SFL, regardless of where this repository will be sited. Tools here refer to measurement instruments and data systems including calculation tools and software used for visualisation and modelling, and the strategies associated with their application. Strategies and methodology for verification of requirements for post-closure safety (technical design requirements), such as control of the excavation damaged zone (EDZ), the occurrence and quantification of spalling, and hydraulic criteria for accepting deposition holes, will be devised and tested.

Other development regarding detailed site investigations in deposition areas will continue during the RD&D period.

12.4.2 Modelling methodology for detailed site investigations

Design, production, site understanding and post-closure safety define the different needs and requirements that site descriptive modelling faces during repository construction and operation. Design and production require continuous predictions on the tunnel scale, which means that modified and in some cases new modelling methodology and new prediction tools have to be developed. The methodology is focused on an early and close co-interpretation and integration of geological and hydrogeological information for identification and classification of critical structures (Section 12.4.3), and of the hydraulically connected fracture network. The fracture network constitutes the geometric basis for continued site descriptive modelling on the facility part scale (e.g. accesses, central area or an individual deposition area) or the facility scale (the entire repository with its accesses). The other disciplines, i.e. hydrogeochemistry, rock mechanics, thermal properties, transport properties and surface systems, continuously include and incorporate new information from monitoring in existing boreholes and characterisation of new pilot boreholes and tunnels. In addition to serving as a basis for discipline-specific description modelling, this information provides additional support for modelling in the form of single-hole interpretation and single-tunnel interpretation, which constitute fundamental building blocks in the continued integrated deterministic geometric modelling in 3D. The latter is linked to the stochastic description of fractures (DFN), see Section 12.1.4, in the rock mass between the deterministically modelled deformation zones. Some fractures/structures, which were initially stochastically described, are expected to gradually obtain such informational support that they can be (re-) interpreted as deterministic structures, mainly in the description of deposition areas on a tunnel scale.

An important starting point for further development of the modelling methodology for all disciplines is the transient and disturbed environment experienced during construction and operation of the repository. Effective surveillance and measured changes relative to established reference levels comprise important prerequisites for effective follow-up of the effects of the gradual development of the repository and for calibration of quantitative models.

Current situation

At present, there are preliminary methodology reports for geological and hydrogeochemical modelling during detailed site investigations. Work is under way with the corresponding documents for the other disciplines.

Programme

Continued development of discipline-specific and integrated modelling methodology is planned to take place during the RD&D period. Application and testing of the preliminary methodology is planned in conjunction with continued work in the Äspö HRL, in Forsmark and in cooperation with Posiva in Onkalo. The results of these tests serve as a basis for further refinement and mutual adaptation of the discipline-specific methodology reports.

12.4.3 Critical structures and volumes

The concept of critical structures and volumes features geological structures or rock volumes with properties that may have a negative impact on the constructability and post-closure safety of a KBS-3 repository. Critical structures are divided into three classes with respect to their influence on the repository area layout (Munier and Mattila 2015). Class 1 (CS1) includes structures and volumes with properties such that they cannot be accepted within the repository footprint. These have been crucial in the site selection for the Spent Fuel Repository and constitute boundaries for the repository area. Class 2 (CS2) includes structures and volumes with properties such that they can be accepted between deposition areas but not within deposition tunnels, and which thereby control the layout of individual deposition areas within the repository area. Class 3 (CS3) includes structures and volumes with properties such that they cannot be allowed to intersect deposition holes, and which thereby control the location and acceptance of deposition holes. Furthermore, the concept critical structure also includes the structure's stability as manifested in a tunnel and the structure's significance for flow and retention (Posiva SKB 2017). Methods for effective characterisation and classification of critical structures have to be developed, as well as QA systems to make this classification traceable.

Current situation

The methodology for assessment of the effect of earthquakes on a KBS-3 repository has been continuously developed since its first application in SR-Can and thereafter in SR-Site (Fälth 2018). In particular, an extensive matrix of case studies has been analysed with the intention of addressing the conservatism that influenced previous modelling work (see also Section 12.2.3). The results of these efforts indicate that the critical radii, i.e. the smallest fracture that may pose a threat to the post-closure safety, are essentially much larger than what has previously been demonstrated. The critical radii are more than approx. 200 metres in all parts of the final repository. In practice, this means that the structures that are critical for canister integrity (CS3, as per Munier and Mattila 2015) are in fact minor deformation zones, which can be readily identified by traditional mapping and investigation methods. The difficulties that were previously linked to the identification of relatively small, singular fracture planes are therefore no longer considered relevant. Specially developed methods for identification of the latter type of structures will therefore probably not be required. This is also confirmed by fracture studies in Olkiluoto, Finland (Nordbäck and Mattila 2018) showing that single fracture planes do not become larger than 150 metres in radius before they connect with networks of other fracture planes to form minor deformation zones.

A remaining challenge is the methodology for categorisation of critical structures related to groundwater flow and retention, for example with respect to the transport resistance (F-factor). This type of analysis requires that the individual conductive structure is analysed as modelled in its context, i.e. in a hydrogeological flow model based on deterministically modelled deformation zones and stochastic DFN. Thus, the summed-up F-factor for the relevant flow path, where the potentially critical structure is included, is the critical factor for categorisation of the structure in question. This implies that even if the individual structure exhibits a high F-factor, it is the integrated F-factor for the whole flow path that is crucial for whether the structure is to be categorised as critical or not.

Programme

Results of both modelling and mapping of potentially critical structures show that they correspond in size to minor deformation zones, for which traditional mapping methods (Cosgrove et al. 2006, SKB 2016) are considered adequate for safe identification in the context of a KBS-3 repository. Specific research efforts for development of new or improved mapping methods are therefore not deemed to be necessary.

For more in-depth analyses of critical structures related to shearing of copper canisters with spent nuclear fuel, including the effects of multiple earthquakes, see Section 12.2.3.

Continued development of the methodology for DFN-based modelling of groundwater flow, including both deterministically modelled deformation zones and the stochastically described naturally fractured rock mass, see Section 12.1.1. Critical structures related to groundwater flow and retention can be assessed by means of these models.

12.4.4 Earth currents in the Forsmark area

Low-frequency and static electric fields in the Earth's crust are generated by naturally occurring processes and electrical infrastructure. Since soil and rock layers, to a varying degree, are electrically conductive, these electric fields will be associated with electric currents. These static earth currents may increase the corrosion of metal in underground facilities under certain conditions. In the Forsmark area, an electrode is installed as a part of the high-voltage direct-current cable Fenno-Skan, which connects the Swedish and Finnish electrical power grid. Depending on the current load in the cable, electric current flows in or out of the electrode, typically a couple of hundred amperes. The flows from Fenno-Skan provide a substantial contribution to the static earth currents observed in the area.

Current situation

Corrosion on equipment for hydrological monitoring installed in boreholes in Forsmark has been linked to earth currents from Fenno-Skan. Problems with corrosion of this equipment in certain boreholes have led to a number of investigations over the years (see Thunehed 2017, 2018 for a summary). The problems have been observed ever since the measurements in Forsmark started (Nissen et al. 2005), but

they have escalated after the extension of Fenno-Skan (Fenno-Skan 2 was put into operation in 2011). In order to reduce the problems, cathodic protection has been installed in the boreholes. This action has proven insufficient, and, moreover, it has turned out that it may lead to increased gas formation in connection with certain operational states of Fenno-Skan; additional actions have therefore been discussed (Sandberg and Taxén 2017).

The significance of Fenno-Skan for the post-closure safety of SFR has been investigated (SKB 2014d, Löfgren and Sidborn 2018). After closure, the metal in SFR will corrode (aerobically or anaerobically), regardless of whether an earth current field is present or not. An increased corrosion rate is not deemed to have any effect on the conclusions in the assessment with respect to these components. The significance of Fenno-Skan for the post-closure safety of SFR is therefore low (SKB 2014d).

The effects of earth currents caused by existing and planned future high-voltage installations on the corrosion of the copper canisters in a KBS-3 repository in Forsmark have been previously studied (Taxén et al. 2014), with the conclusion that stray currents give negligible contributions to the total corrosion.

Programme

The equipment in instrumented boreholes will now be modified according to the proposal that has been formulated as a part of SKB's cooperation with Swerea KIMAB (Sandberg et al. 2017). An investigation of the soil system at the drilling sites was initiated in 2019. The purpose is to devise additional measures that can be adopted in parallel with those mentioned above. Continued mapping of the sources and extent of the earth currents in the Forsmark area is planned to take place during the RD&D period, including the possible effects on the post-closure safety of the final repositories.

12.5 Tunnel production

The different parts of the underground facilities must comply with technical design requirements and other requirements depending on their function during operation and after closure. The rock vaults in SFR will be excavated with conventional technology, which means that no particular technology development is required. For excavation in accesses (ramp and shafts) and the central area, no further technology development is required; methods for cautious blasting have been developed. The special requirements and technical design requirements that apply for deposition tunnels and deposition holes in the Spent Fuel Repository differ from the requirements that apply to other construction objects. This entails challenges, both for the technical execution of the deposition holes and for the investigation methods that will be used for selecting their location, and for how to verify that the criteria have been met. A large part of the work with technology development for tunnel production involves identifying techniques and materials that are previously known and proven, and which meet the post-closure safety requirements as well as the technical design requirements for the Spent Fuel Repository.

12.5.1 Grouting

Grouting of underground openings in the Spent Fuel Repository is more challenging than in normal infrastructure projects. To ensure that the grouting will not have any negative effects on the rock as a barrier, there are strict requirements regarding which grouting materials can be accepted, how the grouting fan should be designed and that grouting is performed with a limited and very controlled spread. The high groundwater pressure at repository depth also requires another grouting design to prevent the erosion of grout, and to prevent the hydraulic widening of fractures.

Current situation

Experience from previously completed grouting (Funehag and Emmelin 2011, Johansson et al. 2013, Funehag 2016) has been analysed and evaluated, and incorporated into updated method descriptions for grouting with low-pH materials, adapted to the different type domains that can be expected to occur during the excavation of underground openings in the Spent Fuel Repository (see Jansson et al. 2019). The work also included devising a strategy for inspection and verification of grouting results.

Programme

Guidelines for grouting are implemented in SKB's programme for the Spent Fuel Repository and will be incorporated into the management system.

12.5.2 Deposition tunnel excavation

Current situation

Based on the deposition sequence in the Spent Fuel Repository, where several different machines and equipment are used, the tunnel floor must be sufficiently level to permit easy access to the tunnel and to reduce maintenance and wear of the machines. The current method for excavation of deposition tunnels is drill-and-blast, but alternative methods such as mechanical mining are being studied in order to achieve a more level tunnel floor, a smoother tunnel contour, and a reduced excavation-damaged zone.

Programme

During the RD&D period, the study of methods for mechanical mining will continue in order to gather data to support the technical decision to change the method for excavation of deposition tunnels.

12.5.3 Drilling of deposition holes

Current situation

SKB has followed Posiva's development work for drilling of deposition holes. Posiva has drilled ten vertical, full-scale experimental holes in Onkalo, distributed in two demonstration tunnels (Railo et al. 2015, 2016). In order to make the method for drilling of deposition holes more effective with respect to production, machine suppliers have been contacted to study several potential drilling techniques.

Programme

A drilling machine for drilling of deposition holes will be manufactured and the method for drilling of deposition holes will be tested and subsequently verified in conjunction with an integration test that is planned at the Äspö HRL or another suitable site (Section 11.6).

13 Surface ecosystems

13.1 Overview

SKB's research programme for surface ecosystems primarily intends to create a basis for calculations of the potential radioactive dose to humans and the environment in the assessment of safety after closure of the different repositories. The programme also provides a basis for environmental impact assessments, environmental monitoring and for assessing safety in the facilities in operation. Furthermore, the programme contributes to long-term maintenance of competence in the area, which is necessary for SKB's future work with safety assessments for the existing and planned repositories. The current research issues for the three different repositories on the topic of surface ecosystems are largely overlapping. SKB believes that there are no remaining critical research issues in the area that must be resolved prior to the PSAR for the Spent Fuel Repository, nor for an extension of SFR. However, the Radiation Safety Authority's reviews identify a number of questions which SKB has to resolve in future reports. In the safety evaluation of SFL, SKB has acted on SSM's review comments and shows possible ways forward for the continued work with applying SKB's methodology to describe biosphere processes and calculate accumulation in the surface system. This work requires supplementary data and further development of methods for future safety assessments.

There are several 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 this necessary in order to reduce uncertainties in future safety assessments. The most important remaining issues related to surface ecosystems are found within four different areas: 1) uptake pathways and uptake mechanisms for various organisms; 2) temporal and spatial heterogeneity in the landscape; 3) transport and accumulation processes; and 4) radiological, biological and chemical properties of important substances in the repositories.

An overview of SKB's work in this field in recent years can be found in the special issue of *Ambio* that was published in 2013 (Kautsky et al. 2013). As a part of the work with the most recently completed safety assessment, SR-PSU, SKB has published reports which describe assumptions in the modelling of the near-surface ecosystems (SKB 2015a), data and models used for the dose calculations (Grolander 2013, Saetre et al. 2013, Tröjbom et al. 2013) and the application of the models used in the safety assessment (SKB 2014a). In the safety evaluation of SFL, efforts have been made to reduce uncertainties in the assessment 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 the safety evaluation of SFL (SKB 2019a).

Review comments to the RD&D Programme 2016 from 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 for maintaining high scientific quality, with efficient utilisation of limited resources. An important part of fulfill this goal is collaboration with different research groups, both nationally and internationally. On the national level, SKB is collaborating with groups working at Krycklan (Laudon et al. 2013) and Skogaryd, sites that are included in the Swedish Research Council's national network SITES (<http://www.fieldsites.se>), and internationally, SKB participates in several different networks (for instance BIOPROTA, <http://www.bioprota.org/>).

A short description is given below of the issues related to surface ecosystems that are deemed to be most important in the RD&D period, and SKB's plans for working with these issues. A complete account of the results of research and development in the area in recent years can be found in the reports from the latest safety assessments and from the recently completed safety evaluation of SFL. The activities that are planned in conjunction with the SAR will be specified in detail in the research programme for the Spent Fuel Repository.

13.2 Uptake pathways and uptake mechanisms for radionuclides in various organisms

Current situation

Issues concerning uptake pathways and uptake mechanisms for radionuclides include both organisms and the aquatic and terrestrial ecosystems which they are a 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). Measurements have previously been carried out for a number of elements in the relevant ecosystems. However, there are gaps in these measurements, as previously pointed out by SSM, most recently in the review of SR-PSU, in which SSM emphasised the need of paired plant and soil samples for certain ecosystems.

The 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). In previous safety assessments, the carbon uptake in plants and animals has been based on the specific activity of inorganic carbon, i.e. the activity relative to the stable substance. In SR-PSU, the uptake of chlorine was assumed to be limited by the plants' nutritional needs, and in the safety evaluation of SFL, SKB also used a model for controlled uptake for two other plant nutrients (potassium and calcium). In conjunction with previous RD&D reviews, SSM has encouraged this work, and most recently in the review of SR-PSU, SSM expressed a positive view of SKB's continued work on deepening the understanding of the biological uptake of radionuclides.

For most large animals, the uptake of radionuclides is mainly linked to food ingestion; SKB's work has therefore focused on describing the food web. Initially, the 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.

More recent research indicates that certain radionuclides can accumulate near running waters (Lidman et al. 2017a, Tiwari et al. 2017, Ingri et al. 2018, Ledesma et al. 2018), but also that there are other processes that may affect the cycling and transport of radionuclides in running waters, for example gas exchange (Natchimuthu et al. 2017a, b, Wallin et al. 2018). The processes are affected directly or indirectly by the metabolism of plants or microorganisms, but also by chemical and abiotic processes (Sections 13.4 and 13.5), and SKB does not rule out that running waters may have a greater importance for radionuclide cycling in surface ecosystems in some future scenarios than previously assumed.

General mechanisms for uptake of radionuclides in terrestrial ecosystems have previously been studied with the Coup and Tracey models (Gärdenäs et al. 2009). These models have been developed further in recent years to include leaf uptake of wet deposition in agricultural systems (Gärdenäs et al. 2017), and in the safety evaluation of SFL, a simple representation of leaf uptake and allocation of radionuclides in plants was implemented in the models used for calculating consequences in conjunction with irrigation (SKB 2019a). During the work with SFL, a semi-empirical model was also used to better describe the processes that control chlorine uptake in plants (SKB 2019a). The model was originally developed for a forest in France (Montelius et al. 2016), and has now been parametrised with field data from Forsmark.

During the work with the 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 cycling of carbon-14. The biosphere model was updated in accordance with this in SR-PSU (Saetre et al. 2013). After comments from SSM's external reviewers (Walke et al. 2017), SKB has within the safety evaluation of SFL continued the work on developing the integrated modelling of stable and radioactive carbon. In the preliminary review of SR-PSU, SSM was of the opinion that SKB should continue the development of carbon-14 modelling and verify it by comparing modelling results with data and other carbon-14-models. SKB is participating actively in the international working group for carbon-14 in BIOPROTA. In the safety assessment, the cautious assumption is made that all carbon-14 is available for fixation via photosynthesis (in the form of carbon dioxide or carbonate). That this is a reasonable assumption in unsaturated soil layers has recently been shown experimentally (Hoch et al. 2014), but in anoxic environments, carbon-14 might also reach the biosphere in the form of methane.

This could affect the proportion of a carbon-14 release that is available for uptake in plants and the rest of the food chain, and if it were to be considered in the transport and dose modelling, it could thereby have an effect on the calculated doses.

In the last few years, SKB has initiated efforts to describe the cycling of methane in natural ecosystems (Natchimuthu et al. 2015), and SKB has access to a large and active scientific network via a collaboration with Linköping University. Innovative method development has resulted in new measurement methods (Bastviken et al. 2015, Gålfalk et al. 2015) and valuable knowledge of methane flows in different biotopes (Natchimuthu et al. 2015, 2017b, Denfeld et al. 2016, Wallin et al. 2018). 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 transport via air channels in aquatic plants. Carbon-14 from radioactive waste can thus either reach the atmosphere as methane or carbon dioxide (CO₂), or be taken up in the food web via methane-oxidising microorganisms or via plant uptake of carbon dioxide formed by methane oxidation. Methane emissions show large dynamics and variation in, for example, running waters (Natchimuthu et al. 2017b, Lupon et al. 2019) and ice-covered lakes (Denfeld et al. 2016). At the same time, the vegetation type does not seem to play any decisive role for the cycling of methane in wetlands (Milberg et al. 2017). Preliminary results from methane measurements in Forsmark show a very high methane emission from one wetland, indicating that methanogenesis and methane discharge may be important in the type of environment where radionuclide releases from a repository could occur. The transport and uptake of gas may also be significant for other elements that can evaporate, such as selenium, iodine and chlorine (Hardacre and Heal 2013). In natural water, chlorine most often occurs as the relatively mobile inorganic chloride ion, but under certain conditions, it may form organic compounds, which have other uptake and storage pathways than inorganic chlorine. Therefore the residence time for chlorine in terrestrial systems is very long, and there are several different pools of chlorine that behave in completely different ways (Bastviken et al. 2013). The transformation and uptake are controlled by biological processes in the soil and plants. The knowledge of evaporation (volatilisation) of chlorine has been summarised in a report (Svensson 2019). The new picture of chlorine that is emerging is based on studies in terrestrial environments, but there are no corresponding studies in aquatic systems, see the safety evaluation of SFL (SKB 2019a) and also Section 13.5.

The assessment of doses to non-human organisms that was carried out in SR-Site has thereafter been updated by Jaeschke et al. (2013). The updated methodology has been used in the assessment for SR-PSU, and there it was integrated with transport and accumulation calculations (SKB 2014a). In the review of SR-PSU, SSM reacted positively to this methodology development (SSM 2019). SKB notes that models that describe uptake and accumulation of substances in food webs and ecosystems provide an opportunity for independent comparisons with the Erica tool.

SKB has tested new methods for describing the composition of organisms on the sites, in order to thereby be able to identify organisms that may be exposed to or transfer radionuclides, or be disturbed by other activities at the repository. The presence of the pool frog in ponds that may be occupied by frogs was investigated by eDNA analyses on water samples. The study showed that this may be a useful method for general inventories (Eiler et al. 2018).

Programme

SKB will continue its long-term efforts aimed at replacing or supplementing the concentration ratios for organisms with mechanistic models. A validation with field data (for example nutrients and other stable substances) is planned for existing models. In the recommendations for a continued monitoring programme in Forsmark (Berglund and Lindborg 2017), supplementary sampling in different biotopes is proposed. During the coming five-year period, new and coordinated plant and soil sampling is planned within the operational detailed site investigation programme (Section 12.4). One purpose of the sampling is to fill in the gaps that exist today regarding elements (Section 13.5). The analyses are planned to be carried out using milder leaching processes than previously to better represent what is available to plants. It also provides a basis for the development of mechanistic models for substances that are taken up passively (for example with water) and actively (for example substances similar to nutrient salts). Furthermore, a comparison of results from the CR-based methods used thus far with results from allometric methods (for example Beresford et al. 2016) is planned.

As a continuation of the work with the lake ecosystem model, a model for running waters will be developed. Although running waters in many ways resemble lake ecosystems, differences related to hydrology, the potential for chemical precipitation of different substances (Section 13.5), and biological uptake mechanisms may provide other conditions for accumulation of radionuclides and doses. In parallel, work with development of the terrestrial ecosystem model will continue, with a focus on root uptake in plants and on links between radionuclide uptake and water transport or primary production.

Ongoing studies of the properties of chlorine in terrestrial and aquatic environments (wetlands, lakes and running waters) will continue with the purpose of developing the models for transport and accumulation of chlorine-36. In the continued work, new measurements of organically bound chlorine are planned (Section 13.5), which will be subsequently linked to the flow of chlorine through the system and to biological production, to thereby permit modelling of the chlorine cycle. The models will also include evaporation of chlorine as described in Section 13.5.

The importance of gas transport, for example of methane and carbon dioxide, in water, ground and atmosphere will be investigated further, along with uptake processes for gas via primary production and microbial metabolism. The methane flow from deep soil layers in natural ecosystems will be studied in cooperation with Linköping University with the aid of newly-developed measurement methods, in order to assess the extent to which the flow is diluted by methane produced closer to the surface, and how large a fraction is oxidised or released to the atmosphere. Continued work is planned to study the emissions and quantify the sources of methane in Forsmark.

SKB will also continue to actively participate in BIOPROTA's work with carbon-14.

13.3 Temporal and spatial heterogeneity in the landscape

One of the most important factors for the calculated dose after a release is the variation of the landscape in space and in time. Depending on the type of ecosystem in which a release occurs and the properties of the ecosystem in relation to the point in time at which the release occurs, the calculated dose may be affected by several orders of magnitude.

Current situation

As a basis for safety assessments in Forsmark and Laxemar-Simpevarp, SKB has formulated a description of the events and processes that have determined the development of the landscape to date (Söderbäck 2008). The historical description, combined with a description of the present-day landscape and the understanding of its function (SKB 2008, 2009b), has been used to describe probable landscape evolution under different assumptions regarding future climate and shoreline displacement. The landscape evolution in Forsmark is determined mainly by climate variation and shoreline displacement (Chapter 14). On top of these large-scale and slow changes, the landscape is also affected by the deposition and reworking of sediment, which entails, among other things, 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.

Flooding areas and hydrological discharge areas for deep groundwater are located in topographical low-lying points in the landscape, for example around lakes, streams and wetlands (Section 13.4). The conditions for transport and accumulation of radionuclides in these areas are determined on the one hand by the topography and properties of the local and regional catchments, and on the other, by discharge area properties such as size and layer thicknesses. The digital height model and the regolith depth model have been updated in the site modelling (Sohlenius et al. 2013). These updated geometric models should be incorporated into the landscape models (see below).

There is a natural covariation between the size and properties of the discharge areas that may receive releases, and this covariation is strongly dependent on where in the landscape the objects are localised. The location of an object is also linked to the size of the catchment area, and it affects the time periods in which a release to the object is possible (Berglund et al. 2013). Variations in local topography, soil stratigraphy and ecosystem succession can also be expected to create heterogeneity within a discharge area, which was pointed out by SSM in the reviews of SR-Site and the RD&D Programme 2013.

SR-SFL describes 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 2019a, d).

In order to investigate how small-scale variation in topography and soil layer thickness influences the transport and accumulation of radionuclides reaching a discharge area, SKB has initiated simulations with the COMSOL tool (von Schenck et al. 2015, Silva et al. 2015, Abarca et al. 2016b). The studies include the main discharge area for SFR, a landscape profile in the Krycklan area in Västerbotten (Abarca et al. 2017), and a discharge area in Laxemar. Since tools have recently been developed that link water flows to chemical and physical processes such as sorption and precipitation in spatially distributed models, there are now good prospects for studying how for example flow paths, layer thickness and redox reaction zones affect the accumulation of various substances, and how dissolved organic matter influences the transport of substances (see further Section 13.4).

The PhD study in the Krycklan area previously funded by SKB (Lidman 2013) has continued in the form of a postdoctoral position. Several studies related to this show that the landscape's large-scale mosaic of forest and wetlands has a large impact on the mass transport of various substances, and that gradients in thin layers may have a strong influence on both the forms of occurrence and the accumulation of various substances (Lidman et al. 2016, 2017a, Tiwari et al. 2017, Ingri et al. 2018, Ledesma et al. 2018). Furthermore, the work has provided several important background reports for SKB's safety assessments (Lidman et al. 2017b, SKB 2019a).

In order to understand the landscape effects on groundwater flows and transport of substances in a cold-climate domain, an area on Greenland in front of the ice sheet margin has been studied within the GRASP project (Section 12.3.4). The study included, among other things, the hydrological interaction between the active layer, lake, and talik, and the importance of windblown deposition for the formation of lake sediments and for the chemical properties of these sediments. Two PhD theses have been published (Johansson 2016, Lindborg 2017). They describe the hydrology and material cycling in the area (Lindborg et al. 2016a, b, Rydberg et al. 2016, Petrone et al. 2016). The new knowledge from the GRASP project has not yet been fully utilised for the understanding of our sites in Sweden.

SKB has participated actively in the IAEA projects EMRASII and MODARIA I and II. A working group within MODARIA I and II has worked to achieve a consensus regarding the description of future climate evolution (IAEA 2016, Lindborg et al. 2017, 2018). The project is important for establishing a common view of climate evolution and how it might affect near-surface ecosystems. The results from work in MODARIA are being used in an ongoing revision of the IAEA's biomass methodology (Lindborg 2018) within the framework of BIOPROTA.

As described above, SKB has since the start of the site investigations developed a landscape model to describe the evolution of sea basins, lakes and soil layers. The model has been used in the safety assessments SR-Site and SR-PSU (Brydsten and Strömngren 2010, 2013, Lindborg et al. 2013), but the tools and methods used need to be updated for future safety assessments, both in order to meet future needs and to secure competence in the area. In order to describe landscape evolution, SKB has therefore initiated work on an alternative model, Untamo (Pohjola et al. 2014), which is developed by Posiva and partially based on SKB's previous landscape modelling.

In the review of SR-PSU, SKB received some criticism for a lack of transparency in dose calculations as they are driven by complex support models that are difficult to overview (Walke et al. 2015). However, experience from SR-PSU shows that it is difficult to predict and simplify processes that are dependent on landscape evolution before the dynamics of release are known (Kautsky et al. 2016). SKB therefore intends to proceed with simplified models, but this will primarily be carried out in connection with the interpretation of results (e.g. SKB 2019a, see also Section 13.4). The safety evaluation of SFL shows that a more complex model, which uses the understanding of how agricultural land ages, results in lower doses than the simplified model that was used in the base case (SKB 2019a). The reason for lower doses differs depending on radionuclide.

Programme

The insight from the work with COMSOL models in the safety evaluation of SFL will be developed further for potential discharge areas in Forsmark, to comprehensively shed light on the importance of different assumptions concerning the spatial variation and resolution of the calculated dose to the most exposed group. SKB judges that continued work with the COMSOL tool will provide an opportunity to resolve or illustrate several issues concerning spatial resolution.

Modelling of landscape evolution or the evolution of individual ecosystems will be studied by comparing SKB's previous results with the landscape model Untamo. Knowledge from the GRASP project and from Krycklan will be applied to the Forsmark area to illustrate the effects of a colder climate on the landscape level.

SKB will also continue to actively participate in the international work aimed at describing the long-term evolution on the landscape level within BIOPROTA and in a potential continuation of the IAEA programme MODARIA II.

13.4 Transport and accumulation processes

Transport processes here mainly refer to abiotic transport by water, particles and, to some extent, gas. The term accumulation processes here refers to processes in loose deposits (for example sorption), but not to the uptake of radionuclides in organisms (Section 13.2).

Transport modelling is a central component in both dose modelling and ecosystem modelling. Hydrology affects which areas may be contaminated by a release of radionuclides, and the amounts and concentrations of nuclides that reach these areas, whereas sorption determines the amount which can be accumulated in the soil layers or be associated to particles. The uncertainties in sorption (K_d) are often very large, which affects the dose calculations.

Current situation

In the reporting for the safety evaluation of SFL, it is described how the current understanding of transport and accumulation processes in the biosphere has been used in the safety evaluation (SKB 2019a).

In the review of SR-Site, SSM noted the importance of the division of soil layers. In the safety evaluation of SFL, it was investigated how the division of soil layers affects the modelled transport of radionuclides. When the discretisation of the lower ground layers in the biosphere model was increased, the dispersion in the calculations was also reduced so that it corresponded to the macro dispersion that can be observed in field tests. This affected accumulation and time, but the effect was strongly dependent on the properties of the biosphere object and the radionuclides (SKB 2019a). Even though the effects of the model update are limited with respect to most of the dose-contributing radionuclides in areas with high groundwater discharge, SKB believes that the new model gives a more accurate picture of the effects of radioactive decay during transport through layers of soil, and therefore intends to use the finer discretisation in future safety assessments.

The decay products of the uranium chain exhibit transport properties that differ from each other, and in the safety evaluation of SFL, a modelling study was carried out that demonstrated the importance of the half-life, the sorption, and the transport rate in the different ground layers (SKB 2019a). In addition to providing an opportunity to refine temporal and spatial resolution (Section 13.3), the COMSOL tool also constitutes a complement to the modelling carried out with the hydrological tool MIKE SHE. COMSOL makes it possible to study in detail mechanisms driven by physical, chemical and biological processes, similar to what can be done with CoupModel (Gårdenäs et al. 2009, 2017), at the same time as landscape can be used as a driving factor for water flows. A project studied the possibility of using COMSOL to calculate K_d values that vary over time and space for some well known elements (Silva et al. 2015, Abarca et al. 2016b).

In the safety evaluation of SFL, radionuclide transport was studied in detail in the object that gave the highest doses. A three-dimensional COMSOL model calculated water flows and sorption in different regolith layers (SKB 2019a). Modelling showed that for several layers, flows and concentra-

tions were concentrated to a few areas, while other layers efficiently spread radionuclides in a large volume. The same model was used for comparison with a simplified model in Ecolego (SKB 2019a). A more detailed MIKE SHE model studied the hydrology in the same area (Johansson and Sassner 2019), which resulted in a similar pattern of discharge areas. A difficulty in several of these detailed studies is that the results are strongly affected by how the physical boundaries are set, which makes further work necessary to provide comparable results.

The well-studied so-called S profile in Krycklan (Lidman et al. 2017a) has been implemented in the COMSOL tool. The model shows the speciation of uranium in different organic complexes and is compared with measurements in the profile. The database in the chemical modelling tool PHREEQC has been updated in order to handle organic acids better, since complexation with these is crucial for the mobility of uranium and many other radionuclides in these environments. A PhD project was initiated in the same area with a focus on solute transport near the surface. In a first step, the importance of ground frost for the interaction between surface water and groundwater was studied (Jutebring Sterte et al. 2018), and continued work is planned related to reactive transport (Section 12.3.1).

In the transport of radionuclides from terrestrial to aquatic ecosystems, the near-stream or littoral zone often plays a crucial role because it often has, compared to normal forest soils, a lower pH, more reducing conditions and a higher concentration of both solid organic materials, which can bind many radionuclides, and dissolved organic matter, which can transport radionuclides that would otherwise not be mobile. This can lead to the concentration of certain substances, for example thorium, being a hundred times higher in the soil water in the near-stream zone than in normal forest soils. However, no clear differences have been detected related to the uptake in vegetation, which is probably connected to the fact that these substances are bound to dissolved organic carbon to a high degree and thereby not bioavailable (Lidman et al. 2017a). For the long-term transport of radionuclides through the landscape, however, it is an important environment, since it controls what is retained in the terrestrial systems and what is transported further to the surface waters (Ledesma et al. 2018). The importance of these organic soils for the transport of different main and trace elements is also visible up to a landscape level (Tiwari et al. 2017). The lanthanide series is of particular interest for understanding the long-term mass balance in these environments, in particular europium, which also in the Forsmark area exhibits a diverging behaviour. Europium therefore has the potential to act as a tracer in the dissolved phase and as a tool for quantifying weathering and accumulation processes in soils.

SSM and its reviewers have requested simpler radionuclide transport models as a complement to those used in the safety assessment (SSM 2019). In the safety evaluation of SFL, development of an equilibrium model of the existing radionuclide transport model was initiated so as to simply and quickly be able to investigate how different parameters and assumptions affect the modelling results. A simplified hydrological model was also implemented in the safety evaluation of SFL, describing the vertical flows as a function of the discharge (SKB 2019a).

To calculate the effect of nitrogen dispersion from releases of remnants of explosives during the construction and operation of the repositories in Forsmark, the oceanographic models have been updated with the knowledge collected through site investigations and research (see summary in Aquilonius 2010). The results from these studies, together with the results from the recently completed Predo project, a joint project for new calculation methods for doses from the operation of nuclear facilities, will provide valuable insights regarding water transport in the sea area outside Forsmark.

Programme

With respect to hydrology and transport, several activities that also touch upon the surface systems are described in Section 12.3. An important question pointed out by SSM in the reviews of the latest safety assessments is how to transform a detailed hydrological model to parameters that describe hydrological flows, which can then be used in the safety assessment to describe the main transport characteristics by means of a simple box model. In the safety evaluation of SFL, a study (SKB 2019a) was initiated which will continue with data from the Forsmark area. Alternatives to the current methodology will be investigated, with the objective of developing a methodology that captures the transport of water and solutes during longer time periods and in an accurate, simple and comprehensible manner while maintaining transparency regarding which actual mechanisms control the transport.

As mentioned in Section 13.2, continued development of the model for running waters is planned. In addition to the movements of water to and in water courses, there are several chemical processes that take place in the transition from land to water (Section 13.5), and this can greatly affect the conditions for accumulation of radionuclides and thereby the potential dose. The intention is to develop a radionuclide transport model for this heterogeneous ecosystem.

Further studies of the Krycklan catchment are expected to provide valuable insights regarding the processes leaching, mobilisation and accumulation of substances in a landscape perspective. An interim goal is to model, with the aid of chemical and hydrological boundary conditions, accumulation, leaching and solute transport in a cross-section of a catchment area.

In the safety evaluation of SFL, a study of the uranium decay chain was carried out in conjunction with the modelling of the surface ecosystems. This study will be developed during the coming period. The work is planned to include the transport of and exposure to radon, and aims at shedding light on the consequences of different transport rates for the calculated dose. In the site modelling, the oceanographic model is planned to be updated and adapted for solute transport in the area.

13.5 Radiological, biological and chemical properties of important elements

In addition to different elements being taken up in various organisms and food chains (Section 13.2) and transported and accumulated in different ways in loose deposits (Section 13.4), they also have various chemical and radiological properties. These properties, in interaction with the chemical environment in which the substances occur, affect both their mobility and radiotoxicity. This section intends mainly to describe the programme for measuring or compiling these fundamental properties.

Current situation

For the safety assessment SR-PSU, a compilation was made of the concentration ratios (CR) and sorption data (K_d) used in the analysis of near-surface ecosystems in Forsmark (Tröjbom et al. 2013), and these data have thereafter been updated prior to the safety evaluation of SFL. The values are based primarily on measurements from the site and secondarily on available international data. During the period 2009–2019, SKB has participated in the IAEA programmes EMRASII and MODARIA I and II. Within these programmes, aggregated CR and K_d values have been compiled, and they will be published in a forthcoming IAEA report. Parts of this work have already been published in other contexts (Wood et al. 2013, IAEA 2014, 2016). Posiva has also compiled interesting chemical data from near-surface ecosystems in Finland (e.g. Haavisto and Toivola 2017, Kuusisto 2017, Lahdenperä 2017), which permits comparisons with SKB's data.

There is a need to supplement the completed compilation with data for certain substances that are judged to be important for SKB's various final repositories (e.g. radium, molybdenum, nickel, chlorine, and carbon-14).

In order to get an overview of the properties of molybdenum and its path from the source to the repository, SKB arranged a workshop with approximately 30 participants from the nuclear power industry, SKB, and invited researchers (Lidman et al. 2017b). In the work with the safety evaluation of SFL, new data for molybdenum and nickel have been compiled from SKB's site investigations, the Krycklan catchment and the literature (SKB 2019a).

A thorough survey and synthesis of available chemical data has been carried out for surface waters (lakes, streams and sea), near-surface groundwater, soils, sediments and biota from the site investigation areas, with a special focus on molybdenum and nickel (Lidman et al. 2017b, SKB 2019a). 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.

An initial study of the cycling of chlorine in Forsmark indicates unusually high chlorine concentrations in the field layer (Svensson et al. 2018). Other new studies of chlorine cycling and forms of occurrence (Montelius et al. 2016, Svensson T et al. 2017, Svensson 2019) are summarised in SKB (2019a).

The chemical environment of an element is crucial for its chemical speciation, which in turn determines the element's properties in this environment. The chemical environment can be characterised by means of pH, redox, concentrations of main components (i.e. dominant anions and cations) and the amount of dissolved organic matter. The chemical environment at a site can change over time through a number of different processes, such as land uplift, climate changes, weathering and ecosystem succession.

The occurrence of wetlands and other organic soils is important for the mobility of many radionuclides. This depends mainly on an increase in the concentration of dissolved organic carbon (DOC), which in turn leads to lower pH and an 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 13.4) by affecting both pH and DOC concentrations, and thereby the mobility of many radionuclides (Lidman et al. 2016).

The Forsmark area is characterised by an abundant occurrence of calcite in the soil, which among other effects leads to a high pH and high carbonate concentrations in the water. By means of leaching processes, the effect of calcite on the water chemistry is expected to gradually decrease during the coming millennia, which can be expected to have significance for the mobility of several important radionuclides. Among other elements, uranium is expected to become much less mobile in a future landscape with less calcite influence. Studies of the Krycklan catchment in northern Sweden show that the ratio of uranium-234/uranium-238 can act as an indicator for deeper groundwater (Lidman et al. 2016), and in Forsmark as well, clear differences between uranium from the bedrock and uranium from the Quaternary deposits are seen.

Aside from organic matter, iron is potentially important for both sorption (for instance ferrihydrite) and colloidal transport of several radionuclides in oxidising environments (Dahlqvist et al. 2007). The natural landscape evolution in the site investigation areas with the cut-off of lakes and the subsequent formation of wetlands may, however, together with climate change, lead to previously oxidising environments getting waterlogged. This can in turn destabilise iron(III) precipitations and thereby mobilise elements that are bound to them (Ingri et al. 2018).

The waste in SFL and SFR also contains potential toxic pollutants. In 2016, SKB arranged a workshop with representatives from other waste organisations in order to get an overview of which substances are of concern, which methods can be used and which problems can arise in attempts to assess the effects of these toxic pollutants (Thorne and Kautsky 2016). SKB has also participated in BIOPROTA's programme and other workshops where toxic pollutants in repositories for radioactive waste are discussed (NRPA 2018).

Programme

In the plans for the continued sampling and monitoring programme for Forsmark, supplementary sampling of different elements is proposed (Berglund and Lindborg 2017). Within the operational programme, supplementary sampling of elements and radionuclides for which there are currently no site data is planned. This data will permit a more detailed description of sorption properties (K_d) utilising other softer leaching methods (than total content, see also Section 13.2) and taking the chemical environment in the discharge areas into account in a better way.

Chloride, which has previously been regarded as the dominant form of chlorine in nature, has proven to be more reactive than previously believed, and the quantity of organic chlorine is much higher in many environments than the quantity of mobile chloride (Section 13.2). SKB is therefore continuing an in-depth evaluation of the distribution pattern of chlorine, with the objective of relating the observed pattern to processes in the ecosystems. This means that the occurrence of inorganic and organic chlorine in water, sediments, benthic animals, plants, zooplankton and fish will be determined. Furthermore, an estimation of the cycling of organic chlorine in sediments and water is planned, and a study for describing the total flows of volatile organic chlorine compounds from different environments.

In the continued work on estimating uptake and transport processes for methane and carbon dioxide (Section 13.2), measurements will be carried out to determine concentrations of stable isotopes of these elements and thereby be able to separate transport from deeper soil layers and horizontal surface flows. The measurements are supplemented with new methods that can effectively survey flows of methane and carbon dioxide (e.g. with the aid of sensor networks and a hyperspectral camera for methane).

Within the operational programme, investigations are planned for the purpose of describing how the soil chemical environment in Forsmark will change over time, in order to provide by extension a picture of the mobility and immobilisation of various elements in a future landscape. This will be carried out by means of studies of representative ecosystems in the vicinity of Forsmark, but located at greater height above the current sea shoreline and thereby older than the ecosystems in Forsmark.

Continued studies in the Krycklan area are aimed at gaining a better understanding of leaching and enrichment of different elements in a near-stream zone. Iron colloids (together with DOC) are important for transporting more insoluble radionuclides. They affect mobility and bioavailability and thereby the uptake in biota. A study on how to use lanthanides as tracers for leaching has been initiated in Krycklan and will continue in the Forsmark area. Regarding uranium and its decay products, a more thorough survey of isotope data from the natural decay chains from the site-investigation areas will be carried out.

14 Climate and climate-related processes

Section 4.9 describes a number of general questions within SKB's climate programme that require further research, either because reasons have emerged in the comments from the regulatory authorities during the review of the submitted applications, or because SKB has judged it necessary in order to reduce uncertainties in ongoing and upcoming safety assessments. These remaining questions concern:

- i) Historical climate change – climate during the Weichselian and the Holocene.
- ii) The dynamics and behaviour of ice sheets.
- iii) Sea-level variations and shoreline displacement.
- iv) Age and long-term stability of the bedrock surface in Forsmark, including quantification of glacial erosion.
- v) Validation of the permafrost model.
- vi) Variability in climate and ice sheets over the coming one million years.
- vii) Description of ice sheet hydrology from the Greenland Analogue Project (GAP).

Most of these questions concern all three repositories, i.e. the Spent Fuel Repository, SFR and SFL. This chapter describes the current level of knowledge and ongoing or planned activities within the above areas more thoroughly.

14.1 Historical climate change

The current climate change results in elevated temperatures both globally and in Scandinavia as a result of anthropogenic emissions of greenhouse gases (IPCC 2013). Several studies have shown that anthropogenic influence on the climate in the form of elevated greenhouse gas concentrations and higher temperatures may be visible for tens to hundreds of thousands of years (Clark et al. 2016, Ganopolski et al. 2016, Lord et al. 2019). The exact timescale associated with anthropogenic climate change is difficult to assess since it is strongly linked to, among other things, future anthropogenic greenhouse gas emissions, and thereby how large the warming ultimately will be (e.g. Zickfeld and Herrington 2015). This is also highlighted in SKB's climate reports produced within the various safety assessments, which describe the expected climate evolution globally and at Forsmark for the coming 10 000–100 000 years under different emission scenarios, based on the latest scientific literature in the field.

Even though current emissions of greenhouse gases may affect the climate for up to hundreds of thousands of years, the climate in Forsmark is expected *at some time point* within the coming one million years, which is the relevant timescale for the safety assessments for the Spent Fuel Repository and SFL, to be dominated by natural climate variability in the form of repeated glacial cycles. SKB therefore uses a reference glacial cycle for describing the development of climate, ice sheets, permafrost, sea level, and denudation (erosion plus weathering) during a typical glacial cycle. In SKB's safety assessments, the effects of the reference glacial cycle on hydrogeology, geochemistry, geosphere, landscape evolution, repository barriers and, finally, the function and safety of the repository after closure are analysed. The reference glacial cycle also constitutes a scientific starting point for analysing the effect of other possible climate evolutions that could have a larger impact on the safety of the different repositories after closure. The reference glacial cycle consists of a future scenario where reconstructed conditions from the last glacial cycle (the Weichselian glaciation and the Holocene interglacial) are repeated for the coming 100 000 years. During the past RD&D period, SKB's methodology for handling climate was part of an international IAEA programme to address the issue of climate change and landscape evolution in assessments of long-term repository safety for disposal of radioactive waste (Lindborg et al. 2018).

SKB's future climate scenarios need to be well-founded and well-described to be able to form a basis for the assessment of the post-closure safety of the different repositories. For this purpose, SKB uses both historical climate archives and simulations with climate models and other computer models.

14.1.1 Climate during the Weichselian and the Holocene

Current situation

During the RD&D period, the work with reconstructions of the climate during different periods of the Weichselian, the Holocene and the last interglacial the Eemian, has continued through 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 could change during different phases of a glacial cycle, and quantification of variations in temperature, and some cases precipitation.

In addition to previous scientific papers presented in the RD&D Programme 2016, the studies of lake sediments from Sokli in northern Finland have resulted in seven scientific papers during the past RD&D period (Pliikk et al. 2016, 2019, Shala et al. 2017, Sánchez Goñi et al. 2017, Kylander et al. 2018, Helmens et al. 2018, Salonen et al. 2018), and in a large number of abstracts presented at international symposiums. A PhD project that studied the climate during the Eemian interglacial based on data from Sokli was completed during the same period and resulted in a PhD thesis at Stockholm University (Pliikk 2018).

The results from all completed studies at Sokli have been summarised and contextualised in Helmens (2019). Helmens (2019) provides a detailed reconstruction of the climate and environment of north-eastern Fennoscandia from multi-proxy data for i) the current Holocene interglacial (past 11 000 years), ii) warm and cold periods during the early Weichselian (Marine Isotope Stages (MIS) 5c-d for about 115 000 to 90 000 years ago) and iii) the preceding interglacial, the Eemian (the period MIS 5e for about 130 000 to 115 000 years ago). The results complement previous results from the same study, which provide corresponding information on the climate and environment of the mid-Weichselian (MIS3 for about 50 000 years ago) (Helmens 2009) and overall for the entire Weichselian (MIS 5-2, 130 000 to 15 000 years ago) (Helmens 2013).

By using a variety of proxy data (geochemical data, analysis of the amount of organic matter (Loss On Ignition), carbon-to-nitrogen ratio (C/N), pollen, diatoms, chironomids, and macrofossils) together with various quantitative methods for reconstructing climate, detailed reconstructions of environmental conditions (for instance sedimentation environment, azonal and zonal vegetation development) and climatological conditions (temperature and precipitation) have been made for the investigated time periods.

The results from Sokli partially revise previous reconstructions of historical environmental and climatological conditions during the past 130 000 years in northern Fennoscandia (see for example Helmens 2009, 2019).

The results of the studies above are used directly in the descriptions and justifications of the climate cases used in SKB's safety assessments, most recently in the work with the PSAR for the Spent Fuel Repository. The results will be used to provide a better picture of the different climates that may exist in the transition from a warm interglacial climate to cold glacial conditions, e.g. at the beginning of SKB's reference glacial cycle based on the Weichselian. The results of the fluctuations in temperature, precipitation and vegetation during these climate transitions can also be used in the field of surface ecosystems in SKB's safety assessments. The studies provide a better understanding of SKB's reference evolution and the methodology for handling climate and climate-related processes in a long-term perspective. These types of studies also contribute to improving the dating of terrestrial climate archives in Fennoscandia and Europe, especially when the results are compared with climate archives from other sites as made in Wohlfarth (2013), Helmens (2019), and Schenk and Wohlfarth (2019).

As a part of SKB's work with historical climate based on borehole temperatures, thermal and petrographic data from a bedrock borehole in Lake Vättern has been presented (Sundberg et al. 2016). The results have also given information on the glacial history and the quaternary geological and hydrological conditions in the Lake Vättern region (Preto et al. 2019). During the RD&D period, SKB has also concluded and published a study of historical climates based on measured rock temperatures in boreholes in Forsmark and Laxemar (Rath et al. 2019). The historical climates reconstructed from borehole temperatures can be used as independent data for assessing the realism in some of the air temperatures previously reconstructed for the Weichselian period, i.e. the temperature data that are linked to SKB's reference glacial cycle, including its uncertainty range.

Programme

The results from the study of lake sediments from Sokli in northern Finland have very 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 last glacial cycle, the Weichselian). Therefore, the study is planned to be complemented with a corresponding shorter study of a remaining warm interstadial during the Weichselian called the Odderade interstadial (MIS 5a for about 85 000 to 74 000 years ago). In this way, the reconstruction of warm and cold periods (interstadials and stadials) for the early Weichselian will become complete and will thus provide a complete description of the events when the climate most recently changed from warm interglacial conditions to glacial conditions. The results have so far shown that this took place over the course of a very long time and with large fluctuations between warm and cold periods (Helmens 2013, 2019).

14.1.2 Transitions between climate domains at the end of a glaciation

Current situation

The RD&D Programmes 2013 and 2016 described a planned and then ongoing study with the objective to investigate of transitions between climate domains (in contrast to the more static climate model studies previously conducted by SKB for the glacial, periglacial and temperate climate domains, see Kjellström et al. 2009). The study used both geological climate archives and climate modelling to provide information and examples of what the climate may be like within different climate domains and in the transitions between them. This study has now been completed and the results are reported in Schenk and Wohlfarth (2019). In addition to the previously published papers presented in the RD&D Programme 2016, the study has resulted in a number of scientific papers during the past RD&D period (Muschitiello et al. 2016, 2017, Wohlfarth et al. 2017, 2018, Ahmed et al. 2018, Schenk et al. 2018), and it has also been presented at international symposiums. The study also included a PhD project at Stockholm University, which was presented in 2016 (Muschitiello 2016). Since the final report of the study has now been produced, a summary of the main results is provided below.

The end of the most recent glaciation, the Weichselian, was characterised by a sequence of several rapid transitions between relatively cold periods (stadials) and relatively warm periods (interstadials). These rapid transitions indicate a large instability of the climate during the general warming trend of the last deglaciation. The fast evolution of these changes is in clear contrast to the gradual increase in insolation at high latitudes in summer and the global increase of greenhouse gases at the end of the glaciation, indicating that the climate system responds non-linearly to a general slow warming. This instability in the climate has been linked to large and sometimes abrupt variations in the strength of the North Atlantic heat transport.

In order to obtain more detailed information about and understanding of the climate evolution and variability in Scandinavia and Europe during the transition from a cold glacial climate to a warm interglacial climate, an extensive multi-proxy study was carried out on historical climate archives, combined with new simulations using a high-resolution, global climate model. The study focused on climate transitions during the last deglaciation, in particular the warm Bølling-Allerød interstadial and the cold Younger Dryas stadial.

The study also compiled published quantitative historical climate data, including published data from pollen and plant macrofossils from all over Europe. In addition, abrupt climatological and environmental changes in southern Sweden during the transition from glacial to interglacial conditions were quantified by a detailed analysis of two lake sediment cores from Attemosse and Hässeldala Port.

The multi-proxy data obtained from the two lake sediment cores provided information on the response of the terrestrial environment to rapid transitions in hydroclimate and were used to quantify the minimum mean July air temperature and the mean July surface water temperature during pre-Bølling, Bølling-Allerød, Younger Dryas and the early Holocene. Based on information from several Swedish lakes, it was concluded from the analysis of plant macrofossils that remarkably high summer temperatures of at least +16 °C prevailed from the time of deglaciation to the early Holocene (approximately 10 000 years ago).

In contrast to the high air temperature during summer periods obtained from plant macrofossils, the lake-water temperature reconstructed through chironomids shows distinct shifts between cold stadials and warm interstadials. These shifts co-vary with proxy data from other biomarkers, which indicate dry conditions and precipitation from a less saline source during stadials, and humid conditions and a more saline source of precipitation during interstadials.

The new high-resolution climate simulation, which took into account the presence of ice sheets, glacio-isostatic vertical movements of the Earth's crust and a low sea level during glacial conditions, surprisingly indicated that summer temperatures during the Younger Dryas stadial, which has traditionally been regarded as a cold period, remained at least as high as during the preceding warm Allerød interstadial. This result is also consistent with the historical climate data from southern Sweden and from the other 120 sites in Europe. The sustained high summer temperatures during all phases of the deglaciation are a result of an atmospheric blocking of westerly winds over the Fennoscandian ice sheet during the summer periods. The atmospheric blocking, which was caused by the presence of the ice sheet, was intensified by the low surface temperatures in the North Atlantic during the Younger Dryas stadial. The resulting warm summer climate explains why heat-demanding plants continued to thrive in Scandinavia during the Younger Dryas. The situation with atmospheric blocking and the associated high temperatures is however only stable during a short summer period, as colder conditions dominate in the spring, autumn and winter periods during the Younger Dryas. The results from the climate simulation therefore support findings in previous studies, which have indicated that abrupt transitions in climate at high latitudes are dominated by changes in the seasonal variations, with an increased continentality of the climate during stadial periods.

The surprising results with warm summers even during the “cold” periods of the deglaciation, and other results, are presented in detail in Schenk et al. (2018) and Schenk and Wohlfarth (2019). The results will be used by SKB to provide a considerably more nuanced picture of the climate changes associated with climate transition from glacial to interglacial conditions, as in SKB's reference glacial cycle. The results may also be used in the biosphere programme, where landscape and vegetation development during the deglaciation are described. Like the studies described in Section 14.1.1, this study contributes to improving the dating of terrestrial climate archives in Fennoscandia and Europe.

Programme

After the Schenk and Wohlfarth (2019) report, no further detailed study is planned of climate transitions in connection with the transition from cold glacial climates to warm interglacial conditions. However, transitions between climate domains are indirectly included in several other studies described in this chapter, for example in the study of the variability of climate and ice sheets in a one million-year perspective (Section 14.6) and the study of ice sheet dynamics and behaviour (Section 14.2).

14.2 Ice sheet dynamics and behaviour

Current situation

Within the framework of the most recent RD&D programme, two dedicated modelling studies were carried out with the purpose of estimating the maximum ice thickness over Forsmark, and thereby also the maximum hydrostatic pressure at repository depth, which could affect the Spent Fuel Repository during a future glacial maximum. These studies modelled the glacial maximum during the penultimate glacial cycle (the Saalian glaciation), which is the period with the largest ice sheet in Northern Europe according to geological and geomorphological data. The glacial maximum of the Saalian glaciation, which was simulated here, occurred during the Marine Isotope Stage 6 about 140 000 years ago. The first study (Colleoni et al. 2014) investigated the change in ice thickness over Forsmark as response to i) different climates over Eurasia based on two climate model simulations, and ii) systematically varying, one after the other (univariate analysis), parameters in the ice sheet model that affect the simulated ice thickness. The analyses of ice sheet thickness were based on steady-state simulations, i.e. the climate was held constant-in-time and the ice sheet was allowed to grow to the maximum size given that climate.

In the follow-up study (Quiquet et al. 2016), the parameter analysis was performed in more detail by varying several parameters of the ice sheet model simultaneously (multivariate analysis) in order to investigate how the influence of these different parameter combinations affected the simulated steady-state ice thickness. The combined results of both studies led to the conclusion that the ice sheet over Forsmark can be at most 4000 metres thick (Figure 14-1), which can be compared with the assumption in SR-Site where 3400 metres was postulated. The results of the two studies have also been published in scientific journals (Colleoni et al. 2016, Wekerle et al. 2016).

The studies of the Saalian glaciation described above are an important complement to the ice sheet reconstruction of the last glacial cycle (the Weichselian glaciation) that is included in SKB's reference glaciation.

Programme

In an ongoing study, complementary ice sheet simulations of the entire Saalian glaciation are carried out, i.e. not only of the glacial maximum as in the two studies described above. The simulated transient ice sheet evolution in this study, together with several other ice sheet reconstructions, will be used as a boundary condition for calculations of the stress field in the Earth's crust during different ice load histories, see Section 12.2.3 and Lund et al. (2019).

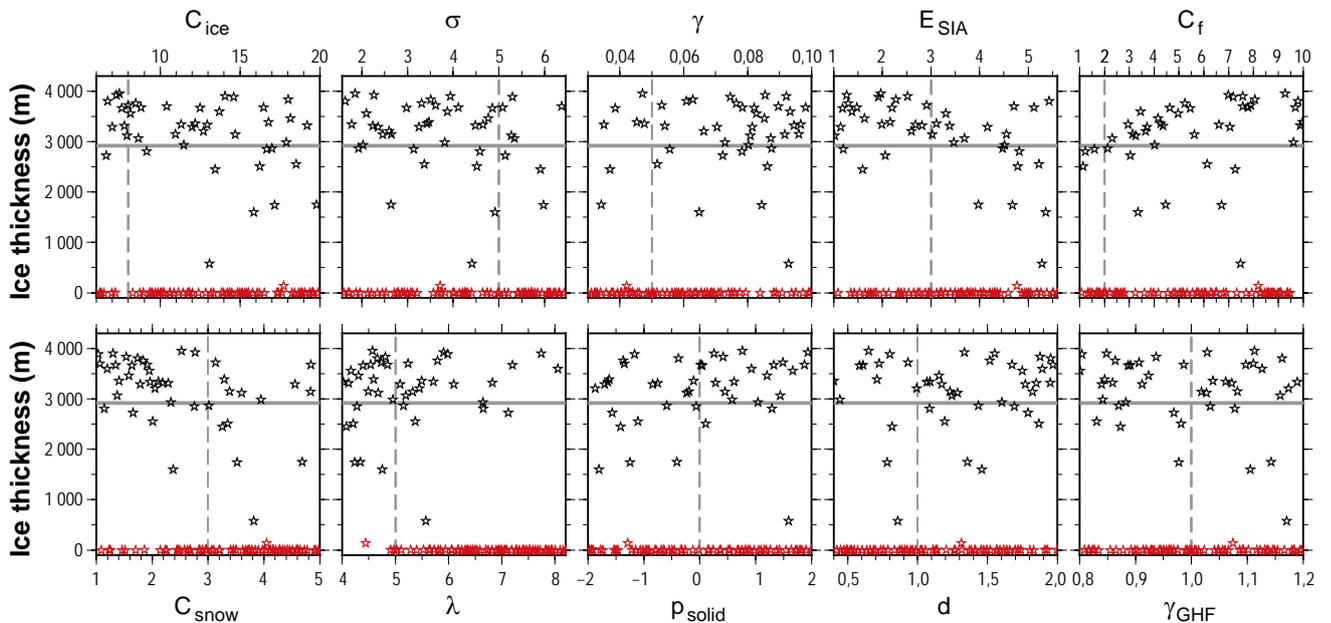


Figure 14-1. Simulated ice sheet thickness over Forsmark during the glacial maximum of the Saalian glaciation (about 140000 years ago) as a function of different parameter combinations in the ice sheet model (from Quiquet et al. 2016). The black stars represent individual steady-state simulations that resulted in ice thicknesses over Forsmark of at least 500 metres, while the red stars represent simulations that yielded no ice sheet or ice sheets thinner than 500 metres. The horizontal lines show the simulated ice thickness according to a reference run (Colleoni et al. 2014), while the vertical lines show the value of each parameter in this reference run. The investigated parameters are: C_{ice} = melting coefficient for ice (unit: $\text{mm } ^\circ\text{C day}^{-1}$), C_{snow} = melting coefficient for snow ($\text{mm } ^\circ\text{C day}^{-1}$), σ = standard deviation of the daily mean air temperature ($^\circ\text{C}$), λ = rate of air temperature decrease with elevation ($^\circ\text{C km}^{-1}$), γ = coefficient controlling the precipitation change with temperature ($^\circ\text{C}^{-1}$), p_{solid} = air temperature at which precipitation changes from snow to rain ($^\circ\text{C}$), E_{sia} = enhancement factor of ice flow (unitless), d = coefficient controlling the amount of refreezing of meltwater (m), C_f = coefficient controlling the ice flow's friction against the ground surface in areas with fast ice flows (unitless) and γ_{GHF} = factor controlling the geothermal heat flow from the ground surface (unitless). For a detailed description of these parameters, see Quiquet et al. (2016). The results show that the ice thickness over Forsmark reaches at most 4000 metres.

SKB plans to conduct a review of published paleo-reconstructions of ice sheets for Fennoscandia to evaluate the ice sheet reconstructions for the Weichselian and Saalian glaciations used in SKB's safety assessments. This study is planned to form a basis for an assessment of the effect of relatively fast climate changes (on a timescale of about 1 000 years, observed e.g. on Greenland during the Weichselian period, see for example Dansgaard et al. 1993) on the ice sheet margin's location and variability in Fennoscandia.

This study on ice sheet variability is also linked to the analysis of glacial erosion (Section 14.4). The results from the ongoing glacial erosion studies, together with estimates of the number of advances and retreats of the ice sheet margin over Forsmark, can be used to better estimate the extent of glacial erosion during the coming million years.

14.3 Sea-level variations and shoreline displacement

Current situation

The current level of knowledge concerning sea-level variations of relevance for Forsmark is described in the climate reports for the PSAR for the Spent Fuel Repository and SFR (the reports are planned to be published in 2020). The reports compile data on possible relative sea-level variations, including their uncertainties, for the period up until the year 2100 and on time scales of 10 000 and 100 000 years. The information on future sea-level variations is highly relevant both in SKB's safety assessments and for the construction and operation of the Spent Fuel Repository and the operation of SFR until closure.

The relative sea level at Forsmark, and its variation over time, is determined by the net result of isostatic changes (crustal movement, which is dominated by rebound following the latest glaciation in Forsmark) and eustatic sea-level variations (changes in the volume and distribution of sea water in the world's oceans). The present-day isostatic uplift at Forsmark is 6.7 mm/year (Vestøl et al. 2019). The isostatic uplift is expected to be relatively unchanged until year 2100 and will therefore compensate for a considerable portion of the eustatic sea-level rise.

In SKB's work, sea-level variations until the year 2100 AD are divided into: i) slow, persistent, and (in a relatively near future) non-reversible processes; and ii) fast, short-duration and reversible processes. The former consists of the rise of the mean sea level (globally and/or locally) due to for instance the melting of ice sheets and glaciers and the thermal expansion of sea water, and the latter consists of the temporary sea-level rise that occurs during storm surges.

There are large uncertainties related to the extent of future sea-level variations, both related to the slow, persistent rise of mean sea level and the fast, short-duration variations during storms. Uncertainties in the evolution of the mean sea level are related to the amount of global warming and how current ice sheets will react to this warming. To take these uncertainties into account, probabilistic models are often used to estimate how much sea level will change in the future. These models take uncertainties into account by estimating probabilities of the projected sea-level rise given a certain probability distribution and climate scenario. Uncertainties related to future variations in sea level during storms are primarily linked to how low-pressure areas and storms in the Baltic Sea and the Bothnian Sea will behave in the future, but also to the frequency distribution of the storms.

Figure 14-2 summarises the estimated total change in the relative sea level in Forsmark until 2100 AD and the individual contributions to total change from storms, mean sea-level rise, and isostasy. The figure shows data from different sources, among others, estimates of the global mean sea-level rise from the IPCC (2013) and from Grinsted et al. (2015), who used a probabilistic model to investigate changes in the relative sea level in northern Europe for the case of strong global warming (IPCC emissions scenario RCP8.5). The Grinsted study was used as the *worst case* for the relative sea level change until 2080–2100 AD. Figure 14-2 presents the maximum relative sea-level rise in Forsmark within the confidence ranges 5–95 percent, 1–99 percent and 0.1–99.9 percent under the RCP8.5 emissions scenario. The total sea-level rise in Forsmark was then calculated by adding the relative sea-level rise from Grinsted et al. (2015) to the contribution from storms calculated with a recurrence time of 100 years (Meier 2006) and an expected increase of the wind speed at Forsmark by the year 2100 (Nerheim 2008). The total, pessimistically calculated, maximum sea level in Forsmark in the

year 2100 in this scenario is estimated to be between +2.8 and +3.9 metres during heavy storms, depending on the probability level for the rise of the mean sea level (Figure 14-2). It is important to note that wave height was not included in this analysis.

In relation to the construction and operation of the repositories in Forsmark, it is very important to note that the slow, persistent rise of the mean sea level, which has the largest uncertainty, is observable over time. This means that during the coming decades it will be possible to monitor and observe how the sea level develops, and thus also to take further measures beyond those planned today, if necessary.

In addition to the analysis of the possible relative sea level rise in the short term (until 2100 AD), a thorough analysis has been carried out on the potential relative sea level change globally and at Forsmark during the coming 10000 years as a result of different scenarios of global warming. The results of this analysis for Forsmark are exemplified in Figure 14-3 for the climate scenario RCP8.5 (high emission scenario). The green curve shows the estimated isostatic change in Forsmark based on GIA (Glacial Isostatic Adjustment) modelling (SKB 2010a), while the blue curves show the development of the global sea level over the coming 10000 years based on the studies Clark et al. (2016) and Levermann et al. (2013). These studies constitute the maximum and minimum published sea-level rise projections on longer timescales for the RCP8.5 climate scenario, according to the present level of knowledge, and have therefore been used to estimate the uncertainty related to the relative sea-level evolution.

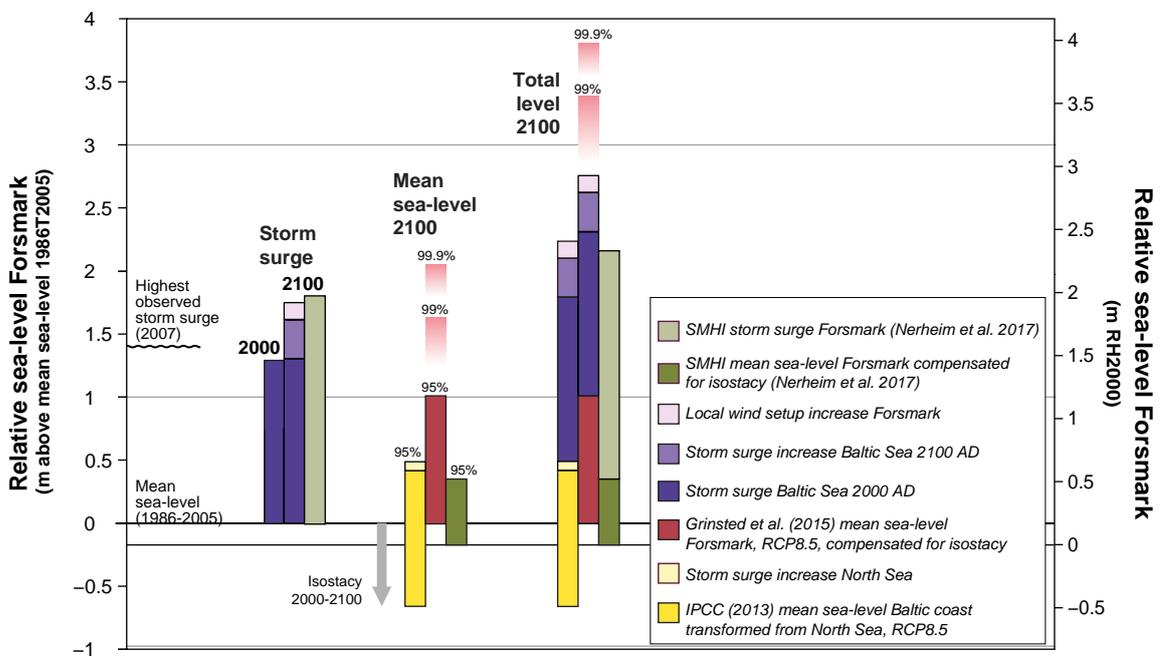


Figure 14-2. Estimated maximum relative sea level at Forsmark by 2100 AD. The leftmost group of bars show the highest sea level caused by fast, temporary processes during storms for the years 2000 and 2100 AD. The bars in the middle show the local rise in mean sea level by 2100 AD from three different studies, compensated for isostatic uplift, caused by slow, persistent processes. The rightmost bars show the total maximum sea-level rise in Forsmark in the year 2100 taking into account both the rise of the mean sea level and the temporary rise during storms. Note that isostatic uplift compensates for about 0.67 metres of the sea-level rise from 2000 to 2100 AD. The sea level is presented in metres above the current mean level (defined as 1986–2005 reference period) (left y-axis) and in the RH2000 reference height system (right y-axis).

The change in relative sea level at Forsmark during the next 10000 years, based on Clark et al. (2016) and Levermann et al. (2013) combined with the isostatic change, is illustrated by the red curves in Figure 14-3. As a result of the large uncertainty in global sea-level rise on these time scales, the relative sea level at Forsmark during the next 10000 years may either be subject to a continued regression, where the ground continues to rise from the sea like today (lower red line) or a transgression, where the sea level initially rises culminating in a maximum relative sea level of up to 30 metres around the year 4500 (upper red line), after which the relative sea level slowly declines again. According to this sea-level projection, the conditions with a relative sea level higher than present-day might persist up to around 15700 AD (outside the graph). In view of this great uncertainty in future sea levels, the two cases are treated as two possible variants in SKB's climate cases. A corresponding analysis has also been made for the IPCC climate scenario RCP4.5 (medium emission scenario).

Programme

SKB will continue to monitor the research area concerning future sea-level variations. An update of the level of knowledge concerning the mean sea-level rise in the near future (until 2100 AD, Figure 14-2) is planned to be carried out during the RD&D period. Furthermore, a study is planned with the purpose of calculating the contribution to the sea-level rise from storms with recurrence times of up to 10000 years or more, i.e. much longer recurrence times than the 100 years that are used in Figure 14-2. The study is also planned to include a probabilistic estimate of how much the total sea level (the rise of the mean sea level plus the rise of storm surges) will rise at Forsmark until 2100 AD for the climate scenarios RCP2.6 (low emissions), RCP4.5 (medium emissions) and RCP8.5 (high emissions). The results of this study are expected to provide a more detailed picture of how the maximum sea level during heavy storms, under pessimistic assumptions, will change up to the years 2050, 2080 and 2100.

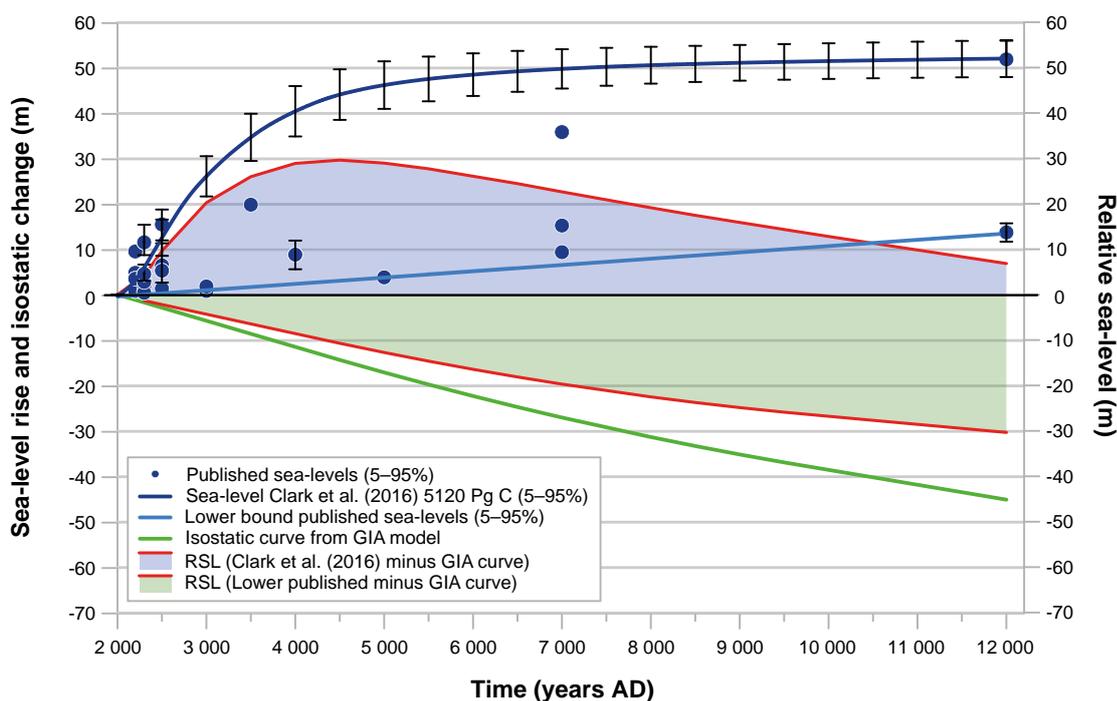


Figure 14-3. The change in relative sea level at Forsmark during the next 10000 years for the IPCC emission scenario RCP8.5 (strong global warming). The blue dots show projections of the global mean sea level from different studies in the scientific literature published up until the middle of 2018. The blue curves show the evolution of the global mean sea level during the coming 10000 years based on Clark et al. (2016) (upper blue curve) and Levermann et al. (2013) (lower blue curve). These two studies constitute the maximum and minimum projection of global sea-level rise during the coming 10000 years according to present knowledge. The green curve shows the isostatic evolution in Forsmark calculated from GIA modelling (SKB 2010a). The red curves show the resulting change of the relative sea level in Forsmark, i.e. the sum of the blue curves and the green curve. The very large difference between the two relative sea-level evolutions, illustrated by the two red curves, clearly demonstrates the great uncertainty that is associated with the future sea level in Forsmark under strong global warming until 12000 AD.

Furthermore, an update is planned of the most recent information from the scientific literature concerning sea-level changes over the perspective of several thousands of years (Figure 14-3), including the highest level/longest period with submerged conditions that may be caused by strong global warming. This affects, among other things, the future landscape evolution in Forsmark, which is included as an important part of SKB's safety assessments for the Spent Fuel Repository and SFR.

14.4 Age and long-term stability of the bedrock surface in Forsmark, including quantification of glacial erosion

Current situation

Previous studies have shown that the extent of glacial erosion and denudation in the Forsmark area, generally, has been limited in the past, and that it will continue to be so also in the future (Olvmo 2010). A primary reason for this conclusion is the very flat bedrock topography in the area. The GIS study in Olvmo (2010) was carried out at a relatively coarse spatial resolution, which means that the results are not necessarily applicable on a detailed scale. Furthermore, the study used only one method, even though new methods have been developed in recent years and new data sets have been collected, which can be used for an improved quantification of the extent of glacial erosion that has occurred to date. In the comments on previous RD&D programmes, SSM writes, among other things: *In order to support the current assessment of freezing at the Spent Fuel Repository depth, SSM judges it urgent that SKB carries out the planned studies to clarify the uncertainty in the estimated values of glacial erosion and the evaluation of the permafrost model reliability.* SSM's review of SR-Site also pointed out the need for further investigation and justification of SKB's view that future denudation and glacial erosion in Forsmark will be limited.

In view of the above points, an extensive study was initiated in 2015 concerning the age and stability of the bedrock surface, including the depth and rate of previous glacial erosion, in the Forsmark area and Uppland. Examples of questions included in the study are: i) How extensive has the denudation (erosion plus weathering), and specifically the glacial erosion, been in Forsmark during all Late Quaternary glacial periods, and during the entire period since the Sub-Cambrian peneplain was formed, when the area is studied in detail? ii) Which glacial erosion processes have been active in the landscape to date in Forsmark? iii) At which rate could the documented area with more glacial erosion 20–30 km southeast of Forsmark (see Olvmo 2010) potentially extend towards the site for the Spent Fuel Repository? and iv) What extent of glacial erosion can be expected during the coming 100 000 and one million years in Forsmark and at the site for the Spent Fuel Repository?

The study has now been completed and is published in Hall et al. (2019). Results from the study have also been presented continuously at scientific symposiums (Moon et al. 2017, 2018, Hall et al. 2018, Heyman et al. 2018, 2019, Krabbendam et al. 2019).

Four different methods were used, each linked to a sub-study. i) geomorphological analysis of the bedrock: using the Sub-Cambrian peneplain as a reference surface against which estimations of the extent of glacial erosion during the Pleistocene are made; ii) mapping and geomorphological analysis of the distribution and properties of glacial land forms, primarily in Quaternary deposits, to understand spatial patterns and active processes of glacial erosion; iii) fracture analysis: studies of bedrock fractures, including fracture mapping and modelling of the stress field in the upper part of the bedrock (linked to the fracture contribution to the bedrock's potential erodibility); and iv) cosmogenic exposure dating for estimation of the age of the bedrock surface and quantification of glacial erosion rates during previous glaciations.

The results from these studies have provided a detailed picture of the extent of the total glacial erosion that has occurred in Forsmark to date. When these results are combined with the simulated periods with an ice sheet over Forsmark during the coming one million years (from the study reported in Figure 14-5, Section 14.6), the total glacial erosion during the coming one million years can be preliminarily estimated to between a few and several tens of metres, depending on the position in the landscape. For detailed descriptions of the extensive results of the study, see Hall et al. (2019).

Programme

As mentioned above, southeast of Forsmark there is an area with observed stronger glacial erosion. The sub-project that studied the potential lateral growth rate of this area was not carried out fully within the above project. This part is therefore planned to be carried out and finalised during this RD&D period. The study aims to estimate how quickly this relatively strongly glacially affected area potentially could extend towards the repository area during future periods with glacial conditions in Forsmark (for example according to Figure 14-5). The study encompasses detailed mapping and analysis of bedrock forms and sediments, some remote sensing with field checks, and possibly cosmogenic dating.

Complementary studies on the identified glacial erosion process “glacial ripping” are also planned to be carried out during the RD&D period.

The results from these studies are expected to contribute to a scientifically more substantiated estimate of possible changes at the repository depth of the Spent Fuel Repository during the next one million years and to better evidence for the current assessment of freezing at the Spent Fuel Repository depth. In addition, the results will include information on the fracturing of the near-surface rock, which is planned to be used in the areas of rock mechanics and hydrogeology (Section 12.1.4).

14.5 Validation of the permafrost model

Current situation

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. The occurrence of permafrost greatly affects the flow pattern of the groundwater in the geosphere, both at depth and near the surface. Groundwater composition may also be affected by salt exclusion.

The RD&D Programmes 2013 and 2016 presented plans and ongoing studies for validating the permafrost model that was used in the safety assessments for the Spent Fuel Repository (Hartikainen et al. 2010, SKB 2010a) and SFR (Brandefelt et al. 2013).

This model validation study proceeded during the RD&D period. The study has been presented at an international symposium (Hartikainen et al. 2018). The permafrost simulations performed within the framework of SKB’s safety assessment work, and the corresponding work for Posiva, have also been presented in a PhD thesis at the Aalto university in Finland during the RD&D period (Hartikainen 2018).

Programme

The validation of the permafrost model will continue and is planned to be concluded during the RD&D period. The work includes verification of current and former versions of the permafrost model by applying them on the GAP study area in western Greenland (Section 14.7), where the permafrost depth and the rock temperature are partially known.

Different repository concepts, with different types of barriers, have different sensitivities to degradation by freezing. For example, concrete barriers suffer larger physical degradation than bentonite barriers if they were to freeze. Moreover, different repository depths provide different preconditions for freezing, as freezing always starts at the surface. The proposed repository concept for SFL (Section 2.1.1) includes placing the repository at a sufficient depth for avoiding negative impacts of freezing on the engineered barriers. In a future complete assessment of the safety after closure for SFL, the maximum depth for freezing during periods of cold climate must therefore be evaluated for the site selected for SFL.

14.6 Variability in climate and ice sheets over the coming one million years

Current situation

SKB's climate studies for the Spent Fuel Repository and SFL has so far almost exclusively focused on historical climate and the first 120 000 years of the future. On the one hand, the reference glacial cycle has been constructed (Section 14.1), based on the most recent glacial period (the Weichselian glaciation and the Holocene), and on the other hand, a number of complementary climate cases have been defined, some of which are based on the reference glacial cycle, describing alternative climate evolutions for this time period. Two examples of complementary climate cases are the *global warming* and *extended global warming* climate cases, for which it is assumed that the current interglacial period of temperate climate is extended by 50 000 and 100 000 years, respectively, due to the ongoing process of global warming and natural variations in insolation. Other examples of complementary climate cases are the *extended ice-sheet duration climate case* and *maximum ice-sheet thickness climate case*, where the ice sheet in the first case covers the repository for a longer time than in the reference glacial cycle, while the ice sheet in the latter case is much thicker than in the reference glacial cycle, resulting in maximum isostatic pressures (see Section 14.2).

The remaining period after the first 120 000 years up to one million years has been covered in the safety assessments by assuming a repetition of the reference glacial cycle. Although this methodology, complemented by climate cases with extended ice-sheet duration and maximum ice-sheet thickness, covers the extremes in parameters of importance for repository safety related to glaciation, and thereby is sufficient from a safety assessment perspective, it yields a simplified picture of the climate variability that can be expected in a one million-year perspective. This is illustrated in Figure 14-4, which shows the relative variation of ^{18}O from ocean sediments, which in turn gives an estimate of how the global ice volume has varied during the last one million years. It is evident from Figure 14-4 that there is a significant variability in the length of different glacial periods, the regularity with which they occur (including how often they occur), and the size (volume) of the ice sheets. The variability in the length of the glacial periods and the size of the ice sheets is to some extent covered by the climate cases used in the safety assessments, while the actual future variability related to when and how often the glacial periods may occur is not taken into account with the current methodology.

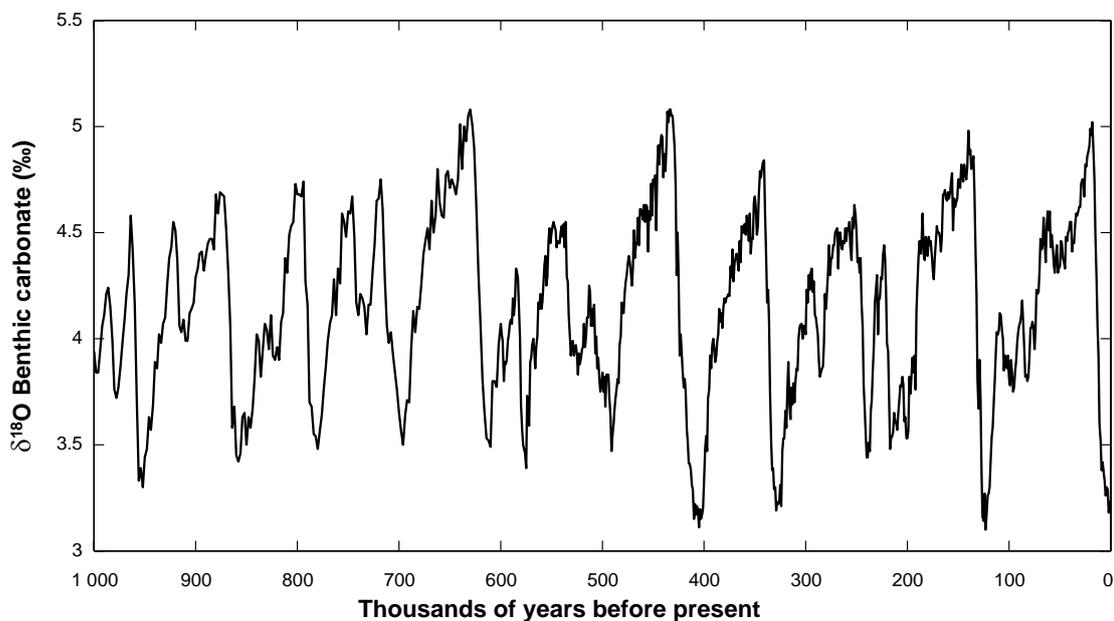


Figure 14-4. Variations of $\delta^{18}\text{O}$ from deep-ocean sediments for the last one million years (Lisiecki and Raymo 2005). High values of $\delta^{18}\text{O}$ indicate a relatively large global ice volume on the continents and low temperatures in the deep ocean, while low values of $\delta^{18}\text{O}$ indicate a relatively small global ice volume and high ocean temperatures.

In the RD&D programme 2016, a planned joint project with Posiva was described that was aimed at investigating the climate (air temperature and precipitation) and the occurrence of ice covered periods over the repository sites at Forsmark and Olkiluoto during the coming one million years. The study is now finished and has been published in Lord et al. (2019). The study has also been presented at two international symposiums (Lord et al. 2018a, b).

The results show that with moderate global warming (corresponding to the IPCC's climate scenario RCP4.5 and SKB's *global warming climate case*), the next phase of glaciation in the Forsmark area will occur in about 150 000 years, whereas with strong global warming (corresponding to IPCC's climate scenario RCP8.5 and the *extended global warming climate case*), it will occur in about 400 000 years, see Figure 14-5. Note that the glacial cycle, and glacial conditions, will have started earlier at other locations, before the ice sheet reaches Forsmark.

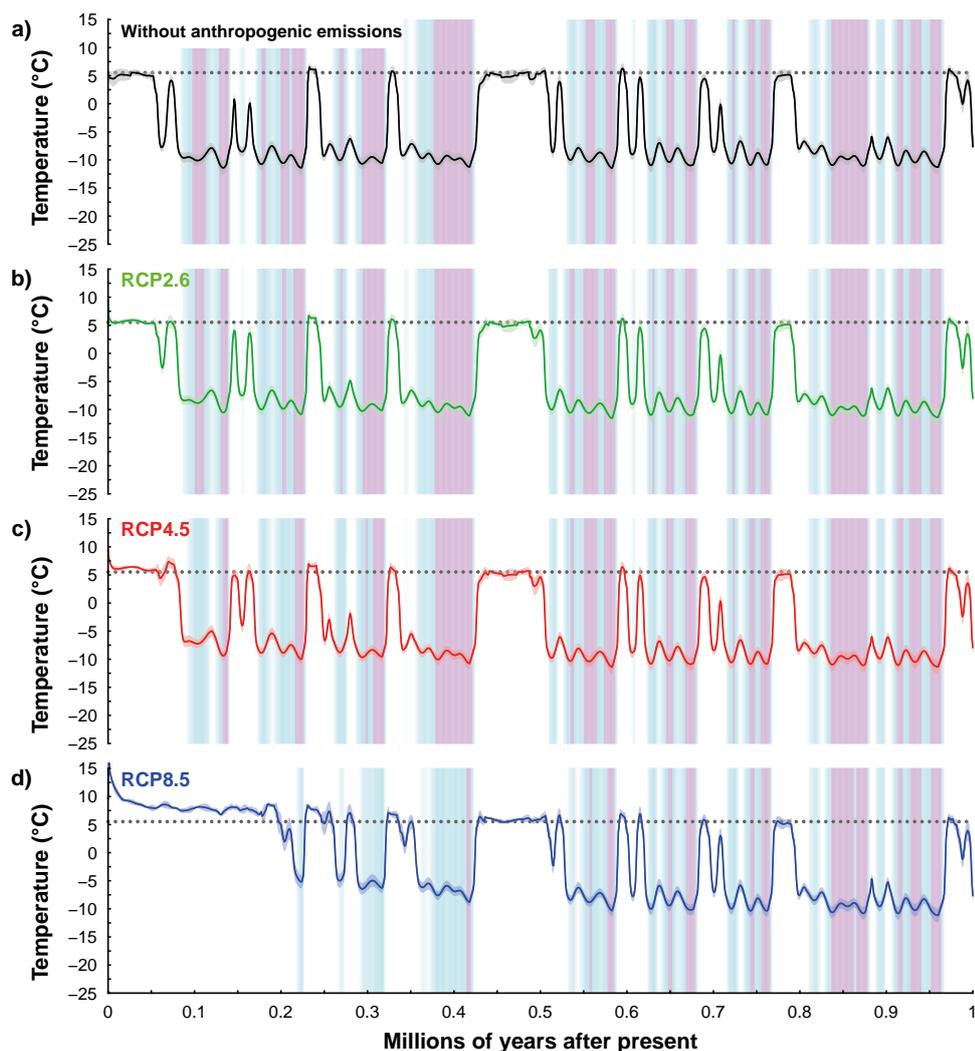


Figure 14-5. Annual mean surface air temperature ($^{\circ}\text{C}$) for the coming one million years at Forsmark for (a; black line) a case with no anthropogenic greenhouse gases emissions, (b; green line) IPCC's emission scenario RCP2.6 (low emissions), (c; red line) emission scenario RCP4.5 (medium emissions) and (d; blue line) emission scenario RCP8.5 (high emissions). For periods of ice-covered conditions at Forsmark, the surface air temperature is computed as if the ice sheet was not present. The air temperature has been modelled for every one thousandth year. The narrow coloured uncertainty band around the air temperature curves shows the uncertainty (one standard deviation). There is also corresponding information for precipitation (not shown here). The vertical bands show an estimate of periods when the Fennoscandian ice sheet may cover Forsmark. Purple bands show periods with high confidence of ice-covered conditions, while cyan-coloured periods have a lower confidence, see Lord et al. (2019). A more intense cyan colour indicates higher confidence. Periods without coloured vertical bands indicate ice-free conditions at Forsmark. The model has been evaluated by simulating the preceding 800 000 years including the latest glacial cycle and comparing the result with independent historical climate data for these periods (see Lord et al. 2019). The dotted grey lines show present-day temperature in the Forsmark area.

The results hence show that the air temperature and precipitation at Forsmark may be strongly affected by current anthropogenic emissions of greenhouse gases for several hundred thousand years. After that, the climate will again be dominated by natural variability driven by variations in insolation caused by variations in the Earth's orbit around the sun. The latter half of the one million-year period in Lord et al. (2019) is characterised by repeated glaciations in Forsmark (Figure 14-5), similar to the Weichselian reconstruction, with one or two periods of ice-covered conditions per glacial cycle.

The simulated future periods with ice sheet over Forsmark from Lord et al. (2019) have been used for quantification of future glacial erosion together with results from the study of historical glacial erosion in Forsmark (Section 14.4).

SKB has also participated in a study where geothermal data for Finland and Sweden were combined and evaluated (Veikolainen et al. 2019). The Swedish data were produced earlier within the context of SKB's climate work (Näslund et al. 2005). The data set will be used as input data to future ice sheet simulations, described below.

Programme

The study by Lord et al. (2019) will play an important role in illustrating and investigating the effect of more realistic climate variability in SKB's studies on the timescale of one million years.

The simulated climate and the future periods of glaciation at Forsmark during the coming one million years from Lord et al. (2019) will be used in a complementary climate case, which in a first step can be used to evaluate the realism in the climate cases that were previously included in SKB's safety assessments and safety evaluations for the Spent Fuel Repository, SFR, and SFL. Depending on the results of this evaluation, the simulated air temperature and precipitation from Lord et al. (2019) may subsequently be used as forcing in a dynamic 3D ice sheet model integrated one million years into the future. The result from such a simulation is expected to provide, among other things, an indication of the effect of more realistic climate variability on ice sheet extent, and on the possibility of an ice sheet passing Forsmark on several occasions within the same glacial cycle (Section 14.2).

14.7 Description of ice sheet hydrology from the Greenland Analogue Project (GAP)

Current situation

The Greenland Analogue Project (GAP) was carried out between 2008 and 2013. The results are analysed and published continuously. The two final reports from GAP (Claesson Liljedahl et al. 2016, Harper et al. 2016) were published in 2016, at the same time, in 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. At the time of publication of the final reports, the GAP had generated 43 scientific papers published in international peer-reviewed journals (see compilation in Appendix A in Claesson Liljedahl et al. 2016), 19 reports in SKB's, Posiva's and NWMO's report series (Section 1.3 in Claesson Liljedahl et al. 2016), and a large number of contributions to symposiums. For references to these publications, the reader is referred to the reports above. Additional reports and scientific papers and contributions to symposiums have been published during the period 2016–2019 (for example Drake et al. 2017, Puigdomenech et al. 2017, Vidstrand 2017, Hasholt et al. 2018, Henkemans et al. 2018, Ruskeeniemi et al. 2018, van As et al. 2018). At present, the GAP project has generated more than 50 scientific publications, 8 PhD theses and 20 reports.

The GAP was initiated to gain a better understanding of how climate change, and particularly glaciations, may affect the long-term function of a final repository and the development of its environment. The GAP has resulted in a developed and refined understanding of a number of different processes that are of great importance in assessments of long-term repository safety, primarily those concerning boundary conditions for groundwater modelling, bentonite stability and landscape evolution.

The GAP project has contributed to an improved process understanding within four areas. One area concerns transient melting processes and where and how groundwater is formed beneath an ice sheet. A new conceptual model of ice sheet hydrology, with four distinct areas with different hydrological properties, has been developed (Figure 14-6). Field observations and modelling results suggest, among other things, that more than 75 percent of the studied part of the Greenland ice sheet is warm-based, with presence of basal water. The central part of the ice sheet is cold-based. For a detailed description of the hydrological properties of the different zones in the conceptual model, to be used as boundary conditions in future hydrogeological simulations of glaciation, see Figure 14-6 and, above all, Section 5.1 in Claesson Liljedahl et al. (2016).

Another area of improved process understanding is related to the temporal and spatial variations in water pressure at the base of the ice sheet. Using pressure and thermal data from 23 boreholes drilled through the ice sheet, the conceptual understanding of the basal groundwater drainage system has been revised (Claesson Liljedahl et al. 2016). The hydraulic measurements and analyses from the boreholes indicate that the ice overburden pressure (i.e. a water column corresponding to 92 percent of the ice thickness) is a good approximation of the basal hydraulic pressure over a large part of the ice sheet and for most of the year.

GAP has also contributed to improved process understanding of the potential depths that glacial meltwater can penetrate into the bedrock, and on the chemical composition of this water. The stable water isotopic signatures ($\delta^2\text{H}$ and $\delta^{18}\text{O}$) show that groundwater from two of the bedrock boreholes in front of the ice sheet originates from glacial meltwater. Penetration of glacial meltwater has probably been facilitated by the predominantly glacial conditions in the GAP area for many millions of years, by the local geology and fracture distribution, and by the presence of high hydraulic gradients. The relatively low concentrations of sodium and chlorine in the groundwater are probably a consequence of water/rock interactions and diffusion, whereas calcium and sulphate in these waters are a result of the dissolution of the mineral gypsum, which occurs as a fracture filling mineral at depths greater than 300 metres.

Measured helium concentrations in the water indicate residence times exceeding hundreds of thousands of years. This, together with the extensive occurrence of the highly soluble mineral gypsum at depths greater than 300 metres, indicates stable conditions in the deep borehole at the ice sheet margin, with limited groundwater flow beneath the permafrost. Below the permafrost, reducing conditions prevail.

Permafrost and taliks (unfrozen parts in an otherwise frozen area) have also been studied within the project. The study area is located in an area with continuous permafrost, which at the ice sheet margin reaches a thickness of 350–400 metres. It is likely that permafrost does not exist under most of the large warm-based parts of the ice sheet. An exception is the ice sheet margin, where a wedge of permafrost probably extends in under the ice. Unfrozen taliks that extend through the entire permafrost thickness, and thereby permit exchange of deep groundwater and surface water, are commonly occurring in the area. The project has, for the first time ever, observed and confirmed the presence of such a talik, located underneath the lake next to which one of the GAP bedrock boreholes was drilled. Sampling of this borehole has further resulted in the first information on groundwater composition and hydrological data from a talik close to an ice sheet.

For more detailed information on the above results, see Claesson Liljedahl et al. (2016).

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 on Greenland. ICE was initiated in 2014 and completed in 2016, which generated field data from three seasons. The final report for ICE was published in 2019 (Harper et al. 2019). The project entailed that a block (700×700×700 metres) of the ice sheet was studied in detail in the same area as where GAP was performed. In this block, nine boreholes were hot-water drilled to the base of the ice sheet and an extensive data set was generated from sensors in these boreholes during three years (2014–2016). Data from this block were analysed and interpreted in relation to the approx. 50 km long data transect studied in GAP, aiming at gaining a better understanding of the processes that were studied within ICE (Harper et al. 2019). The project has so far resulted in five published articles, which are summarised in Harper et al. (2019).

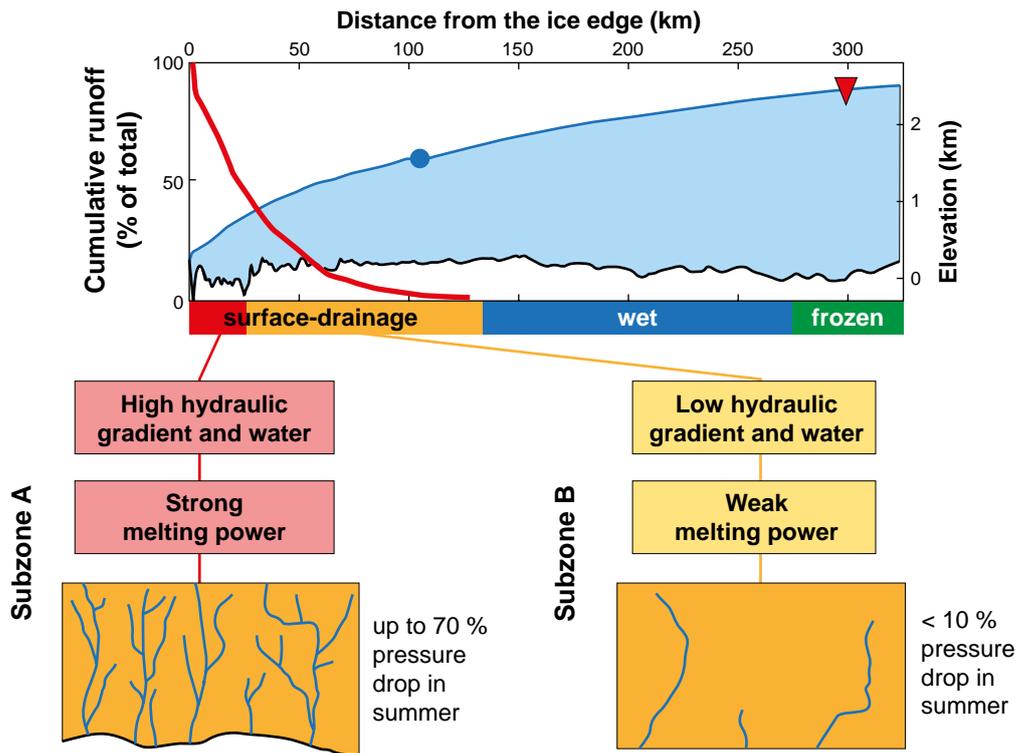


Figure 14-6. Recently developed conceptual model of ice sheet hydrology, with four distinct zones of fundamentally different hydrological properties (from Claesson Liljedahl et al. 2016). Note that the zone dominated by surface drainage is divided into two parts: i) subzone A and ii) subzone B. In subzone A, the basal hydrology is dominated by a transient network of drainage tunnels developed during the summer, up to 15–20 km from the ice sheet margin. The pressure in the tunnels has a very strong daily variation during the melting season (30–105 percent of the ice overburden pressure). The hydrology in subzone B still has a strong water flow due to the supply of surface meltwater, but the lower gradient of the ice sheet’s upper surface leads to a general lack of discrete, larger drainage tunnels. This is a main result in the GAP study: that the growth of drainage tunnels during the summer, from the ice sheet margin towards the inner parts, is highly limited by the reduced melting ability of the basal hydrological system (Section 5.1 in Claesson Liljedahl et al. 2016).

The following processes have been studied within ICE:

- **Short-duration very high subglacial pressures;** Research on valley glaciers has shown that very high but short-duration pressure pulses may occur in the basal drainage system. No such high pressure pulses have been documented from the Greenland ice sheet, but this has not previously been possible to study since the instrumentation in the ice boreholes did not have the capacity to record such short-duration pressure pulses. The ICE project developed special sensors in order to be able to study short-duration pressure pulses.
- **Pressure gradients on a scale corresponding to the ice thickness;** Results from the GAP show pressure gradients at the base of the ice sheet in the order of 16 kPa/m. However, it is not known how far these high gradients can be maintained, nor is it known whether the gradients can exist over length scales of tens to hundreds of metres. The ICE project was designed for studying pressure gradients on this scale.
- **Transmissivity and infiltration capacity;** A considerable amount of information was collected in the GAP regarding the pressure situation at the base of the ice sheet, but how large a portion of the underlying bedrock that is affected by this water pressure is unknown, and this was studied further within ICE. This is important information since the proportion of bedrock covered with water determines the size of the recharge area for groundwater.

The ICE project has resulted both in a documented and refined understanding of the questions that were studied within the GAP, and in a more detailed and more well-quantified understanding. Furthermore, the project has led to new results concerning processes that were not investigated in the GAP (Harper et al. 2019). The results from the ICE and GAP projects show that the top surface of the rock beneath the ice sheet consists of rock with a thin sediment layer, which indicates that the subglacial drainage system is hard rather than a porous medium similar to till. The hydraulic measurements in ICE confirm the results from the GAP and the conclusion that the ice overburden pressure is a good approximation of the basal hydraulic pressure over a large part of the ice sheet and for most of the year. Two types of pressure gradients have been identified: i) a primary gradient driven by large-scale longitudinal differences in the ice overburden pressure, and ii) secondary local gradients dependent on the dynamics in the subglacial drainage system (Figure 14-7). Data from the ICE project show that pressure changes in the drainage system are dependent on the basal flow dynamics and mechanical adaptation to the drainage system's volume. Spatially localised and very rare pressure pulses, corresponding to magnitudes of 0.72 MPa, were registered in the subglacial drainage system. Reliable estimates of water volume changes in the basal drainage system are inhibited by the ice deformation, since it is difficult to observe height changes of the upper ice surface with sufficient accuracy. The large-scale geometry of the ice sheet reflects increased basal sliding, which likely reflects the availability of water at the interface ice/rock.

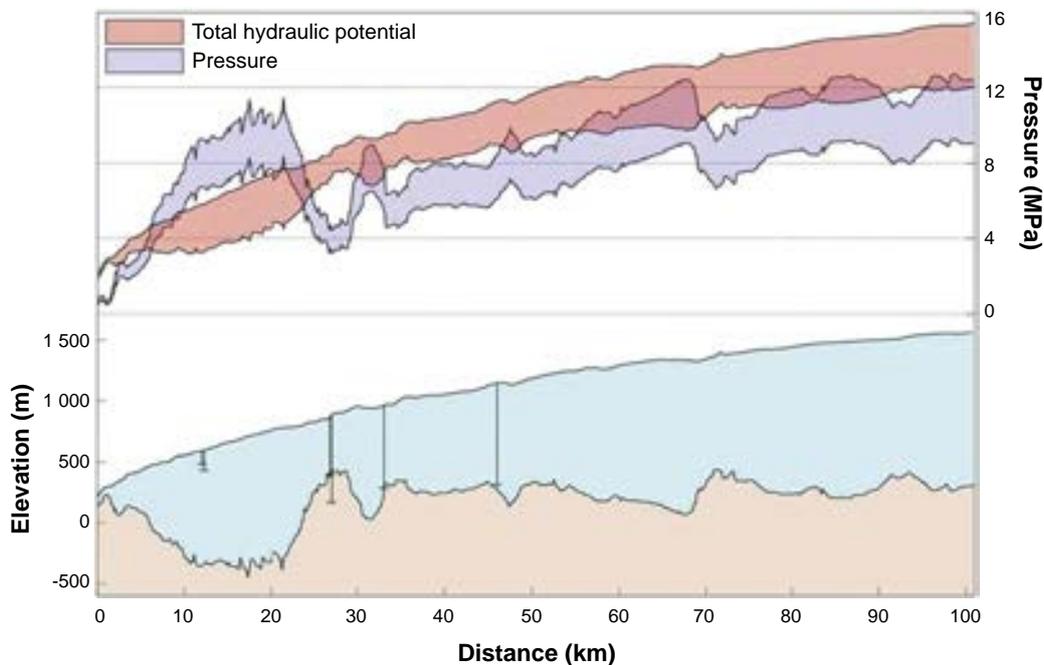


Figure 14-7. Basal hydraulic pressure and total hydraulic potential along the transect through the GAP/ICE area on Greenland, which extends from the ice sheet margin approx. 100 km into the ice. The topography of the upper surfaces of the ice and of the rock along the profile is here illustrated together with positions and depth of the boreholes that have been drilled into and through the ice (vertical lines). The field shaded in blue in the upper panel shows water pressure, which is obtained from the ice thickness and measured water pressures in the boreholes and which corresponds to 0.8–1.1 times the ice overburden pressure. The total hydraulic potential (field shaded in red) includes basal water pressure and the height potential. The gradients of the total hydraulic potential therefore reflect changes in ice thickness and in the height of the rock surface, combined with the local pressure gradients originating from basal processes (figure from Wright et al. 2016). This type of information will be used together with other information to set boundary conditions in future groundwater simulations under glacial conditions.

Programme

In the review of the RD&D Programme 2016, SSM writes: *SSM takes a positive view of SKB's completed studies on Greenland. Furthermore, the Authority takes a positive view of the fact that SKB will continue to monitor the deep borehole in front of the ice sheet margin. However, it is not entirely clear to SSM what SKB means by the statement that some level of monitoring is to continue during the coming RD&D period. SSM believes that the monitoring should include temporally repeated measurements, i.e. a monitoring programme, to ensure the results obtained from the deep borehole.*

The large field studies and modelling efforts within the GAP and ICE projects are completed, but monitoring of the deep borehole at the ice sheet margin, which reaches a depth similar to that of the repository, will continue during the period 2019–2021. The monitoring includes periodic registration of pressure, temperature and electrical conductivity in the borehole. Furthermore, it is possible to take water samples and gas samples from the borehole when necessary and to carry out temperature measurements along the borehole in order to obtain thermal data for permafrost studies etc. The network of weather stations on and in front of the ice sheet will be maintained during the same period.

The newly developed conceptual model of ice sheet hydrology will be adapted and applied in future hydrogeological modelling of the area in Forsmark. The boundary conditions for the hydrogeological models can thereby be improved for simulations of groundwater flow during glaciated periods. Other knowledge concerning hydrology, geochemistry and glacial hydrology will also be applied in SKB's safety assessment work in scenarios with glacial conditions. Also the extensive knowledge obtained from the GAP and GRASP projects concerning permafrost/taliks/surface hydrology in periglacial areas will be applied in the assessment of safety after closure in Forsmark, especially in the areas of hydrogeology and surface ecosystems. During the RD&D period, SKB will be a part of an international permafrost network together with several nuclear waste organisations and universities, in order to further study the hydrology in the periglacial landscape (Section 12.3.4). Within the framework of this network, one field of study will consist of system understanding where the connection between the glacial and subglacial systems and their interaction with the periglacial hydrological system are studied numerically and conceptually.

Part III

Decommissioning of nuclear facilities

- 15 Prerequisites for decommissioning of nuclear facilities
- 16 Planning for decommissioning at Uniper
- 17 Planning for decommissioning at Vattenfall
- 18 Planning for decommissioning of SKB's nuclear facilities
- 19 Continued activities within decommissioning

Part III – reading instructions

Part III of the RD&D Programme 2019 presents the planning and division of responsibilities for executing the decommissioning of the Swedish nuclear power reactors and SKB's facilities. It also gives a summary of the development work that is linked to decommissioning, which in some cases has been described in Part II.

15 Prerequisites for decommissioning of nuclear facilities

This chapter describes in general terms the set of requirements that apply during decommissioning of nuclear facilities, as stated in SSM's regulations and in the Environmental Code. It also gives a description of how the set of requirements affects the structure of a generic decommissioning project in the form of different stages and phases. Furthermore, the division of roles between the nuclear power companies and SKB regarding decommissioning and management of the resulting radioactive waste is presented. The subsequent chapters provide a more detailed description of how the work will be pursued within each corporate group and within each decommissioning project, with a focus on strategies and planned measures.

15.1 Concepts and requirements

Decommissioning of a nuclear power reactor includes defueling, possible shutdown operation and dismantling and demolition. Defueling is the activity from final shutdown of the nuclear power reactor until all nuclear fuel has been removed from the plant. In cases where dismantling and demolition cannot commence immediately after defueling, a period of service operation follows, during which the facility is maintained awaiting the start of dismantling and demolition.

During dismantling and demolition, activities will be carried out for disposing of the radioactively contaminated facility parts in the form of process systems, buildings and any contaminated soil. The dismantling and demolition phase is concluded when the site has reached a state that makes it possible to release from regulatory control. When SSM has approved an application for site clearance, the activities at the site are free from obligations under the Nuclear Activities Act and the Radiation Protection Act. The site thereby ceases to be considered as 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 15-1 shows schematically how the decommissioning of a nuclear power reactor is carried out in relation to the requirements on the facility during its life cycle. The upper part of the figure presents the activities that are planned to take place at the facility and the lower part shows the requirements according to the Environmental Code and SSM's regulations.

The main licensing processes that govern a decommissioning project are: a licence under the Environmental Code and approval under the Nuclear Activities Act and the Radiation Protection Act. According to the Environmental Code, an Environmental Impact Statement (EIS) 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 15-1. In conjunction with final shutdown, the existing EIS is expected to be possible to revise as defueling and shutdown operation will be largely similar to the previous operation. Prior to dismantling and demolition, on the other hand, a new EIS is required. The environmental impact statement, together with consultations, constitutes the basis for a licence under the Environmental Code.

In accordance with the Nuclear Activities Act and the Radiation Protection Act, as well as relevant ordinances, license conditions and regulations, the following documents must be prepared prior to and in some cases continuously during the execution of decommissioning:

- Decommissioning plan and decommissioning strategy.
- Waste management plan.
- Safety analysis report.
- Documentation according to the Euratom treaty, Article 37.
- Step/sub-project notification.
- Decommissioning report.
- Inspection programme for clearance.

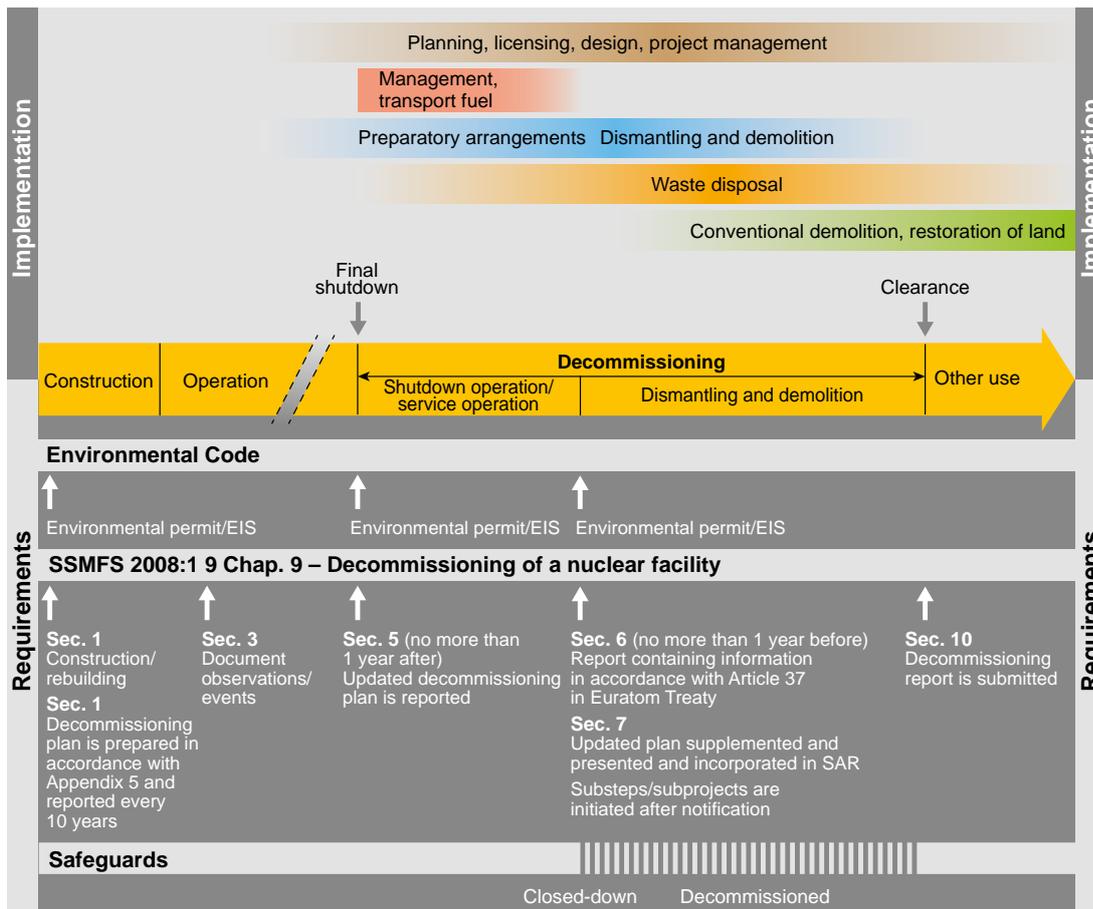


Figure 15-1. Overview of the different phases for execution of reactor decommissioning, and SSM's and the Environmental Code's requirements on decommissioning during the life of a nuclear facility.

The decommissioning plan is reported to SSM, while the waste management plan and safety analysis report (SAR) must be notified for formal approval according to SSMFS 2008:1. Furthermore, documentation will be prepared for informing the European Commission in accordance with the Euratom treaty, Article 37. The formal report is submitted to the European Commission by SSM, while each decommissioning project prepares background reports to SSM.

During dismantling and demolition, a notification of the measures that will be adopted in the facility is required. These are distributed practically in different steps/sub-projects which are notified progressively as decommissioning progresses. Each step notification should contain information on for example protective measures, choice of technology and risk assessments and is to be formally safety reviewed.

When dismantling and demolition are completed, a decommissioning report must be submitted to SSM, where experience from decommissioning and the final state of the site are described.

During decommissioning, reports are also required within the area of safeguards. The reports include accounting reports, establishment of activity programmes and annual updates of the site description. The requirements on reporting cease after a written verification from SSM that the facility has reached the status "decommissioned" according to the Ordinance (Euratom) No. 302/2005.

In addition to the above-mentioned requirements, there are also requirements related to decommissioning made by other regulators. These requirements, which are not described in detail in this document, include for example laws and regulations monitored by the Ministry of Environment, the Swedish Work Environment Authority, the Swedish Civil Contingencies Agency, the Swedish Transport Agency and the Building Committee in the municipality concerned.

Decommissioning of the Swedish nuclear power plants is furthermore affected by the planned operating times for 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.

15.2 Responsibility and division of roles

The licensee for a nuclear facility is responsible for its decommissioning according to the Nuclear Activities Act, 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 SSM has approved of sealing the final repository in question and the Government has granted exemption from responsibility under Section 10 of the Nuclear Activities Act (SFS 1984:3).

Vattenfall AB is the main owner of Ringhals AB and Forsmarks Kraftgrupp AB. Uniper is the main owner of OKG Aktiebolag and Barsebäck Kraft AB. The licensees for 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 for the Ågesta reactor. SKB is responsible for its facilities Clab and SFR, and the future facilities Clink, the Spent Fuel Repository and SFL. In order to pass on the nuclear licence to another actor, a government decision is required.

For more efficient work with decommissioning and waste issues, work areas have been divided between actors on both the company level and the group level. The joint commitments within the management of radioactive waste are normally coordinated by SKB, whereas the practice for handling decommissioning issues varies slightly within the two industrial groups, Vattenfall and Uniper. This section describes the working methods and distribution of work tasks.

15.2.1 Division of roles between the licensees and SKB

The licensees are responsible for decommissioning of their nuclear power reactors. SKB's principal task is to provide final disposal of decommissioning waste according to the licensees' needs. 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 costs as reported by the licensees.

Under the Nuclear Activities Act, the nuclear power companies, working in consultation, prepare or arrange for a programme for the comprehensive research and development work and the other measures needed to manage and 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 nuclear power companies, prepares the RD&D programmes and submits them to SSM, see Section 1.2.

General methods and procedures for demolition

SKB is contracted to coordinate the general methods and procedures for transportation and final disposal of radioactive waste that are required for decommissioning. SKB's tasks are, among other things, to develop waste type descriptions that describe how the waste meets the applicable acceptance criteria for each repository. The waste producer develops waste type description specifications that form a basis for the waste type descriptions but they are also incorporated into the waste producer's SAR and are to show how the waste meets the acceptance criteria for manufacturing and interim storage in the facility. Furthermore, SKB has been tasked with developing, when necessary, new waste containers for the decommissioning waste, which are common for all the nuclear power companies, see Section 7.5.

In order to achieve optimal national coordination, the nuclear power companies have jointly agreed on the tasks that SKB coordinates in connection with waste management, for example development of joint guidelines for clearance (SKB 2011a, Berglund et al. 2016), and guidelines for reporting of decommissioning plans (Calderon 2014).

Each nuclear power company will be responsible for the future decommissioning waste inventory, while SKB will be responsible for compiling the inventory and specifying requirements on the waste so that it can be transported and disposed of in the appropriate final repository.

Through close cooperation between the nuclear power companies and SKB, management of the radioactive waste can be carried out optimally across the entire chain from dismantling and demolition to disposal and closure of the final repositories.

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 type as the waste produced during operation, with the difference that the volumes will be larger during dismantling and demolition. The waste comes from different parts of the facility with great variations in the degree 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 conventionally since it stems from facility parts 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 the handling has been preceded by an initial assessment where the waste has been categorised as having an extremely low risk of contamination (ELR). Categorisation of waste is described in the report SKB (2011a) and in Berglund et al. (2016).

Waste that cannot be classified as zero-grade waste is sorted according to the appropriate management route. A compilation of forecasted decommissioning waste from the Swedish nuclear power plants per management route and principles for managements routes for nuclear waste during dismantling and demolition are presented in Figures 15-2 and 15-3.

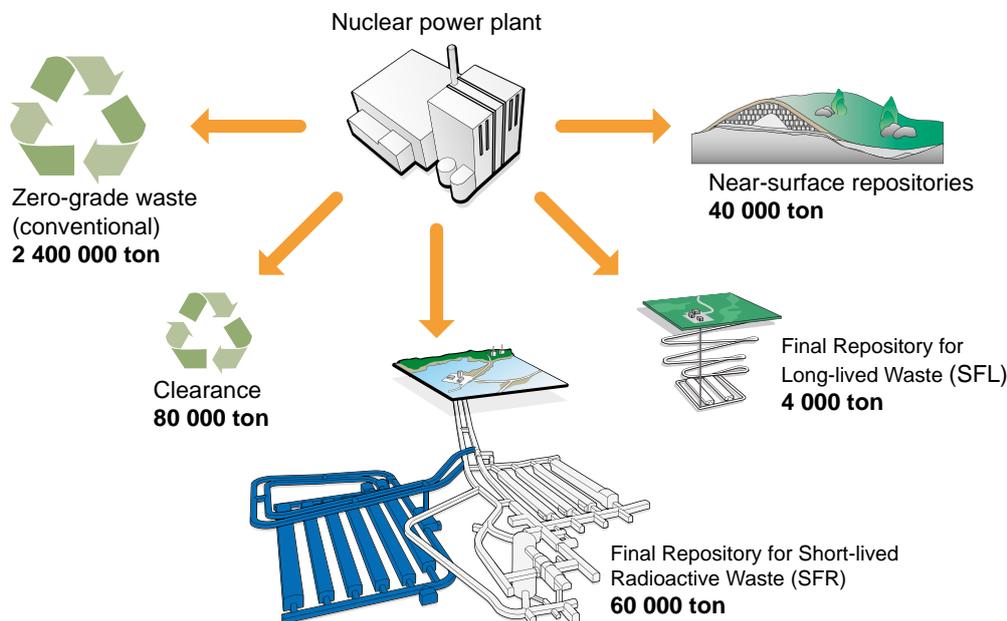


Figure 15-2. Compilation of forecasted decommissioning waste from the Swedish nuclear power plants.

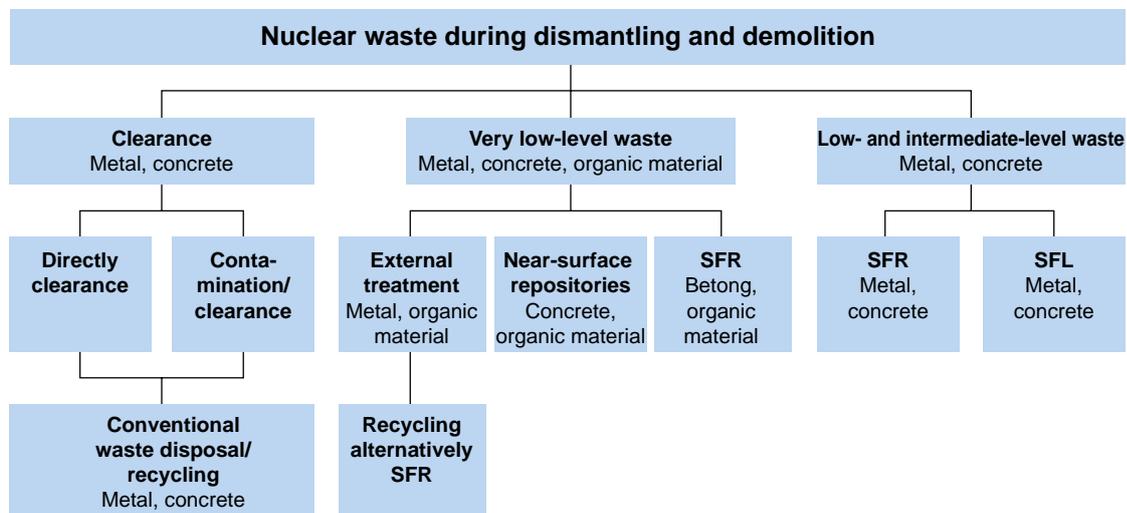


Figure 15-3. The nuclear power plants' principles for management routes for nuclear waste during dismantling and demolition.

Costs for decommissioning

Under the Nuclear Activities Act, the nuclear power companies are obliged to bear the costs for the measures needed to manage and dispose of the nuclear waste and the spent nuclear fuel and to decommission their facilities. On behalf of the nuclear power companies, SKB ensures that a cost calculation is prepared every three years, in accordance with the Financing Act, see Section 1.3. Paid-in fees are managed by the state Nuclear Waste Fund.

The nuclear power companies produce cost estimates for the decommissioning of their nuclear power reactors and submit them to SKB. SKB compiles the estimates and produces an overall cost estimate and risk analysis, which then serve as a basis for the fees and guarantees determined by the Government.

Transportation 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 transportation system consists of the ship *m/s Sigrid*, special vehicles and different types of transport casks and containers, see Section 2.3. The ship and the vehicles are used for transportation of both low-level and intermediate-level waste and spent nuclear fuel. The different transport casks and containers are specifically developed for each waste type. If new transport casks or containers need to be developed, SKB is responsible for the execution of this.

Nuclear waste can, when considered justified, be transported outside of SKB's system. This could for example involve large components.

15.2.2 Division of roles within Uniper

Uniper's nuclear decommissioning portfolio includes the decommissioning of OKG Aktiebolag's and Barsebäck Kraft AB's facilities. The division of roles is described in each licensee's decommissioning strategy and decommissioning plans, which show that Uniper's goal is to coordinate decommissioning of the shutdown reactors Barsebäck 1, Barsebäck 2, Oskarshamn 1 and Oskarshamn 2.

Since the spring of 2017, the strategic work for decommissioning within Uniper is coordinated functionally within the framework of Project Performance Center (PPC). PPC has the purpose of identifying potential synergies for optimising the decommissioning of the facilities. PPC's task is also to support and follow up the execution of the decommissioning programmes.

The steering committee for PPC consists of representatives from Sydkraft Nuclear Power (SNP) and the licensees OKG Aktiebolag (OKG) and Barsebäck Kraft AB (BKAB). PPC reports both to SNP and to Uniper's central function for project management (UTG). The principles are based on the fact that the CEO of each licensee ultimately decides in matters concerning the execution of decommissioning. Decommissioning at OKG is managed by department A (Decommissioning) with support from the rest of OKG, where certain functions are shared (with Oskarshamn 3 and the shared facilities that are still in operation). The organisation at BKAB is streamlined for the current decommissioning phase, during which all facilities at the site will be decommissioned. Adaptation of the organisation to large-scale dismantling and demolition are in progress in both OKG and BKAB. The reorganisation is planned to be completed in the autumn of 2019. The organisation will be based on the main functions during decommissioning in the form of waste management, dismantling and demolition, operation and maintenance, and protection and safety.

15.2.3 Division of roles within Vattenfall

In 2015, a unit was created within Vattenfall AB with the purpose and responsibility to coordinate and pursue decommissioning issues within the Vattenfall Group, the Business Unit Nuclear Decommissioning (BUND). With the establishment of BUND, the group's decommissioning activities were separated from its power production and the focus of the activities was made clear for operation and decommissioning. The establishment of BUND also permitted centralisation of the group's decommissioning competence to an organisational unit dedicated for the purpose.

Within BUND, it has been decided that the planning and execution of the group's decommissioning task will primarily be conducted in programme form. The background for this decision is that nuclear decommissioning entails a multi-faceted and multi-disciplinary challenge extending over long periods

of time, which means that target outcomes are determining rather than explicitly measurable project goals. By applying programme control, the complex network of issues that need to be addressed is possible to handle in a transparent and synchronised manner.

BUND's task is to ensure that safety is in focus and that decommissioning planning is carried out efficiently with clear priorities within and between the decommissioning programmes. BUND is also responsible for group-wide optimisation of decommissioning programmes and for securing strategic decommissioning-specific competence and strategic development support in the activities.

All analysis and planning for operational decommissioning activities are carried out by BUND on behalf of, and in close collaboration with, the licensee. The licensee makes sure that competence regarding the facility and its history is available for the decommissioning programmes and is responsible for providing clear and realistic financial and technical prerequisites for decommissioning. The licensee also ensures that final shutdown takes place in an optimal way, that the fuel is transported away and that physical separation of any facilities or systems that will remain at the site is performed.

Existing nuclear licences for Ringhals 1 and 2 are planned to be transferred to Vattenfall AB immediately after the facilities are free from nuclear fuel.

15.3 National and international coordination

15.3.1 Industry-wide coordination

From a national perspective, there is a need to coordinate decommissioning 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 several forums to support this work, where also certain international forums are important.

On a national level, there is SKB's RD&D and Plan group, in which the nuclear power companies participate. The group coordinates SKB's work with the RD&D programmes and the Plan reports.

To further support line managers within SKB and the nuclear power companies when it comes to decommissioning and waste management, a forum for waste managers has been established where SKB is the primary convener. This makes it possible to make decisions, within the mandate of the participants, to bring important issues into focus and prioritise the work carried out by SKB and the nuclear power companies.

In conjunction with the premature 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 nuclear power companies and SKB. The cooperation includes for example interim storage of waste awaiting that planned final repositories are commissioned or optimising the management of certain waste fractions.

SKB and the nuclear power companies are also participating in international forums in the field of decommissioning. Participation in the OECD-NEA's collaborative programme and the IAEA's programme are two examples. The main work within the OECD-NEA that has previously been carried out in the Working Party for Dismantling and Decommissioning (WPDD) will now be pursued within the OECD-NEA in the Committee on Decommissioning of Nuclear Installations and Legacy Management (CDLM) and within the IAEA in the group International Decommissioning Network (IDN).

Barsebäck Kraft AB and AB SVAFO are also members of the OECD-NEA's CPD (Co-operative Programme for the Exchange of Scientific and Technical Information concerning Nuclear Installation Decommissioning Projects) and its subgroup TAG (Technical Advisory Group).

15.3.2 Coordination in Uniper

PPC, which has the functional responsibility to coordinate the decommissioning portfolio within Uniper, consists of, in the practical work, a number of working groups with representatives from OKG and BKAB consisting of personnel with discipline-specific expertise and project managers within the

disciplines. The focus of the work groups varies over time. The result of the work within the framework of PPC provides, at a strategic level, a basis for the continued implementation of OKG's and BKAB's decommissioning programme.

15.3.3 Coordination in Vattenfall

In Vattenfall, several forums have been created to coordinate decommissioning within the group. The Decommissioning Council constitutes a forum for decisions on strategy and approach for the decommissioning of Ringhals 1 and 2. The purpose is to achieve safe and cost-effective decommissioning of Ringhals 1 and 2 while ensuring continued safe and stable operation of Ringhals 3 and 4. Issues that are handled in the Decommissioning Council include strategic issues concerning for example licensing, planning, infrastructure and personnel.

A strategic forum for radioactive waste management and decommissioning, Strata, has been formed in order to ensure that different activities' strategies and plans for decommissioning and waste management are coordinated and optimised. This is based on different aspects such as safety, regulatory requirements, cost-effectiveness and other external and internal stakeholder requirements.

Within the framework of the Collaboration Area – Radioactive Waste (SO Avfall), group-internal collaboration is conducted within the waste and decommissioning field, for example regarding clearance, waste documentation and general exchange of experiences. Within the group, there is also a competence centre in the waste area, with the principal purpose of working to meet the competence and resource needs in the area in the long term.

An important part of building competence within Vattenfall is working with experience feedback from the decommissioning in a structured and thought-out manner. Decommissioning competence 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 Ringhals 1 and 2. Similarly, there are ongoing initiatives on exchanging experience and utilising synergies, for example within the procurement area, between Vattenfall's decommissioning activities in Germany and in Sweden.

There are also a number of councils and forums within Vattenfall that address more operative issues within the decommissioning of Ringhals 1 and 2 and the Ågesta reactor.

15.4 Resources and competence

Decommissioning of nuclear facilities in Sweden will continue up until the mid-2070s, when the facilities for management and disposal of spent nuclear fuel and other radioactive waste will be decommissioned. These decommissioning activities will not take place in a steady flow, but will be carried out in three main stages; one in the 2020s (now), one in the 2040s and the final one in the 2070s. A challenge is thus securing access to competence for all three decommissioning stages, since the need for decommissioning competence will be limited in Sweden between stages.

To secure competence in areas important for final disposal of nuclear waste, as well as dismantling and demolition, will therefore be a strategically important issue for the companies in both the short and long term. In order to secure competence in the country, collaboration with universities and university colleges will be a part of the long-term competence management, as will the continued encouragement of cooperation within the industry to both retain personnel and attract new personnel in these areas.

Current situation

A key to completing the first stage is to work in a structured manner with building competence and experience feedback in the decommissioning area. An important part is the organic development of competence; it must be ensured that the decommissioning competence and the experience obtained during decommissioning of for example previous research reactors can be used in the decommissioning of the commercial reactors.

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.

Furthermore, the companies have established contact with international suppliers with considerable experience of practical decommissioning activities, and different forms of collaboration have been initiated with the purpose of acquiring and building the relevant competence.

This permits experience and competence to be gradually built up and secured in the industry to the final phase of the first decommissioning stage.

Another example is the educational activities in the backend area established by the industry within KSU, including education in both waste management and decommissioning.

Programme

Ensuring that competence and experience is preserved for future stages is a significant challenge. An important constituent is to carefully and in a structured manner document the practical experience gained by the companies during decommissioning, to clearly describe how organisation and management should be designed depending on the work task, how a facility in operation is best prepared for decommissioning regarding for example documentation, different forms of decommissioning concepts, the experience linked to them 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, managed and carried out with respect to safety and efficiency.

In addition to this, it is critical to secure facility knowledge and formal competence within, for example, radiation protection, in the long term. This requires both industry-wide collaboration and collaboration with the educational system.

An example of industry-wide collaboration is the Swedish Centre for Nuclear Technology, SKC, a research centre where the nuclear power plants, Westinghouse Electric Sweden AB, KTH, Chalmers University of Technology and Uppsala University since 27 years supports education, research and development within nuclear applications at universities and university colleges in Sweden.

Another example of industry-wide collaboration is that Vattenfall and Uniper in Sweden and TVO in Finland have established a network for future competence management, BWR Future, where advanced engineering competence will be available as a flexible resource. The assignments will be conducted in the form of projects. Each company in the network can, when necessary, request expertise and participation in the activities is voluntary. The resource network will focus on technology for the boiling water reactors in Olkiluoto, Forsmark and Oskarshamn.

Finally, it is important to note that even if the need for decommissioning competence in Sweden will be limited between decommissioning stages, international decommissioning activities will continue on a large scale and for a long time. Hence, the global need will be great over time, which means that the motivation to establish and preserve competence over time will be significant, which most likely means that the relevant competence will be available internationally throughout the Swedish decommissioning period.

16 Planning for decommissioning at Uniper

The planning of Uniper's decommissioning portfolio is carried out on a strategic level within the framework of Project Performance Center, PPC, in order to identify and make use of synergies between the decommissioning programmes at OKG Aktiebolag and Barsebäck Kraft AB. With the aim of safe and cost-effective decommissioning, a concept has been developed on the portfolio level with a demolition sequence in which a critical path lies in sequence (after each other) between the decommissioning programmes and the different facilities. The approach allows the benefits of learning effects to be maximised in both planning and execution. The technical sequence is also developed to create a high degree of flexibility.

16.1 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 (BKAB). The Barsebäck nuclear power plant produced electricity between 1975–2005. The reactor in unit 1 (B1) has been permanently shut down since 1999 and the reactor in unit 2 (B2) since 2005, see Figure 16-1.

The decommissioning of BKAB's nuclear facilities is described in the Decommissioning Plan for the Barsebäck plant.

At the end of 2019, all necessary licences for the start of dismantling and demolition are expected to have been obtained. Dismantling and demolition of Barsebäck 1 (B1) and Barsebäck 2 (B2) will begin in 2020. During dismantling and demolition, the different work packages/steps will be reported according to current licence conditions.

The reactor internals have been segmented. Preparatory measures have been carried out according to the current licence conditions. The facilities have been surveyed radiologically and the waste in each step has been assigned preliminary solutions for management and disposal.



Figure 16-1. View of the Barsebäck NPP with the two BWR reactors B2 (closest in the photo) and B1. Photo Jenny Eliasson, Malmö Museums.

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 thereafter, a sequence for work packages/steps has been established. Based on that sequence, detailed planning and procurement for each work package/step is carried out progressively. The approach is based on reducing technical and safety-related risks. During decommissioning, BKAB acts as a purchaser organisation. Early dialogues are held with suppliers, which are also involved in the planning to optimise the execution.

Handling lines for radiological and clearable waste are established. Waste logistics are designed based on the demolition sequence and the radiological survey. BKAB has redundancy in the handling lines in order to reduce the risk of disturbances in the waste logistics, see Figure 15-3.

Since decommissioning of B1 and B2 will be carried out before the extended SFR facility is commissioned, BKAB needs to build a new interim storage facility for the low-level radioactive waste. For the long-lived radioactive waste that is interim-stored in the existing storage facility awaiting disposal in SFL, alternatives for external interim storage are investigated. The interim storage facility for long-lived waste is utilised to about 50 percent, so it will also accommodate the waste packages from the segmented reactor pressure vessels. BKAB 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 opportunity to use OKG's near-surface repository is planned to be an alternative to disposing of waste in SFR.

Decontamination and clearance of buildings is planned to be completed in 2028.

Conventional demolition and restoration of land takes place in connection with transportation of the radioactive waste to SFR.

The general timeplan for decommissioning of the Barsebäck nuclear power plant is presented in Figure 16-2.

Estimated waste quantities from the decommissioning are presented in the decommissioning plan, waste management plan and in more detail in each step report. The presented waste quantities per waste type and activity category are based on results from the radiological survey that will also be updated continuously during dismantling.

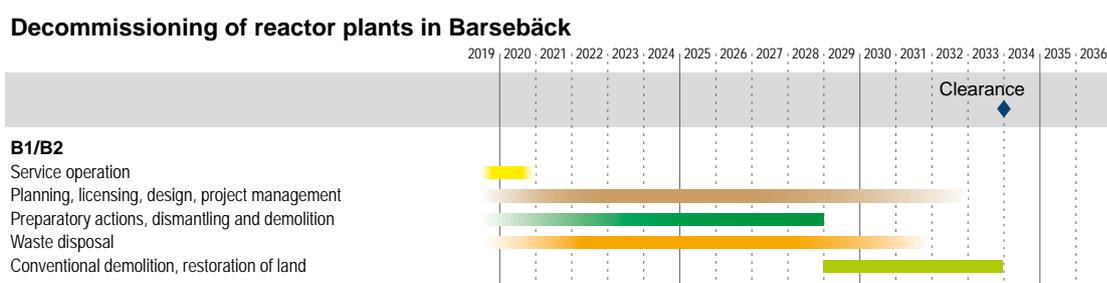


Figure 16-2. Schematic overview of Barsebäck Kraft AB's timeplan for decommissioning.

16.2 OKG Aktiebolag's planning for decommissioning

OKG Aktiebolag (OKG) owns the nuclear power reactors Oskarshamn 1 (O1), Oskarshamn 2 (O2) and Oskarshamn 3 (O3) and a number of shared service facilities called Unit 0 (O0). The reactor site is located on the Simpevarp Peninsula next to the Baltic Sea, about 30 km north-east of Oskarshamn. O1, O2 and O3 are boiling water reactors (BWR), see Figure 16-3.

In 1966, a licence to build O1 was obtained. The facility was phased into the power grid for the first time in 1971 and was inaugurated in 1972. In 1969, a licence to build O2 was obtained and the facility was commissioned in 1974. O3 was commissioned in 1985. O1 and O2 are situated adjacent to each other while O3 is situated slightly more north of the others.

The execution of decommissioning of OKG's nuclear facilities is described in the current decommissioning strategy and decommissioning plans.

16.2.1 Oskarshamn 1 and Oskarshamn 2

Dismantling and demolition of O1 and O2 are under way. O1 was finally shut down in the summer of 2017. Regarding O2, which has undergone an extensive modernisation programme, a decision was taken not to restart the facility and the final shutdown was established at the end of 2016. All necessary licences for the start of dismantling and demolition have been obtained for both reactors. During the execution of dismantling and demolition, the different work packages/steps will be reported according to current licence conditions.

The completed and continued planning for decommissioning of O1 and O2 (including the shared waste facility, OAVF) is based on planning of large projects, where standardised planning processes based on Lean principles are used (Lean – methodology for maximising benefits and minimising waste of resources). Technical, safety-related and organisational dependencies have been identified, after which a sequence for the execution of the different work packages has been determined. Detailed plans for large-scale dismantling and demolition are then gradually developed for each work package/step. The approach is based on reducing technical and safety-related risks. During decommissioning, OKG intends to act as a purchaser organisation. Early dialogues are held with suppliers, which also will be involved early in the planning to optimise the execution.



Figure 16-3. View of OKG's nuclear power plant with the three BWR reactors O1, O2 and O3 from left to right in the picture.

The facilities have so far been emptied of fuel and segmentation of reactor internals will be concluded in the beginning of 2020. A number of preparatory measures have also been taken. A radiological survey of the facilities has been made, as well as a 3D scan. Building components and materials from the facilities have been divided into different routes for material and waste management.

Materials and waste that is produced in connection with decommissioning is of the same type as the waste produced during operation, with the difference that the waste volumes will be larger during decommissioning. This means that material and waste management needs to be adapted but that available techniques 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 technical demolition sequence and the radiological survey.

SFR and SFL are unavailable during the decommissioning of O1 and O2 (including OAVF). Material and waste management have to be optimised from an overall perspective, where possible and reasonable management routes, such as geological final repositories, local clearance, external treatment and near-surface repositories at the sites, are weighed against each other. This means that redundancy in management routes is pursued when possible, and that the preferred management route is based on an overall evaluation of factors such as cost, risk, environment, ALARA principles etc.

During the decommissioning of O1 and O2 (including OAVF), the waste that will be disposed of in the extended SFR or SFL must be interim-stored on site, until the final repositories are in operation and available. 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 is to be transported to SFR. However, interim storage of intermediate-level waste that is to be transported to SFR and long-lived waste intended for SFL is expected to be possible to accommodate in the existing interim storage facility BFA (waste vault for interim storage of low-level and intermediate-level waste).

The existing near-surface repository MLA (near-surface repository for final disposal of very low-level waste) has to be extended to be able to receive the very low-level waste from the decommissioning of O1 and O2 (including OAVF). The extension of the near-surface repository is also necessary for 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 BKAB's facilities.

The capacity of both interim storage facilities and near-surface repositories is governed by the need based on an optimal distribution of the different management routes on the basis of specified factors. The extensions of interim storage facilities and near-surface repositories are handled in separate licensing processes.

Estimated waste quantities from the decommissioning are presented in each decommissioning plan, and for the facilities under dismantling and demolition also in the current waste management plan and more specifically in each step report. For the facilities in dismantling and demolition, there are more 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. The total amount of material and waste from the decommissioning of O1 and O2 (including OAVF) is estimated to about 300 000 tonnes, of which radioactive material and waste are estimated to constitute a small fraction (about 5–10 percent).

Decontamination and clearance of buildings will be carried out in parallel with dismantling and demolition. Clearance of buildings is planned to be completed in 2028.

Conventional demolition including the reactor buildings for O1 and O2 and restoration of land takes place in connection with the decommissioning of O3 and shared facilities around 2050–2055.

The general timeplan for the decommissioning of the Oskarshamn nuclear power plant is presented in Figure 16-4.

Decommissioning of reactor plants in Oskarshamn



Figure 16-4. Schematic overview of OKG's timeplan for decommissioning (O0 are common facilities that are described separately).

16.2.2 Oskarshamn 3 and shared facilities

Oskarshamn 3 (O3) is planned to be operated until 2045. Thereafter, dismantling and demolition are planned to start 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 occurred in view of the fact that several decommissioning projects will have been carried out by then. This, together with experience from decommissioning of O1 and O2 guarantees an optimisation of the decommissioning.

Regarding decommissioning of the shared facilities associated with Unit 0, the joint waste facility (0AVF) will be decommissioned in conjunction with the decommissioning of O1 and O2. 0AVF will be decommissioned when the facility is no longer needed for the decommissioning of O1 and O2. The start of dismantling and demolition of the joint waste facility is planned to take place in 2025. Licensing processes and planning for 0AVF are handled in connection with the decommissioning of O1 and O2. Other facilities within Unit 0 will be decommissioned in connection with the decommissioning of O3.

The decommissioning planning for O3 and Unit 0 will continue during the operating time according to regular procedures, where experience from decommissioning of O1 and O2 0AVF will be utilised. Estimated waste quantities from decommissioning are presented in each decommissioning plan and are based on the quantities from previously completed decommissioning studies. The total amount of material and waste from decommissioning of O3 and the shared facilities is estimated to be about 400 000 tonnes. Of these total quantities, radioactive material and waste are estimated to constitute a small fraction (about 5–10 percent).

Conventional demolition of O3 is planned for around 2054–2055. Dismantling and demolition of the shared facilities are planned in two stages. The first stage, in 2046, concerns those facilities that are not required for the decommissioning of O3, while the next stage concerns the remaining facilities, which are demolished in connection with the demolition of O3. Remediation of all sites occurs around 2050–2055 in such a manner and to the extent that other industrial activities may be established at the sites.

17 Planning for decommissioning at Vattenfall

This chapter describes Vattenfall's decommissioning planning on a general level. The plans thus apply to both the Ågesta reactor and the reactors in Forsmark and Ringhals.

During defueling, transportation of fuel, shutdown of systems and management of operational waste are planned to be carried out as well as preparatory activities for dismantling and demolition. These preparatory activities, for example decontamination of primary system, 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 improving the possibilities for flexibility in critical activities during dismantling by moving the steps from time-critical paths (steps that affect the project's end date). Another possibility is to manage certain non-radioactive or low-level contaminated systems, such as turbine and generator systems, in order to optimise waste management prior to the start of dismantling and demolition on a large scale.

The decommissioning programme organisation will have as a task to plan and execute the constituent projects in which contractors are used as the main work force during the execution phase. The objective is to keep the programme organisation small and effective.

In order to effectively execute the decommissioning programmes, prerequisites and scope must be well defined in advance. The time for dismantling and demolition will be optimised on the basis of the prerequisite that the decommissioning will begin after shutdown without delay with a continuous execution phase. This means that service operation is to be minimised and that dismantling and demolition will last until the facility's final state is reached.

The execution of dismantling and demolition will be divided into a number of suitably delineated work packages and stages. In order to achieve efficiency, dismantling and demolition activities are planned to proceed, wherever possible, in parallel in the entire facility. The plans for the execution and chronology of work operations will be optimised throughout decommissioning based on ALARA and BAT in order to minimise dose and maximise efficiency.

SKB's owner companies have, since the RD&D Programme 2016, changed the strategic focus regarding the design of the extended SFR, which leads to consequences for management of reactor pressure vessels from the boiling water reactors. The updated strategy entails that the reactor pressure vessels from Vattenfall's BWR plants will be segmented on site in the facility. Segmentation of reactor internals takes place covered by water in the handling pool. Depending on the final choice of method, segmentation can be done mechanically (e.g. sawing, cutting, waterjet cutting) or thermally (e.g. plasma cutting, laser cutting, electrical discharge machining) and the like. For Ringhals 2, which is a pressurised water reactor, there is a policy decision on segmentation of both the reactor pressure vessel and internals. Both reactor internals and pressure vessel contain sufficiently high concentrations of long-lived radionuclides for the waste to require final disposal in SFL. Since SFL is only commissioned in 2045, segmentation will facilitate the long period of interim storage that must be localised at Ringhals NPP until the waste can be managed by SKB.

A crucial difference from the operation of a facility is the considerably larger amount of waste that is produced during decommissioning. This means that the capacity for handling certain waste streams must increase substantially during decommissioning in order not to hinder progress in the project. The increased capacity can be achieved in several ways if necessary, for example by adapting existing waste management buildings or other buildings that are not needed after shutdown. 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 should be justified and efficient. This is achieved by classifying the waste before it is produced and sorting it directly at production. Processing of nuclear waste must be minimised and wherever possible carried out in connection with producing the waste. Creation of secondary waste must be justified, in other cases avoided. Large components must be possible to dispose of whole where this is justified from a cost and dose perspective. Time-consuming waste processing is only performed if benefits can be ensured.

Waste management, including interim-storage and final disposal, is to be optimised from a cohesive operational and decommissioning perspective on the group level. Different waste scenarios are analysed within the group where near-surface repositories are a disposal alternative but also other disposal options are analysed. The point of departure is to increase the capacity for management of the waste where it is most appropriate from a group perspective.

The progress of the decommissioning programmes should, to a reasonable level, be independent of the capacity for waste treatment, waste transport and final disposal. For transport, SKB's transportation system will primarily be used, but the programmes are not exclusively bound to it. Coordination of nuclear fuel shipments should take place on the group level, with the goal of minimising the impact on the decommissioning programmes.

The conventional waste stream constitutes approximately 95 percent of the total waste volume. The majority of this waste consists of demolition materials from buildings. In order to minimise the requirements on treatment of the waste, decontamination of buildings is carried out prior to demolition. For the demolition materials that are produced in system dismantling, a clearance procedure is established in line with the existing clearance manual (Berglund et al. 2016).

The end state of decommissioning is a site that is released from regulatory control and can be used for other industrial purposes. The end states for the different decommissioning programmes can, however, be an industrial site where buildings and infrastructure that are of use for continued activities, and are possible to release from regulatory control without demolishing them are left, while other installations are dismantled. Clearance can be handled in a separate project or by line organisation depending on what is optimal for each facility. The 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 status required by continued industrial activities on the site.

17.1 Ringhals AB's planning for decommissioning

Ringhals NPP is situated on the Värö Peninsula in Varberg Municipality, 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. Besides R1 to R4, the site also contains shared buildings and facilities for waste management, offices, workshops, storage buildings, access roads etc, and occupies a total of 2.5 square kilometres, see Figure 17-1.



Figure 17-1. Ringhals NPP with reactors 1 and 2 to the right and reactors 3 and 4 to the left. In the centre of the site, there are office buildings and a lunch restaurant. Just above R1 in the image is the plant's waste area, which includes handling, conditioning and storage of nuclear waste.

Ringhals 1 and 2 were built in the 1970s and placed in a joint operations area. In the 1980s, Ringhals 3 and 4 were built in an area originally separated from R1 and R2, but later linked together by a transport road. This provides good prospects for separating the units again and permits parallel operation and decommissioning in different operations areas after final shutdown of R1 and R2.

The site is relatively large, as it was originally planned for a number of additional reactors. This means that areas are available for, for example, temporary storage of waste and there are possibilities for several transport roads, which permits effective logistics. Decommissioning of R1 and R2 will also be facilitated, in terms of waste logistics, by the existing waste facility being situated adjacent to R1.

Videberg harbour is situated close the site and is used for transportation of nuclear fuel and radioactive waste.

Overall planning

The decommissioning of the reactors at Ringhals is described in two decommissioning plans, one for R1 and R2 and one for R3 and R4. These decommissioning plans are based on the overall strategy and goals presented in the introduction of the chapter. Ringhals AB (RAB) is planning to operate the reactors R1 and R2 until the end of 2020 and 2019, respectively. This is based on the decision that was made in 2015 on taking R1 and R2 out of service earlier than planned. For the reactors R3 and R4, the current planning prerequisite of operation until 2041 and 2043, respectively, remains. Defueling is currently estimated to take about 18 months for R1 and about 26 months for R2, taking into account cooling requirements and the capacity for fuel transport.

In December 2015, the work with planning of dismantling and demolition of Ringhals 1 and 2 was initiated, today compiled in a programme called R12D. The focus so far has been on assessing the technical, legal, temporal and commercial preconditions for decommissioning and evaluating in detail how the specific activities during decommissioning may be best performed. This includes assessments of, for example, demolition methodology, organisation and waste management. Furthermore, reducing the dose rate in the facility is planned by means of a so-called system decontamination. The decommissioning plan describes the activities that will be assessed or conducted during the different programme phases.

In 2018, Ringhals AB obtained a licence under the Environmental Code for final shutdown and shutdown operation of Ringhals 1 and 2, and has thereafter submitted an application for a modification licence where the reactors are allowed to transition from defueling and shutdown operation to complete dismantling and demolition.

In conjunction with the shutdown decision for Ringhals 1 and Ringhals 2, the project STURE, Safe and Secure Phase Out of Ringhals Reactors 1 and 2, was initiated, with the purpose of identifying, assessing and implementing measures required to continue the operation of Ringhals 3 and 4 when Ringhals 1 and 2 are decommissioned. The project will ensure that the reactor units that will continue to be operated are physically separated from the one 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. The project will also ensure that measures during defueling are carried out. Examples of such measures are for example system decontamination after final shutdown and other preparatory activities needed for dismantling and demolition. In the spring of 2019, Vattenfall AB submitted an application for transferring the nuclear licences for Ringhals 1 and 2 from Ringhals AB to Vattenfall AB. The plan is that the licences will be transferred after completed defueling. Ringhals AB has submitted a declaration of intent to SSM, in which an account is given of the company's measures to prepare for a transfer of these licences.

Close cooperation takes place between project STURE and programme R12D during the preparations and during the execution of defueling operation, including the preparatory measures.

The general timeplan for decommissioning of the Ringhals NPP is presented in Figure 17-2.

Decommissioning of reactor plants in Ringhals

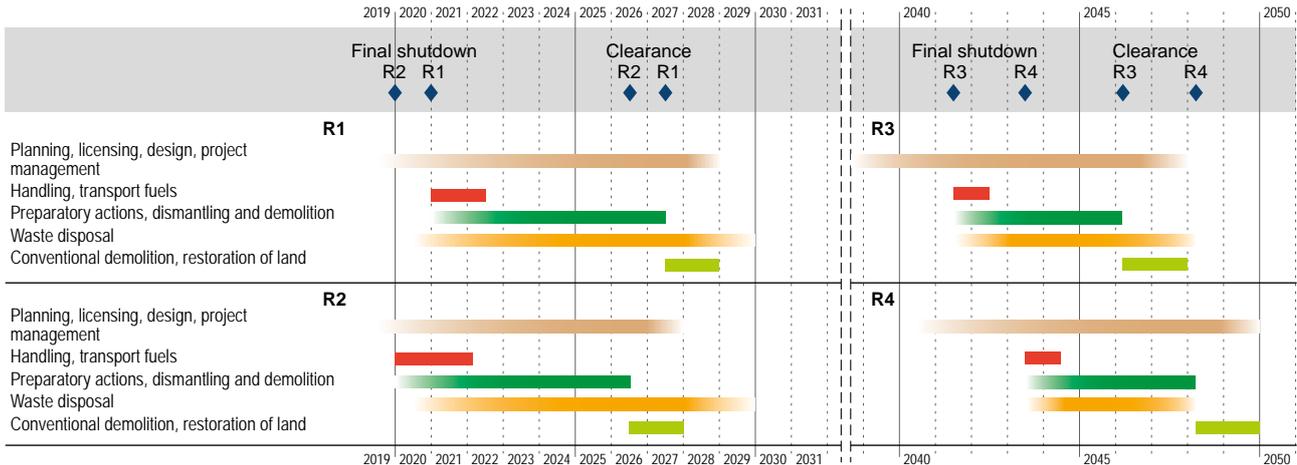


Figure 17-2. Schematic overview of Ringhals AB's timeplan for decommissioning.

Waste management

Since the final shutdown of R1 and R2 occurs at a time when the extension of SFR is not yet in operation, interim storage of the arising radioactive decommissioning waste is required. This can take place at Ringhals NPP and/or externally. By interim storage at the nuclear power plant, the external dependencies linked to waste management are minimised, which makes this alternative the first choice for most waste streams. It is estimated that interim storage at the nuclear power plant can take place using existing infrastructure, i.e. no new interim storage facilities are planned.

Reactor pressure vessel segments from R1 can, because of the relatively low activity content, be disposed of in SFR. The most neutron-activated internals are segmented and packaged for interim storage prior to future disposal in SFL. Reactor internals that have been some distance away from the core region such as the steam dryer and steam separator will be segmented and disposed of in SFR.

The reactor pressure vessels and internals of R2, R3 and R4 require, because of their larger activation, disposal in SFL. For R2, there is a policy decision on segmenting the reactor pressure vessel and internals. The final management for Ringhals 3 and 4 is not yet decided. Interim storage takes place at Ringhals until extended SFR and SFL is commissioned.

17.2 Forsmarks Kraftgrupp AB's planning for decommissioning

The Forsmark power plant is located on the east coast, about four kilometres north of Forsmarks Bruk, in Östhammar Municipality 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 17-3. The NPP also contains buildings for temporary accommodation, storage and workshop buildings and administrative buildings. A harbour has been constructed, which is used 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 north-west of these. Shared facilities such as access roads, harbour, water and sewage treatment plants, water tower and administrative buildings are utilised by all three units and by SFR. The area provides good prospects for parallel operation and decommissioning. Large surfaces are available for buffer storage and for the establishment of different transport alternatives.



Figure 17-3. The Forsmark nuclear power plant with the three BWR reactors F1, F2 and F3, from left to right in the photo.

Overall planning

Forsmarks Kraftgrupp AB (FKA) plans for 60 years of operation for each of the three reactors, which entails final shutdown for F1 in 2040, for F2 in 2041 and for F3 in 2045. Defueling operation begins at final shutdown; its length should be minimised as far as possible and is at present predicted to 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 of Chapter 17. F1 and F2 are expected to be dismantled and demolished in a way that maximises synergy benefits and minimises the need for facility separation or service operation. At the end of the decommissioning projects for F1 and F2, defueling is started for F3, which means that decommissioning is expected to take place at the site without interruption from the start of the first project until the last reactor is finally dismantled. Figure 17-4 presents the general timeplan for decommissioning of the Forsmark nuclear power plant.

The current planning assumes that SFR is in operation at the time for dismantling and demolition of the Forsmark facilities, which means that the need for interim storage can be limited to the long-lived waste that will be disposed of in SFL. The short-lived low-level and intermediate-level waste is sent directly to SFR after packaging.

Waste management

When dismantling and demolition is started in 2040, SFR is expected to be extended and SFL close to commissioning. This means that interim storage of waste intended for SFL might be necessary during a limited period.

Decommissioning of reactor plants in Forsmark

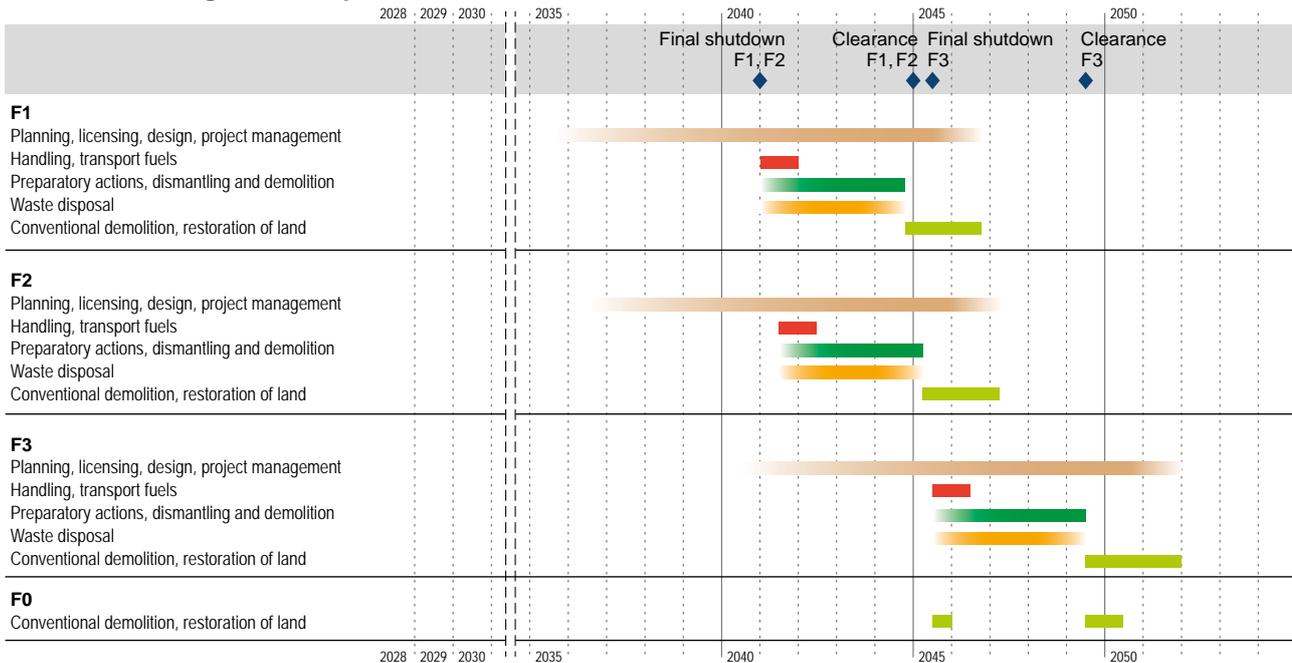


Figure 17-4. Schematic overview of Forsmarks Kraftgrupp AB's timeplan for decommissioning (F0 are common facilities that are described separately).

Decommissioning will yield a number of different waste streams. Fuel will be transported to Clink and then to the Spent Fuel Repository. Other waste will be sorted in short-lived waste and long-lived waste (to SFL), the short-lived waste that cannot be released from regulatory control will then be sorted depending on activity content and disposed of in near-surface repositories or in SFR.

The RPVs from F1–F3 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 SFR and some internals such as the core grid, core instrumentation, parts of the core shroud, core shroud head and core plate will be deposited in SFL.

17.3 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 with a commercial production of electricity. The Ågesta reactor was a heavy water moderated PWR reactor of 80 MW which provided Farsta with 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 17-5. The rock cavern together with a steel shell functioned 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 shell, are the control room, the control and switchgear building, and the transport tunnel and emergency exit.

Because the Ågesta plant is situated in a rock cavern, space and opportunities are limited for handling and interim storage of waste on site. Ågesta's location means that all transport must be on roads, with a possible exception for the RPV, which must initially be transported on roads but thereafter possibly by ship. The location near a densely populated area also means that the transports that need to be carried out will affect nearby residents and facilities to some degree.

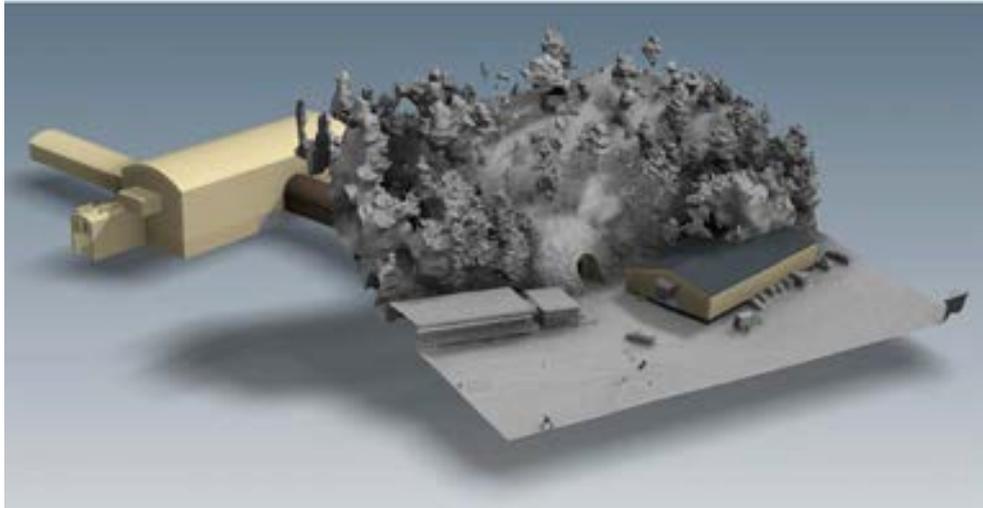


Figure 17-5. 3D view of the Ågesta reactor, which is situated in a rock cavern.

Overall planning

Vattenfall has chosen to carry out the decommissioning of Ågesta in programme form. A programme consists of a number of interconnected projects and studies. The work packages complement each other in order to gradually reach a long-term target outcome. The work packages in Vattenfall's decommissioning programme for Ågesta includes, among other things, preparation of licence documentation, procurement of demolition contractors, survey and characterisation of the facility. All definable work packages complement each other in order to achieve the target outcome of the programme: safe and effective decommissioning. The programme is controlled by the Business Unit Nuclear Decommissioning, BUND, where a part of the personnel is leased from AB SVAFO.

Vattenfall's ultimate goal is to remove all radioactive material so that the facility, i.e. the rock cavern, turbine hall, land and underground facilities can be released from regulatory control and the nuclear license can be revoked.

The current Environmental permit for the ongoing service operation of the Ågesta reactor expires in 2020. In 2018, all application documents that are required for the application for an environmental licence, Environmental Impact Statement and consultation process have been submitted to the Land and Environment Court. The aim is that the necessary licences and approvals according to the Environmental Code, the Nuclear Activities Act and the Radiation Protection Act will be in place during the latter part of 2019. Figure 17-6 shows the timeplan for planned decommissioning of the Ågesta reactor.

Dismantling and demolition begins when all licences have been obtained and the work can begin with regard to the time required for preparatory measures. The work is divided into two major steps, segmentation of the RPV on site in Ågesta and dismantling of other contaminated systems, components and structures. The work efforts within the two steps will be possible to perform at the same time to a large extent, since they do not disturb each other with regard to logistics, staffing, radiation doses etc. The opportunity to use competence and resources, as well as experience, efficiently can also be optimised as personnel is available in the facility for a number of projects.

Decommissioning of reactor plant in Ågesta

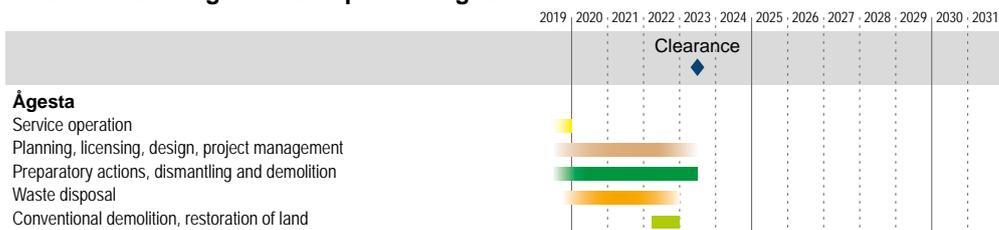


Figure 17-6. Schematic overview of the timeplan for planned decommissioning of the Ågesta reactor.

The RPV in Ågesta will be segmented on site. The step includes work such as removal of insulation and sanitation of asbestos, packing of waste in standard waste containers as well as scanning and transport to Studsvik for interim storage or further handling by Cyclife Sweden AB.

Waste management

The different waste streams that will be generated in conjunction with dismantling and demolition of the Ågesta reactor have been determined and for each waste stream different management steps have been identified until final clearance or disposal through one of the available disposal alternatives. The plan is to transport the waste from the area to interim storage in AB SVAFO's facilities in Studsvik awaiting commissioning of final repositories.

18 Planning for decommissioning of SKB's nuclear facilities

SKB's facilities are among the last nuclear facilities to be decommissioned in Sweden and the decommissioning lies far in the future. The decommissioning planning is for that reason described in very general terms and refers to existing decommissioning plans for more detailed information.

18.1 Central facility for interim storage and encapsulation of spent nuclear fuel

SKB is the licensee for Clab and will continue to be so when the integration of the planned encapsulation facility is completed and the plant is renamed Clink. The decommissioning plan for Clink was updated in 2013, in conjunction with the compilation of supplementary material for the licence application for Clink. During the RD&D period, another update will be made in conjunction with the development of the PSAR for the facility. Clink will be decommissioned when all spent nuclear fuel has been encapsulated and disposed of in the Spent Fuel Repository. The timeplan depends on when the last nuclear power reactor is finally shut down. According to current planning, decommissioning of Clink can commence around 2065 and be concluded within 5–7 years.

During the work of preparing the decommissioning plan for Clink, no reason has emerged why the decommissioning should be more complicated than for the other nuclear facilities whose decommissioning lies closer in time. It should be possible to carry out dismantling and demolition with a low dose to personnel, and the quantity of short-lived and long-lived radioactive waste that arises is expected to be limited. According to current plans, waste from dismantling and demolition will be transported to SFR for final disposal.

The goal of decommissioning is to remove all radioactive material and release the site from regulatory control. This means that all buildings, including equipment and land, will be released from regulatory control.

In 2013, SKB conducted a decommissioning study for Clink in order to provide waste inventory data as a basis for the extension of SFR and to estimate the costs for the Plan report (Edelborg et al. 2014).

In 2016, SKB updated the decommissioning plan for Clab. The update was made to harmonise with current regulations from SSM and to follow the established structure for a decommissioning plan (Calderon 2014).

18.2 Final Repository for Short-lived Radioactive Waste

During 2012–2013, in preparation for the application under the Nuclear Activities Act for the extension of SFR, SKB developed a new decommissioning plan for the facility (Calderon 2013).

Decommissioning of SFR will begin when operation ceases. 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. The parts of the facility that may be demolished in connection with decommissioning (the surface parts) are regarded as conventional, since they do not contain any radioactive material. A radiological survey of the facility will have to be done in order to rule out possible contamination of building parts that have been in contact with waste packages during operation, for example the terminal building. The goal of decommissioning is to release the facility from regulatory control. How far demolition should be carried beyond that depends mainly on how the site will be used in the future.

The timeplan for decommissioning of SFR is linked to when the last currently existing nuclear power plants and SKB's other nuclear facilities are dismantled and released from regulatory control. With current planning, decommissioning of SFR will commence at the end of the 2060s.

In 2018, prior to the periodic overall assessment and report of the facility's safety and radiation protection, SKB formulated a new decommissioning plan for the existing SFR (Calderon 2018). The update was also made to harmonise with SSM's regulations and to follow the industry-wide structure for a decommissioning plan.

During the RD&D period, SKB plans to update the decommissioning plan for the extended SFR facility. The update is made in conjunction with the preparation of a preliminary safety analysis report (PSAR) for the facility.

18.3 Final Repository for Long-lived Waste

No decommissioning plan has yet been prepared for SFL, since the design of the facility is in the concept stage. A decommissioning plan will be prepared in conjunction with the preparatory PSAR (F-PSAR) for the facility. The decommissioning will start in conjunction with repository closure, which is planned to take place in the mid-2050s, see Section 3.3.4.

18.4 Spent Fuel Repository

A decommissioning plan was initially prepared for the applications under the Nuclear Activities Act for final disposal of spent nuclear fuel and under the Environmental Code for the KBS-3 system (Hallberg and Tiberg 2010).

An update of the decommissioning plan was made in 2017. The update was made to harmonise with current regulations and to follow the industry-wide structure for a decommissioning plan.

Decommissioning begins after operation is concluded, i.e. when all spent nuclear fuel has been disposed of and the deposition tunnels have been backfilled and sealed. Decommissioning entails sealing the remaining parts of the underground facility and demolishing the surface facilities. Sealing the underground facility is a part of the repository's barrier function and is of importance for post-closure safety. SKB's work with closure is described in Section 11.8.1.

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

19 Continued activities within decommissioning

This chapter provides an overview of the completed and planned development work related to decommissioning of nuclear facilities. Section 19.1 presents completed and planned industry-wide development work. Sections 19.2 and 19.3 present completed and planned development work at Uniper and Vattenfall.

19.1 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. Several of the activities are therefore described in more detail in Part I and part II, see references.

19.1.1 Long-lived waste

Current situation

In order to be able to release the Barsebäck nuclear power plant from regulatory control after completed decommissioning, there has been interest in a solution to avoid interim storage of nuclear waste at the facility for a long time. During the past RD&D period, the possibility to interim-store waste in SFR until SFL is commissioned has been analysed. The conclusion of the analysis is that interim storage will not be carried out in SFR but must be localised at the nuclear power plants or externally. Interim storage of long-lived waste is also described in Section 3.3.4.

Before final conditioning of long-lived waste can be carried out, acceptance criteria for the waste have to be established. During the past RD&D period, a safety evaluation of the proposed final repository concept for SFL was carried out. The safety evaluation constitutes a basis and is a prerequisite for the development of acceptance criteria. The work of developing preliminary acceptance criteria will continue during the RD&D period, see also Section 3.3.4.

The power companies that have initiated segmentation of reactor internals and have a need to store long-lived waste have, during the past RD&D period, worked with developing waste type description specifications (TBS). Waste type description specifications for the waste types B100 and O100, at BKAB and OKG, have been developed and a decision by SKB to allow waste packages to be manufactured provided that the waste cannot be finally conditioned has been obtained.

Programme

A large portion of the long-lived waste from the nuclear power plants will arise before the planned commissioning of SFL. This means that sufficient planning premises need to be established in conjunction with the project planning for decommissioning at the nuclear power plants in order to minimise the risk of needing to recondition the waste after it has been placed in interim storage facilities. The ongoing planning of SFL provides important conditions for the decommissioning planning of the nuclear power plants.

19.1.2 Management of BWR pressure vessels and large components

In 2017, Vattenfall and Uniper decided not to dispose of whole BWR pressure vessels in SFR; they will instead be segmented before disposal.

During the RD&D period, efforts have been made, based on the decision to segment the BWR pressure vessels for SFR, to optimise the handling chain of the BWR pressure vessels from handling at the nuclear facility to final disposal in SFR. The efforts have identified possible waste containers for the licensees to use for waste to the waste vault for reactor pressure vessels (BRT) in the extended SFR facility.

19.1.3 Development of concrete structures and materials for SFL

During the past RD&D period, SKB has conducted a study concerning the choice of materials and methods for constructing the foundation for concrete structures and backfilling the waste vault for core components with concrete, see further Section 10.3. SKB has also commissioned a study concerning interactions between cement and bentonite for the proposed design of the waste vault for legacy waste (BHA) in SFL, see further Section 10.1.5.

Given the long periods of time that remain until construction of SFL, SKB currently sees no need for a special experimental development programme concerning the design of materials and technology for the construction of concrete structures and backfilling of SFL.

19.1.4 Very low-level waste

In recent years, environmental issues have received increased focus in the industry and several development and improvement measures linked to the very low-level waste have been adopted by the waste producers. The measures aim to better follow the waste hierarchy through an increased focus on preventive measures, reuse and recycling, and to improve the long-term safety. The industry has also worked to increase the dialogue with regulatory authorities and other stakeholders. Optimising the conditions for continued development of the management of very low-level waste requires a number of different activities, which are described in Section 3.2.

19.1.5 Waste containers and waste transport containers

Work has been carried out to develop waste containers and waste transport containers to facilitate optimal management of the waste during dismantling and demolition. This work is described in Section 7.5.

Programme

ATB 1T, which will be used for transportation of steel tanks during the decommissioning projects, is under development in cooperation with the American company Holtec. The new waste transport container is planned to be certified and commissioned by 2020, see Section 7.5.3.

A prerequisite for transporting the nuclear waste from the nuclear power plants to the final repositories is that there are appropriate waste transport containers. A study to identify whether SKB's transportation system should be supplemented with additional waste transport containers is intended to be initiated during the RD&D period.

19.1.6 Non-regular fuels

A prerequisite for being able to begin dismantling and demolition in a decommissioning project is that all spent fuel has been transported from the facility. This includes failed fuel in the facility. A project with the objective of managing all failed fuel in the facilities has been in progress during the RD&D period.

The project has managed and removed all failed fuel from the Oskarshamn reactors O1 and O2. The failed fuel has been transported to the Studsvik site for treatment, after which a subset has been encapsulated in special containers (so-called transport boxes) and transported to Clab for interim storage. In connection with transporting failed fuel to Studsvik, studies of selected parts of the failed fuel will be initiated during the RD&D period, see Section 8.1.

In Ringhals, failed fuel from the reactors R2, R3 and R4 has been managed and placed in five specially designed containers (Quivers). Three of these Quivers have been sealed and are ready for transport to Clab for interim storage. Consequently, R2 is emptied of failed fuel. R3 and R4 each have a Quiver which has been kept open in order to manage failed fuel rods in the future. Ringhals 1 was emptied of failed fuel in a pilot project before the start of project Failed Fuel.

In Forsmark, a number of failed fuel rods from the reactors F1 and F2 have been managed using Quivers.

With the introduction of Quivers, SKB has established a new, practical, safe and long-term cost-effective method for managing failed fuel rods at the nuclear power plants, see also Section 8.3.

Programme

During the RD&D period, the remaining main tasks in project Failed Fuel will be completed. This includes, among other things:

- Managing failed fuel rods in Oskarshamn 3 and Forsmark 3 using the specially designed Quiver containers.
- Managing failed fuel rods from Forsmark 2 and 3 using the specially designed containers transport boxes.
- Transporting filled Quivers from Ringhals, Forsmark and Oskarshamn to Clab and transporting filled transport boxes from Studsvik to Clab.

19.1.7 Harmonised licensing

A fundamental prerequisite for being able to execute the planned decommissioning projects is that the necessary licences are obtained. Since several projects will be carried out in parallel, and it is important that milestones are reached in the specified time, there is a benefit to all licensees together with the concerned bodies having a uniform process.

During the past RD&D period, work has been carried out within a number of areas in order to achieve harmonised licensing. Work has mainly been carried out through the Nuclear power industry's safety coordination group (KSKG) and the development of common positions, so called position papers (KPP). Position papers have been developed in the following areas:

- Safety analysis reports during decommissioning of a nuclear facility – overall requirements and content.
- Principles for categorisation of extremely low risk of radioactive contamination (ELR).
- Steps.
- Inspection programmes for clearance.

Programme

The work with finding common working methods for priority issues among the licensees will continue during the RD&D period. Examples of areas where it is necessary to agree upon common approaches and work processes within the industry are:

- Methodology and consensus for the preparation of waste acceptance criteria, waste type descriptions and waste type description specifications WAC/TB/TBS for extended final repository parts and decommissioning waste.
- Decommissioning report/archiving prior to the expiration of a licence under the Nuclear Activities Act and the Radiation Protection Act.

19.1.8 IT support for decommissioning documentation

During the past RD&D period, SKB has initiated a dialogue with the licensees to coordinate the work with radiological surveys and characterisation of decommissioning waste for the licensees that are about to initiate dismantling and demolition of nuclear power reactors.

Programme

The information required for characterisation of decommissioning waste will, in some respects, differ from the information required for operational waste. For decommissioning waste, information regarding radionuclide and material composition will be largely based on results from the radiological survey that is performed before the waste is produced. It is therefore important to document information that makes it possible to map waste in individual packages to information obtained during the radiological survey. SKB is working to make it possible to receive survey information and updated forecasts for decommissioning waste to thus devise a suitable format for a future waste registry for decommissioning waste, see also Section 7.2.1.

19.1.9 International development work

During the period, SKB and the nuclear power companies have monitored 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's collaborative programme but also the IAEA's programme is of importance. The latter focuses to a greater extent on the development of the IAEA's Safety Standards, which underlie the member countries' requirements.

Within the OECD-NEA, the work within the area of decommissioning has intensified as the first generation of commercial reactors approach the end of their service life. A focus area for OECD-NEA during the period is how to optimise the management of very low-level waste and clearable materials during decommissioning of nuclear facilities.

Collaboration between the Swedish nuclear power industry and the Spanish Enresa started during the RD&D period with the purpose of exchanging experience and knowledge regarding decommissioning of nuclear facilities.

Programme

SKB and the nuclear power companies will continue to participate in the international networks on decommissioning, which gives both benefits and opportunities to contribute experience. There is potential for a deeper exchange in the future as several decommissioning projects have started in Sweden. More experience is then built up within the country and the need to gather information increases.

For example, participation will take place within OECD-NEA and CDLM (Committee on Decommissioning of Nuclear Installations and Legacy Management). Since CDLM is newly started, there is a possibility to influence the development within the areas where CDLM will conduct work.

The collaboration and knowledge exchange between Enresa and the Swedish nuclear power industry will also continue during the RD&D period.

19.1.10 Method development for difficult-to-measure nuclides

Programme

Within the framework of the radiological survey conducted prior to dismantling and demolition, the models for radiological calculations will be updated. Furthermore, the measurements that are performed within the radiological survey will be used to verify the calculated results and to potentially revise the computational models. An updated inventory for difficult-to-measure nuclides in the decommissioning waste can then be determined from calculated and measured values. SKB has a continuous dialogue with the licensees in order to ensure that the information gathered during radiological survey can be utilised for characterisation of the decommissioning waste intended for to be disposed of in SFR or SFL, see also Section 7.2.2.

19.2 Development within Uniper

The Project Performance Center (PPC) was established in 2017 with the purpose of coordinating Uniper's decommissioning portfolio. PPC has so far focused on different scenarios for decommissioning regarding the execution time and how these scenarios relate to the technical sequence, organisation, waste management and financing. The work has led to a preferred alternative where the decommissioning programmes for B1/B2 and O1/O2 are carried out with a critical path that is sequential between the facilities.

Barsebäck Kraft AB

BKAB has, based on the collaboration with OKG, developed the decommissioning planning concerning dismantling, radiological survey, waste management, interim storage, organisation and control. This is described in the supplemented decommissioning plan, the waste management plan and the SAR for dismantling and demolition.

Examples of completed activities:

- The reactor internals are segmented at B1 and B2. The waste is stored in an interim storage facility on site. Planning for external interim storage is in progress.
- Objects (buildings or part of building) that have been categorised as having an extremely low risk of contamination (ELR) have been surveyed and verified.
- An in-depth radiological survey of materials is under way with the purpose of directing the waste to the most appropriate management route.
- Waste type description specifications have been prepared for decommissioning waste.
- The technology and process for management of the radiological waste have been determined.

OKG Aktiebolag

Adaptation of the facilities and separation measures are ongoing. The intensive planning work that was initiated prior to the previous RD&D period has continued. Several studies have been carried out on both the strategic and the tactical levels, in some cases with associated execution of actual activities in the facilities. The details are described in OKG's decommissioning planning, which was presented in conjunction with the licensing processes that have taken place. They can be summarised as follows:

- There is a dedicated department for decommissioning (department A) of OKG's facilities. The department was established in the autumn of 2016. The design of the department has been flexible based on its main activities and has been adapted thereafter. Adaptation of the organisation will also continue.
- The strategic work and the overall decommissioning planning have continued and will continue. The work is being pursued in cooperation with BKAB within the framework of the PPC.
- The work with licence management has continued and all licences for starting dismantling and demolition have been obtained for O1 and O2 including OAVF.
- Actual physical activities are, for example:
 - The facilities have been emptied of fuel.
 - Survey of the facilities in the form of characterisation and categorisation of materials. The survey has thereafter continued with buildings and building parts.
 - Reactor pressure vessel internals have been segmented at O2 and thereafter at O1 (in progress at O1).
 - Complete system decontamination has been carried out at O2. Planning is under way for execution at O1.
 - Several adjustments of the facilities and separation measures have been carried out and will continue.

Programme

The development need is considered to be the same, in principle, for both BKAB and OKG, where the focus lies on continuing the adaptation to the process for decommissioning and then further development of this process. The main activities are:

- The strategic work will continue during decommissioning but with a changed focus to consensus-building, implementation and follow-up.
- The previous overall planning is gradually moving to design and detailed design and thereafter to execution, as the technical sequence progresses.
- Further development of the different functional sub-processes of the decommissioning – material and waste management, dismantling and demolition, operation and maintenance plus protection and safety. This can for example include development and optimisation of facility surveys, near-surface repositories, interim storage facilities, clearance, logistics etc.
- Adaptation and development of the organisation and management system will continue.
- Preparation of readily accessible and functional experience feedback between BKAB and OKG in order to achieve learning benefits in the technical sequence, and long-term experience feedback to the decommissioning of O3 and O0.

19.3 Development within Vattenfall

Vattenfall has made a strategic decision that all decommissioning should be coordinated within the Vattenfall Group through the business unit BUND. This creates opportunities for synergies between presently ongoing decommissioning work, the R2 reactor in Studsvik, and the future decommissioning of Ågesta, Ringhals 1 and Ringhals 2. Furthermore, it creates good conditions for development and coordination between the decommissioning in Germany and Sweden.

Since the previous RD&D programme, the organisation and management has been built up and a management system is implemented in the organisation. Moreover, different collaboration forms/projects within the group have been developed as support for management and control, for example:

- Strategic forum for competence and staffing issues. The goal is to arrive at the best possible solutions for the coming years' need for competence within operation and decommissioning.
- In conjunction with the shutdown decision for Ringhals 1 and 2, project STURE (Safe and Secure Phase Out of Ringhals Reactors 1 and 2) was started. Apart from physical separation measures, STURE was also responsible for the licensing process required for shutdown of Ringhals 1 and Ringhals 2. This licence was obtained in 2018.

Vattenfall has also devised a training programme together with KSU called "Decommissioning basis". The purpose of the programme is to strengthen the competence within the organisation and provide a common platform for decommissioning issues.

Furthermore, Vattenfall has developed decommissioning concepts both technically and commercially and decided that decommissioning of Ågesta and Ringhals will be carried out in programme form. The strategic focus of the programmes is also decided.

Programme

A large part of the development work within the Vattenfall Group will be carried out as an integrated part of the programmes prior to and during decommissioning of Ringhals 1, Ringhals 2 and Ågesta. Prior to the dismantling and demolition phase for Ringhals 1 and Ringhals 2, during the coming three-year period, a considerable amount of preparatory waste studies will be pursued. It will be particularly urgent to study the following areas:

- Clearance (of systems, buildings and land), including decontamination and measurement system.
- Internal transportation, as well as interim and decay storage.
- Waste process optimisation.
- Packaging and conditioning optimisation.
- Volume optimisation and final repository links (very low-level, short-lived and long-lived waste).

In addition to waste-related issues, a range of other development activities are also planned, such as:

- Developing the cost model for calculating and evaluating costs for decommissioning of a nuclear facility. The plan is that the model will present risks and uncertainties and provide an opportunity to carry out sensitivity analyses.
- Further development and adaptation of portfolio, programme and project processes. This aims to ensure, among other things, that the decommissioning activities are optimised regarding both time and pace so that the actual work is conducted safely and efficiently.
- Further development of the management system in order to ensure functional control of the decommissioning activities so that the work can always be carried out safely and efficiently.
- In all relevant areas, gathering external and internal experience and adapting it to decommissioning activities to thus establish the best practice. Experience outside the nuclear sector is also of interest here, not least regarding safety and working environment during decommissioning.

- As a step in ensuring that there are sufficient resources with the appropriate competence available at Vattenfall in order to complete the planned activities for managing the waste, a strategic competence plan will be devised. This also includes the organic build-up of competence; decommissioning competence and experience obtained within the group during decommissioning of the R2 reactor in Studsvik should be possible to transfer to the decommissioning of Ågesta, Ringhals 1 and Ringhals 2.

During the subsequent three-year period, the planned decommissioning of Ågesta will be completed and decommissioning of Ringhals 1 and 2 will be in progress. The development needs for the period will thereby be strongly linked to the experience gained during decommissioning of Ågesta, and the needs identified during the final planning for the decommissioning of Ringhals 1 and 2.

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Abbreviations

OAVF	Shared waste facility at the Oskarshamn Nuclear Power Plant
ABM	Alternative Buffer Materials. Experiments in the Äspö HRL in which potential buffer materials are studied.
ALARA	As Low As Reasonably Achievable. Limiting radiation doses as far as reasonably possible considering both financial and societal factors.
AM	Active interim storage facility at the Studsvik site.
Andra	Agence nationale pour la gestion des déchets radioactifs. Organisation responsible for final disposal of radioactive waste in France.
Asha	Indian bentonite from the Kutch region.
ATB	Waste transport container.
ATB 1T	A new container for transport of long-lived, low-level and intermediate-level waste in BFA tanks.
B1	Nuclear power reactor Barsebäck 1.
B2	Nuclear power reactor Barsebäck 2.
Beacon	Bentonite Mechanical Evolution. EU project.
BELBaR	Bentonite Erosion: effects on the Long term performance of the engineered Barrier and Radionuclide transport. EU project.
BFA	Rock vault for interim storage of low-level and intermediate-level waste on the Simpevarp Peninsula in Oskarshamn.
BHA	Waste vault for legacy waste in SFL.
BHK	Waste vault for core components in SFL.
BKAB	Barsebäck Kraft AB.
BLA	Waste vault for low-level waste in SFR. SFR contains one waste vault for low-level waste (1BLA) and four additional waste vaults (2 – 5BLA) are planned in the extended part of SFR.
BMA	Waste vault for intermediate-level waste in SFR. SFR contains one waste vault for intermediate-level waste (1BMA) and one additional waste vault (2BMA) is planned in the extended part of SFR.
BMWi	Bundesministerium für Wirtschaft und Energie. Department responsible for nuclear energy in Germany.
BTF	Concrete tank repository in SFR, mainly intended for dewatered ion exchange resins.
BRIE	Bentonite Rock Interaction Experiment. Experiment in the Äspö HRL.
BUND	The Business Unit Nuclear Decommissioning in Vattenfall.
BWR	Boiling Water Reactor. The reactors in Forsmark and Oskarshamn and reactor 1 in Ringhals are boiling water reactors.
Calcigel	Bentonite from Germany.
CEC	Cation Exchange Capacity.
CEMI	Centre for Excellence in Mining Innovation. Canada.
Clab	Central interim storage facility for spent nuclear fuel.
Clink	Central facility for interim storage and encapsulation of spent nuclear fuel.
COMSOL	Calculation tool for modelling and simulation of complex physics-based systems. COMSOL Inc.

0AVF	Shared waste facility at the Oskarshamn Nuclear Power Plant
COVRA	Centrale Organisatie Voor Radioactief Afval. Organisation responsible for final disposal of radioactive waste in the Netherlands.
CR	Concentration Ratios.
CSH	Calcium Silicate Hydrate.
DBD	Deep Borehole Disposal
DFN	Discrete Fracture Network.
Disco	Modern Spent Fuel Dissolution and Chemistry in Failed Container Conditions. EU project.
DOC	Dissolved Organic Carbon.
Domplu	Dome Plug Experiment. Full-scale test in the Äspö HRL to test and demonstrate the complete plug system.
EIS	Environmental Impact Statement.
ELR	Categorisation of contamination – extremely low risk of contamination.
EmrasII	Environmental Modelling for Radiation Safety. IAEA project.
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.
ERICA	Environmental Risk from Ionising Contaminants. Calculation tool for assessing biological effects of ionising radiation in habitats and ecosystems.
ESS	European Spallation Source. Spallation facility in Lund, which is constructed, owned and operated by a European consortium for research infrastructure (ERIC).
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.
FEP	Features, Events and Processes. Features, events and processes that may affect the safety after closure of a final repository.
F-PSAR	Preparatory preliminary safety analysis report.
FSW	Friction Stir Welding.
GAP	Greenland Analogue Project.
GIA	Glacial Isostatic Adjustment.
GIS	Geographic Information System.
GPR	Ground Penetrating Radar.
GRASP	Greenland Analogue Surface Project.
HM	Hydromechanical properties, factors or processes.
HPC	High Performance Computing.
IAEA	International Atomic Energy Agency.
ICE	Greenland ICE Project. Glacial-hydrological study that complements the studies in GAP.
IPCC	Intergovernmental Panel on Climate Change. United States climate panel.
ISA	Isosaccharinic acid.
ISO container	Containers in sizes standardised by the International Organization for Standardization (ISO), which can be loaded on railroad cars, trucks and cargo ships.
Kaeri	Korea Atomic Energy Research Institute. Nuclear research institute in South Korea.

0AVF	Shared waste facility at the Oskarshamn Nuclear Power Plant
KBS-3 method	The KBS-3 method has been given its name because it is based on the third report in the KärnbränsleSäkerhet (Nuclear Fuel Safety) Project.
KBS-3H	The KBS-3 method with horizontal deposition.
K_d	Sorption coefficient, distribution coefficient.
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.
LOT	Long term test of buffer material. Experiment in the Äspö HRL aimed at finding out how primarily bentonite clay behaves in conditions similar to those in a final repository for spent nuclear fuel.
LTDE-SD	Long term diffusion experiment – Sorption-diffusion. Concluded experiment in the Äspö HRL.
LVDT	Linear Variable Differential Transformer.
MARFA	Migration Analysis of Radionuclides in the Far Field. Calculation tool for radionuclide transport modelling.
MIKE SHE	Calculation tool for modelling of hydrogeology and surface hydrology.
Mind	Microbiology In Nuclear waste Disposal EU project.
MiniCan	Miniature Canister Corrosion Experiment. Experiment in the Äspö HRL.
MIRARCO	Mining Innovation Rehabilitation and Applied Research Corporation. Canada.
MIS	Marine Isotope Stages.
MMD	The Land and Environment Court.
MODARIA	Modelling and Data for Radiological Impact Assessments. IAEA project.
Modern2020	Development and Demonstration of monitoring strategies and technologies for geological disposal EU project.
MoFrac	Calculation tool for DFN modelling.
MOX	Mixed Oxide Fuel.
MX-80	Sodium bentonite from Wyoming, USA.
Nagra	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.
NWMO	Nuclear Waste Management Organization. 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 is constructing in Olkiluoto since 2004. Onkalo is used for research and development, but will also provide access to the actual final repository.
PAN	Polyacrylonitrile.
PHREEQC	Calculation tool for transport and geochemistry modelling.

0AVF	Shared waste facility at the Oskarshamn Nuclear Power Plant
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.
PURAM	Public Limited Company for Radioactive Waste Management. Organisation responsible for final disposal of radioactive waste in Hungary.
PWR	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.
RCP	Representative Concentration Pathways. Scenarios concerning conceivable future concentrations of atmospheric greenhouse gases, providing different cases for how much the greenhouse effect may increase.
Redox	A redox reaction is a chemical reaction where an element is reduced while another element is oxidised.
RK&M	Preservation of Records, Knowledge and Memory across Generations Concluded project within the OECD-NEA.
RWM	Radioactive Waste Management. Organisation responsible for the management of radioactive waste in the UK.
RWMC	Radioactive Waste Management Committee. OECD-NEA.
SAR	Safety Analysis Report.
Scale	Calculation tool for criticality and decay heat calculations.
Scip	Studsvik Cladding Integrity Project. Project led by the OECD-NEA and Studsvik Nuclear AB.
SFL	Final Repository for Long-lived Waste.
SFR	Final Repository for Short-lived Radioactive Waste.
SKB	Svensk Kärnbränslehantering AB.
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 safety after closure for the SFR extension. Published in August 2015.
SR-Site	Report on the safety after closure of the Final Repository for Spent Nuclear Fuel in Forsmark.
SSM	Swedish Radiation Safety Authority.
SSMFS	Regulations 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 final repository.
Task Force GWFTS	The 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.

0AVF	Shared waste facility at the Oskarshamn Nuclear Power Plant
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.
TBS	Waste type description specifications.
THM	Thermal, Hydraulic, Mechanical (properties, factors or processes).
TURVA-2012	Report on the safety after closure of the final repository for spent nuclear fuel in Olkiluoto, Finland. Published by Posiva in 2012.
UFM	Nearly universal fracture model. Method for generating a fracture network in modelling of a rock volume.
U.S. DOE	United States Department of Energy.
U.S. DOT	United States Department of Transport.
U.S. NRC	United States Nuclear Regulatory Commission.
WAC	Waste Acceptance Criteria.
WP-cave	A concept for a compact geological final repository for spent nuclear fuel, which has previously been proposed as an alternative to the KBS-3 method.
XRD	X-ray diffraction.

