Technical Report
TR-16-15
September 2016

RD&D Programme 2016

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

SVENSK KÄRNBRÄNSLEHANTERING AB

SWEDISH NUCLEAR FUEL
AND WASTE MANAGEMENT CO

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

ISSN 1404-0344
SKB TR-16-15
ID 1568615
September 2016

RD&D Programme 2016

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

Preface

The licensees for the Swedish nuclear power plants are responsible for safe management and disposal of nuclear waste and spent nuclear fuel and also for safe decommissioning and dismantling of facilities. Under the Nuclear Activities Act, the licensees for the Swedish reactors shall every third year present a programme for the research and development activities and other measures needed to manage and dispose of the nuclear waste and the spent nuclear fuel in a safe manner and to decommission the nuclear power plants (this is presented in the RD&D programme). SKB, Svensk Kärnbränslehantering AB, is owned by the nuclear power companies and manages and dispose of nuclear waste and spent nuclear fuel from the reactors. The licensees have delegated to SKB to, on their behalf, develop and submit the RD&D programmes to the Swedish Radiation Safety Authority (SSM). To comply with the requirements of the Nuclear Activities Act SKB is now presenting the RD&D Programme of 2016 (programme for research, development and demonstration) which has been produced in collaboration with the nuclear power companies.

As part of the Governments decision on the RD&D programme of 2013, terms on future RD&D programmes were expressed. To meet these terms in the RD&D Programme of 2016, consultation meetings with SSM have been conducted. SSM has presented their expectations on the RD&D Programme of 2016 in a memorandum which has been discussed at these consultations and addressed during the work on the RD&D programme of 2016. These expectations are one of the reasons for modifying the structure of the RD&D programme of 2016. The new structure includes a refined plan of action for research and development activities, based on the needs for facilities and final repositories. Furthermore, decommissioning is presented in a separate part.

Another reason for the modified structure of the RD&D Programme of 2016 is the special situation with two simultaneously ongoing licensing matters in the Land and Environment Court and at SSM. One application for the KBS-3 system (final repository system for spent nuclear fuel) and one application for the extension of the final repository for short-lived radioactive waste, SFR. To avoid redundancy in the present RD&D programme, references are made to the applications or other information which have been submitted to SSM and the Land and Environment Cour where necessary.

In parallel with these licensing processes, SKB is conducting the research and technology development needed to design, construct and operate the KBS-3 system and the extended SFR in a rational manner while requirements on post-closure safety, low radiation dose during operation of the facilities and limited impact on the external environment are met. Progress in ongoing licensing process and its outcome, including any licence conditions, may affect SKB's timetables and prioritisation of research and development over the coming years. In this context, it should be noted that SKB in the RD&D Programme of 2016 has not been able to fully evaluate and incorporate the comments which SSM presented in its statement on the KBS-3 application to the Land and Environment Court in June 2016.

Of SKB's repositories, the final repository for long-lived waste (SFL) is planned to be commissioned last. In order to be able to manage and dispose of the long-lived waste, the development and research that is required for this work to progress according to plan is under way.

When it comes to decommissioning of nuclear facilities, the measures for dismantling and demolition of the nuclear power reactors and the development work that is planned are presented in this programme.

Stockholm, September 2016
Svensk Kärnbränslehantering AB

Christopher Eckerberg
CEO

Anders Ström
Sponsor RD&D Programme 2016

Summary

The Swedish system for management and disposal of radioactive waste consists of two main parts: one part for low- and intermediate-level waste and one part for spent nuclear fuel. The interim storage facility for spent nuclear fuel (Clab) and the final repository for short-lived radioactive waste (SFR) have been in operation since the 1980s, along with the transportation system. Other facilities or facility parts remain to be built. At the same time, planning and development of methods for decommissioning of the nuclear power reactors continue.

In the RD&D Programme2016, the licensees for the Swedish nuclear reactors and SKB present plans for research, development and demonstration during the period 2017–2022. In order to fulfil the conditions imposed by the Government on clearly stating the arguments for planned efforts as well as developing the information on decommissioning, the reporting of the RD&D Programme 2016 has a new structure compared with the RD&D Programme 2013. Consultations were held with the Swedish Radiation Safety Authority (SSM) regarding the new structure and planning for decommissioning.

The RD&D programme 2016 consists of four parts:

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

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

Part II describes planned research and technology development during the RD&D period. The focus lies 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 the low- and intermediate-level waste, the spent nuclear fuel and the different parts of the repository system, which means that the research and technology development is described in an integrated way for the three repositories. The state of knowledge is described in general terms and provides references to more detailed reporting of results in the background reports.

Part III presents plans for decommissioning of the nuclear power reactors and SKB's facilities and the dependencies and flexibility in the system, as well as the development work that is planned.

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

Part I Activities and plan of action

The Swedish power industry has been producing electricity by means of nuclear power for more than 50 years. A large part of the system needed to safely manage and dispose of the radioactive waste and the spent nuclear fuel from operation of the reactors has already been built up.

The facilities in the Swedish system that are in operation today 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, plus the ship m/s Sigrid. The present report provides an overall description of these. SKB and the nuclear power operators have as licensees for the facilities in operation an obligation to, for each of them, submit various information and documents to SSM that are not discussed in this RD&D programme.

What remains to be done for final management of spent nuclear fuel is to construct and commission the system of facilities needed for final disposal. This includes a new facility part for encapsulation of the spent nuclear fuel adjacent to Clab (the integrated facility is called Clink), transport casks for shipping encapsulated spent nuclear fuel and a final repository for spent nuclear fuel.

For safe management and disposal of the low- and intermediate-level waste, SFR needs to be extended, an additional final repository – the final repository for long-lived waste (SFL) needs to be established and waste transport containers for shipment of long-lived waste be obtained. Furthermore, the NPPs' waste management needs to be adapted so that decommissioning can be carried out according to plan.

Planning premises

Prior to the RD&D Programme 2016, the premises are unusual with two simultaneously ongoing licensing matters in the Land and Environment Court and at SSM, one for the KBS-3 system (final repository system for spent nuclear fuel) and one for the extension of SFR for operational and decommissioning waste. In order to avoid redundancy in the present RD&D programme, references are, if necessary, made to the applications or other information which has been submitted to the Land and Environment Court or SSM.

Furthermore, during 2015 decisions have been made on a premature shutdown of four nuclear power reactors (Oskarshamn 1, Oskarshamn 2, Ringhals 1 and Ringhals 2). This means that the total amount of fuel that will be managed within the framework of the programme decreases. However, a larger quantity of fuel will arrive earlier to Clab when the final fuel cores from the four reactors are transported there. Another consequence is that dismantling and demolition of seven reactors will begin before the extension of SFR is commissioned. This means that larger quantities of radioactive waste need to be interim-stored at the nuclear power plants. The load on the transportation system will increase when the extension of SFR and the Spent Fuel Repository is commisioned.

Stepwise process

When establishing new facilities, planning follows a stepwise decision process based on requirements in the Nuclear Activities Act and SSM's Regulations. The procedure implies that SKB needs to submit increasingly detailed safety analysis reports to SSM.

- First, *applications under the Nuclear Activities Act and the Environmental Code* to construct, own and operate the facility must be submitted to SSM and the Land and Environment Court. The Government decides on permissibility and licensing, whereafter conditions are determined by SSM and the court.
- *Before a nuclear facility may be constructed*, a preliminary safety analysis report (PSAR) should be approved by SSM.
- When systems and processes in the facility have been tested and work as intended, the safety analysis report (SAR) should be renewed. When SSM has approved the SAR, *trial operation* can commence. Before a plant can be put into *regular operation*, the SAR must be supplemented with experience from trial operation and approved by SSM.

Plan of action

There are large uncertainties in the time schedules for both the KBS-3 system and the SFR extension, particularly during the licensing phase when SKB does not control the course of events. Nevertheless, SKB has a thorough plan of action based on current premises.

Licensing of the KBS-3 system's facility parts for encapsulation and final disposal of spent nuclear fuel is under way. SKB has answered questions and requests for supplementary information to both

SSM and the Land and Environment Court reviewing bodies. An announcement of the applications under the Environmental Code and the Nuclear Activities Act was done in January 2016. SSM offered its statement in the Environmental Code case to the Land and Environment Court in June 2016. SKB is currently developing the basis for assessment so that the PSAR for the Spent Fuel Repository and Clink can be submitted to SSM during the present RD&D period.

The spent nuclear fuel is interim-stored in Clab. During the time it takes to construct and commission the encapsulation part, the quantity of spent nuclear fuel that needs to be interim-stored in the facility will exceed the current licence of 8,000 tonnes. SKB has therefore, as part of the application to construct and operate Clink, also applied to increase the interim storage capacity to 11 000 tonnes. During the RD&D period, SKB plans to submit a PSAR for Clab with interim storage capacity of 11 000 tonnes to SSM.

SKB submitted in 2014 licence applications under the Nuclear Activities Act and the Environmental Code to extend SFR to be able to dispose of short-lived operational and decommissioning waste. The licensing process continues and SKB is regularly answering questions and requests for supplementary information. The next step is to develop the basis for assessment so that a PSAR for the extended SFR can be submitted to SSM during the present RD&D period.

The Spent Fuel Repository and the extended SFR are planned to be commissioned around 2030.

SFL is the last repository to be commissioned. Up to commissioning, there are several important milestones that must be passed, such as safety evaluation of the repository concept, the site selection, assessment of post-closure safety, preparation of licence applications and construction of the repository. SKB plans to submit licence applications under the Nuclear Activities Act and the Environmental Code to construct, own and operate the repository so that SFL can be commissioned around 2045.

Further research and technology development

SKB's and the nuclear power plant licensees' planning of future research and technology development for the final repositories is based on the stepwise decision process. Milestones related to decisions steps in the form of applications and safety analysis reports control when knowledge and development of the technology needs to have reached a certain level, while SSM's approval controls when SKB can commence construction and operation of the facilities.

SKB has as part of the applications for a licence to construct the Spent Fuel Repository, the encapsulation part of Clink and the extension of SFR compiled reviews of the state of knowledge and the status of technology development. In conjunction with this, the significance of remaining uncertainties on fulfilment of the requirements on protection of human health and the environment against radiation after closure of the repository have been evaluated. For SFL a concept study has been carried out and a repository concept has been chosen for which a safety evaluation is currently under way. The safety evaluation will be finished in 2018 and the results will clarify the need for future research and technology development for SFL.

This information, together with the comments that have been submitted in conjunction with the reviewing of the applications and reviews of previous RD&D programmes, are basis for the planned activities for research and technology development within the different disciplines. 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 repository system and thereby the basis in order to judge their importance for post-closure safety.
- The need for knowledge and competence around design, construction, manufacture and installation of the components to be used in the facilities.
- The need for knowledge and competence of inspection and testing to verify that barriers and components are produced and installed according to approved specifications and thereby fulfil the requirements.

Furthermore, better information of the inventory of radionuclides in the low- and intermediate-level waste is needed, as well as the properties of the spent nuclear fuel and development of technology for handling of both nuclear waste and spent nuclear fuel.

Based on this, the research and technology development needed to address the repository design and construction issues, as well as the research that is needed to carry out assessment of the safety of the repositories after closure, has been identified and justified.

Part II Waste and final disposal

The comprehensive research, development and planning conducted over four decades has led to many issues of importance for the nuclear waste programme being resolved. The identified need for research and development for the remaining parts of the nuclear waste programme is presented here.

Low- and intermediate-level waste

Research on low- and intermediate-level waste is being carried out to obtain a deeper process understanding for the waste to be disposed of in SFR or SFL. The research includes degradation of organic material, gas formation and the effects of swelling material. Handling and conditioning of, above all the long-lived low- and intermediate-level waste will be developed during the RD&D period, and preliminary acceptance criteria for long-lived waste need to be formulated for handling and conditioning of waste.

Prior to the decommissioning of the nuclear reactors, waste containers and waste transport containers for the long-lived waste must also be developed. Technology development for handling of reactor pressure vessels and other large components will be carried out. Information of the radionuclide inventory of each final repository needs to be improved to reduce the uncertainties in projections of waste quantities and content of radionuclides in the waste packages. Special efforts are made for the nuclides which quantities are difficult to measure, but are of importance in the post-closure risk assessment.

Spent nuclear fuel

The programme for research and technology development with regard to the spent nuclear fuel includes both initiatives for a better process understanding and development of technology and inspection methods for handling of the fuel.

If a canister is breached and water enters, the properties of the fuel are crucial for determining if and when radionuclides can be released. Generally the fuel dissolves very slowly, which delays the dispersion of radioactive elements. The ability of the fuel to dissolve and above all, the rate at which this takes place is essential when assessing the safety of the repository after closure. Dissolution data for new types of so-called doped fuel and for high-burnup fuel is for example needed. The fraction of radionuclides that is not embedded in the fuel matrix, and may thereby be relatively quickly released needs to be quantified more accurately as well as the rates of dissolution of metal parts of the fuel assemblies and control rods. Furthermore, deeper understanding of speciation and the solubility of radionuclides in a damaged, water-filled canister is needed.

The programme for handling of fuel comprises several parts, from requirements of information on the properties of the fuel before it is used, to devising a programme for safeguards that is internationally approved. In the coming years, further studies will be carried out on how information on the spent nuclear fuel best can be handled and stored prior to commissioning of the complete KBS-3 system. In preparation for final disposal of damaged fuel as well as fuel samples and residues that need special handling in one or more separate processes further development is required. The development activity also includes determination of decay heat and criticality verification for all facilities and transports in the KBS-3 system.

Canister

In the Spent Fuel Repository the copper canister provides the containment barrier. Further work concerns both research on properties of the copper canister in the repository environment and technology development in order to be able to manufacture canisters, verify them against specified requirements and handle them in the KBS-3 system.

For the assessment of post-closure safety in the Spent Fuel Repository, there are issues regarding corrosion and copper creep that require further research. In the long-term, sulphide is the most important copper corroding agent in a KBS-3 repository. A better understanding of the details of this corrosion process strengthens the scientific basis of the safety assessment. The understanding of creep of copper under mechanical loads is incomplete. A better understanding is needed of how an addition of phosphorus leads to favourable creep properties. The results are needed to improve the modelling of creep when assessing canister strength.

Regarding the canister design, SKB will update the design and defect analysis based on updated design premises. For this, an update of the material model for the cast iron insert is for example required. Additional process understanding is needed to determine the requirements on hydrogen, oxygen, sulphur and phosphorus in canister copper and requirements on the copper content in the cast iron insert.

A prerequisite for a proper production system for the canister is that technology and methods for production of canisters are developed, and that they can be applied on industrial scale. Furthermore, technology and methodology for welding and for inspection of canisters needs to be well developed. The equipment that will be used in Clink needs to be adapted to the nuclear environment there.

Cementitious materials

Cementitious materials exist in SFR in the waste matrix, engineered barriers and structural components. These materials have a central function in maintaining post-closure safety for SFR and for the planned SFL. Therefore the knowledge and ability to model the evolution of material properties over time need to be strengthened for future assessments of post-closure safety. This concerns changes in the material caused by chemical processes such as interactions with groundwater or components dissolved in the groundwater, and of the mechanical process like pressure from swelling material, internal gas pressure or freezing of the pore water in the concrete.

SKB plans to carry out extensive work linked to the design of concrete structures and materials for the different repositories. The focus during this RD&D period will be on development of structural concrete for the caissons in the rock vaults for intermediate-level waste in the extended part of SFR, and on developing robust low-pH materials for plugs, grout and rock support in the Spent Fuel Repository.

Buffer, backfill and closure

Clay material is used in all three repositories: as buffer and backfill in the Spent Fuel Repository, in the silo filling in SFR and as a barrier in the rock vault for legacy waste in SFL. For the Spent Fuel Repository design of buffer, backfill and closure need to be further developed prior to the continued design of the final repository, as well as the production system for bentonite components. Quality assurance during manufacturing, handling and installation needs to be further developed.

The design of the backfill in the Spent Fuel Repository requires above all efforts regarding verification of the ability of the backfill to serve as constraints against upwards swelling of the clay buffer in the deposition holes. The closure sequence for the Spent Fuel Repository will be developed.

For the assessment of post-closure safety, there are issues concerning the clay barriers that require further research, including homogenisation and water uptake, erosion processes such as piping and colloid release, microbial sulphate reduction, mineral stability and hydromechanical properties of bentonite barriers.

In SFL, the interaction between cement-solidified waste and the bentonite is a process requiring further action. The outcome of the safety evaluation will also provide information to help specifying what additional efforts needed. For SFR, the state of knowledge is considered to essentially be satisfactory regarding the silo filling, whereas some efforts are required for backfilling and closure of boreholes.

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 must be isolated. For this to be achieved, access to methods for investigating and characterising the rock at a sufficiently detailed level is needed. Furthermore, the final repositories' underground openings need to be designed in such a way that the long-term stable conditions are not jeopardised, and in order to be able to evaluate safety after closure, understanding of processes that alter the mechanical and chemical conditions in and around the repository is also needed.

The assessment of post-closure safety for the Spent Fuel Repository shows that shear movements in the rock due to earthquakes could cause canister failure. In order to reduce the negative effect of earthquakes, SKB is developing methods for identification of critical structures in the rock so that deposition of canisters adjacent to these can be avoided. Understanding of shear movements in the rock is also important for being able to make sufficient requirements on the canister's resilience. The largest remaining uncertainty concerns the relationship between the earthquake frequency and magnitude and how these vary during a glacial cycle. Therefore, studies are planned of paleoseismic events (prehistoric earthquakes) and in-depth studies of measurement data from earthquakes recorded by the Swedish national seismic network. Efforts in characterisation and modelling of the mechanical properties of the rock mass are also planned, along with further efforts concerning the effect on processes in the geosphere. Special initiatives are planned within DFN modelling, i.e. modelling of the fracture network.

Technology development in the rock area will continue during this RD&D period to enable the start of construction of the Spent Fuel Repository. A detailed characterisation programme with associated modelling will be devised prior to the construction of the repository. Methods, instruments and modelling methodology are being prepared for this.

An update of methods for tunnel production is also planned. The reference method for tunnel production of deposition tunnels is blasting but studies of alternative methods are being conducted, among other things in order to achieve a sufficiently leveled tunnel floor. Methods for excavation need to be established before detailed design of the Spent Fuel Repository's deposition areas. The rock excavation which will be relevant for the extension of SFR may, however, be performed with existing technology.

Surface ecosystems

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

The most important remaining issues in the surface ecosystems concern the uptake paths and uptake mechanisms for different organisms, temporal and spatial heterogeneity in the landscape, transport and accumulation processes as well as radiological, biological and chemical properties of certain substances. For SFL, supplementary data for certain radionuclides needs to be produced before applications for licences to construct the repository can be submitted.

Climate and climate-related processes

The overall purpose of the work on climate issues is to present scientifically substantiated scenarios for the future evolution of the climate, as bases for the evaluation of repository safety after closure. Important parts in the work on climate issues are gaining a better process understanding, validating the modifications of climate models that are used to describe the span of the climate that the repositories may be subjected to during the coming 100 000 to 1 million years. Issues that need to be further studied concern primarily; age and stability of the bedrock surface in Forsmark, palaeoclimate during the last glacial cycle as well as climate variations which make up transitions between different climate domains. Further studies of sea-level variations in the near future and in the long term will be carried out, as well as validation of permafrost models.

Part III Decommissioning of nuclear facilities

The licensees' decommissioning planning for of the nuclear power reactors in Barsebäck, Forsmark, Oskarshamn, Ringhals and Ågesta has been further developed since the RD&D Programme 2013. This applies to both planned measures for dismantling and demolition and strategies for technology development.

Planning premises

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. 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.

Plan for execution

Currently the reactors Barsebäck 1 and Barsebäck 2 are in service operation and Oskarshamn 2 is not planned to be re-started after a period of shutdown. Three additional reactors will be finally shutdown within the next few years: Oskarshamn 1 in conjunction with the maintenance outage in the summer of 2017 and Ringhals 1 and Ringhals 2 in conjunction with the maintenance outage in2020 and 2019. Ågesta NPP has been in service operation since 1974 and the aim is to commence dismantling and demolition at the latest in 2020.

The other six reactors are planned to finally be shutdown after 60 years' operation, which means that between 2040 and 2045 three reactors in Forsmark, two in Ringhals and one reactor in Oskarshamn will be shutdown.

In general, the timeline for decommissioning shall be kept as short as possible, for example by preparing as much as possible prior to dismantling and demolition and utilising coordination effects. The original strategy for decommissioning assumed that the extension of SFR would be commissioned before dismantling and demolition begin. As a result of the postponement of SFR's extension the strategy has changed and the decommissioning waste will be interim-stored at the nuclear power plants or externally awaiting transportation to SFR.

From a national perspective, there is a need to coordinate decommissioning issues both within the nuclear power companies, and between the nuclear companies and with SKB in order to ensure that the whole chain from decommissioning planning to final disposal of the waste is carried out in an optimal manner. Various forums exist to support this, where international forums are also important.

For more efficient work with decommissioning and waste issues, work areas have been divided between actors on both the company level and group level. The joint commitments within the management of the radioactive waste is normally coordinated by SKB, whereas the practice for handling decommissioning issues varies slightly within the two industrial groups, Uniper (the reactors in Barsebäck and Oskarshamn) and Vattenfall (the reactors in Ringhals and Forsmark and the Ågesta reactor).

Development efforts

Based on previously carried out development work and current decommissioning planning, there is a need for a number of activities in order to carry out the decommissioning of the Swedish nuclear power plants in a safe and efficient manner. The efforts concern areas that are largely common for the licensees and relate more to fuel and waste management, licensing processes, logistics, resources and coordination than to basic research and pure technology, although adjustments of available technology will be required. Decommissioning of the nuclear facilities that SKB is the licensee for lies fairly far in the future.

Part IV Other issues

Questions relating to the preservation of information and knowledge on final repositories and to the evolution of other concepts for final disposal, especially disposal in deep boreholes, have been constantly recurring during all years of the consultations prior to the applications for the encapsulation plant and the Spent Fuel Repository. SKB is therefore moving forward with the work on how to preserve documents, information and knowledge of the final repositories across generations far into the future and continues monitoring of developments in the areas of drilling and disposal in deep boreholes.

Preservation of information and knowledge across generations

Questions regarding preservation of information and knowledge on closed repositories for radioactive waste for future generations can be considered most urgent for the Spent Fuel Repository, but also need to be considered for SFR and SFL.

Practically, the solutions for information preservation need to be in place when the Spent Fuel Repository is sealed, which is estimated to take place around 2085 at the earliest. It is not possible, either for SKB, regulatory authorities or other parts of society, to determine definitively how best to proceed so far in the future. SKB considers that the only meaningful plan of action is to have a way or working that aims to keep the issue updated, develop the work and disseminate knowledge on the need. SKB is participating among other things in OECD-NEA’s working group on issues of how to preserve information and knowledge on the final repository for radioactive waste across generations.

Disposal in deep boreholes

In the case of disposal of spent nuclear fuel in deep boreholes, the most important safety function is the isolation and retardation of elements that the rock offers. This is based on the assumption that groundwater is stagnant at great depths due to the fact that it has high salinity and thereby high density which counteracts remixing with the lighter fresh water above. The great depth, however, brings with it cons such as difficulties with characterisation of the surrounding rock, and poor control over the handling during deposition or possible retrieval.

Since the RD&D programme in 2013, SKB has, in the supplementary information submitted to the Land and Environment Court in licensing matters for the Spent Fuel Repository, among other things compared disposal in deep boreholes with deposition according to the KBS-3 concept regarding the deposition process, closure and nuclear safety during operation and after closure.

The assessment from previous RD&D programmes remains, i.e. that the concept of disposal in deep boreholes is not today developed sufficiently to serve as a realistic methodology for final disposal of spent nuclear fuel. SKB intends however to continue to follow and in relevant contexts participate in international forums regarding deep boreholes.

Contents

Part I Activities and plan of action

1 Introduction 21
1.1 Premises 21
1.1.1 Relevant regulatory framework and SKB's mission 21
1.1.2 Fundamental principles 22
1.1.3 Planning premises regarding reactor operation 23
1.1.4 The radioactive waste and spent nuclear fuel 24
1.2 Programme for research, development and demonstration 25
1.3 RD&D Programme 2016 and its relation to other reports submitted to SSM 27
1.4 Collaboration with Posiva 29
1.5 Financing 30

2 Description of the waste system 31
2.1 Facilities in the system for low- and intermediate-level waste 31
2.1.1 Facilities for short-lived waste 31
2.1.2 Facilities for long-lived waste 34
2.2 Facilities in the KBS-3 system. 35
2.3 The transportation system 39

3 Plan of action 41
3.1 Effects of premature shutdown 41
3.2 Main timetable for the nuclear waste programme 41
3.3 Plan of action for low- and intermediate-level waste 44
3.3.1 Current situation 44
3.3.2 Overall planning 44
3.3.3 Short-lived waste 44
3.3.4 Long-lived waste 48
3.4 Plan of action for spent nuclear fuel 52
3.4.1 Current situation 53
3.4.2 Overall planning 54
3.4.3 Interim storage 55
3.4.4 Encapsulation 56
3.4.5 Transportation of fuel 58
3.4.6 Final disposal 59
3.5 Plan of action for decommissioning of nuclear facilities 60
3.5.1 Decommissioning overview 61
3.5.2 Current situation and overall planning 61
3.6 Alternative actions for changed conditions 62
3.6.1 Operating times of the nuclear power reactors 62
3.6.2 Commissioning of the extended final repository for short-lived radioactive waste 64
3.6.3 Interim storage capacity for low- and intermediate-level waste 64
3.6.4 Commissioning of the final repository for long-lived waste 64
3.6.5 Siting of the final repository for long-lived waste 65
3.6.6 Commissioning of the Spent Fuel Repository and Clink 65
3.6.7 Horizontal deposition – KBS-3H 66

4 Procedures, resources and competence 67
4.1 Research 67
4.1.1 The goals of research 67
4.1.2 Management of research 67
4.1.3 Strategy 67
4.2 Technology development 69
4.2.1 The goals of technology development 69
4.2.2 Control of technology development 69

4.2.3 Technology development process 69
4.2.4 Design premises 70
4.2.5 Quality management and testing 71
4.3 Work tools 72
4.3.1 Databases 72
4.3.2 Model and calculation tools 72
4.3.3 Site models 73
4.3.4 Quality assurance 73
4.4 Resources and competence 74
4.4.1 Within SKB 74
4.4.2 Competence network and suppliers 75
4.4.3 Collaboration 75
4.5 SKB's facilities for research, development and demonstration 76
4.5.1 The Äspö HRL 76
4.5.2 The Canister Laboratory 78
4.5.3 Other laboratories 78

5 Further research and technology development 81
5.1 Future efforts 81
5.2 Overview of the final repository and Clink 82
5.2.1 Final repository for short-lived radioactive waste 82
5.2.2 The final repository for long-lived waste 83
5.2.3 The Spent Fuel Repository and Clink 84
5.3 The low- and intermediate-level waste 87
5.3.1 Radionuclide inventory 87
5.3.2 Process understanding 87
5.3.3 Handling of the low- and intermediate-level waste 88
5.3.4 Waste packaging and waste transport container 88
5.4 The spent nuclear fuel 89
5.4.1 Process understanding 89
5.4.2 Handling of the spent nuclear fuel 89
5.4.3 Fuel information, criticality and safeguards 89
5.5 Canister for spent nuclear fuel 90
5.5.1 Process understanding 90
5.5.2 Design and manufacturing 91
5.5.3 Inspection and testing 91
5.6 Cementitious materials 91
5.6.1 Process understanding 92
5.6.2 Construction, manufacture and installation 93
5.7 Buffer, backfill and closure 94
5.7.1 Process understanding 94
5.7.2 Construction, manufacture, installation and inspection 95
5.8 Rock 97
5.8.1 Process understanding 97
5.8.2 Production, verification and inspection 98
5.9 Surface ecosystems 99
5.10 Climate and climate-related processes 100
5.11 Decommissioning 100
5.12 Other questions 101

Part II Waste and final disposal

6 Low- and intermediate-level waste 105
6.1 Radionuclide inventory 105
6.1.1 Reference inventory 105
6.1.2 Method development for difficult-to-measure nuclides 106
6.1.3 Uncertainties in radionuclide inventory 107
6.2 Acceptance criteria for long-lived waste 108
6.3 Conditioning of long-lived waste 108

	6.3.1	Stabilisation of waste in steel tanks	108
	6.3.2	Transloading of waste	109
6.4	Management of reactor pressure vessels and large components		109
6.5	Waste containers and waste transport containers		110
	6.5.1	Waste containers for waste from AB SVAFO and Studsvik Nuclear AB	110
	6.5.2	Waste containers for decommissioning waste	110
	6.5.3	Transport containers for new waste containers	111
6.6	Degradation products from organic materials and their interactions with radionuclides		112
	6.6.1	Degradation products from cellulose	112
	6.6.2	Degradation products from filter aids	112
	6.6.3	Degradation products from cement additives	112
6.7	Corrosion of aluminium and zinc		113
6.8	Microbial gas production		114
6.9	Swelling waste – bitumen-solidified ion exchange resins		114
7	**Spent nuclear fuel**		115
7.1	Non-regular fuels		115
7.2	The ageing of fuel		116
7.3	Decay heat and fuel measurement		116
7.4	Fuel information		117
7.5	Criticality		118
7.6	Safeguards		119
7.7	Fuel dissolution and radionuclide chemistry		120
	7.7.1	Fuel dissolution	120
	7.7.2	Radionuclide speciation and solubilities	123
8	**Canister**		125
8.1	Corrosion		125
	8.1.1	Sulphide corrosion	125
	8.1.2	Localised corrosion	127
	8.1.3	Copper corrosion in pure, oxygen-free water	128
	8.1.4	Radiation-induced corrosion	130
	8.1.5	Stress corrosion cracking	130
	8.1.6	Verification of different copper materials for corrosion sensitivity	131
8.2	Creep of copper		132
	8.2.1	Impact of phosphorus	132
	8.2.2	Deformation and failure	133
8.3	Design		135
	8.3.1	Design analysis	135
	8.3.2	The role of hydrogen in copper	137
	8.3.3	Requirements on maximum copper content in nodular cast iron	138
8.4	Manufacturing		139
	8.4.1	Copper components	139
	8.4.2	Canister insert	140
	8.4.3	Welding	142
	8.4.4	Inspection and testing	143
9	**Cementitious materials**		145
9.1	Cementitious materials – development after closure		145
	9.1.1	Groundwater impact	145
	9.1.2	Modelling of gas transport	146
	9.1.3	Impact of degradation of organic waste	146
	9.1.4	Impact of corrosion of metallic waste	147
	9.1.5	Impact of bentonite on cementitious materials	148
	9.1.6	Impact of mineral additives	149
	9.1.7	Freezing	150
	9.1.8	Internal and external loads	151
9.2	Design of concrete structures and materials in SFR		151

9.2.1 Waste vault for intermediate-level waste 152
9.2.2 Waste vault for reactor pressure vessels 153
9.2.3 System for gas transport 155
9.3 Design of concrete structures and materials for SFL 156
9.3.1 Rock vault for core components 156
9.3.2 Grouting of waste packages 156
9.4 Design of concrete structures and materials for the Spent Fuel Repository 157
9.4.1 Plugs for deposition tunnels 157
9.4.2 Low-pH cement materials for grouting and rock support 158

10 Buffer, backfill and closure 159
10.1 Evolution of the bentonite material after installation until saturation 159
10.1.1 Piping/erosion 159
10.1.2 Water uptake in the buffer 161
10.1.3 Swelling, homogenisation and self-healing 163
10.1.4 Water vapour circulation 165
10.1.5 Microbial sulphide formation 168
10.2 The bentonite material properties in the saturated state 169
10.2.1 Material composition 170
10.2.2 Swelling pressure and hydraulic conductivity 170
10.2.3 Shear strength 171
10.3 Evolution of the bentonite material after water saturation 172
10.3.1 Buffer loss due to colloid release/erosion 172
10.3.2 Sulphide formation and sulphide transport 175
10.3.3 Self-healing of bentonite 177
10.3.4 Long-term stability of bentonite 177
10.4 Barrier design 178
10.4.1 The buffer in the Spent Fuel Repository 178
10.4.2 Backfill in the Spent Fuel Repository 180
10.4.3 Design of clay barriers in SFL 181
10.5 Manufacturing, inspection and testing of buffer and backfill components 181
10.5.1 Material supply and quality assurance of bentonite materials 181
10.5.2 Manufacturing of buffer components 182
10.5.3 Manufacturing of backfill components 183
10.6 Deposition and installation of buffer and backfill 184
10.6.1 Deposition in the Spent Fuel Repository 185
10.6.2 Buffer 186
10.6.3 Backfill 187
10.7 Borehole sealing 188
10.7.1 The Spent Fuel Repository 188
10.7.2 Final repository for short-lived radioactive waste 189
10.8 Closure 189
10.8.1 Closure of the Spent Fuel Repository 189
10.8.2 Closure of the final repository for short-lived radioactive waste 189

11 Rock 191
11.1 Detailed characterisation 191
11.1.1 Methodology for detailed characterisation 192
11.1.2 Critical structures 192
11.1.3 Modelling methodology within detailed characterisations 194
11.2 Tunnel production 194
11.2.1 Grouting 195
11.2.2 Tunnel excavation for deposition tunnels 195
11.2.3 Boring of deposition holes 196
11.3 Modelling of discrete fracture networks 196
11.4 Hydrochemistry, and transport modelling 198
11.5 Link between near-surface and deep groundwater 200
11.6 Development of hydrogeological calculation tools 201
11.7 The impact of the ice load on the flow and transport properties of the rock 202

11.8 Effect of freezing on the flow and transport properties of the rock 202
11.9 Handling of glacial cycle in hydrochemical and transport modelling 203
11.10 Effect of freezing on the mechanical properties of the rock 205
11.11 Seismic impact on post-closure safety 205
11.11.1 Seismic monitoring 206
11.11.2 Investigations of glacially induced faults 207
11.11.3 Investigation of possible tsunami 210
11.11.4 Modelling of seismic impact on the final repository 210
11.12 The mechanical properties of the rock mass 211
11.13 Induced deformation in the rock mass caused by thermal, seismic or glacial load 213
11.14 Rock stress in Forsmark 214

12 Surface ecosystems 215
12.1 Uptake paths and uptake mechanisms for radionuclides in various organisms 215
12.2 Temporal and spatial heterogeneity of the landscape 218
12.3 Transport and accumulation processes 220
12.4 Radiological, biological and chemical properties of potentially important elements 222

13 Climate and climate-related processes 225
13.1 Age and long-term stability of the rock surface in Forsmark 225
13.2 Climate variations 226
13.2.1 The climate in SKB's reference glaciation 226
13.2.2 Climate change: transitions between climate domains 227
13.2.3 Onset of future cold climate, permafrost and ice sheet growth 228
13.3 Sea-level variations, isostasy and shoreline displacement 230
13.4 Validation of the permafrost model 231
13.5 Newly developed description of ice sheet hydrology from GAP 232

Part III Decommissioning of nuclear facilities

14 Premises for decommissioning of nuclear facilities 237
14.1 Concepts and requirements 237
14.2 Responsibility and division of roles 239
14.2.1 Division of roles between the licensees and SKB 239
14.2.2 Distribution of work within the groups 240
14.3 National and international coordination 240

15 Planning for decommissioning at Uniper 243
15.1 Barsebäck Kraft AB's planning for decommissioning 243
15.2 OKG Aktiebolag's planning for decommissioning 245

16 Planning for decommissioning at Vattenfall 249
16.1 Vattenfall's decommissioning planning 249
16.2 Ringhals AB's planning for decommissioning 251
16.3 Forsmarks Kraftgrupp AB's planning for decommissioning 253
16.4 Vattenfall's planning for decommissioning of the Ågesta reactor 254

17 Planning for decommissioning of SKB's nuclear facilities 257
17.1 Central interim storage facility and encapsulation plant for spent nuclear fuel 257
17.2 Final repository for short-lived radioactive waste 257
17.3 Final repository for long-lived waste 257
17.4 The Spent Fuel Repository 258

18 Dependencies and flexibility 259
18.1 Decommissioning activities during this RD&D period 259
18.2 Flexibility concerning external dependencies 261
18.2.1 Final shutdown 261
18.2.2 End of defueling 262
18.2.3 Start-up of sub-projects/steps 262
18.2.4 Release of facility 263

18.3 Flexibility concerning internal dependencies 263
18.3.1 Separation of facilities prior to decommissioning 263
18.3.2 Preparatory measures 264
18.3.3 Spent fuel transportation 264
18.3.4 Interim storage capacity in Clab 265
18.3.5 Management of reactor pressure vessels 265
18.3.6 Management of long-lived waste 265
18.3.7 Clearance and management of very low-level waste 266
18.3.8 National planning for critical resources and functions 266

19 Continued activities within decommissioning 267
19.1 Current situation 267
19.1.1 Industry-wide development work 267
19.1.2 Development work at Uniper 268
19.1.3 Development work at Vattenfall 269
19.2 Programme 270
19.2.1 Industry-wide development work on waste and final disposal 270
19.2.2 Other industry-wide development 272
19.2.3 Development needs at Uniper 272
19.2.4 Development needs at Vattenfall 273

Part IV Other issues

20 Preservation of information and knowledge through generations 277
20.1 Complete work 277
20.2 Current and future work 278
20.2.1 Records, Knowledge and Memory across Generations 278
20.2.2 Assembling Alternative Futures for Heritage 278

21 Disposal in deep boreholes 281
21.1 Current situation 281
21.1.1 Carry out deposition 281
21.1.2 The evolution of the repository after closure 281
21.1.3 Work carried out since RD&D 2013 282
21.2 Evaluation and further work 284
21.2.1 SKB's judgement 284
21.2.2 Continued work 284

References 285

Appendix Abbreviations 307

Part I

Activities and plan of action

1 Introduction

2 Description of the waste system

3 Plan of action

4 Procedures, resources and competence

5 Further research and technology development

1 Introduction

The Swedish power industry has been generating electricity by means of nuclear power for more than 50 years. The country's first nuclear power reactor was commissioned in 1964. A large part of the system that is needed to safely manage and dispose of the spent nuclear fuel and the radioactive waste from operation of the reactors has already been built up. The system consists of the 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 containers for transport.

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 finally disposed of the facilities can be decommissioned and the final repository can be sealed and closed. The KBS-3 system includes a facility part for encapsulation of the spent nuclear fuel adjacent to Clab, containers for shipping spent nuclear fuel and a final repository for canisters with spent nuclear fuel. In addition to these facilities, systems for production of canisters as well as buffer and backfill material is needed.

For disposal of the low- and intermediate-level waste, SFR needs to be extended, an additional repository – the final repository for long-lived waste (SFL) – needs to be established and containers for transport of long-lived waste need to be procured.

A decision on the premature shutdown of the reactors Oskarshamn 1, Oskarshamn 2, Ringhals 1 and Ringhals 2 was taken in 2015. Since decommissioning of the reactors will commence before the extended SFR is ready to receive and deposit decommissioning waste, the licensees must arrange for interim storage of decommissioning waste.

SKB's plan of action in this RD&D Programme describes the overall plans for implementing the remaining parts of the waste system and adapting the existing facilities in such a manner that human health and the environment are protected today and in the future.

1.1 Premises

1.1.1 Relevant regulatory framework and SKB's mission

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

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, Svensk Kärnbränslehantering AB, SKB, is responsible for management and final disposal of the radioactive 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 licensees are responsible for decommissioning of their nuclear power reactors. In this context, SKB has been contracted by the nuclear power companies to participate in the planning and execution of the future decommissioning. SKB's participation mainly pertains to compilation of the development needs identified by the licensees, coordination of general methods and procedures for transportation and final disposal of radioactive waste, and compilation of the decommissioning costs as reported by the licensees.

Under the Nuclear Activities Act, the nuclear power companies, working in consultation, shall present a programme for the research and development activities and other measures needed to manage and dispose of nuclear waste and spent nuclear fuel in a safe manner and to decommission the nuclear power plants. Such a programme (RD&D programme) shall be submitted every three years to the Swedish Radiation Safety Authority (SSM). The programmes are reviewed and evaluated by SSM after extensive referral for consideration and comment. They are also reviewed by the Swedish National Council for Nuclear Waste. SSM and the Council submit comments to the Government, which decides whether the programme meets the requirements set out in the Nuclear Activities Act and if any guidelines should be given for the continued activities.

SKB, on behalf of and in cooperation with the nuclear power companies, is responsible for preparing the RD&D programmes and submitting them to SSM.

Under the Nuclear Activities Act, the nuclear power companies are obliged to pay the costs of the measures needed to manage and dispose of the nuclear waste and the spent nuclear fuel and to decommission the facilities. According to the Act (SFS 2006:647) on Financial Measures for the Management of Waste Products from Nuclear Activities (the Financing Act), the nuclear power companies are obliged to pay a fee for future waste management and decommissioning. On behalf of the nuclear power companies and pursuant to the Financing Act, SKB present cost calculations every three years, see Section 1.5. Paid-in fees are managed by the state Nuclear Waste Fund.

In addition to the radioactive waste SKB receives from the nuclear power companies, SKB also receives some radioactive waste from other companies. This is regulated by commercial agreements between SKB and the respective companies. SKB currently has agreements with AB SVAFO, Studsvik Nuclear AB, Cyclife AB[1] and European Spallation Source ERIC (ESS). Westinghouse Electric Sweden AB and SKB have made a declaration of intent regarding final disposal of long-lived radioactive waste.

1.1.2 Fundamental principles

The management of radioactive material is regulated by legislation. The focus of the work with management of radioactive waste has furthermore 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 radioactive waste from the Swedish reactors shall 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.
- The final repository shall be established by the generations that have derived benefit from Swedish nuclear power.

SKB plans for geological disposal of the nuclear waste and the spent nuclear fuel. Other more or less unrealistic strategies have also been studied and discarded, such as launching the spent nuclear fuel into space, disposing of it beneath the seabed or burying it in the continental ice sheet. Most countries and organisations, such as the IAEA and the OECD/NEA, are agreed today that geological disposal is a solution that satisfies all requirements on safe final disposal and feasibility. Geological disposal is also supported in the EU's community framework for the responsible and safe management of spent fuel and radioactive waste[2].

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.

[1] Previously Studsvik Nuclear Environmental AB.

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

- Repository safety shall be based on multiple barriers.
- Engineered barriers shall primarily consist of naturally occurring materials that are long-term stable in the repository environment.
- Barriers shall work passively, i.e. without human intervention and without input of energy or materials.
- Repositories shall be designed in such a manner that safety is not dependent on active measures such as maintenance and repairs after closure.

The multiple barrier principle 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 or barrier functions which are needed in a final repository is 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 lower than those on the repositories for spent nuclear fuel and for long-lived radioactive waste.

The above principles, along with a number of other considerations, such as that construction of the repository must be technically feasible, have led SKB to choose the KBS-3 method for final disposal of spent nuclear fuel. SKB has within the framework of the RD&D Programme on several occasions conducted evaluations of different strategies and systems for disposal of spent nuclear fuel. In the most recent evaluation (SKB 2014j), SKB explains the background and reasons for the choice of the KBS-3 method in relation to other methods. The evaluation was carried out against stipulated requirements, both overall societal requirements and environmental, safety and radiation protection requirements.

The method, whose development began at the end of the 1970s, is characterised as follows:

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

Internationally, the KBS-3 method is one of the methods for final disposal of spent nuclear fuel where development has progressed furthest. The method is used in Finland (see Section 1.4) and is being considered as a method for final disposal in several other countries, including Canada, South Korea, the UK, Taiwan and the Czech Republic.

1.1.3 Planning premises regarding reactor operation

The long-term planning is based on the nuclear power companies' current planning premises. During 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. The reason for the premature shutdown is that the nuclear power companies no longer deem it economically viable to operate these reactors. For the reactors in Oskarshamn the decision entails the shutdown of Oskarshamn 1 by mid-year 2017 while Oskarshamn 2, which had been shut down for modification, will not be restarted. For the reactors in Ringhals, it entails that Ringhals 1 and Ringhals 2 will be shut down by mid-year 2020 and 2019, respectively. For the other six reactors the planned operating time is, as in the RD&D Programme 2013, 60 years. This applies to the reactors Forsmark 1, Forsmark 2 and Forsmark 3, Oskarshamn 3 as well as Ringhals 3 and Ringhals 4.

The reactors' planned operating times are an important factor in the planning of the nuclear waste programme. Based on operating times, predictions are made of the quantities of radioactive waste and spent nuclear fuel that will be managed and when the need for interim storage and final disposal will arise. Section 3.1 provides a description of how the premature shutdown affects SKB's and the nuclear power companies' plans for execution of the nuclear waste programme.

1.1.4 The radioactive waste and 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 (low-, intermediate- or high-level) as well as the life of the radioactivity (short- or long-lived waste). The level of radioactivity determines how the waste is handled before final disposal. The intermediate-level waste and the high-level spent nuclear fuel require radiation-shielded handling, while the low-level waste can be handled without radiation shielding. The design of final disposal is largely determined by whether the 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 premises in the planning of the waste system. The planning is based on the waste from the current nuclear power programme. The waste quantities are dependent on the reactors' operating times as well as availability and other operating conditions. Estimated quantities of radioactive waste and spent nuclear fuel are based on predictions by the nuclear power companies.

Low- and intermediate-level waste

The low- and intermediate-level waste is divided into short-lived and long-lived waste. Short-lived waste contains a significant quantity of radionuclides with a half-life shorter than 31 years and only a limited quantity of radionuclides with a longer half-life. Long-lived waste contains significant quantities of radionuclides with long half-lives.

Low- and intermediate-level waste is produced 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 deposited in SFR or near-surface repositories. The near-surface repositories, where waste with very low-level radioactivity is deposited, are operated by the nuclear power companies, while SFR is operated by SKB. 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. AQccording to current forecasts, about 170 000 cubic metres of waste plus nine reactor pressure vessels from BWRs will be disposed of in SFR. Most of the short-lived waste origins from the nuclear power plants. Other waste origins from Clab and Clink (central interim storage facility and encapsulation plant for spent nuclear fuel) and from Studsvik Nuclear AB and AB SVAFO.

Long-lived waste from the NPPs consists of used core components, reactor pressure vessels from pressurised water reactors (PWRs) and control rods from boiling water reactors (BWRs). 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- and intermediate-level waste is estimated to about 16 000 cubic metres, about one-third of which comes from the NPPs. The rest comes from facilities operated by other companies (see 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. However, the spent fuel contains by far most of the radioactivity, both short- and long-lived. Spent nuclear fuel is high-level and requires radiation shielding in conjunction with all handling, storage and final disposal. Final disposal is planned to take place in the Spent Fuel Repository.

The spent fuel generates heat even after it has been removed from the reactor (decay heat). Because of the decay heat, the fuel must be cooled to avoid overheating. The amount of decay heat depends above all on burnup and how long the fuel has decayed. Burnup is specified in megawatt-days per kilogram of uranium (MWd/kgU). Due to technical advances and changes in the operation of the reactors, fuel burnup has increased steadily since the reactors were commissioned. The reason for these changes 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 amount of spent nuclear fuel is given as the weight of the uranium that was originally present in the fuel.

In addition to all the spent nuclear fuel from the Swedish nuclear power plants (including fuel from the Ågesta reactor), the amount of spent nuclear fuel to be deposited in the Spent Fuel Repository also includes fuel residues from testing programmes at Studsvik, and MOX fuel (Mixed Oxide Fuel). These fuel types comprise a very small fraction of the total. 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 currently being interim-stored in Clab. 23 tonnes of MOX 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 Programme. The programme shall provide an overview of all measures that are needed to manage the radioactive waste and describe in greater detail the measures intended to be adopted within a timespan of at least six years. The programme shall be submitted every three years to SSM, who conducts a review and evaluation of planned research and development activities, reported research results, alternative management and disposal methods and planned measures. After extensive referral for consideration and comment, SSM hands over the matter to the Government, which rules on whether the programme meets the requirements in the Nuclear Activities Act and any guidelines for the continued activities. The Swedish National Council for Nuclear Waste also submits its independent evaluation of the programme to the Government.

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 new 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. In June 1984, the Government granted the nuclear power companies 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 has been found essentially acceptable with regard to safety and radiation protection." The KBS-3 method has since served as a basis for SKB's RD&D programmes. SKB has also followed the development of other methods and has on a number of occasions evaluated them in relation to the KBS-3 method.

The focus of the RD&D programmes has varied through the years, depending on where the emphasis has been in SKB's activities. The RD&D Programme 2010 included a brief summary of the RD&D programmes presented by SKB up to 2007 and the RD&D Programme 2013 includes a brief summary of the RD&D Programme 2010. All programmes have been circulated for consideration and comment and then approved by the Government, in some cases with demands for supplementary information or directives on how SKB should respond to comments expressed by the reviewing bodies. Figure 1-1 shows an overview of presented research and development programmes and other milestones in the development work.

RD&D Programme 2013

The RD&D Programme 2013 consisted of five parts: SKB's activities and plan of action, Low- and intermediate-level waste, Spent nuclear fuel, Research for assessment of long-term safety and Social science research. Research and technology development is largely being carried out under SKB's own auspices, but also in cooperation with universities and institutes of technology all over the world. SKB is also collaborating with sister organisations in other countries, and in 2013, planning was initiated to deepen SKB's collaboration with Posiva in Finland.

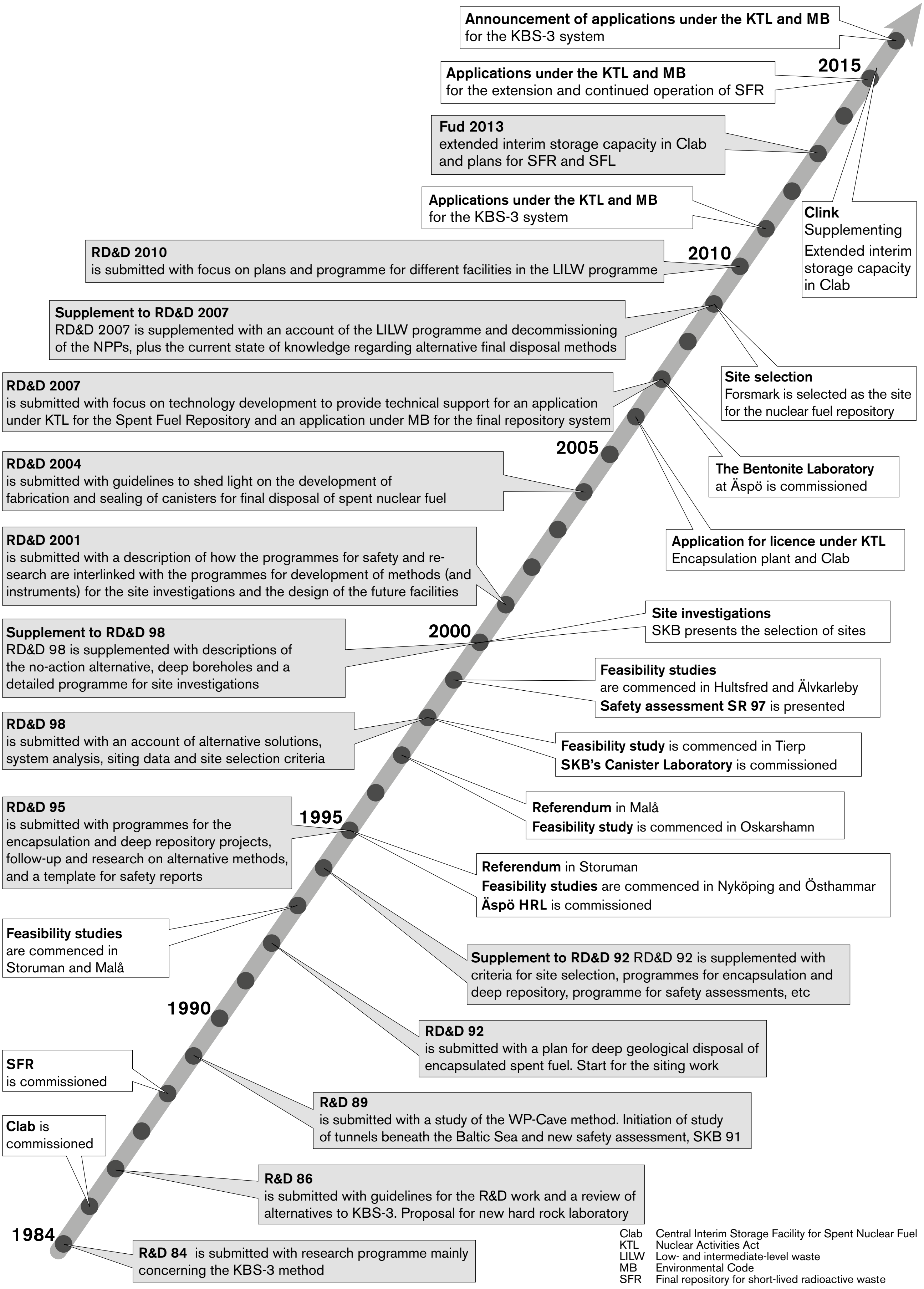


Figure 1-1. *Milestones in SKB's RD&D programme.*

The first part of the programme gave a general description of SKB's activities and the measures SKB plans to carry out. In the plan of action, SKB presented its plans for applying for a licence for increased storage capacity in Clab. Today SKB has a licence to store 8 000 tonnes of fuel in Clab.

Part two and three gave more detailed descriptions of measures for low- and intermediate-level waste and spent nuclear fuel.

At the time of the RD&D Programme 2013, a siting study for the extension of SFR had been completed. SKB presented the fact that the extension will be built at a greater depth than the existing facility. The work with SFL was further detailed and in particular the work of selecting a final repository concept was described. Furthermore, the nuclear power companies' and SKB's plans for decommissioning of their nuclear facilities were described.

Technology development for the spent nuclear fuel programme was described with the goal of proceeding from schematic solutions to solutions tailored to an industrialised process. A large part of the remaining development work consists of building a production system with quality control.

The conclusion of the safety assessment SR-Site, presented in 2011, is that it is possible to construct a spent fuel repository that meets SSM's requirements on post-closure safety. The programme for future research focused on the factors which SR-Site has shown to contribute to risk and where conditions are considered possible to improve by further research. For low- and intermediate-level waste, the safety assessment for the extended SFR and the concept study for SFL were under way. It was noted that the results from the safety assessment for the SFR extension and the concept study for SFL will guide further research to a great extent.

After its review, SSM found that the reported activities were sufficiently comprehensive and the planned measures were sufficiently appropriate and that the RD&D Programme 2013 thereby met the requirements of the Nuclear Activities Act. The Swedish National Council for Nuclear Waste stated that SKB had made great progress compared with previous years. The Government decided that the programme met the requirements of the Nuclear Activities Act and stipulated the requirement that SKB continue to consult with SSM in matters concerning decommissioning plans and decommissioning studies. SKB were also to ensure that future RD&D programmes would be clearer, more structured and would clarify how research and development initiatives are planned, justified and evaluated in order to fulfil the requirements of the Nuclear Activities Act. The reactor owners and SKB should also carefully consider other comments given on review of the programme by SSM, the Swedish National Council for Nuclear Waste and other reviewing bodies.

1.3 RD&D Programme 2016 and its relation to other reports submitted to SSM

In addition to the RD&D programmes, SKB and the nuclear power companies submit other reports to SSM where plans for the management and final disposal of radioactive waste and spent nuclear fuel as well as decommissioning of the nuclear facilities are described. Prior to the RD&D Programme 2016, the situation is unusual with two simultaneously ongoing licensing matters in the Land and Environment Court (MMD) and with SSM, one for the final repository system for spent nuclear fuel and one for the extension of SFR for operational and decommissioning waste.

In the initial part of the work on the RD&D Programme 2016, consultation meetings with SSM were held. The purpose was to ensure that the requirements by the Government on the RD&D Programme were met. The requirements relate to the development of decommissioning plans and decommissioning studies as well as a clearer and more structured RD&D Programme that clarifies how research and development initiatives are planned, justified and evaluated in order to satisfy stipulated requirements. Further, SSM presented its expectations on the RD&D Programme 2016 in a memorandum (SSM 2016). This memorandum has also been discussed at the consultation meetings.

In preparing this RD&D programme, SKB has considered the comments that were expressed in consultations. SKB has particularly observed the Authority's general opinion on the RD&D Programme 2013 where SSM as well as the Government stated that the RD&D Programme 2016 should be more clearly structured and clarify how research and development is planned, justified and evaluated on the basis of the measures that are planned to meet the requirements in the Nuclear Activities Act. For this purpose, SKB has made significant changes in the structure of the RD&D Programme 2016 compared with previous programmes.

SKB's goal is a more strategically focused RD&D Programme where the planned measures are linked to the overall goals. SKB has today a refined and more detailed plan of action for implementation of new facilities which is described in Part I, Activities and plan of action. Further research and technology development is justified based on the milestones that are planned for the new facilities as well as other measures and remaining issues regarding these facilities. Chapter 5 provides clear links to the programme in Part II (Waste and final disposal) which describes in more detail the programme for the next six years. This is the most important change made to achieve a well-justified and structured RD&D Programme 2016.

In summary, the RD&D Programme 2016 consists of four parts:

Part I Activities and plan of action, summarises the prerequisites for the RD&D Programme and the overall planning to fulfil legislative requirements on safe management and final disposal of the radioactive waste and the spent nuclear fuel and decommissioning of facilities. It also describes how prioritisation of research and technology development is made and how the needs for competence and resources are met.

Part II Waste and final disposal, describes in greater detail the measures planned during this RD&D period (2017–2022) in light of the results obtained thus far. The results are summarised and references are given to more detailed descriptions.

Part III Decommissioning of nuclear facilities, presents the plans for the decommissioning of the nuclear facilities.

Part IV Other issues, presents plans for other issues such as information preservation and deep boreholes.

The programme presented in the RD&D Programme 2013 has largely been carried out according to plan and the increased state of knowledge has contributed to progress in ongoing licensing issues. As always, adaptions have been made to the needs in licensing issues and on the basis of new knowledge gained in SKB's current projects and in the world. The point of departure for the RD&D Programme 2016 has been to systematically evaluate the outcome since the RD&D Programme 2013 in order to subsequently define the programme based on current needs according to the plan of action.

The RD&D Programme 2016 is mainly intended for experts and decision-makers at the regulatory authorities but also for other stakeholders with knowledge of nuclear waste issues. Experts' needs for information on specific issues are met in references to the programme.

In order to avoid redundancy in the present RD&D programme, references are given where necessary to the following other documents:

- The applications submitted by SKB for the Spent Fuel Repository and encapsulation plant and supplements to these.
- The applications submitted by SKB for extension of SFR and supplements to these.
- Studies and plans for decommissioning of the nuclear power reactors and other nuclear facilities.
- Safety analysis reports (SAR) and periodic overall evaluations for SKB's commissioned facilities Clab and SFR.
- The recurring Plan reports.

The application for final disposal of spent nuclear fuel under the Nuclear Activities Act 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, along with reports of operational safety and post-closure safety.

An application under the Nuclear Activities Act for the encapsulation plant was submitted in 2006. It was supplemented in 2009 with regard to integration of the encapsulation plant with Clab to a single facility, Clink. In March 2011 it was supplemented again with regard to those parts dealing with the KBS-3 system. In its work with reviewing the application, SSM in autumn 2012 made a request for supplementary information, which SKB met 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 volume in Clab to 11 000 tonnes of spent nuclear fuel.

In December 2014, SKB submitted applications under the Nuclear Activities Act and the Environmental Code for the extension of SFR. These applications are currently under review and SKB regularly responds to requests for supplementary information from regulatory authorities and reviewing bodies.

Regarding the facilities in operation, Clab and SFR, SKB is obligated as licensee to submit various reports to SSM that are not discussed in this RD&D programme. The reports mainly linked to the RD&D Programme are the SAR for the facilities that are constantly kept up-to-date and the periodic overall evaluation of facility safety and radiation protection that under the Nuclear Activities Act must be carried out at least every tenth year.

Each nuclear facility has a decommissioning plan that must be produced before the facility is constructed and thereafter kept up-to-date until the facility is decommissioned. Each licensee reports significant changes to SSM and presents an updated plan to the Authority in conjunction with the periodic overall evaluation. Decommissioning plans for the nuclear power plants have recently been updated and submitted to SSM. Decommissioning plans for the Spent Fuel Repository, Clink and the extended SFR were included as supporting material in the applications to construct these facilities.

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

Another report linked to the RD&D Programme is the Plan report, see Section 1.5. Here the future cost for disposing of the radioactive waste and the spent nuclear fuel and decommissioning of the nuclear power reactors is calculated. The cost calculation is based on the plans presented in the RD&D programme.

1.4 Collaboration with Posiva

SKB's Finnish sister organisation Posiva has also 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. Since 2004, Posiva has been constructing a hard rock facility (Onkalo) in Olkiluoto. Now, the construction has reached the potential repository level. Onkalo is used for research and development, but will also provide access to the actual final repository.

At the end of 2012, Posiva submitted an application for a licence for construction of an encapsulation plant and 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 comments on this application and recommended that the Finnish Government grant a licence. In its review statement, STUK identified a number of issues that Posiva needs to solve and report before it is possible to provide the operating permit. In November 2015, the Finnish Government issued their licence for construction of the final repository and the encapsulation plant. Posiva plans to start constructing the facilities in 2016 and to begin trial operation, if licences for this are granted, around 2023.

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. The purpose of the collaboration is to solve the remaining issues regarding design as well as quality control and inspection for primarily the engineered barriers so that the regulatory requirements are met and deposition can begin according to Posiva's timetable. The cooperation means that about half of all technology development projects in the KBS-3 system, mainly concerning canister and bentonite issues, are now co-funded by SKB and Posiva. Although this entails that some remaining technical issues are solved earlier than needed according to SKB's timetable, it also permits more optimised designs to be available when SKB's final repository is commissioned. Moreover, SKB and Posiva are studying the prospects for cooperation on the actual construction projects (encapsulation plant and final repository), as well as the prospects for common production of canisters and buffer components.

1.5 Financing

The costs for disposing of the operational waste are paid by the nuclear power companies as they arise, but financing the rest of the nuclear waste programme is based on the payment of fees to a special fund, the Nuclear Waste Fund. This is regulated in the Financing Act and the Financing Ordinance. Management and disposal of waste from other companies with which SKB has agreements are not funded via the Nuclear Waste Fund but rather directly by these companies.

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

At the end of 2015 there was about SEK 58 billion in the nuclear power companies' shares of the Nuclear Waste Fund (market value). In addition, some SEK 39 billion (current price level) has been spent in the creation and operation of today's system and for the research and development. During the period 2015 to 2017, the average fee is 4.1 öre (100 öre = 1 Swedish krona) per kilowatt-hour of electricity produced for the nuclear power plants that are in operation. Barsebäck Kraft AB pays an annual fee of SEK 1 042 million during the same period.

Besides paying fees, the nuclear power companies' parent companies pledge guarantees to cover the fees that have not yet been paid. For the reactors that are in operation, a guarantee is also pledged for the eventuality that the Fund proves insufficient due to unplanned events.

2 Description of the waste system

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

SKB is responsible for the transportation system, which is the same for both low- and intermediate-level waste and 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 Ågestaverket where transport of decommissioning waste will occur 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 waste producers via interim storage and treatment plants to different types of final repositories. Solid lines represent transport flows to existing or planned facilities. Dashed lines represent alternative handling pathways. The following section gives a description of all existing and planned facilities in the system.

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

The system for short-lived low- and intermediate-level waste is partially already existing. SKB's final repository for short-lived radioactive waste, SFR, went into operation in 1988. And there are a number of facilities operated by the nuclear power companies, AB SVAFO and Studsvik Nuclear AB. These facilities include treatment plants, interim storage facilities and near-surface repositories.

SKB plans to extend SFR to obtain space for additional short-lived operational and decommissioning waste.

For long-lived waste there is a need for extended interim storage capacity. This will be solved partly with interim storage facilities at the power plant sites and partly with a shared interim storage facility in the extended SFR. 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

There are treatment plants for short-lived low- and intermediate-level waste at the nuclear power plants and at the Studsvik site. At the treatment plants, the waste is treated and packaged so that it meets the requirements for disposal in SFR or near-surface repositories. The purpose of the treatment may be to 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 low- and intermediate-level waste.Today, these are used as buffer storage for operational waste prior to further handling such as treatment, packing and transport to SFR for disposal.

Dismantling and demolition of the first seven reactors[3] is planned to start before the extended SFR can receive decommissioning waste. This means that the existing interim storage capacity for short-lived waste will need to be extended. Plans for this are described in Section 3.3.3. A new interim storage facility for low-level waste can consist of a paved surface or a simple construction for arrangement of ISO-containers. For intermediate-level waste a construction that provides radiation shielding is required.

[3] Barsebäck 1 and Barsebäck 2, Oskarshamn 1, Oskarshamn 2, Ringhals 1 and Ringhals 2 and the Ågesta reactor.

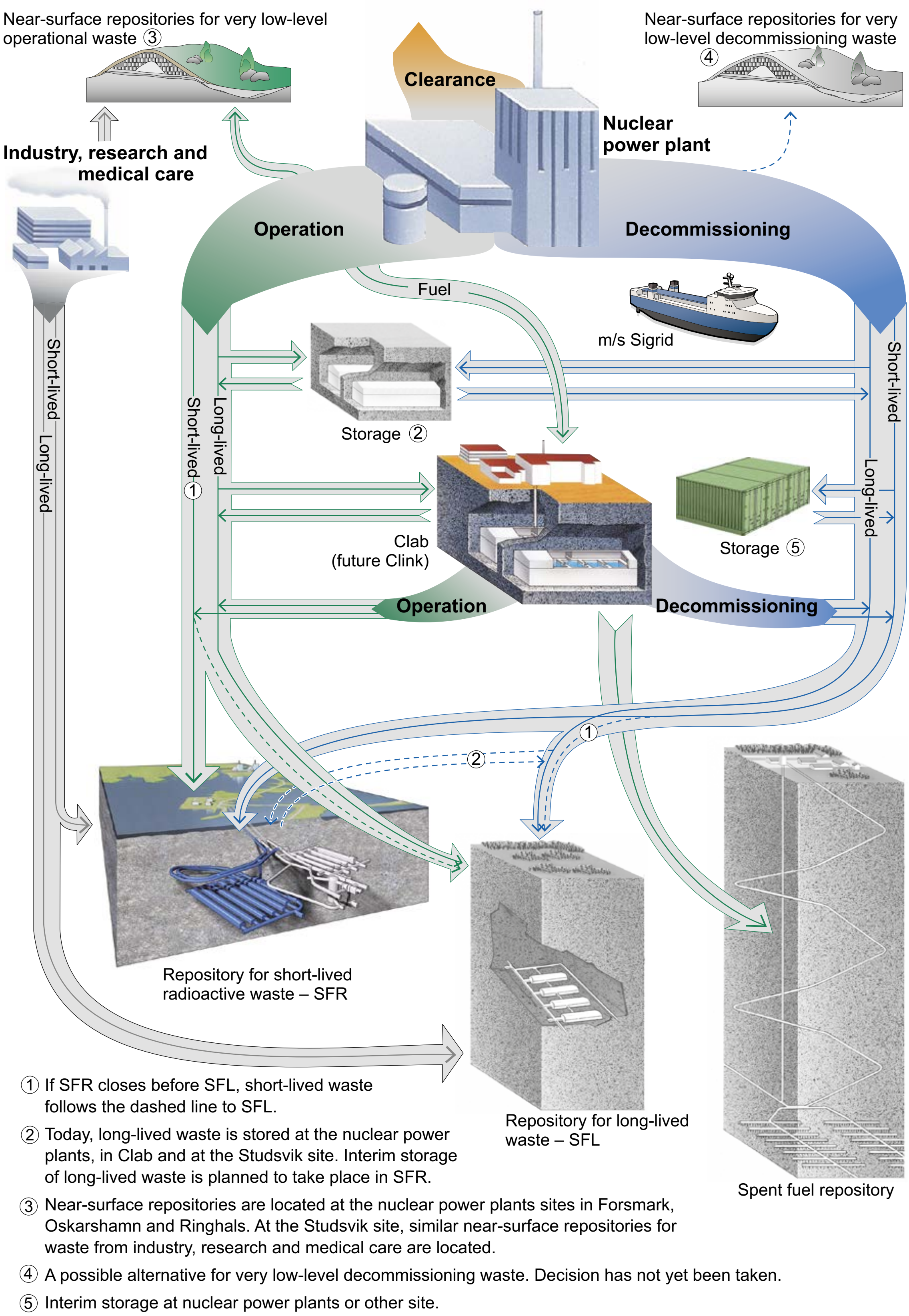


***Figure 2-1.** The system for management and disposal of Sweden's radioactive waste and spent nuclear fuel. Solid lines represent waste flows to existing or planned facilities. Dashed lines represent alternative handling pathways.*

Near-surface repositories

Near-surface repositories are used for very low-level waste. After roughly 50 years, the radioactivity in this waste has decreased to such low levels that it could have been cleared from a radiation protection perspective. There are near-surface repositories today on the industrial sites at the nuclear power plants in Forsmark, Oskarshamn and Ringhals as well as at the Studsvik site.

The existing near-surface repositories at the power plant sites are only licensed for operational waste. As the repositories have limited storage capacity, OKG Aktiebolag and Ringhals AB are investigating the possibility of extending their near-surface repositories. The extension is primarily for operational waste, but the use of the near-surface repositories for parts of the low-level waste from the decommissioning of the nuclear power plants may be possible.

Final repository for short-lived radioactive waste

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

The design of each waste vault is adapted based on the activity level of the waste that is deposited. Low-level waste is disposed in one of the four rock vaults. Intermediate-level waste with lower activity levels is disposed in two of the rock vaults. The intermediate-level waste with the highest activity levels is disposed in the fourth rock vault. The silo will contain most of the radioactive elements in SFR.

The waste in SFR comes mainly from the nuclear power plants, Clab, Studsvik and Ågesta, whereas a minor part comes from industry, research and medical care. At the end of 2015, 38 000 cubic metres of waste had been deposited.

Figure 2-2. The final repository for short-lived radioactive waste, SFR consists of two rock 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) Rock vault, d) View of the silo top.

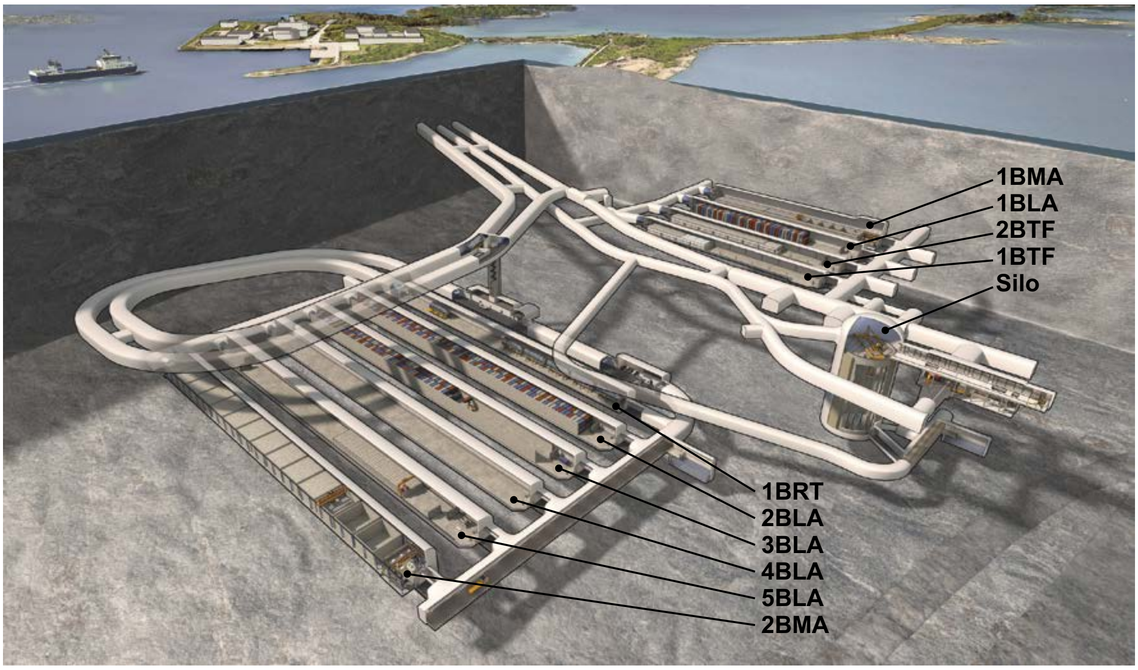


***Figure 2-3.** When SFR is extended, it will consist of four additional waste vaults for low-level waste (2–5BLA), one additional waste vault for intermediate-level waste (2BMA) and one waste vault for reactor pressure vessels (1BRT).*

When SFR was built, the intention was that the facility would receive waste up to 2010. Due to the prolonged operation time of the nuclear power plants also, SFR's operating phase will be prolonged, which imposes new demands on the maintenance of the facility. The maintenance programme includes, in addition to remedial and preventive maintenance, also identification, handling and prevention of age-related deterioration and damage. In recent years a number of maintenance projects have been carried out in SFR. These have included installation of a waterproofing membrane to protect barriers and waste in the rock vault for intermediate-level waste (1BMA) and the silo and the addition of a sprinkler in the operations building. Within the framework of renovation work, projects continue for replacement of fire alarms, evacuation alarms, fibre-optic networks, systems for monitoring and control (SCADA systems) and gates and doors in the underground part of the repository.

Today only operational waste is disposed of in SFR. SFR's storage capacity will be extended to provide room for additional short-lived waste from both operation and decommissioning. SKB has therefore submitted an application for a licence to extend the facility to hold in total about 170 000 cubic metres of waste plus nine reactor pressure vessels (RPVs) from BWRs. The RPVs from PWRs will be disposed of in SFL. Figure 2-3 illustrates SFR according to current plans when it is fully extended.

2.1.2 Facilities for long-lived waste

Treatment of waste

Currently, the possibility exists of segmenting certain used core components in order to be able to place these in steel tanks for storage at the nuclear sites.

If a need for increasing the capacity for interim storage of the spent nuclear fuel arises, SKB can eventually segment the BWR control rods that are interim-stored in Clab. A study to analyse suitable technique and a site for segmentation as well as a site for further interim storage is under way, see Section 3.4.2.

AB SVAFO is currently studying the possibility of constructing a handling facility at the Studsvik site for legacy-waste that stems from nuclear power's early development. The facility will be used to screen and characterise the waste and prepare and pack it prior to final disposal. According to AB SVAFO's plans, the facility would be possible to be commissioned around 2022.

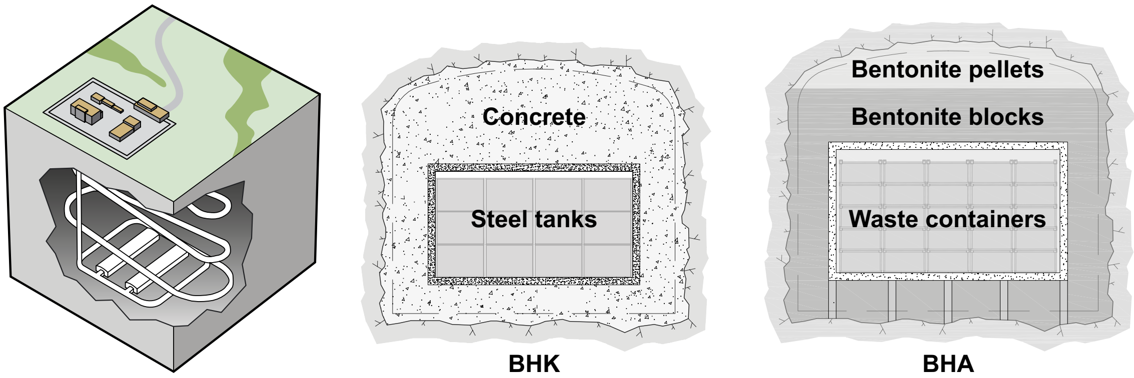


Figure 2-4. *Preliminary layout and the proposed repository concept for SFL, one rock vault for core components (BHK) and one rock vault for legacy-waste (BHA).*

Interim storage facilities

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

The long-lived waste produced when decommissioning the first reactors might be accommodated in the existing interim storage facility. To increase the capacity for interim storage of long-lived waste in the future, SKB has submitted an application to use a part of the extended SFR for interim storage.

Final repository for long-lived waste

SKB plans to dispose of the long-lived waste at a relatively large depth. SFL will be the last final repository in the nuclear waste system to be commissioned. The design of the repository is in an early stage. A proposed repository concept is being evaluated at present with respect to post- closure safety. Siting of the repository is yet undecided.

The storage capacity of SFL will be relatively small compared with SKB's other final repositories. The total storage capacity is estimated to be about 16 000 cubic metres. The proposed repository concept includes two repository parts, one for core components from the NPPs and one for legacy-waste from AB SVAFO and Studsvik Nuclear AB. The core components, which are metallic waste, comprise about one-third of the volume, but contain (initially) the main part of the radioactivity. The repository part for core components will have an engineered barrier of concrete. The other repository part, for long-lived legacy-waste from AB SVAFO and Studsvik Nuclear AB, is suggested to have an engineered barrier of bentonite. The repository concept is illustrated in Figure 2-4.

2.2 Facilities in the KBS-3 system.

SKB's central interim storage facility for spent nuclear fuel, Clab, has been in operation since 1985.

SKB is planning to construct a facility part for encapsulation of the spent fuel adjacent to Clab in Oskarshamn and a final repository, the Spent Fuel Repository, in Forsmark. In addition to these facilities, SKB plans to construct a facility for machining and assembly of the copper canisters.

Central interim storage facility for spent nuclear fuel

The interim storage facility for spent nuclear fuel, Clab, is situated adjacent to the nuclear power plant in Oskarshamn. 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 taken down in a fuel elevator to the storage section where the spent fuel is stored in water pools, see Figure 2-5.

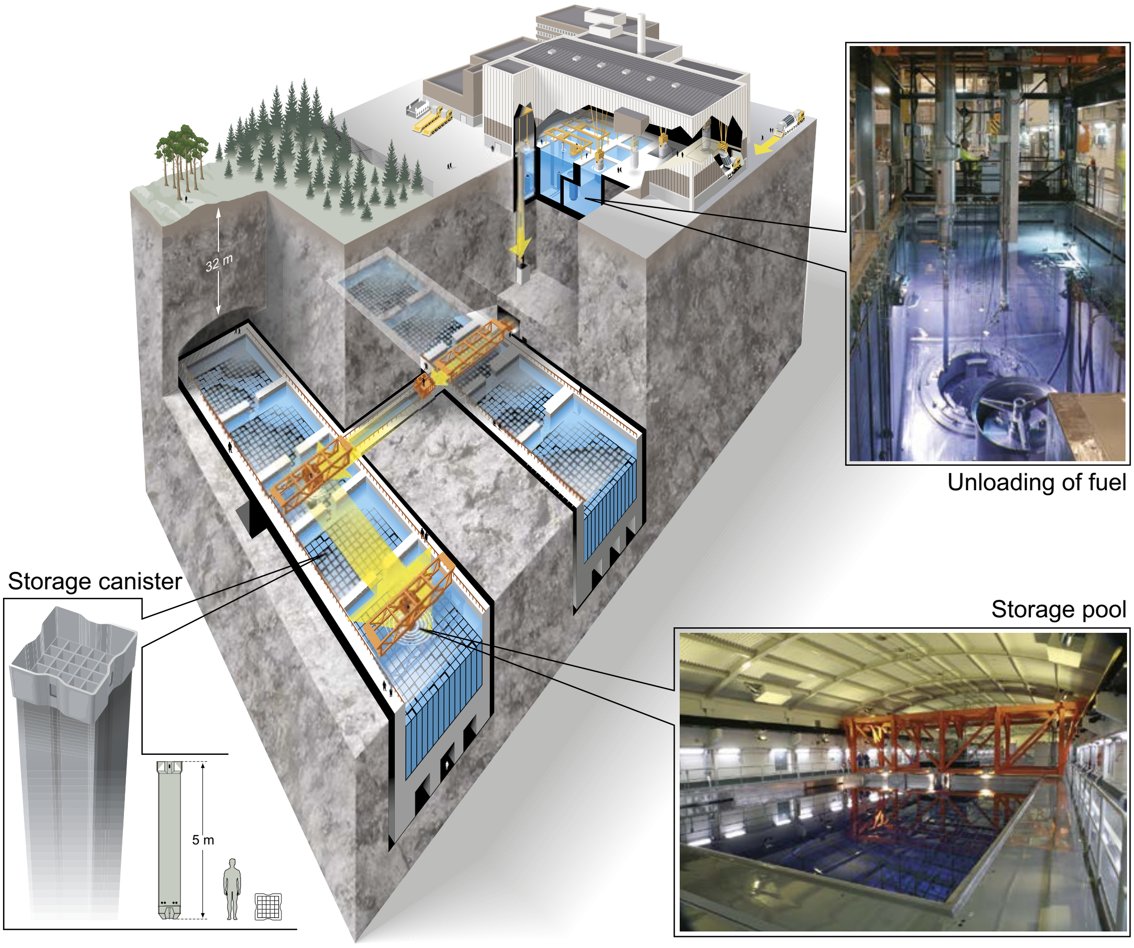


__Figure 2-5.__ The central interim storage facility for spent nuclear fuel, Clab.

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 and 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 the personnel can stand there without radiation protection.

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 under way or have recently been completed, including an upgrade of the cooling chain in order to obtain increased cooling capacity and redundancy, galvanised fire-water pipes being replaced with stainless steel pipes, and a membrane filtration system being installed for improved treatment of effluent. An inventory of the systems in Clab has been commenced to investigate the long-term need for improvements. A number of improvement projects are scheduled today. These include modernisation and upgrading of the fuel elevator and facility modifications for handling of new fuel transport casks. Further modifications being considered are new electricity supply pathways and modernisation of overhead cranes. Planning and execution of the alterations in Clab will be integrated with planning and construction of the new facility part for encapsulation.

At the end of 2015 there were 6 352 tonnes of fuel (weight of the uranium originally in the fuel) in the facility. SKB has a licence to store 8 000 tonnes of fuel in Clab. According to current forecasts, this amount is projected to be reached in 2023. The pools can accommodate a total of about 11 000 tonnes of fuel under the assumption that the core components that are stored in Clab today are unloaded. During 2015 SKB applied for licences to extend the allowed amount to 11 000 tonnes of fuel.

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 part adjacent to Clab, see Figure 2-6. When this encapsulation part has been connected with Clab, the two parts will be operated as an integrated facility, Central interim storage and encapsulation plant for spent fuel, Clink.

The canister consists of a copper shell and an insert of nodular iron, see Figure 2-7. There are two types of inserts, one that holds twelve fuel assemblies from BWRs and one that holds four fuel assemblies from PWRs. There are other fuel types to be disposed of as well, see Section 1.1.4. They can be placed in one of the two insert types.

The canister's different components, such as the insert, copper shell and lid will be produced by different subcontractors. A facility will be needed for final machining, assembly and quality assurance of the canister components. The canister factory will not be a nuclear facility.

In the encapsulation plant, there will be a number of stations for different operations where all handling of fuel occurs remotely and with radiation shielding. The encapsulation process begins with the fuel being placed in a transport canister 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 copper canister is sealed by means of friction stir welding (FSW). The quality of the weld is inspected, and if it is approved the canister is taken to the machining station, where excess material is removed. Finally, a new quality inspection of the weld is performed. If necessary, the canister is cleaned before being placed in a special transport cask for transport to the Spent Fuel Repository. Clink is designed for encapsulation of 200 canisters per year.

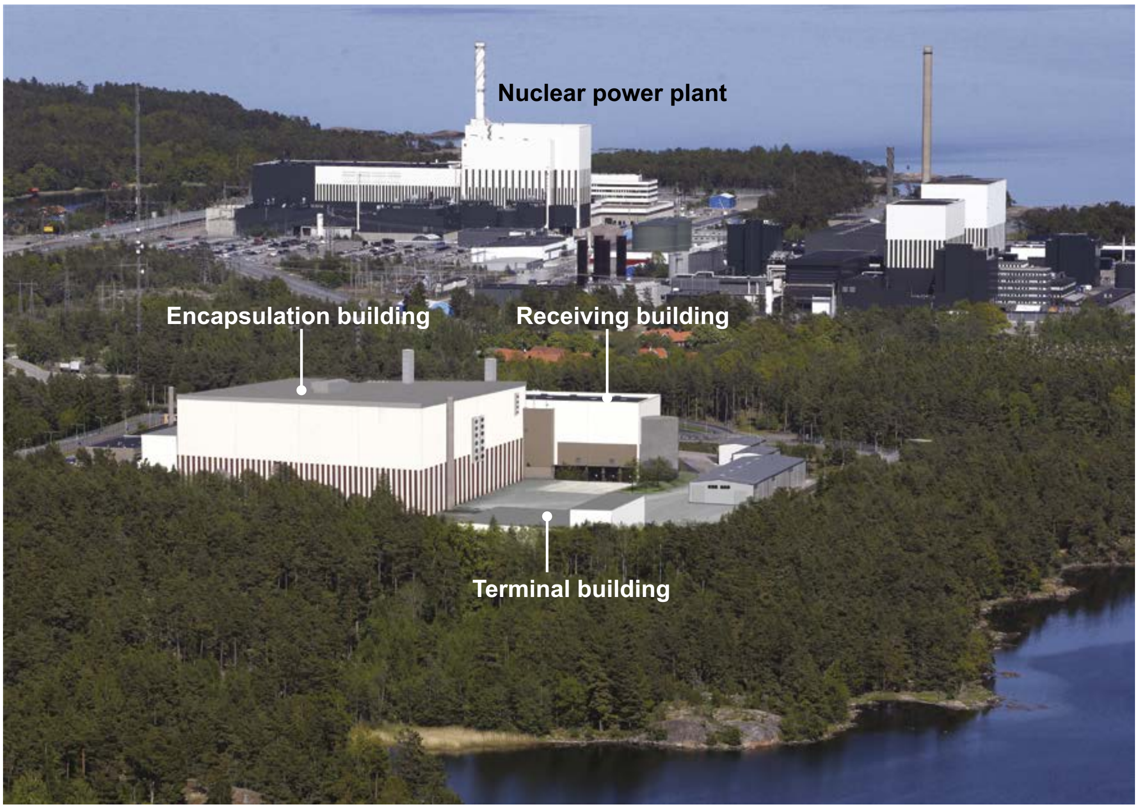


Figure 2-6. *Photo-montage that demonstrates the integrated facility for interim storage and encapsulation of spent nuclear fuel, Clink.*

Figure 2-7. *Copper shell and nodular iron insert (the inset photo shows the copper lid).*

The final repository for spent nuclear fuel

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 Forsmark in Östhammar municipality and Laxemar in Oskarshamn municipality. After evaluations of the site investigations, SKB selected Forsmark as the site for the Spent Fuel Repository. A decisive factor in the selection of Forsmark was that the prospects of achieving long-term safe disposal were judged to be better there.

The final repository will consist of a surface facility and an underground facility, see Figure 2-8. The underground facility consists of a central area and a number of deposition areas plus 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 comprise the repository area, will be located about 470 metres below ground and consist of a large number of deposition tunnels with bored deposition holes at the floor of the tunnels. The positioning of the deposition tunnels, as well as the spacing between the deposition holes and the design of infrastructure at the repository level, is determined on the basis of the properties of the rock, for example the location of large deformation zones, the occurrence of large or highly water-conducting fractures, and the thermal conductivity of the rock. The surface facility consists of an operations area, rock heap, ventilation stations and storeroom. The facility is designed for a maximum deposition capacity of 200 canisters per year.

The canisters are transported to the deposition level via a ramp with a specially built transport vehicle. There the canisters are transloaded to the deposition machine to be carried out to the deposition area and finally deposited. After the canisters have been emplaced in the deposition holes, surrounded by bentonite clay, the tunnel is backfilled with clay that will swell in contact with water and sealed with a concrete plug. When all fuel has been deposited, other openings are also backfilled and the surface facilities are decommissioned.

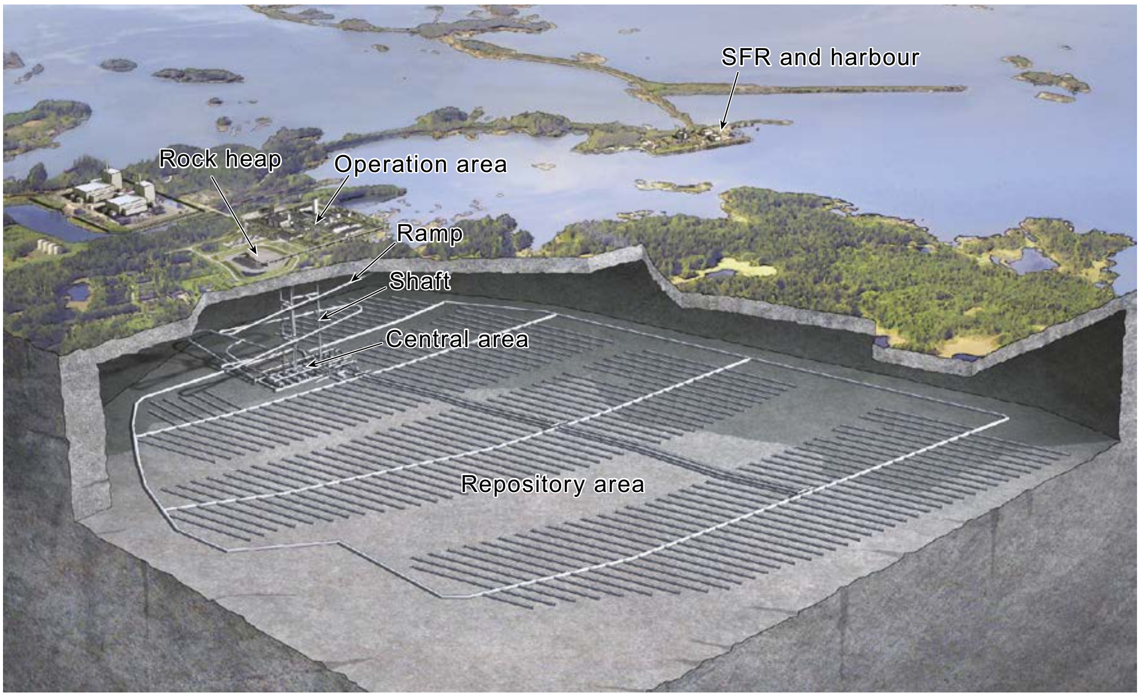


Figure 2-8. Illustration of possible layout of the Spent Fuel Repository in Forsmark.

2.3 The transportation system

SKB's transportation system was built up during the 1980s. It consists of the ship m/s Sigrid, special vehicles for overland shipments and different types of transport containers for fuel and radioactive waste. The ship and the vehicles are used both for shipments of low- and intermediate-level waste and spent nuclear fuel. The different transport containers are developed 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. Sigrid has been constructed to have low fuel consumption and provide low releases to air and water and generally has a lower environmental impact than her predecessor. She can carry twelve fuel- or waste- containers. Normally, the ship, which is operated by a contractor, makes between 30 and 40 trips per year between the nuclear power plants, Studsvik, SFR and Clab.

Short-lived low- 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 shipped in transport containers with 7-20 centimetres thick walls of steel, depending on how radioactive the waste is, see Figure 2-9.

Today part of the long-lived waste, control rods from BWRs, is transported from the nuclear power plants to Clab. The waste is shipped in a transport cask with approximately 30 cm thick walls of steel. The spent nuclear fuel is shipped from the nuclear power plants to Clab in casks with roughly 30-centimetre-thick steel walls. These casks have cooling fins to remove the decay heat generated by the fuel. In view of the amended set of requirements for fuel transport casks, SKB started in 2013 a project to develop new casks. A contract has been signed with an American supplier and completion of a licensing basis for regulatory review is under way. The new casks are designed with double lids to protect the fuel from water penetration after an accident. The casks are constructed with an increased capacity, which means that fewer casks are needed to meet the transport demands in the Swedish system.

A new type of waste transport containers is being developed for shipping long-lived intermediate-level waste. The transport containers are intended for waste that is placed in steel tanks for dry interim storage.

An increased transport volume is envisaged in the future in conjunction with shipments of decommissioning waste and canisters with spent nuclear fuel. There is over capacity in today's transportation system so no major investments are envisaged in order to meet the increased volume.

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

Figure 2-9. *M/s Sigrid and transport container for short-lived radioactive waste (ATB) and transport cask for core components (TK).*

3 Plan of action

This chapter presents plans to construct and commission new and extended facilities. Furthermore, the chapter provides a description of the nuclear power companies and SKB's plans of action regarding decommissioning of nuclear facilities.

The chapter begins with a brief presentation of the effects of the decisions recently taken on a premature shutdown of four reactors. Plans for execution of the nuclear waste programme according to the current main timetable are then presented. Finally, alternative courses of action and measures to handle major changes in the planning assumptions are described.

3.1 Effects of premature shutdown

Since the RD&D Programme in 2013 decisions have been made on a premature shutdown of four reactors (Oskarshamn 1, Oskarshamn 2, Ringhals 1 and Ringhals 2), see Section 1.1.3.

A direct consequence of a premature shutdown is that the total amount of fuel that will be managed in the system decreases. The forecasted amounts, about 11 400 tonnes, still exceed Clab's maximum interim storage capacity of 11 000 tonnes, see Section 2.2.

A large amount of fuel, approximately 340 tonnnes, will arrive earlier at Clab when the final cores from the four reactors being transported there. At the same time, the annual fuel amounts decrease slightly as the four reactors are decommissioned. Clab is calculated to be full according to the current licence, 8 000 tonnes of fuel, around 2023. In early 2015, SKB applied for a licence for increased interim storage capacity to 11 000 tonnes, see Section 3.4.3.

The work load at Clab when the final cores are loaded in will increase as other tasks are performed at the facility at the same time, see Section 3.4.3. Loading in of fuel is therefore planned jointly by SKB and the nuclear power companies so that transport and reception of fuel is ensured. To increase the storage volume for spent fuel, the currently planned measure is transloading of fuel from normal to compact storage canisters. Compact canisters hold more fuel assemblies than normal storage canisters. The corresponding transloading has been done previously when SKB in 1992 received a licence to increase Clab's storage capacity from 3 000 to 5 000 tonnes of fuel. According to the plans, transloading of fuel will begin in 2018.

Another consequence of the premature shutdown is that decommissioning of the first seven reactors will begin before the extension of SFR is commissioned for disposal of the waste. This means that the decommissioning waste needs to be interim-stored at the power plant sites. When the extended SFR is commissioned, the short-lived waste will be deposited while the long-lived waste will either be interim-stored in SFR or at the power plant sites until SFL is commissioned, see Sections 3.3.3 and 3.3.4. This will entail a greater load on the transportation system when the extension of SFR, according to current plans, is commissioned in 2028 and the Spent Fuel Repository in 2030. Today's transportation system has an overcapacity so the increased volume is foreseen to be handled via good planning and high availability in the system.

3.2 Main timetable for the nuclear waste programme

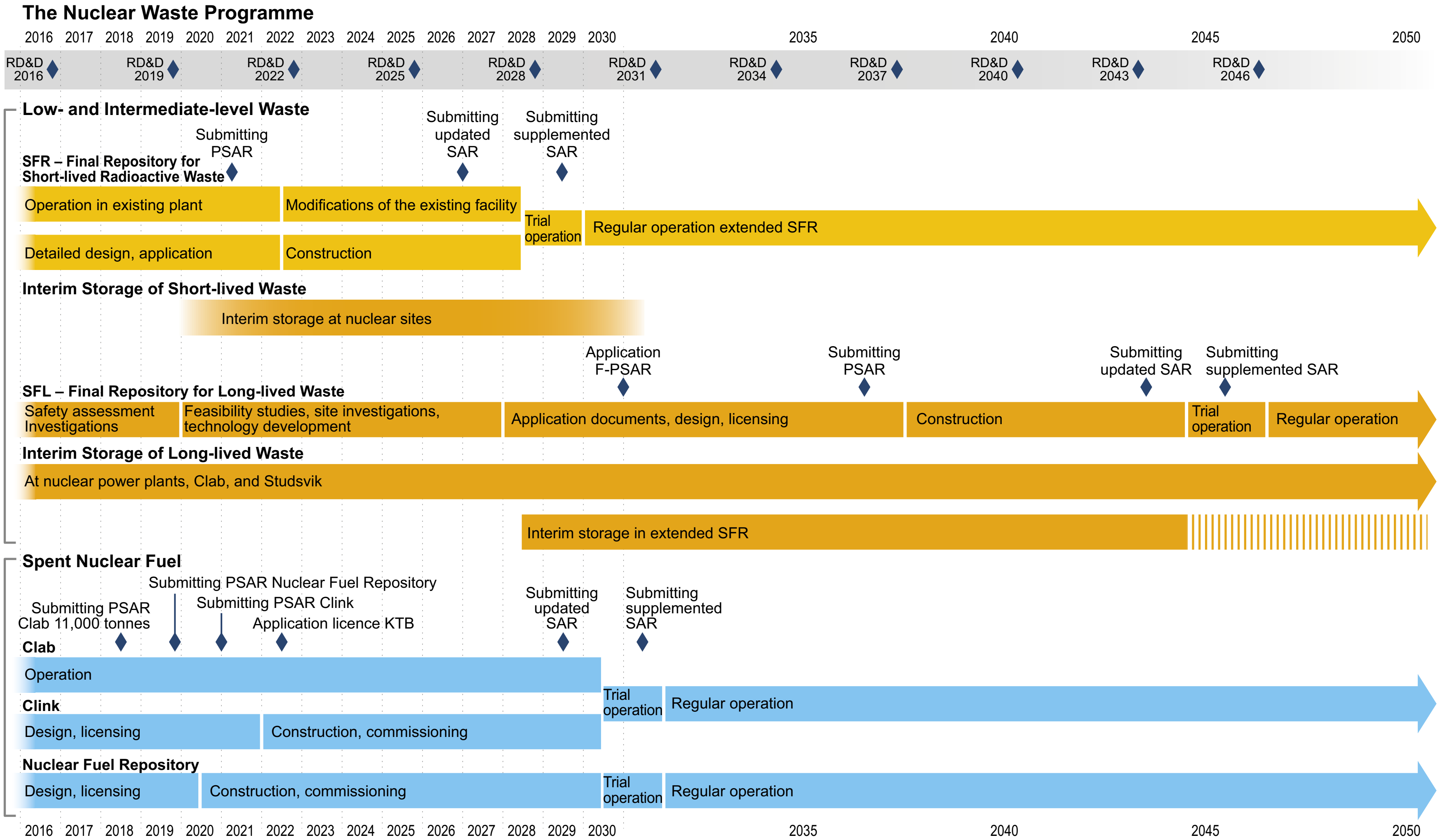
SKB's planning for new facilities involves a stepwise decision process that is based on SSM's regulations. The regulations state, based on international recommendations from e.g. IAEA and OECD/NEA, that development and licensing of nuclear facilities will take place through a process where the requirements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

- **Application for a licence under the Nuclear Activities Act and the Environmental Code to construct a new facility** – in which SKB as a basis for example provides a preparatory PSAR (F-PSAR) where requirements that the facility and activities shall conform to are presented. How the requirements can be met with a suggested reference design and activity is also presented. Consequences for the environment are presented in an Environmental Impact Statement (EIS). During licensing the applications are reviewed by SSM and the Land and Environment Court after they have received comments from different reviewing bodies. The municipalities should, in accordance with the paragraph on the municipal veto in the Environmental Code, approve the activities. Decisions on permissibility under the Environmental Code and a licence under the Nuclear Activities Act are declared by the Government.
- **Approval of the SAR prior to construction** – after licence has been obtained and permission has been granted under the Nuclear Activities Act and the Environmental Code repectively, a preliminary safety analysis report (PSAR) must be approved by SSM before SKB can start construction of the facility. A PSAR shall, in accordance with Chapter 4 § 2 SSMFS 2008:1, give an account of the design of the facility, how operations are arranged and how the requirements are met.
- **Approval of the SAR prior to trial operation and regular operation** – to put the facility into trial operation, which means disposal of radioactive waste under increased monitoring and control, and regular operation, a renewed and a supplemented SAR must be approved by SSM. The SAR shall altogether show how the safety of the facility is arranged to protect human health and the environment from radiological accidents and to prevent unauthorised handling of nuclear material or nuclear waste. The report is supposed to reflect the facility as it is constructed, analysed and verified and demonstrate how the requirements on its construction, function, organisation and operation are fulfilled. To obtain permission for regular operation, the SAR must be supplemented with an account of experience gained from trial operation.
- **Approval of the SAR prior to closure of the repository** – to obtain licences to seal the final repository and decommission the surface facilities, a supplemented SAR and a plan for closure and decommissioning must be approved by SSM.

When trial operation turns into regular operation, the operations enter a management phase. Experience from operations will be utilised systematically. During the operating phase, the studies that are needed to finally select the technology for closure of the repository will also be concluded. Monitoring will also be done of the development in the scientific and technical areas that are of importance for radiation safety. In conjunction with the periodic overall evaluations of safety and radiation protection every ten years, a review will be conducted of the state of knowledge in areas that are essential for radiation safety.

Each of the above milestones is actually two milestones: Firstly SKB's compilation of applications and/or SARs and secondly SSM's and/or other regulatory authorities' approval of these after completed examination. The time points for SARs determine when SKB needs to be finished with their basis for assessment and times for approval when SKB can begin their activities. Both types of milestones are presented below.

Figure 3-1 shows the general timetable, including times for coming applications, for the entire nuclear waste programme. The planning for the additional facilities is largely divided into technology development and siting (SFL), design and licensing, construction, trial operation and regular operation.



Clab Central Interim Storage Facility for Spent Nuclear Fuel
Clink Central Interim Storage Facility and Encapsulation of Spent Nuclear Fuel
RD&D Programme for research, development and demonstration
KTB Transport cask for canisters with spent nuclear fuel

F-PSAR Preparatory Preliminary Safety Analysis Report
PSAR Preliminary Safety Analysis Report
SAR Safety Analysis Report

SFL Final Repository for Long-lived Waste
SFR Final Repository for Short-lived Radioactive Waste

Figure 3-1. *General timetable for SKB's nuclear waste programme.*

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

3.3.1 Current situation

The programme for low- and intermediate-level waste includes partly day-to-day management of existing waste and partly work to realise the remaining parts of the system that is needed to manage and dispose of low- and intermediate-level waste in a long-term safe manner. The activity is primarily led by SKB, but in some respects also by the nuclear power companies.

The activities on low- and intermediate-level waste required to realise the remaining parts of the system can be summarised in the following points:

- Applications under the Nuclear Activities Act and the Environmental Code to extend SFR were submitted at the end of 2014 and the licensing process is currently under way. Supplementary information was submitted in 2015 and 2016.
- In preparation for the extension of SFR, issues concerning licensing are dealt with, and in parallel the technology development, design and building preparations continue.
- An evaluation of the post-closure safety for the proposed repository concept for SFL started during the spring of 2015 and is expected to continue until the middle of 2018. The safety evaluation will constitute an important basis for the continued development.
- An initial study of the process that will lead to the future selection of a site for SFL has been carried out. Further studies will refine the planning and identify needs for expertise and resources.
- Future handling and interim storage of BWR control rods have been studied. To ensure the interim storage capacity for fuel in Clab, a more efficient interim storage of BWR control rods in Clab can eventually be necessary.
- A new container for transport of steel tanks with long-lived intermediate-level waste is under development.

3.3.2 Overall planning

Decisions on the premature closure of four reactors affect the action plan for low- and intermediate-level waste by increasing the need for interim storage and also the need for decommissioning planning to be developed and concretised sooner.

The final repositories that SKB plans to establish for low- and intermediate-level waste include an extension of SFR and construction of SFL. Some of the nuclear power companies have also the intention of arranging temporary interim storage facilities for short-lived decommissioning waste until the extension of SFR is commissioned. The long-lived decommissioning waste will be interim-stored at the power plants and in the SFR extension when it is ready for operation.

SKB has, together with the nuclear power companies, investigated the storage capacity at each facility and the storage needs that will arise until the extended SFR is commissioned. If interim storage facilities for short-lived waste should be located at the site of the nuclear facilities or centrally, has been a part of the study and conducted work shows that SKB should not pursue the issue of central interim storage facilities under its own auspices.

Figure 3-2 shows a general timetable for low- and intermediate-level waste, together with important milestones. In order to visualise the relation to the decommissioning of the nuclear facilities the figure shows preparatory measures as well as dismantling and demolition of reactors, Clink, SFL and SFR.

3.3.3 Short-lived waste

Interim storage of short-lived waste

The work with dismantling and decommissioning the first seven reactors is planned to start before the extended SFR is commisioned. Barsebäck Kraft AB, OKG Aktiebolag and Ringhals AB therefore plan to store the short-lived decommissioning waste at the power plant sites or at another site. There will also be a need for interim storage of operational waste during the period when construction of extension of SFR is in progress. When rock excavation is carried out at the facility SFR will be closed for disposal. If the waste suppliers have large needs for disposal during the work on the extension, a possible time window for disposal could be planned after completed rock excavation.

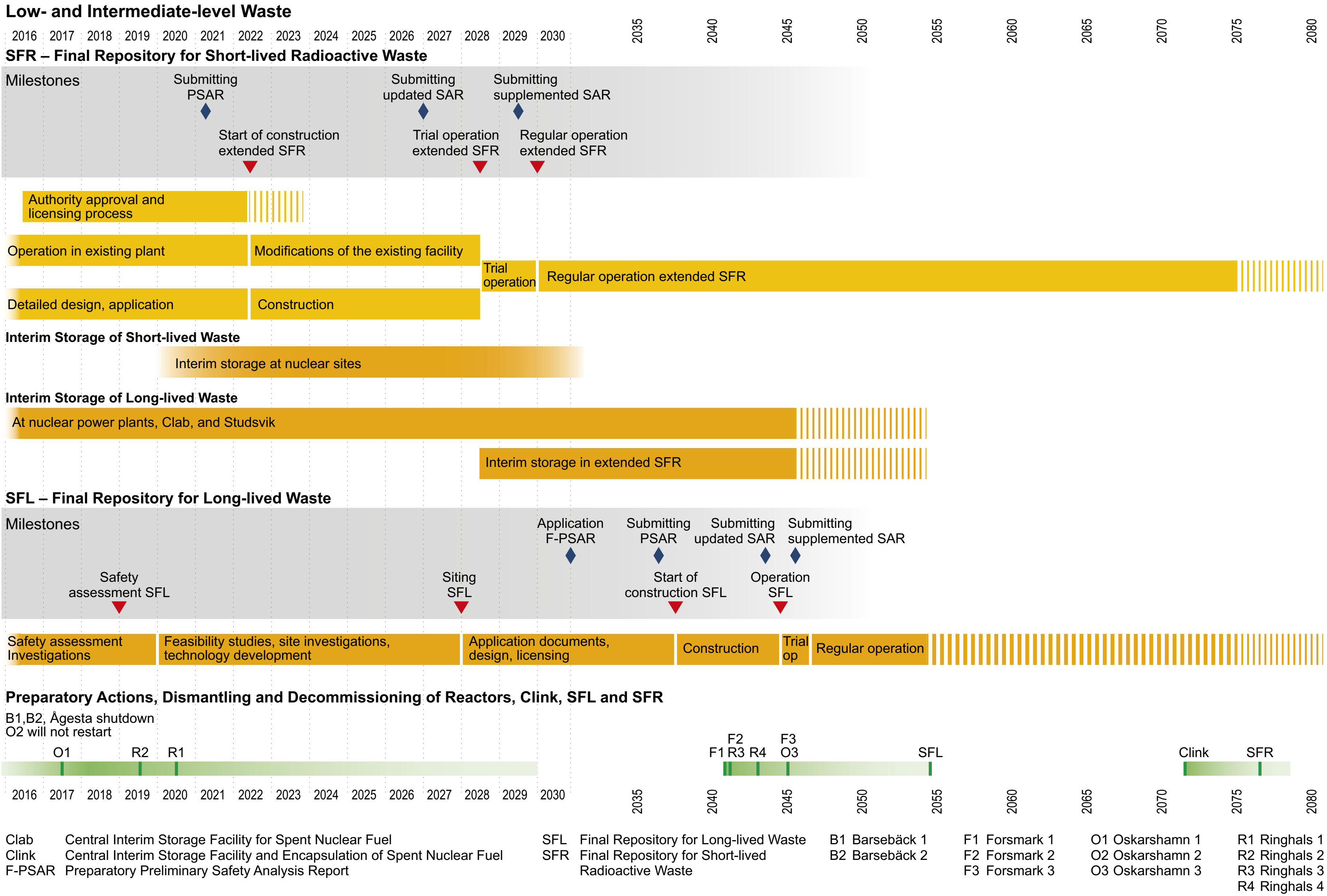


Figure 3-2. *Timetable for low- and intermediate-level waste and decommissioning of the nuclear power plants. Dashed bars mark uncertainties and flexibility in the planning.*

Furthermore, a repository part in the existing SFR, the rock vault for low-level waste, is expected to reach its storage capacity within a couple of years. This means that this waste must also be interim-stored until SFR is extended.

Barsebäck Kraft AB has existing storage facilities that can be used for interim storage. To allow for interim storage of the short-lived waste that arises from decommissioning of Barsebäck 1 and Barsebäck 2, there is a need to increase the existing capacity. Barsebäck Kraft AB has carried out an internal study about the need for interim storage facilities. The study shows that storage capacity can be made available at Barsebäck's facility or externally at another nuclear facility.

For interim storage of short-lived waste from operation and decommissioning of Oskarshamn 1 and Oskarshamn 2 produced before the extended SFR can be commissioned OKG Aktiebolag judges that storage capacity at the facility needs to be increased. Increased capacity is also needed for interim storage of reactor pressure vessels.

Ringhals AB has existing storage facilities that can be used for interim storage but the capacity needs to be increased to handle the short-lived waste from decommissioning of Ringhals 1 and Ringhals 2, including a reactor pressure vessel from Ringhals 1 (BWR).

Forsmarks Kraftgrupp AB has existing storage facilities that can be used for interim storage. The assessment is that there is sufficient capacity for interim storage of operational waste. This is valid provided that space can be made available in the storage facilities before construction of the extension of SFR begins.

AB SVAFO is planning to construct a new building for interim storage of low- and intermediate-level waste from AB SVAFO´s own operations. The interim storage facility will be at the Studsvik site and is planned to be in operation around 2019.

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

Management of very low-level waste

During dismantling and demolition of a nuclear power plant, both conventional and radioactive waste, that need to be managed and disposed of, is produced. The radioactive waste can be deposited in SFR, SFL or near-surface repositories. The waste that will not be deposited in SFL or SFR and contains such small amounts of radioactivity that it is judged possible to release from regulatory control, is handled according to conditional clearance [4] or disposed of in a near-surface repository, see Section 18.3.7.

Extension of SFR

The completion of an application to extend SFR has included detailed investigations, analyses and calculations. 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 estimated, the construction of the extension is expected to start in 2022, with planned trial operation in 2028. A general timetable for the extension of SFR is presented in Figure 3-3.

Licensing and design for the extended SFR

The licensing process for the applications under the Environmental Code and the Nuclear Activities Act is currently under way. Processing of the applications starts with the Land and Environment Court at Nacka District Court and SSM, after which Östhammar Municipality and the Government have to take political decisions. The initiative lies largely with these bodies, and the length of the process is dependent on how long they take for processing and decision. SKB's task consists of aiding the licensing process in different ways through submitting supplementary information when it is requested. At the same time, SKB continues development work on the repository's barriers.

[4] Conditional clearance means that clearance is connected with restrictions regarding how the material may be handled after clearance.

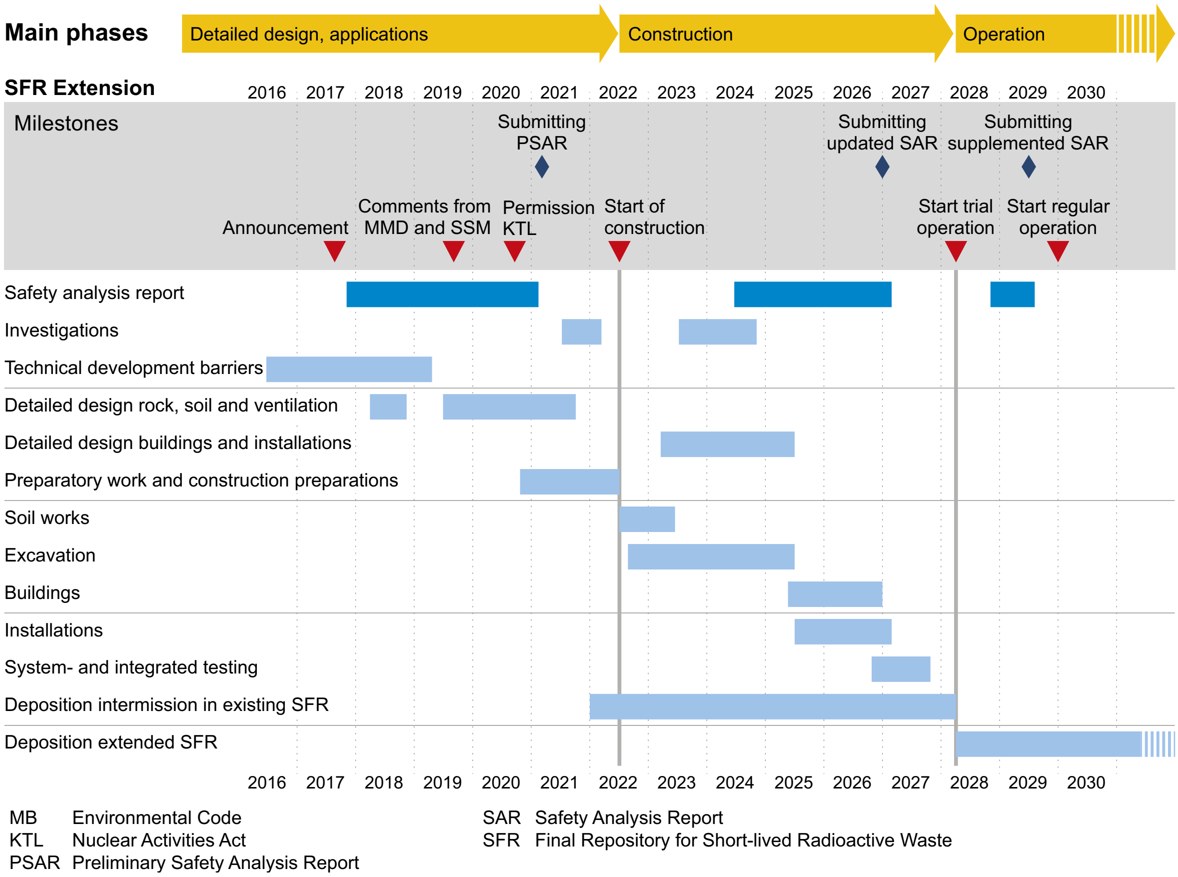


Figure 3-3. *General timetable for extension of SFR.*

During the RD&D period, a PSAR will be prepared. It is submitted to SSM after a licence has been obtained under the Nuclear Activities Act.

During the latter part of the licensing process, detailed design, building preparations and investigations are also carried out.

Provided the licensing process does not take longer than expected, the necessary licences are expected to be obtained so that construction can be commenced in the early 2020s. To start construction SSM must approve PSAR. Furthermore, a building permit is required from the municipality of Östhammar for construction of the surface facility.

Construction and commissioning of the extended SFR

The phase for construction and handover to the operating organisation includes the activities construction, trial operation and handover to regular operation. When the rock excavation work is carried out during the construction phase, the facility will be closed for disposal. If the waste suppliers have large needs for disposal during the construction, a possible time window for disposal could be planned after completed rock excavation. 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 at the end of 2026. Trial operation with disposal of waste in the extended part of SFR is assumed to start about one year later. After about one year of trial operation, SKB intends to submit a supplemented SAR. According to SKB's assessment, an approval for regular operation can be obtained around 2030.

3.3.4 Long-lived waste

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

- Highly neutron-irradiated core components. The waste is produced in connection with both maintenance and with 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.
- Pressure vessels from PWR reactors. The waste is produced during decommissioning of reactors. The reactor pressure vessels may be handled with core components and reactor internals left in the pressure vessel.
- Long-lived waste from Studsvik Nuclear AB's activities and from medical care, research and industry. 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. This waste is managed and interim-stored by AB SVAFO.

The plans for management and disposal of the long-lived low- and intermediate-level waste aim at an integrated system for management and final disposal.

At the end of 2013, SKB presented a tentative system, including a repository concept, for final disposal of the long-lived low- and intermediate-level waste (Elfwing et al. 2013). The proposed repository design is based on the waste and its properties. Since the waste can be roughly divided into two main fractions, with different properties regarding both material and nuclide content, the waste is distributed in two repository parts with different barrier solutions. As the next step in the development, a safety evaluation of the proposed repository concept started in 2015. The safety evaluation will be concluded in 2018. The planning for the different parts of the system is presented under respective headings below.

Interim storage of long-lived waste

SFL is planned to be commissioned around 2045. Since several reactors according to the current plan will be decommissioned before the final repository is comissioned, capacity for interim storage of the long-lived waste from decommissioning is needed.

Today Forsmarks Kraftgrupp AB operates an interim storage facility for dry storage in a building at the power plant site, where for example long-lived waste produced in conjunction with maintenance outages and power uprate is stored. The waste consists of segmented reactor internals placed in steel tanks prior to interim storage.

OKG Aktiebolag today operates an interim storage facility, a rock cavern on the Simpevarp Peninsula (BFA), for dry interim storage of long-lived waste. The operating permit is held by OKG Aktiebolag, but BFA is licensed for interim storage of core components from all Swedish nuclear power plants. So far waste from the Oskarshamn nuclear power plant and Clab is stored in BFA. BFA is deemed to have sufficient capacity for the long-lived waste that arises from decommissioning of Oskarshamn 1 and Oskarshamn 2.

Ringhals AB today 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. Ringhals AB's current planning includes interim storage of whole reactor pressure vessels from PWR at site until SFL has been commissioned.

Barsebäck Kraft AB have recently constructed a new building on their power plant site for dry interim storage of long-lived waste from operation and decommissioning of Barsebäck 1 and Barsebäck 2. The waste will consist of segmented reactor internals that are placed in steel tanks prior to interim storage.

AB SVAFO operates today an interim storage facility for low- 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. AB SVAFO is further planning to construct a new building for interim storage of low- and intermediate-level waste. The interim storage facility will be at the Studsvik site and will be in operation around 2019.

SKB is planning a central interim storage solution for the long-lived waste. As a part of the applications to extend SFR, SKB submitted an application for interim storage of long-lived waste in the extended part of SFR. The application concerns long-lived waste from operation and decommissioning of the nuclear power plants which consists of segmented reactor internals placed in steel tanks prior to interim storage. Interim storage can begin when the extended part of SFR is commissioned, and continue until it is possible to move the waste to SFL or another interim storage facility than SFR. The repository part that is planned for interim storage will then be used for final disposal of short-lived decommissioning waste from the last reactors in the programme.

SKB continuously reassesses the need for interim storage of long-lived waste in relation to existing and planned capacity, to ensure safe and efficient interim storage solutions. Interim storage capacity is judged to be a key factor in planning the management of the long-lived waste. With good interim storage capacity a robust management system is ensured for the long-lived waste that is less sensitive to changes in assumptions and timetables.

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

Figure 3-4 shows a general timetable for interim storage of long-lived waste.

Treatment of long-lived waste

Conditioning of the long-lived waste may be needed before disposal in SFL. Before final conditioning can be carried out, acceptance criteria for the long-lived waste need to be established. The development of preliminary acceptance criteria for the long-lived waste will continue after the ongoing safety evaluation has been completed. Two main alternatives for the long-lived waste are included in the current planning: Stabilisation of metallic waste from the nuclear power plants in steel tanks as well as transloading and conditioning of waste from AB SVAFO and Studsvik Nuclear AB in new waste containers adapted for SFL (Pettersson 2013). According to current plans, the possible conditioning of the long-lived waste will be made in conjunction with disposal in SFL. This is advantageous from a radiation safety perspective, since the waste has by then decayed for a longer time. Furthermore, the acceptance criteria for SFL must have been determined before conditioning.

Transportation of long-lived waste

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

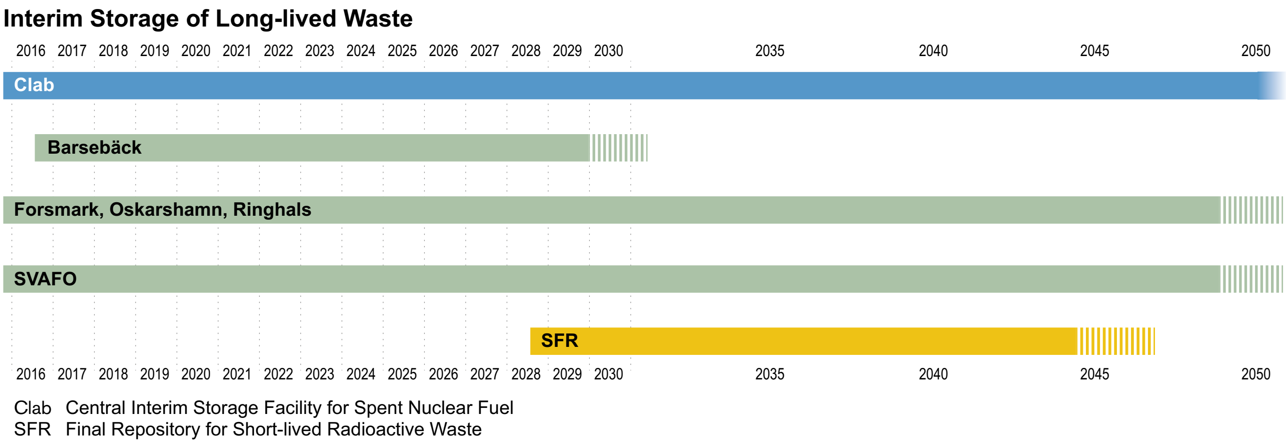


Figure 3-4. *Estimated timetable for interim storage of long-lived waste. SFL will comissioned around 2045.*

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 obtained by the United States Department of Transportation. The approved certificate, type B(U), is then reviewed and validated by SSM. Application documents for licensing were submitted to U.S. NRC in September 2015. U.S. NRC requested in 2015 some supplementary tests, which has delayed the licensing process. According to the revised timetable, the transport container will be delivered in 2020.

The final repository for long-lived waste, SFL

SFL is the repository that will be commissioned the last. Up to commissioning, there are several important milestones that must be passed, such as evaluation and analysis of post-closure safety, siting, preparation of applications, construction, etc. A general timetable for the work with SFL is presented in Figure 3-5. 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 plan, it will be possible for the final repository to be commissioned around 2045. To provide for the needs of the nuclear power companies, it is judged to be necessary to keep the repository in operation for about 10 years.

Repository concept for SFL

In 2013, SKB presented a study where different repository designs were evaluated (Elfwing 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. According to the proposal, SFL will be designed as a geological final repository at a relatively great depth with two different repository parts:

- A repository part for metallic waste from the nuclear power plants, such as core components and reactor pressure vessels from PWRs, which will be designed with a concrete barrier.
- A repository part for the mainly legacy-waste from AB SVAFO and Studsvik Nuclear AB which will be designed with bentonite barrier.

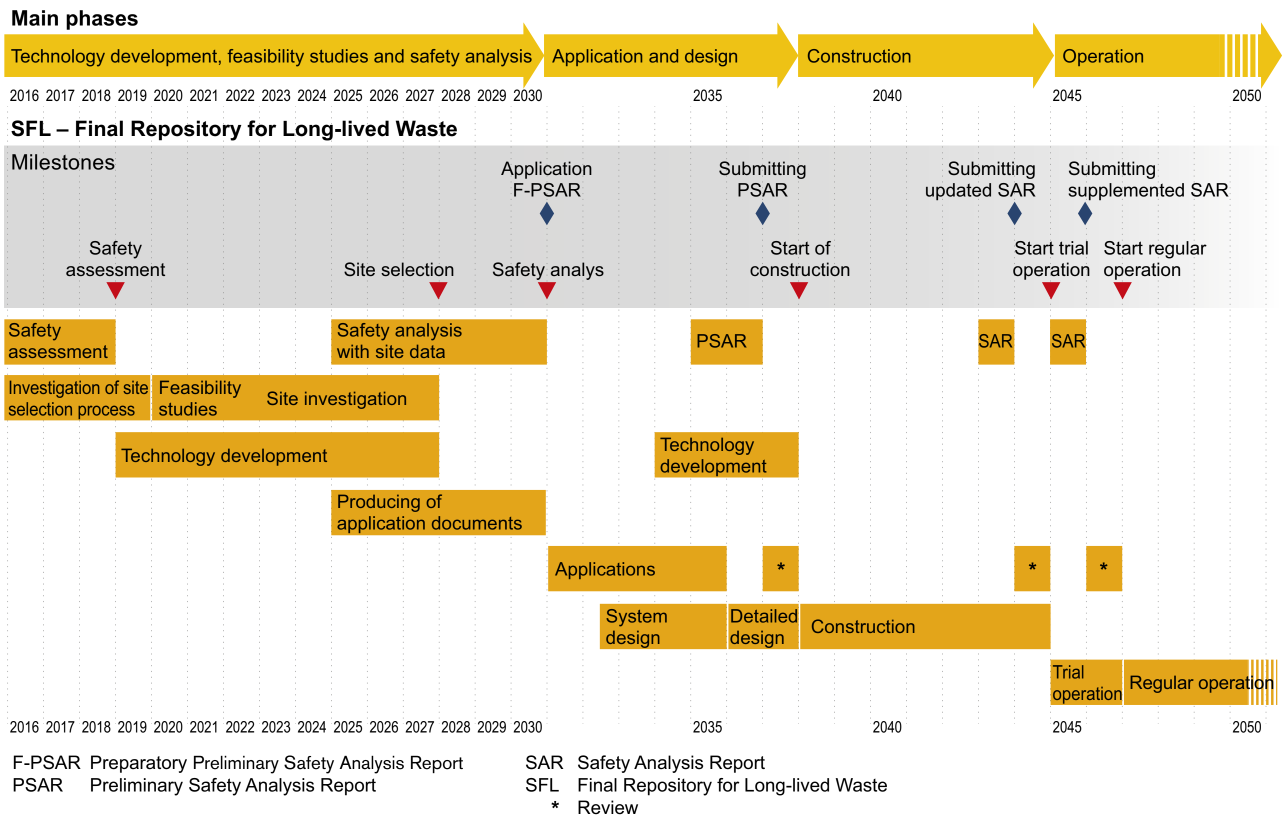


Figure 3-5. Estimated timetable for the activities until regular operation of SFL.

Evaluation of safety after closure of SFL

The next step in the development work is to evaluate the proposed repository concept with respect to post-closure safety. The safety evaluation is one step in the iterative process which SKB follows for development of final repositories for radioactive waste, where technology development and research are followed by evaluation of post-closure safety.

The purpose of the safety evaluation is to provide a basis for SKB to assess whether the proposed concept has the potential to meet requirements on post-closure safety. Furthermore, the safety evaluation should provide a basis for assessing under what conditions the repository concept (waste, barriers and the repository environs) has the potential to satisfy the safety requirements. The results of the safety evaluation are necessary for any modification of the concept, the evolution of the engineered barriers, waste acceptance criteria and for site selection. The evaluation should also provide basis for identifying the areas where SKB needs to improve the state of knowledge to be able to later make complete assessments of safety during operation and after closure of SFL. The safety evaluation is planned to continue until 2018.

Technology development

A feasibility study concerning the design of the final repository has been carried out. The purpose of the feasibility study is to identify, at an early stage, development needs and thereby act as a basis for the technology development plan for SFL.

The feasibility study has used experience and expertise from the process survey and design of the Spent Fuel Repository and the extension of SFR. The results of the study provide an overall description of the facility's design, its technical systems and its overall function. The processes, activities and functions that are expected to be a part of the operation of the facility are described with the aid of for example general function layouts and flowcharts.

With the exception of the waste vaults, the construction of the final repository is not expected to require any technology development specific for SFL. The use of commercially available equipment is judged possible and experience from construction of the Spent Fuel Repository and the extension of SFR will benefit SFL. However, the development of technical solutions for design and construction of the repository parts and management and final disposal of large components such as intact PWR reactor pressure vessels is judged to require targeted research efforts, specific for SFL.

Regarding equipment and systems for the operating period, for example terminal vehicles and overhead cranes, no specific development is planned for SFL as experience from SFR and the Spent Fuel Repository can be used.

For backfill of the waste vaults, it is however deemed necessary to develop technical solutions for installing concrete and bentonite specifically for SFL. In parallel with the safety evaluation, different methods for backfilling are investigated.

Siting of SFL

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

- 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.

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 actors are clarified early on and where the different steps in the process have been agreed upon and communicated. SKB will therefore, as for previous processes, identify and present the siting factors that comprise the basis for evaluation and selection.

As a result of the ongoing safety evaluation, SKB will specify safety-related requirements for the site. Based on these, the siting factors that will be used for site evaluation will be identified and established. The siting factors are not expected to differ much from the factors used in previous siting processes conducted by SKB, but a revision based on the safety evaluation conclusions and previous experience will be made. In addition, during the next three-year period SKB plans studies for example to identify competency requirements and the appropriate organisation for the siting work. The aim is to, in the RD&D Programme 2019, be able to present siting factors, plus a plan for the siting process as a whole.

With the siting factors and the understanding of Sweden's geology that has been gained by SKB during previous siting processes as a basis, the next step in the siting can begin. In addition to safety-related properties, local acceptance is essential for siting, but also other factors such as human health, environment, infrastructure and societal resources are factors taken into account. The concerned municipalities and other stakeholders will be involved in this step in the process, which will result in an overall evaluation of the properties of candidate sites with respect to siting factors.

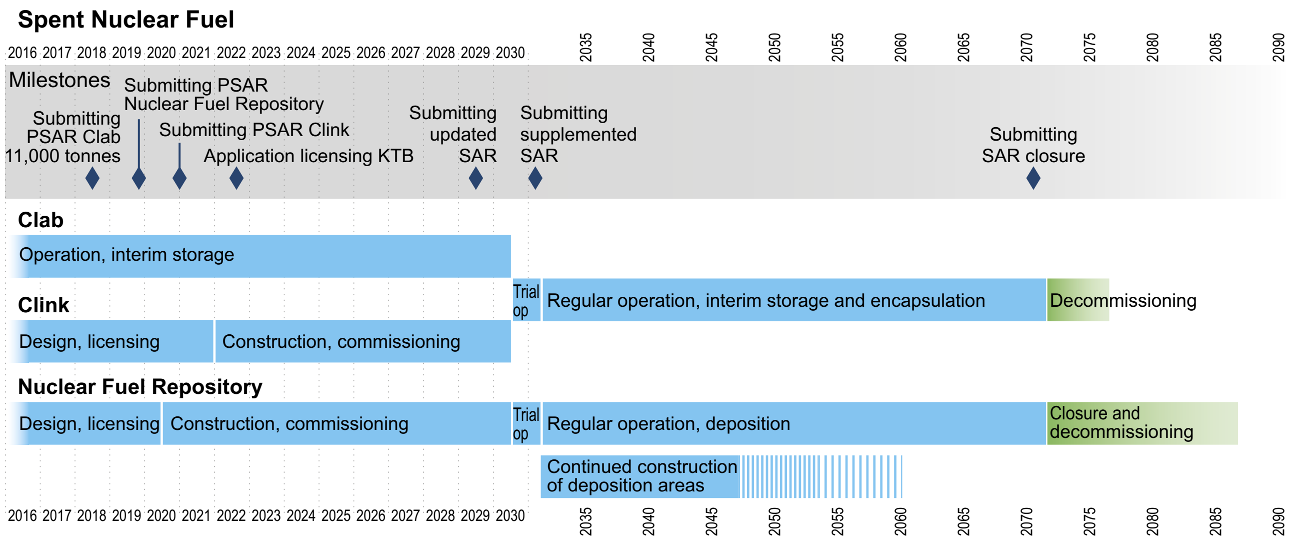
Site investigations constitute the next step in the process with the purpose of learning more about site-specific properties. Site investigations entail both characterisation of rock properties through test drilling and measurements, and an inventory of surface ecosystems. The results of the investigations will serve as a basis for future safety assessments and thereby for site selection.

Applications, construction, operation and closure of SFL

Applications under the Nuclear Activities Act and the Environmental Code for SFL will be submitted around 2030. After the applications have been submitted, work will continue with for example system design. Detailed design will commence when a licence to construct SFL has been obtained. Construction and trial operation will be followed by regular operation. Closure of SFL is planned when all interim-stored long-lived waste and the long-lived waste from decommissioning of the last nuclear power plant has been deposited. Before closure, SKB needs to ensure that the waste from decommissioning of Clink is suitable for SFR and does not need to be disposed of in SFL.

3.4 Plan of action for spent nuclear fuel

This section describes the current situation and the overall plan for future activities for the different facilities in the KBS-3 system. Figure 3-6 shows a general timetable with important milestones.



Clab	Central Interim Storage Facility for Spent Nuclear Fuel	PSAR	Preliminary Safety Analysis Report
Clink	Central Interim Storage Facility and Encapsulation of Spent Nuclear Fuel	SAR	Safety Analysis Report
KTB	Transport Cask for Canisters with Spent Nuclear Fuel		

Figure 3-6. *General timetable for the different facilities in the KBS-3 system.*

3.4.1 Current situation

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.
- SAR 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 and preparations for increasing the interim storage capacity in Clab to more than 8 000 tonnes of fuel.

The two construction projects and the work with 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 what, with current experience, are judged to be realistic timetables for licensing, construction and commissioning and aims at starting operation of the system as soon as possible.

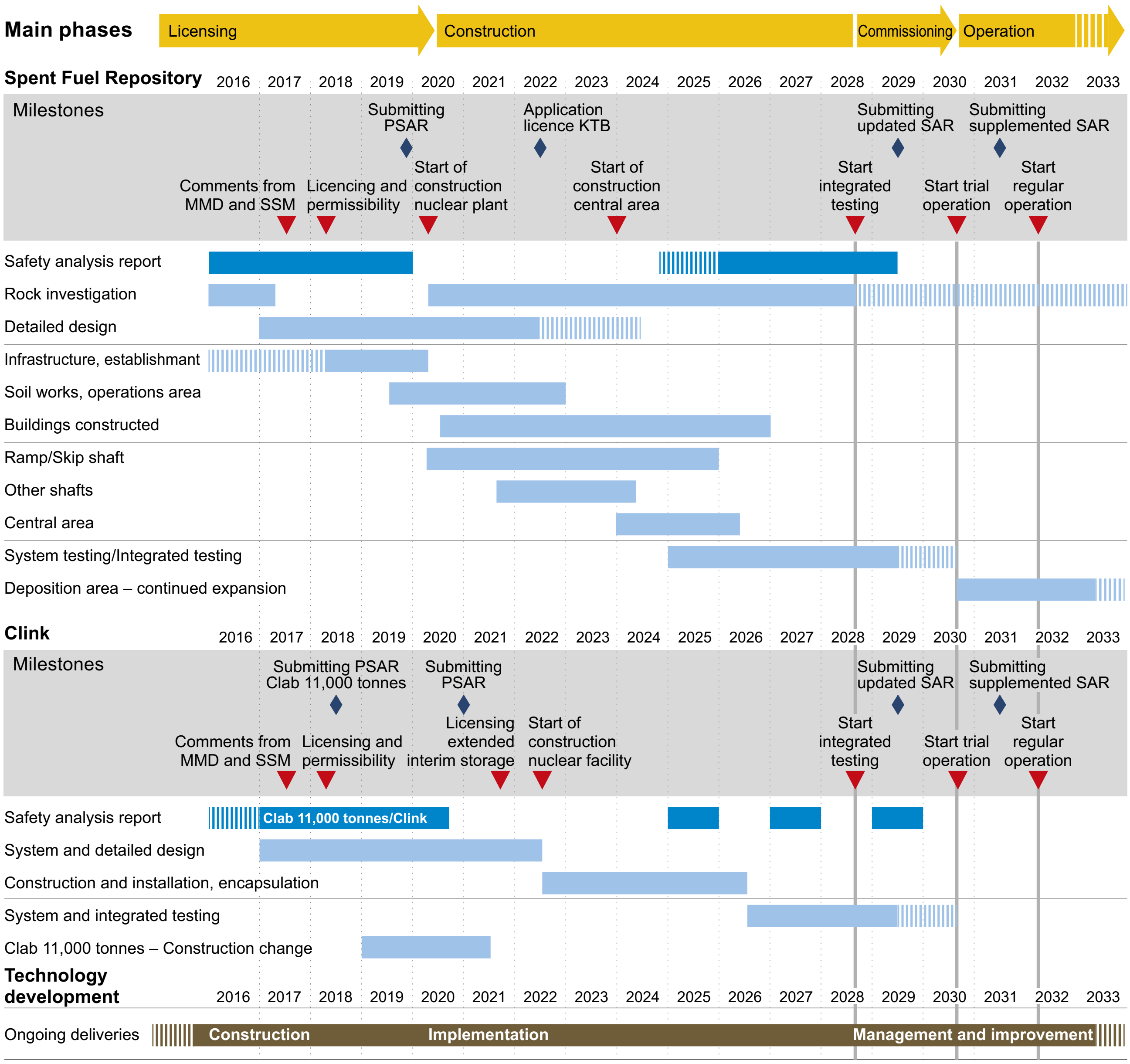
The current situation for the work on the remaining parts and facilities in the KBS-3 system can be summarised in the following points:

- 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.
- Licensing is under way and SKB is responding to questions and requests for supplementary information from both SSM and the Land and Environment Court. An announcement of the applications under the Environmental Code and the Nuclear Activities Act was made in January 2016. SSM offered its statement to the Land and Environment Court in June 2016.
- For the Spent Fuel Repository, the system design of the final repository's facility parts and technical systems has been completed. The preparatory work prior to detailed design is now under way, which, among other things, includes supplementary geotechnical investigations, preliminary design and studies as well as the formulation of requirements prior to detailed design. Compensatory environmental measures have been adopted on the site, and the day-to-day work of monitoring the site and managing buildings, land and drilling sites is under way.
- The work with 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).
- A modernisation of SAR for Clab has been completed and has recently been submitted to SSM.
- In the work with design premises for Clab 11 000 tonnes, a set of requirements is established and the need for changes in the safety concept is specified. Furthermore, an update is made of the analyses that are dependent on the inventory.
- Work with the PSAR for Clab with regard to increased storage capacity to 11 000 tonnes begins in 2016/2017 and will be completed in the summer of 2018.
- Work on system design of the integrated interim storage and encapsulation facility, Clink, has been initiated.

Technology development projects for the Spent Fuel Repository are structured in accordance with the so-called 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 and commissioning. The activities planned during different phases are summarised for each facility in Sections 3.4.3 to 3.4.6. Most of the milestones shown in Figure 3-7 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, see Chapter 4. Other milestones involve the development of the licensing process based on official timetables from the Land and Environment court and SSM and, in some cases, assessments by SKB.



Clab Central Interim Storage Facility for Spent Nuclear Fuel
Clink Central Interim Storage Facility and Encapsulation of Spent Nuclear Fuel
KTB Transport Cask for Canisters with Spent Nuclear Fuel
MMD Land and Environment Court
PSAR Preliminary Safety Analysis Report
SAR Safety Analysis Report
SSM Swedish Radiation Safety Authority

__Figure 3-7.__ Estimated general timetable for establishment of the Spent Fuel Repository and Clink based on the current situation in the licensing process for KBS-3. The technology development needed for the specified milestones is presented in Chapter 5.

Over the next few years, SKB will gradually prepare for construction of the Spent Fuel Repository and the encapsulation part of Clink. According to the plans, construction of the Spent Fuel Repository's accesses will begin around 2020. For Clink, construction will commence around 2022 so that the facilities can be commissioned simultaneously in 2030.

Compared with the plans presented in the RD&D Programme 2013, the time for the start of construction and commissioning of the Spent Fuel Repository and Clink has been delayed by approximately one year. The reason for the delay is an adjustment for the licensing process being estimated to take longer than previously assumed.

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

- main hearing in the Land and Environment Court,
- review reports from the Land and Environment Court and SSM to the Government,
- municipal decision followed by
- the Government's decisions on licences and permissibility.

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

Before construction of the facility parts that are of importance for the safety of the Spent Fuel Repository can be commenced, two documents must be submitted and approved by SSM:

- The preliminary safety analysis report (PSAR).
- An account of how SKB, during the construction of the facility, will address issues of importance for safety during operation and after closure. The account is called Suus (Safety during construction of final repository).

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

3.4.3 Interim storage

The spent nuclear fuel is currently 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. During the time it takes to construct Clink and during the initial operation of Clink, the amount of fuel that is interim-stored in Clab will exceed the currently licensed amount of 8 000 tonnes of fuel. SKB has therefore submitted an application for and is planning steps to increase the capacity for interim storage of spent fuel as a part of the application for construction and operation of Clink.

Steps to increase Clab's storage capacity for spent nuclear fuel

The storage capacity in Clab has two limitations: The permissible quantity of spent fuel in the facility, and the number of physical storage positions in the pools. SKB's current licence for Clab covers interim storage of 8 000 tonnes of fuel. According to current forecasts, this amount will be reached around 2023. During the spring of 2015, SKB submitted a supplement to the application for Clab and Clink under the Nuclear Activities Act. The supplement also includes an additional application to increase the interim-stored amount in Clab to 11 000 tonnes. For increased storage, decisions from both SSM and the Government are needed.

SKB will prepare a PSAR for Clab for the increased storage capacity and submit it to SSM for approval in 2018. Up to 2020, SKB will implement the modifications in the facility that are required for the increased fuel quantity. The currently known modifications in the facility required in order to store 11 000 tonnes includes an upgrade of the cooling chain. Thereafter the SAR for Clab is renewed and SSM's permission for increased interim storage capacity is expected during the latter part of 2021.

If no other measures than a licensed increase in storage capacity are taken, Clab's storage positions will be filled around 2027. This time has not been affected to any appreciable extent by the newly decided premature closure of four reactors. In order to increase the physical storage capacity for fuel, SKB plans transloading of the fuel still stored in normal storage canisters to compact storage canisters. This action means that the storage capacity in Clab is calculated to suffice until 2032.

A second action, which may be taken if the need arises, is segmenting the control rods from BWRs that currently require a lot of space. The control rods from PWRs are integrated in the fuel and require no extra space. After segmentation, the BWR control rods could be packed closer in storage canisters and taken back to Clab's storage pools, or interim-stored at another site. A study was completed at the end of 2015. It assessed suitable technology and site for segmentation and continued interim storage. An alternative given in the study was to conduct segmentation in the pools in Clab's receiving section. No measures are presently planned for the other core components stored in Clab. The only core components that will be received in Clab in the future are BWR control rods and probes. Other core components are interim-stored at the power plant sites. When the SFR extension has been commissioned, there is a possibility, if necessary, to transport these to SFR for continued interim storage there, if a licence for this is obtained.

3.4.4 Encapsulation

An application under the Nuclear Activities Act for a licence to constuct the encapsulation plant and a licence to own and operate it as an integrated facility with Clab was submitted in 2006. After SSM's initial assessment, the authority announced that the application needed to be supplemented and that Clab and the encapsulation plant should be regarded as a single facility, Clink. SKB submitted the requested supplementary information in October 2009. In March 2011, SKB supplemented its application for Clink and at the same time applied for a licence under the Nuclear Activities Act for the Spent Fuel Repository. At the same time, an application was submitted under the Environmental Code for the entire final repository system, including Clink and the Spent Fuel Repository. In January 2015, SKB submitted an additional supplement to the application under the Nuclear Activities Act for Clink.

Current and planned technology development mainly concerns the processes for handling and monitoring of the spent nuclear fuel, fabrication of components for the canister as well as seal welding and non-destructive testing of components and seal welds. This is described in Chapters 7 and 8.

Design, construction and commissioning of Clink will be led by SKB.

Licensing and design

The preparatory PSAR (F-PSAR) for the facility that was submitted with the application in 2015 describes how nuclear radiation safety in Clink will be maintained. The design of the facility and how the requirements are met will gradually be clarified and detailed as technology development progresses and viewpoints are received from the Authority during the licensing process.

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

The facility configuration phase is in principle completed but updating of the set of requirements for Clink may be done based on new or updated regulatory requirements or the viewpoints received from the Authority during licensing.

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

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

Construction

The construction phase will begin when SKB has obtained all licences and conditions needed to start construction of the encapsulation plant. This means for example that a PSAR for the facility has been approved by SSM.

Construction will be divided into two parallel parts, one for modifications to the existing Clab and one for construction of the new encapsulation plant.

Since the operation of Clab proceeds throughout the construction of the encapsulation plant, the contracts will be adapted so that radiation safety in Clab can be maintained and operational disturbances are minimised. The extension of Clab, stage 2, was carried out in a similar manner, and experience from that project is important in designing and building the encapsulation plant.

During the construction work in the existing Clab, periods of reduced capacity for reception of spent nuclear fuel may occur. Planning for construction and commissioning will be particularly important in order to minimise the disturbances concerning regarding reception of fuel.

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

Commissioning

When the facility parts have been connected, initial tests of operation will be conducted. Since the tests are important for personnel training, they will be resource intensive and may affect the capacity for reception of spent nuclear fuel.

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

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

Before trial operation begins, an updated SAR is produced and submitted to SSM for approval. SKB plans to do this in 2029. It is assumed that trial operation will start one year later. Later, the SAR must be supplemented with experience from trial operation and be approved by SSM before regular operation can begin. According to SKB’s assessment, an approval for regular operation can be obtained in the beginning of 2032.

Production system for canisters

SKB’s production system for canisters should ensure the long-term supply of canisters for Clink. The production system will include a number of external suppliers and a facility for assembly of canister components. The suppliers fabricate nodular cast iron inserts and copper components for the canisters according to SKB’s specifications. These are delivered to SKB’s facility where machining, testing against established criteria and assembly of the components is done. 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 timetable for construction and commissioning of the canister factory is substantially shorter than for the other facilities in the KBS-3 system. A feasibility study describing 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 with external suppliers and the scope of the quality work in 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 needs to start considerably earlier in relation to SKB's plans in order to fulfil Posiva's needs.

Current and planned technology development mainly concerns the processes for management and quality control of fabrication of components for the canister as well as seal welding and non-destructive testing of components and the seal weld. This is described in Chapter 8.

3.4.5 Transportation of fuel

Transportation of fuel from the nuclear power plants

Work is under way to replace SKB's current fuel transport casks with new cask that comply with modern requirements. A contract for fuel transport cask was signed in October 2013 with the American supplier, Holtec International Power Division, Inc. The contract includes construction, licensing and manufacture of five transport casks with auxiliary equipment. There is an option for manufacturing of an additional container. The casks are delivered during the period 2019–2021. During 2019, the decision on ordering a sixth transport cask needs to be taken.

From August 2015 requirements are set for a new design of bottom shock absorbers for SKB's existing fuel transport casks. Previously, SKB judged it possible to carry out a limited number of shipments without purchase of new bottom shock absorbers. The strategy was, by taking compensatory measures and minimising the number of shipments, to obtain licences from SSM by special agreement for the shipments remaining before new transport casks that meet the requirements are commissioned. Among other things, the decisions regarding premature shutdown of power plants have changed the assumptions. SKB has therefore decided to acquire new bottom shock absorbers. Awaiting delivery of these, a licence to carry out shipments with existing transport casks without bottom shock absorbers has been obtained from SSM by special agreement. Existing transport casks with new bottom shock absorbers will be validated by SSM before they are used.

Transport of encapsulated fuel

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

SKB has previously conducted feasibility studies concerning the possible design of the canister transport cask. Together with the transport cask suppliers, SKB has presented proposals for design and functionality to the projects that work with the design of facilities and systems in the KBS-3 system.

The final design of the transport cask will be produced in cooperation with the chosen supplier. Design will take place in an iterative process with a focus on satisfying 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 fabricated. Before the container may be used in Sweden, a validation of the licence must be obtained from SSM.

The time required for design and licensing is estimated to be 3–4 years and for fabrication 1–2 years. The canister transport cask will then be tested together with Clink and the Spent Fuel Repository before commissioning.

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 timetable, this testing can be started around 2025. The initial system-specific tests will be conducted one year before the integrated testing of the whole KBS-3 system. The remaining containers will be fabricated and delivered as they are produced during the period 2027–2030, in parallel with integrated testing and trial operation.

3.4.6 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.

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 licensing, the project organisation will be successively staffed according to the work tasks in the project.

In parallel with licensing, the final repository is being designed. SKB has completed the system phase and is now preparing for future detailed design. Preparations also include technical construction and engineering geological investigations, which are needed as a basis for the design work prior to the start of construction. Ground surveys will be done as a basis for the placement of buildings and foundation engineering. 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.

Work centred on the environmental impact of the Spent Fuel Repository will also continue during the licensing process. SKB has applied to the County Administrative Board in 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 along with the entire application.

Detailed design will be carried out progressively as the facility is extended. Detailed design results 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 parts of the surface facilities. When the facility has been commissioned, detailed design of the parts that are extended will continue in parallel to prepare new deposition areas. For the execution of the detailed design work that is carried out in parallel with the extension, especially of deposition tunnels and deposition holes, results and information are also needed from technology development in rock, see Chapter 11.

Construction

The construction phase will begin when SKB has obtained all licences and conditions needed to start construction of the final repository. This means, among other things, that a PSAR and reporting of how issues of importance for safety, mainly after closure, are handled during the construction phase, must have been submitted to and approved by SSM. Construction of underground facilities can be 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 built and technical systems are installed and the third when the first deposition area is established and the facility is commissioned and tested. Construction of the accesses is time-critical for the progress of the entire project. The ramp and the first shaft are excavated in parallel, from the surface downward. Up until the moment in which the shaft has reached the repository level, the rock excavation works are limited to these two fronts. When repository level has been reached, construction of the central area begins. When the rock loading station and rock haulage to the surface via the rock hoist (skip) are commissioned, the capacity of rock handling increases radically and several driving fronts can be established successively. The rock excavation works for accesses and the central area are accompanied by installation works for the equipment that is needed to operate the facility. The surface facilities are built at a pace that is adjusted to the underground works. To start with, parts of the operations area are filled out, handling areas are prepared and temporary construction arrangements are established.

Excavation of the accesses and the central area will yield in-depth knowledge of rock conditions and the experience gained will be translated into for example rock support and rock sealing measures in tunnels or changed repository design.

As the central area is built, investigations are conducted for the first deposition area and a tunnel is driven providing access to this area. From this tunnel a few deposition tunnels are excavated in which deposition holes are bored. One purpose of preparing a deposition area at this early stage is to gather the geoscientific data needed as a basis for an updated SAR prior to trial operation, while another is to create room for integration tests and integrated testing of the whole process prior to trial operation. When operation then commences, the area will be used for deposition of the first canisters with spent nuclear fuel.

Commissioning

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 (rock loading station, skip shaft etc.) will be commissioned before the first deposition tunnels are built.

In connection with commissioning the systems will be tested, first separately and then gradually more interconnectedly. As the different parts of the facility are being commissioned, the operating organisation will be assembled and personnel will be trained for their duties. Running-in of technology and organisation will be concluded with integrated testing of the whole Spent Fuel Repository under realistic conditions. All operational activities will then be carried out, including deposition of a number of canisters, but without any spent nuclear fuel. Deposition takes place in the first deposition area built during the latter part of the construction phase. Finally, integrated testing of the KBS-3 system is 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 installations is presented in Chapter 10.

Commissioning is concluded when SKB obtains SSM's permission for trial operation of the final repository system. All functions and resources, as well as waste vaults in 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 2030. Before a facility can be put into regular operation, the SAR must be supplemented with experience gained from trial operation and approved by SSM.

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, as well as internally in the two groups of companies Uniper and Vattenfall. Furthermore, it presents the dependencies between different functions and between interested parties in the system. Finally, it presents the remaining need for further development work to permit decommissioning of the facilities concerned.

The RD&D Programme 2016 in general and Part III in particular, together with each licensee's decommissioning plans and the industry's joint cost calculation in the Plan report, comprise three interacting, requirement-stipulating main documents describing the planned decommissioning of the Swedish nuclear power plants and other existing or planned nuclear facilities, for example Clink and SKB's final repository. The three main documents supplement each other in terms of content, where the RD&D Programme presents the development activities and other measures needed to safely decommission the nuclear power plants. The decommissioning plans present the planned execution with a focus on radiation safety and strategic aspects. The Plan report presents the estimated cost of decommissioning as described in the RD&D Programme and 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 the respective sites prior to the Plan report in 2013. The decommissioning studies are compiled with a focus on costs but also describe the technical approach and thereby cover everything from organisational to storage volume aspects. 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 reactor facility includes a number of activities to release the facility from regulatory control. Prior to decommissioning, necessary licences must exist. 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 begin. The nuclear power companies' plan is to start dismantling and demolition of the facility as soon as possible after shutdown. When the facility/facility parts have been released from regulatory control, conventional demolition and restoration of land will be carried out.

Because dismantling and demolition of Barsebäck 1, Barsebäck 2, Oskarshamn 1, Oskarshamn 2, Ringhals 1 and Ringhals 2 will commence before the extended SFR is ready to receive short and long-lived operational and decommissioning waste, the licensees need to 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 18-1 (see Part III) shows the general timetable for decommissioning of all nuclear power plants and SKB's facilities. This 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

Shutdown operation of Barsebäck Kraft AB's two reactors, Barsebäck 1 and Barsebäck 2, has been under way since 2006. A decision has been made to segment the reactor pressure vessel internals and store them in steel tanks in a new storage building at the plant. The building has been constructed and segmentation will begin in the autumn of 2016.

Preparation of a radiological inspection programme for clearance of decommissioning material, buildings and land has begun. Furthermore, preparations and establishment of a project organisation for decommissioning have begun.

Dismantling and demolition of the nuclear power plant is planned to start in 2021 and clearance is planned at the end of the 2020s.

OKG Aktiebolag

OKG Aktiebolag has since August 2016 a new department in operation with responsibility for decommissioning activities. In 2015, the Decommissioning Preparation Project (DPP) was started, with the purpose of implementing the new department as well as preparing the documentation and completing the preparations required to put Oskarshamn 1 and Oskarshamn 2 into defuelling operation, followed by shutdown operation and thereafter dismantling and demolition.

Oskarshamn 1 is planned to be finally shutdown in the summer of 2017 and Oskarshamn 2 will not be restarted. Decommissioning of the two reactors will be completed in 2025. Oskarshamn 3 will be in operation up until 2045.

Ringhals AB

In December 2015, Ringhals AB started a decommissioning project for Ringhals 1 and Ringhals 2 with the purpose of dismantling and demolishing the facilities. The focus is presently to assess the prerequisites for decommissioning 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 Reactors 1 and 2) was started, with the purpose of preparing for decommissioning of the two reactors while operation of Ringhals 3 and Ringhals 4 can proceed according to plan.

Ringhals 1 and Ringhals 2 are planned to be finally shutdown in the summer of 2020 and 2019 and will be decommissioned in 2026 and 2025 respectively. Ringhals 3 and Ringhals 4 will be in operation until 2041 and 2043 respectively.

Forsmarks Kraftgrupp AB

Forsmarks Kraftgrupp AB's reactors will be in operation for a total of 60 years, which means that Forsmark 1 will be shut down in 2040, Forsmark 2 in 2041 and Forsmark 3 in 2045.

The nuclear power plant will be fully decommissioned by 2051.

The Ågesta facility

The nuclear power plant in Ågesta has been in shutdown operation since 1974. A decommissioning project started in November 2015. Dismantling and demolition is planned to start no later than 2020 when the environment licence for shutdown operation is no longer valid. Decommissioning will be completed by 2025.

SKB's facilities

Decommissioning of Clink and the Spent Fuel Repository can begin at the earliest when all spent nuclear fuel has been deposited while decommissioning of SFR can begin at the earliest when the waste from decommissioning of Clink has been deposited. SFL can, however, be decommissioned when the long-lived waste from the last reactor has been managed and disposed of.

3.6 Alternative actions for changed conditions

SKB's and the licensees' plan for management and disposal of the waste is based on the conditions and assumptions that currently apply for the nuclear power programme. 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 the consequences may be.

3.6.1 Operating times of the nuclear power reactors

Since the previous RD&D programme, decisions have been made regarding the premature closure of the four reactors from the 1970s: Oskarshamn 1, Oskarshamn 2, Ringhals 1 and Ringhals 2. These reactors will be shut down within the next four years, see Section 1.1.3. The planning for the remaining six reactors, from the 1980s, has not been altered. The planned operating time is, as in the RD&D Programme 2013, 60 years. This means that these reactors will be finally shutdown 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 from the 1980s is described below.

Extension of planned operating times

SKB's facilities in the KBS-3 system will be designed to handle and deposit 6000 canisters. The nuclear power companies' current forecasts, taking into account the premature shutdowns according to the above, contain about 5 700 canisters. This provides a margin for possible extended operating times for the reactors from the 1980s. The design-basis canister quantity of 6 000 canisters is reached if the remaining reactors extend their operating times by about five years, i.e. to a total operating time of about 65 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 need for interim storage capacity in Clab for fuel 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 the spent nuclear fuel. This is the case regardless of any extended operating times, see Section 3.6.6.

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 deposited in near-surface repositories will probably increase at extended operating times. However, it only concerns a small fraction of the total waste amount. The amount of decommissioning waste is not assessed 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, there is a possibility to adjust the final disposal volume in SFL until the start of construction, i.e. 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 therefore lead to reduced storage needs 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 built successively during operation, the size of the deposition areas can be adapted to the actual need. In this case, the number of deposition positions will decrease. If SFR has already been extended to its full size in accordance with current estimated volumes, shortened operating times for the nuclear power reactors will probably mean that the facility will not be fully utilised.

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

An increase in the licensed interim storage capacity for Clab is expected according to SKB's planning during the latter part of 2021. According to current forecasts, 8 000 tonnes of fuel will be reached around 2023. If a license for increased interim storage is not obtained at this time, the spent fuel will need to be interim-stored 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 lead to fuel needing to be interim-stored in pools at the nuclear power plants several years before the final cores can be successively transported to and interim-stored in Clab. Interim storage of spent fuel at the nuclear power plants after final shutdown is not a desirable scenario.

A reduced quantity of spent fuel also entails that the nuclear waste programme in total can be concluded sooner. The extent depends of course on how many reactors would be affected and how much the operating times are shortened.

Shortened operating times would probably also entail an earlier start of decommissioning of the reactors concerned. SFR will, according to the plans, also act as an interim storage facility for long-lived waste. If the last reactor is decommissioned before SFL has been commissioned, alternative interim storage capacity is needed for the long-lived waste. Alternatively, the short-lived decommissioning waste from the last reactor can be interim-stored at the power plants until SFL is finished so that interim storage of the long-lived waste can continue in SFR. Finally, this depends on the total volume of short-lived waste in relation to SFR's final storage capacity and the capacity that will be used for interim storage.

3.6.2 Commissioning of the extended final repository for short-lived radioactive waste

SKB plans to start trial operation for the extended SFR in 2028. According to the plan, decommissioning of the seven first reactors (including Ågesta) will begin before the extension of SFR is finished. This means that other interim storage facilities for the short- and long-lived decommissioning waste are needed, see Sections 3.3.3 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 is transported to SFR for disposal (short-lived waste) or interim storage (long-lived waste). A delay of the extension of SFR thereby also leads to a delay of the facility's clearance, unless the waste has been shipped to another site for interim storage.

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. Interim storage capacity at the power plant sites is judged possible to increase to meet needs in case of a delay of the extension by a couple of years.

As the waste suppliers now plan for interim storage of the waste, the waste system is not as sensitive to minor delays in the commissioning of the extended SFR.

Earlier start

Earlier commissioning of the SFR extension implies a shortened and decreased need for interim storage of short and long-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.

3.6.3 Interim storage capacity for low- and intermediate-level waste

As the decommissioning of the seven first reactors will commence before the extension of SFR is ready to receive waste, the decommissioning waste needs to be interim-stored. According to the discussion in Section 3.3.4, interim storage capacity for the long-lived waste is deemed to be sufficient up until around 2040 when the remaining reactors will be shut down.

When it comes to the short-lived and low-level waste, volumes are large and there are several alternatives for management and disposal of the waste. Some waste is destined for SFR and needs to be interim-stored until the SFR extension can be commissioned. Some waste can be placed in near-surface repositories and some can be handled as conventional waste after clearance. This waste would not need to be interim-stored.

Thus, there are different scenarios regarding how large volumes may need to be interim-stored and where it should be interim-stored. Interim storage of short-lived waste currently exists at the nuclear power plants, see Sections 2.1.1 and 3.3.3, but the capacity is not enough for management and disposal of decommissioning waste.

3.6.4 Commissioning of the final repository for long-lived waste

SKB plans for SFL to be commissioned around 2045 and the last reactors Forsmark 3 and Oskarshamn 3 to be shut down approximately at the same time, see Section 3.3.4.

Postponement

If SFL were to be commissioned later than in the current planning, this implies a prolonged interim storage of the long-lived waste in SFR or at the power plant sites. This could, in turn, affect the nuclear power companies' ability to conclude the nuclear activities on their sites.

A prolonged interim storage of long-lived waste in the extended part of SFR can affect the possibility to dispose of the short-lived decommissioning waste from the last reactors in the programme in SFR. Hence, it is important that SFL can be commissioned as planned.

Earlier start

Prior to the RD&D Programme 2013, SKB made a review of the timetable for SFL based on a time assessment of the necessary development steps. The review showed that there are limited possibilities for commissioning the final repository considerably earlier than planned. A possible earlier start of the commissioning of SFL is not expected to yield any major consequences for the waste system. However, it would entail shorter times for interim storage of the long-lived waste and reduce the quantity of interim-stored waste in SFR.

3.6.5 Siting of the final repository for long-lived waste

SFL will be built as one facility with two different repository parts with different barrier designs. An alternative is to locate the two repository parts at different sites. Such a solution could be possible if it turns out that the requirements on the site with respect to safety after closure differs clearly between the repository parts. The ongoing safety evaluation has the purpose of gathering data as a basis for identifying siting factors as support for the continued siting process.

Siting of the two repository parts at different locations could occur in different ways, through siting the parts individually or through siting one or both to another final repository. The consequences of different sites are handled within the framework of the siting process, where an overall evaluation of various siting factors serves as a basis for site selection. Shared siting is not expected to increase the possibility of earlier commissioning of the repository, unless siting to another existing final repository. In such cases the construction phase could potentially be shortened when joint use of ramp and shafts could shorten the time for excavating to repository depth.

3.6.6 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 2030, which means that SKB begins unloading of the spent nuclear fuel from Clab's storage pools then. For Clab to be able to receive the spent fuel produced up to this point in time, SKB plans to increase the licensed storage capacity to 11 000 tonnes of fuel. For more efficient use of the physical storage space, the fuel must be transloaded to compact storage canisters and, if the need arises, the control rods from BWRs will be segmented for more compact storage, see Section 3.4.3. According to current estimates, which include the premature closure of four reactors, the above measures in Clab will free up volume so that the storage capacity would suffice until around 2038. In order to interim-store 11 000 tonnes of fuel, all core components needs to be moved from Clab's storage pools and interim-stored somewhere else.

Postponement

If delays were to occur so that the Spent Fuel Repository and Clink cannot be in operation until after 2038, additional measures must be taken. Primarily, the core components and control rods stored in Clab would be loaded out for interim storage at another site. The waste can then be reloaded into steel tanks (BFA tanks) for dry interim storage. Steel tanks are already used today for interim storage of this type of waste. If only fuel is stored in Clab, the storage positions will suffice until around 2042. 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. The option of wet storage is a central part of the current Swedish system. Before SKB makes a decision on a possible extension, the option of dry interim storage of fuel will also be explored. Among other things, aspects related to fuel properties after dry interim storage and the possible impact on safety after closure must be analysed. 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 yield any negative consequences for the waste programme.

3.6.7 Horizontal deposition – KBS-3H

Since 2001, SKB has in collaboration with Posiva investigated whether horizontal deposition (KBS-3H) can constitute an alternative to vertical deposition. 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 a number of supercontainers.

A supercontainer consists of a canister surrounded by bentonite buffer and held together by a perforated outer metal shell. Between the supercontainers distance blocks of bentonite clay are placed. 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 on the operations area above ground and the central area and the accesses underground are marginally affected if horizontal deposition is chosen instead of vertical deposition.

A full-scale demonstration of technology for KBS-3H, the Multi Purpose Test (MPT), has been under way since 2013 in the Äspö HRL. The test includes manufacture of full-scale components, assembly and deposition of a supercontainer and distance blocks, plus installation of a compartment plug with associated filling components. The deposition drift and the components are instrumented so that the initial course of events can be monitored. All these steps have been carried out successfully.

An evaluation of how the work with KBS-3H should continue has been carried out together 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 with regard to 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. Remaining development includes for example selection of materials and design of the supercontainer as well as detailed design of packing station, plugs, the intended grouting solution with Mega packer and furthermore design of a deposition machine. Furthermore, the current MPT experiments need to be completed and evaluated and then supplemented with a full-scale test with heaters at full repository depth, which shows that the bentonite swells and homogenises as intended outside the supercontainer.

The evaluation of post-closure safety shows that since KBS-3H entails that the quantity of bentonite material between every canister is considerably less than between canisters in a vertical deposition solution, KBS-3H may be significantly more sensitive to chemical erosion of bentonite followed by copper corrosion due to the sulphide dissolved in the groundwater. This depends above all on the spreading of bentonite erosion at one canister position to other nearby positions.

For the commissioning of the KBS-3 system to progress according to plan, the remaining development work for horizontal deposition is assessed to be too extensive to be able to justify parallel development efforts even if there are large potential economic advantages of horizontal deposition. In any case, it must first be established that changing to KBS-3H would not entail an impairment of post-closure safety.

Technology development will therefore during the next few years focus on completing, industrialising and optimising the system for vertical deposition. Further research in the area of chemical erosion will, however, continue, see Section 5.7. In the longer term, and if safety issues are solved, however, SKB intends to re-evaluate whether the economic advantages of KBS-3H could nevertheless justify a change.

4 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 way to carry out the research, development and demonstration that are needed in order to construct and commission new facilities. For the facilities in operation, SSM's regulations apply and how these are implemented regarding procedures, resources and expertise is not specified here. The chapter focuses on the iterative process of developing, implementing and evaluating the final repository for radioactive waste that 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 competence, the resources and the tools needed are available.

4.1 Research

4.1.1 The goals of research

The objective of SKB's science research programme is to secure safe management and final disposal of nuclear waste by ensuring access to the knowledge that is needed in order to be able to assess a site, design, licence, construct and operate existing and planned facilities. This means that the research 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 sufficient information for the continued technology development and planning that is needed in order to obtain efficient and optimised solutions that at the same time provide safety both during operation and after closure of SKB's final repository.

Research has been one of the pillars in SKB's programme since its start in the 1970s and the remaining research that is now needed concerns primarily SKB's existing and planned final repositories for radioactive waste and spent nuclear fuel (SFR, SFL and the Spent Fuel Repository).

4.1.2 Management of research

To manage the research, SKB has established a research council under the guidance of the head of research. An important function of the research council is to provide a forum where the sections responsible for construction of facilities, through their representatives in the Council, can ensure that SKB's research resources are used in such a way that SKB's goals are achieved efficiently.

The aim of SKB's research programme is largely established in the RD&D programs, which, in accordance with the Nuclear Activities Act, are submitted to the SSM every three years. As an early part in the process of updating SKB's RD&D Programme, SKB experts representing the different research disciplines present their proposals for the plans for the coming three-year period to the research council. The research council acquires, through the reports, an overview of the research needs and make an overall assessment and prioritisation of the need for inputs including usage of resources primarily during the coming three-year period. The plans should be linked to the next decision step for the different repositories.

4.1.3 Strategy

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, see Section 3.2. 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 judge 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, SSM can be expected to make high demands on the level of knowledge regarding the central parts of the information material. This is because these are the first formal decisions on a final repository. Although a central milestone in the level of knowledge is achieved when SKB obtains permissibility and licence to construct a new facility, the need to be able to make assessments on the safety of final repositories both during operation and after closure does not disappear. These assessments entail requirements of knowledge regarding how both the engineered barriers and the natural processes in the rock and on the ground surface interact and evolve in time. According to the regulations, the SAR should be constantly kept up-to-date and, in addition, a periodic overall evaluation of the safety and radiation protection of each facility should be made every ten years according to the requirements of the Nuclear Activities Act.

A current challenge for SKB is the application of scientific knowledge in the design of the new final repository for long-lived radioactive waste (SFL) so that it will have properties that ensure that radionuclides in the radioactive waste are contained and/or delayed in an acceptable manner. Research support is also needed in the continued design work for the extended SFR and for the KBS-3 system. This applies predominantly for the generation of useful design premises (requirements) on the final repositories and to be able to verify that the resultant technical solutions fulfil these requirements.

As a repository programme evolves, the set of questions that require development of knowledge also change. At an early stage of the development work for a repository, fundamental questions regarding for example barrier performance in a geological environment are identified. Some questions can be answered with limited efforts while some central questions require long-term research. In later stages the set of questions matures and few new questions are added. When a repository programme approaches implementation, the degree of detail with which the engineered barriers must be specified also increases and thereby also new questions arise.

Although there now exists extensive knowledge on all the areas that are of importance for the post-closure safety, it can be anticipated that new questions appear. The origin of questions that may require research is of several kinds:

- Most of the questions are generated/identified internally at SKB during the operation of the existing repository facilities, the development of repository concepts and the execution of safety assessments during operation and after closure. Also site investigations can generate questions, and as sites are increasingly being investigated in detail, additional questions can arise. New or modified waste types can also lead to new questions.
- Questions also arise outside of SKB, for example within the scientific community and in SKB's sister organisations in other countries.
- Questions may also be asked by SSM and reviewing bodies in examinations of RD&D programmes and in the licensing process.
- Changes in external requirements may entail that new questions need to be handled.

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 and the results of these evaluations constitute an essential basis for research initiatives that are planned in this RD&D programme. Safety assessments are thus fundamental for the prioritisations of the research programme.

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, touch upon the development of methods for treatment of radioactive waste, reprocessing of the spent fuel, the developement 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. Currently there are no plans for special projects within this category.

4.2 Technology development

4.2.1 The goals of technology development

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 consideration to the fact that the duration of the development of different subsystems may differ.

4.2.2 Control of technology development

Technology development is commissioned by the departments 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 (finished technology development), when it needs to be finished 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 done of how far this technology needs to be developed prior to the previous 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 into a number of production lines linked to the properties of the spent nuclear fuel and the low- and intermediate-level waste as well as the waste system barriers and other components such as the canister, the buffer, the cementitious materials, backfill, closure and rock. This division is reflected in the structure of Chapter 5. Technology development aims at systems, i.e. both physical products such as machines, barriers, measurement instruments as well as processes that describe how the products are used to achieve a certain function and how the system interacts with the surrounding environment, including humans, technology and organisation (MTO).

For the drafting of the RD&D programme, a survey 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 client function for technology development and the functions at SKB that carry out the development work.

4.2.3 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.
- Introduction- and handover phase.

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

The model includes 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 boundaries of the projects are determined on a case by case basis. Normally a project does not run over several phases but it is limited to stay within a certain technology development phase.

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, system design phase and 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, a refinement takes place but more extensive changes of the technical solution may also be needed.

System design includes the definition of the subsystem and its design, premises as well as requirements from and requirements on the whole system. This can in its main features be considered to correspond to what in design planning is called system planning and results in a system design.

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

The introduction 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 is different depending on the system, the product or the method that will be adopted. The phase includes, at least, planning, training/competence transfer, 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 the handover of all documentation.

4.2.4 Design premises

The design premises 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 phase after closure of the final repository are defined. For a technical design that can maintain these safety functions a number of design premises are specified. The design premises 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 design premises and other requirements are stated in the applications for construction of the Spent Fuel Repository, the encapsulation part of Clink and the extension of SFR. Design premises related to post-closure safety for the Spent Fuel Repository are presented in SKB (2009a) and for the SFR extension in SKB (2014c). Design premises and links (relationships) between different requirements, as well as technical specifications have been collected and structured in a database.

Specification of design premises, technical solutions and safety analysis during operation and after closure must be 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 considering whether it fulfils the requirements on post-closure safety. In parallel, possible production and inspection processes and the requirements that need to be set for them are evaluated. Altogether, it leads to an update of the design premises and the technical solution that can be taken to the next phase of technology development.

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 design premises that are related to several barriers are:

- Taken together, the design premises shall lead to agreement with requirements on safety during operation and after closure of the final repository.
- The design premises shall be achievable in practice and verifiable for all concerned barriers.
- Design premises that entail simple, robust and effective solutions are preferred.

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

Regarding the Spent Fuel Repository's barriers, SKB and Posiva, in cooperation, have devised revised and harmonised design premises. These premises are based on hitherto conducted technology development work, as well as 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 2012. The harmonised design premises constitute the basis for coming PSAR even if they may need to be revised slightly as a result of SSM's scrutiny of the licence application. Further revision of the design premises can also be expected in conjunction with the renewal of SAR prior to trial operation.

4.2.5 Quality management and testing

An important goal of technology development is that it should be possible to verify that the developed technical solutions meet the stipulated requirements. "Quality management and testing" refers to the measures that need to be taken in order to ensure and provide confidence that the requirements made on the facilities during operation and after closure of the Spent Fuel Repository are satisfied. The goal is that the results obtained should conform to acceptable values for properties that contribute to safety and radiation protection.

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 the production can begin, the fabrication and testing processes that SKB intends to employ should be proven feasible and must have good ability to achieve and quality-assure products that conform to the established design.

The ability of the fabrication and testing processes to perform their specified tasks is demonstrated in qualifications.

SKB works further to describe how requirements on execution of the process of qualification shall be controlled by the importance of the components for repository radiation safety and by the planned production and the possibilities of using proven technology.

The qualification of each production or testing process is adapted to the manufactured or tested component's importance for radiation safety, available proven technology, available standards and norms, as well as the conditions that will prevail, both physical as organisational, in the execution of the planned production. This means that each process of qualification will be carried out as a unique project, where some, in principle, only reference the standards and norms that should be employed while others require extensive analyses and execution of demonstrations.

Inspections during the construction of the facilities will also be carried out to ensure that the construction is executed in such a way that the requirements on radiation safety, quality and efficiency are met. 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 external environment, programmes for investigations of rock, programmes for inspection of rock excavations and programmes for inspection of occupational safety.

4.3 Work tools

Execution of the research, development and design work needed to decommission the nuclear power plants and dispose of the nuclear waste requires access to a set of work tools. SKB and the licensees of the nuclear power reactors 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 work being done to maintain and further develop these tools.

4.3.1 Databases

Management of radioactive waste and spent nuclear fuel entails management of large quantities of data that is adequately gathered and structured in databases. SKB and the licensees have databases that contain information on the radioactive waste such as DARK for spent nuclear fuel, Draak for long-lived waste, which is stored in Clab, and Triumf for low- and intermediate-level waste deposited in SFR. Work is under way to create a joint waste database to replace Triumf.

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.

The results of SKB's research and development work, material for the work with the siting of the repository and material 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 applies a systematic management of the requirements on the final repositories for spent nuclear fuel and radioactive waste, as well as of the requirements that should be fulfilled during the design of the facilities and at the facilities' construction, operation and decommissioning. This means, among other things, that the requirements and design conditions 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 in databases (Doors).

There are also databases of more administrative character such as Bibas that is SKB's library database and SKBdoc that is a 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.

4.3.2 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 in-house developed model and calculation tools and commercial tools that are purchased and adapted when necessary for SKB's application.

For the execution of the analyses of post-closure safety for the Spent Fuel Repository (SR-Site) and for the extended SFR (SR-PSU), a large number of models and calculation tools (SKB 2010d, 2014d) are used. The calculation tools that are used include both commercial computer 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. To be able to use the software as a part of the safety assessments, SKB has quality requirements that the calculation tools must meet. According to these requirements (Chapter 2 in SKB 2010d), for the results to be usable in the safety assessments there needs to be documentation that proves that:

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

The data used in the calculations comes mainly from SKB's 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 also conducts development work to make the tools more standardised, and strive to use and if necessary make adjustments of commercial products such as for example COMSOL and Matlab. A model database is under construction.

4.3.3 Site models

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 calculations and descriptions of the site's hydrogeochemical and hydrogeological development until today. A site descriptive model is one of the cornerstones in the assessment of post-closure safety for a repository at a site. SKB has developed site descriptive models for Forsmark (SKB 2008, 2013b) and Laxemar (SKB 2009b).

By selecting Forsmark as the site for the final repositories, both for spent fuel and for short-lived low- and intermediate-level radioactive waste, it is only justified to maintain the models for Forsmark. In Forsmark, a monitoring programme where data has been collected on groundwater pressure, groundwater composition, seismic events, precipitation, temperature, development of ecosystems, etc, has been in operation 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 can be used as basis for repository layout and design and will be updated in conjunction with 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 timetable, as discussed in Chapter 3.

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 administers a set of measurement instruments for the execution of site investigations. 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.

To permit modelling and visualising of the geoscientific properties in the rock in three dimensions, SKB has developed the tool RVS (Rock Visualisation System). RVS is strongly integrated with the database Sicada which has also been developed and is administered by SKB.

4.3.4 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, and approval of model and calculation tools.

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

4.4 Resources and competence

SKB has a competence maintenance process established in its management system in order to ensure that enough competence is available in the short and long term for maintaining a high safety and reaching the tasks and goals of the operations.

SKB is as the activity operator obliged to ensure that tasks are completed 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 question. The evaluation made is based on both an assessment of which tasks are of such a strategic importance that they should be taken care of by SKB's own personnel, risks with being dependent on external suppliers and economic considerations. The outcome of the judgements on competency requirements and trade-offs between having in-house personnel and using external suppliers can vary over time.

4.4.1 Within SKB

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 includes that SKB should have the necessary competence to procure and evaluate the services linked to management and final disposal of spent nuclear fuel and radioactive waste that SKB orders from external suppliers.

A large part of the knowledge and technology SKB needs are publically available. Other parts are specifically linked to management and final disposal of nuclear waste. In certain areas there is no external competence available on a commercial basis, for example regarding the development of canisters for spent fuel, and there SKB has chosen to largely have its own resources (for example the Canister Laboratory). In some cases, SKB believes that it is economically or in the long term advantageous to have its own personnel instead of engaging consultants. If there are products or services available commercially or through SKB's owners, the general strategy is to use these.

SKB needs to have sufficient in-house competence in order to maintain its ability to assimilate the knowledge that is present in the community of importance for 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 generated by SKB itself or in international cooperation. 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 can 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 wide and interdisciplinary insight on how the different processes that affect repository safety interact. The group also needs persons with in-depth knowledge of the disciplines that affect safety, i.e. in geoscience (geology, hydrogeology, geochemistry), material issues (canister material, clay material, cement materials), waste and spent nuclear fuel (chemistry, solubilities etc.), solute transport (near-field, rock), surface ecosystems and climate evolution.

The need for competence to manage and carry out technology development is based on the plans that are agreed upon, see Section 4.2.2. In order to be able to evaluate the development needed and drive development needs, SKB needs general competence in the production line linked to the properties of the nuclear waste and the waste system barriers and other components such as the canister, the buffer, the 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 in-house competence for development. This applies in particular in 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.

4.4.2 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 institutes of higher learning. These experts are engaged to larger or smaller extents depending on the need at the time. Many have to a varying extent been associated with SKB's activities for decades. For temporary needs of resources, for example for larger projects regarding design and construction of the facilities, external suppliers are generally engaged. SKB's owners are also an important resource.

SKB is thus collaborating with universities and educational institutions in order to obtain critical knowledge in areas where it is lacking. This is typically research where SKB finances PhD projects while the university supervises the PhD candidate. SKB-funded doctoral candidates constitute a prospective future competence reserve for SKB and others in the industry.

4.4.3 Collaboration

Cooperation with Posiva

As described in Section 1.4, SKB has over a number of years an in-depth collaboration with its sister organisation Posiva in Finland. In addition to the benefits in terms of effectiveness that this collaboration brings, SKB's ability to carry out research and development is also enhanced. The generation of common plans and joint projects entails that these get a broader and more comprehensive preparation and review than if SKB carried out this work on its own. 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 waste organisations

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. An example is BIOPROTA, a joint project concerning the biosphere, which is operated by both waste organisations and regulatory authorities, where SKB is participating actively.

International organisations

SKB is participating in a number of joint projects that are being done with the support of the EU's 7th framework programme and the ongoing framework programme Horizon 2020. SKB is participating in the projects that fit into the overall plan for research and development (see Table 4-1). The benefits of the EU projects are, in addition to the financing these contribute, the breadth of competence and the contact network that participation brings. The EU projects provide for many areas contact with the international research front which is of value for SKB. SKB is an active member of the technology platform for closer European cooperation on nuclear waste disposal, Implementing Geological Disposal Technology Platform (IGD-TP) which was formed in 2009. The purpose of the platform is to work together so that the nuclear waste programme closest to realisation receives support in order to succeed in its efforts. Although the platform does not have financial resources at its disposal, it has an indirect influence on how the EU's research funding is allocated within the area. By cooperation within the platform, a number of larger demonstration experiments have been conducted, for example in the Äspö HRL.

NEA is a collaborative organisation for nuclear energy issues in the OECD. The task of the organisation is through multi-national collaboration in scientific, engineering and legal issues facilitating civilian nuclear power. SKB is participating in a number of committees and groups within NEA. The Radioactive Waste Management Committee, RWMC, consists of representatives from both regulatory authorities and nuclear waste and decommissioning organisations in the NEA States. Under RWMC there are task forces that handle assessments of post-closure safety (Integration

Group for the Safety Case), questions concerning confidence and acceptance (Forum on Stakeholder Confidence) and decommissioning and dismantling issues (Working Party on Decommissioning and Dismantling). Besides committees and groups, projects are also carried out in a number of areas, for example; information preservation across generations (Records, Knowledge and Memory across Generations), sorption of radionuclides (Sorption Project), decommissioning (Co-operative Programme on Decommissioning), database for chemical thermodynamics (Thermochemical Database), database of features, events and processes in geological repositories (International FEP-database) and management of metadata (Radioactive Waste Repository Metadata Management).

Table 4-1. Examples of EU projects that SKB is participating in.

Acronym	Project name	Reference
Lucoex	Large Underground Concept Experiments	www.lucoex.eu
DOPAS	Full Scale Demonstration of Plugs and Seals	www.posiva.fi/en/dopas
BELBAR	Effect of Bentonite erosion on EBS and radionuclide transport	www.skb.se/belbar
PEBS	Term Performance of Engineered Barrier Systems Completed	www.pebs-eu.de/PEBS
Modern2020	Development and Demonstration of monitoring strategies and technologies for geological disposal	www.modern2020.eu
Cebama	CEment BAsed Materials: properties, evolution and barrier functions	www.cebama.eu
Mind	Microbiology In Nuclear waste Disposal	www.mind15.eu
Cast	CArbon-14 Source Term	www.projectcast.eu

The International Atomic Energy Agency IAEA is active in several areas: nuclear energy, nuclear safety, safeguards and technical cooperation. SKB is participating in task forces, networks and projects in these areas. Examples are networks within low- and intermediate-level waste (Disponet), surface ecosystems (MODARIA), underground laboratories (URF), decommissioning of nuclear facilities (IDN), integrated analysis of safety during operation and after closure (Geosaf II), intrusion in repositories (Hidra) and an expert group within safeguards for final repositories (Astor).

Sweden has ratified 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.

4.5 SKB's facilities for research, development and demonstration

4.5.1 The 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, see Figure 4-1. 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 4-2 and current experiments are presented in the Äspö HRL's yearly report (SKB 2016c).

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 characterisation in the Spent Fuel Repository and the extension of SFR in Forsmark.

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 4-1. The Äspö HRL is situated on the island of Äspö north of the Oskarshamn Nuclear Power Plant.

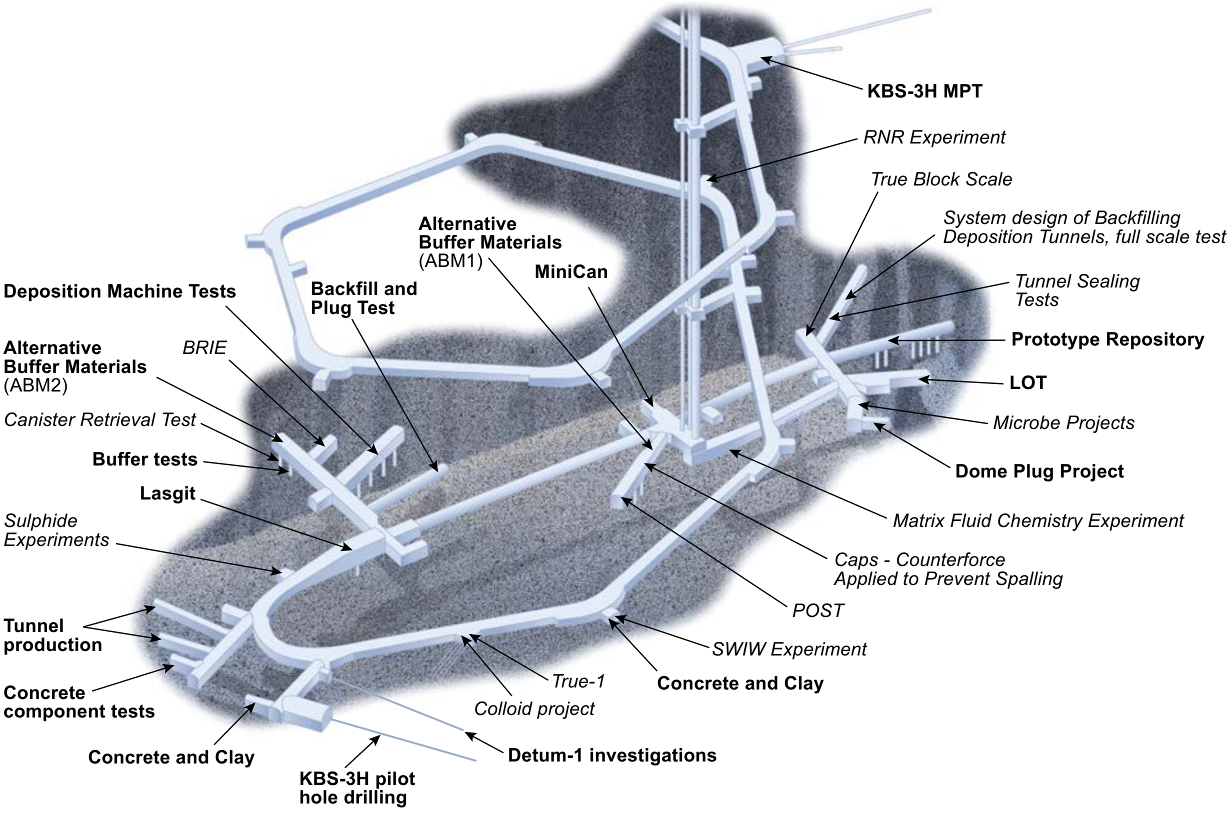


Figure 4-2. The Äspö HRL with ongoing (bold) and concluded (italic) experiments.

After the start of operation in 1995, experiments began gradually to investigate how the barriers and the other components of the Spent Fuel Repository (canister, buffer, backfill and closure) could be designed and managed 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 important material to support 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 and full-scale tests that will be carried out there.

Prior to future analyses of post-closure safety for SFR and SFL, research projects are being carried out focused on studies of interactions between different types of barrier material relevant for these repositories and for different types of materials that are representative of low- and intermediate-level waste. Apart from this, experiments are planned that are connected to evolution of concrete and cementitious materials and technology for construction of the barrier structures in SFR and SFL. Large parts of the ongoing development programme for material and technology for the extension of SFR will be carried out in the Äspö HRL.

In special forums, Äspö Task Force, 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 and SKB Task Force on Engineered Barriers Systems (Task Force EBS). The collaboration is aimed at evaluating different concepts and modelling methods and facilitating collaboration between experimentalists and modellers.

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.

4.5.2 The Canister Laboratory

The Canister Laboratory, situated in the harbour area at 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. The methods that SKB will use to inspect the canister's components and welds are also being developed and demonstrated here. Development of the methods that will be used to fabricate the parts of the canister is being led from the Canister Laboratory. Inspection, testing and evaluation are largely carried out at the Canister Laboratory, while the fabrication trials are conducted at external suppliers. The goal is to develop methods for fabrication and inspection that meet stipulated quality requirements and have sufficiently high reliability to be used in future canister production and in Clink. Important equipment in the laboratory includes a system for friction stir welding with a rotating tool, equipment for non-destructive testing and a handling system for full-size canisters. Figure 4-3 shows the equipment for friction stir welding.

4.5.3 Other laboratories

Bentonite Laboratory

Since 2007, SKB has been conducting research and development in the Bentonite Laboratory located above ground adjacent to the Äspö HRL. The experiments being conducted in the Bentonite Laboratory supplement the experiments being conducted in the underground laboratory.

One of the barriers in all final repositories is a swelling clay, 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 of the tunnels in the repository. In the Bentonite Laboratory, SKB is testing the properties of the bentonite by, for example, simulating water conditions in a controlled manner. Here SKB is also developing methods for backfilling the repository's tunnels and building plugs to seal the deposition tunnels.

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

Water chemistry laboratory

The water chemistry laboratory on Äspö is accredited for assessing the chemical components in the groundwater which are of particular importance for final 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 project in both Forsmark and Laxemar. The combined competence of the laboratory is used for planning and construction of the corresponding laboratory in Forsmark.

The material laboratory

The physical-chemical properties of the bentonite clay can be studied in the material laboratory which is housed in the same building as the water chemistry laboratory on Äspö. In the laboratory, the tests and investigation methods that will be used in control of bentonite deliveries to the final repository are prepared. The laboratory has been equipped with analysis technologies which means that research-like investigations can also be carried out.

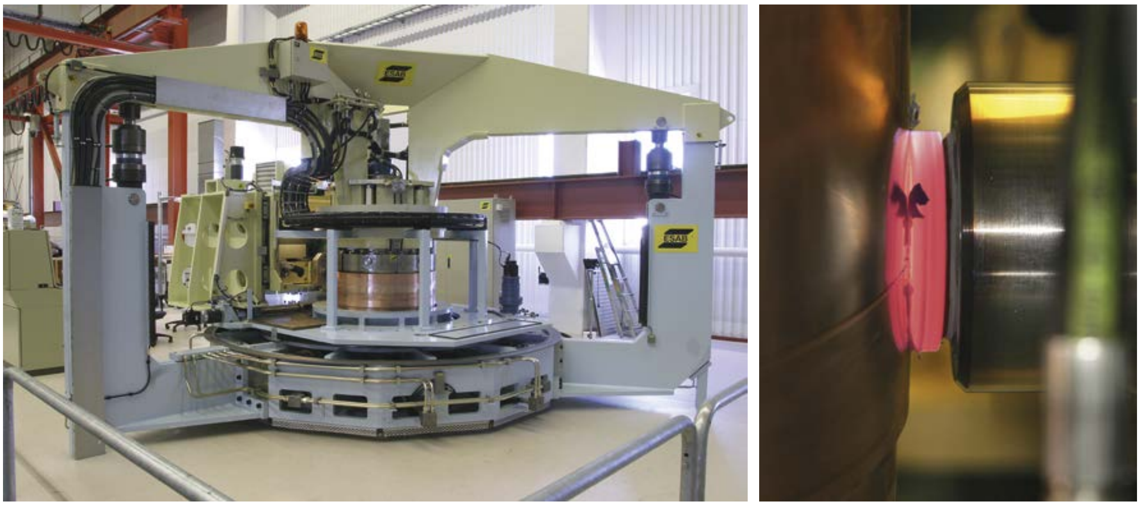


Figure 4-3. *Friction stir welding of copper lids. The image to the left shows the Canister Laboratory equipment for development and the image to the right shows the rotating tool that is pressed into the joint between the parts that are to be combined.*

5 Further research and technology development

In Sweden, the work of studying the handling and final disposal of the radioactive waste from the nuclear power plants began in early 1970s (the AKA-study). The technical solutions and the system that since then have been developed for handling and geological disposal of the radioactive waste from the nuclear power plants is described in Chapter 2. Facilities that remain to be constructed in order to be able to decommission and dismantle the nuclear power plants and final disposal of the resulting waste are also described there. A plan of action is presented in Chapter 3.

This chapter presents an overview of the research and development needs that exist for the remaining parts of the nuclear waste programme. In Section 5.1 the bases for the planning of future research and technology development are described. Section 5.2 gives an overview of the milestones that are relevant for the different repositories and Clink as well as the state of knowledge that is required and how far technology development must have come at each milestone. Sections 5.3 to 5.10 then describe the research and technology development needed for repository design and construction issues, as well as the research that is needed to carry out assessment of the safety of the repositories after closure. Section 5.11 provides an overview of the technology development needs for decommissioning of nuclear facilities. In Section 5.12 other activities and knowledge that are relevant to SKB's mission are summarised.

5.1 Future efforts

SKB's and the licensees' planning of future research and technology development initiatives is based on the stepwise decision process presented in Section 3.2. The milestones relating to decisions steps in the form of applications, dictate when knowledge and development of the technology needs to have reached a certain level while SSM's approval dictates when SKB can commence activities such as construction and operation of the facilities. Milestones linked to submission of the applications will obviously be crucial for the planning of research and development activities as they dictate when different issues must be solved and when reviewing of the current state of knowledge should be carried out and of the technical solutions that will be used.

SKB has as part of the applications for a licence to build the Spent Fuel Repository, Clink and the extension of SFR compiled reviews of the state of knowledge and the status of technology development and evaluated the importance of remaining uncertainties concerning the protection of human health and the environment against radiation after closure of the repository (SR-Site for the Spent Fuel Repository and SR-PSU for SFR).

For SFL a concept study has been carried out and a concept has been chosen for which a safety evaluation is currently under way. The outcome of the safety evaluation that will be available in 2018 will clarify the need for future research and technology development initiatives for SFL. It will also provide a basis for specifying site selection criteria and for devising a siting process.

These reviews and studies, together with the viewpoints provided by SSM in connection with its examination 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 repository system and thereby the basis in order to judge their importance for post-closure safety. The work is being carried out in accordance with the guidance of and strategy for research described in Chapter 4.
- The need for knowledge and competence around **design, construction, manufacture and installation** of the components to be used in the facilities. The work is being carried out in accordance with the technology development process as described in Chapter 4.

- The need for knowledge and competence around **inspection and testing** to verify that the barriers and components are produced and installed according to approved specifications and thereby satisfy the requirements. This also includes development of methods and instruments for inspection of the final repository and the site.

Furthermore, better knowledge of the inventory of radionuclides in the low- and intermediate-level waste is needed, as well as the properties of the spent nuclear fuel and development of technology for handling of both waste and spent nuclear fuel.

5.2 Overview of the final repository and Clink

In Chapter 3, the most important milestones in the planning of new facilities are given. This section further defines the milestones and phases that are relevant for the final repository and Clink. In addition, an overview of the state of knowledge that is required and how far technology development must have reached at the different milestones is given.

5.2.1 Final repository for short-lived radioactive waste

Prior to construction

The preliminary PSAR (F-PSAR) that was prepared and submitted in conjunction with the applications for an extension of SFR will be updated to a PSAR prior to the construction. This includes an update of the safety assessment SR-PSU. In conjunction with this, SKB will also describe how safety in the existing facility will be ensured when constructing the extended part.

Before the construction starts, the technical development concerning the engineered barriers will be conducted to meet the requirements and optimise the design premises, since these, in turn requires on the design of the extension of SFR. Corresponding approach applies also for the closure components, for example plugs, the closure requirements and design premises must be identified which formulates the design and execution of the rock excavation works.

Before the rock excavation can begin, a number of survey boreholes adjacent to the extension will be sealed. The sealing technique will be developed and adapted to the conditions at SFR and a programme for quality control will be designed and established.

The construction phase

Design premises and requirements for the extension are established when the construction starts, Construction will begin with necessary infrastructure being established and prepared. No deposition in the existing SFR will take place during the period that rock excavation is underway. If necessary, a short deposition period could be planned after the completed rock excavation works. Since that the operating time has been prolonged in relation to that originally planned, the existing facility will be upgraded in parallel with the extension.

In the final phase of construction, verification and validation of the systems and functions will be carried out. The construction phase will be concluded with integrated testing of SFR.

Prior to trial operation

Before trial operation begins, an updated SAR will be produced to describe the as-built facility. 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 (SDM-PSU). The new information will provide a more detailed picture of the properties of the rock in the extended area. The positions of deformation zones and ground water conditions will be modelled with improved certainty. Safety-related technical specifications (STF) will also be developed.

The closure plan is reviewed and updated.

Prior to regular operation

Prior to regular operation of the extended SFR, a supplemented SAR will be developed at the same time as the safety-related technical specifications (STF) are updated. The documents are then supplemented with experience from trial operation. As SFR is commissioned, activities shift to a management phase with periodic evaluation of overall safety and radiation protection every ten years.

Trial operation is not expected to lead to any specific technology development needs or research prior to regular operation.

Prior to closure

Prior to dismantling, demolition and closure of the facility, development will be carried out regarding technology for closure and closure components with regard to materials, techniques and installation. The decommissioning plan will be augmented and an updated analysis of post-closure safety will be included in the revised SAR.

5.2.2 The final repository for long-lived waste

The development of the final repository for long-lived waste, SFL, focuses during this RD&D period on the ongoing safety evaluation and on defining a strategy for the site selection process. The safety evaluation constitutes the basis for identifying areas for further research and technology development and then formulating the long-term research programmes that are needed to carry out complete safety assessments. However, a general review of the state of knowledge after the safety assessment, SR-PSU, which was submitted in support of the applications to extend SFR, indicates that the remaining issues identified for SFR are also largely valid for SFL.

Safety evaluation

The starting point of the safety evaluation is the proposed repository concept (see Section 2.1.2) and the data obtained in SR-Site and SR-PSU. Research and development needs identified during the safety evaluation will be used as a starting point for the formulation of a long-term research programme linked to long-lived waste and SFL. The safety evaluation also has an important role for the work with the siting of SFL, since it is expected to clarify the requirements on the rock as a barrier.

Siting

Site selection for SFL will be based on the siting factors, which still are to be decided on, and information regarding different sites and their properties. The siting factors and evaluation parameters are based, among other things, on the set of requirements for the site that the safety evaluation results in. When the siting factors are established, a compilation of existing material of relevance for the siting will be carried out. The concerned municipalities and other stakeholders will be involved in the process (see Section 3.3.4). The next step in the process is site investigations, which involve geosphere and biosphere parameters being investigated and monitored according to the methodology SKB has developed during previous sitings. The work is planned to also include studies of factors such as health, environment, infrastructure and societal resources.

Prior to the licence application

In support of the applications under the Nuclear Activities Act and the Environmental Code to construct SFL, a preparatory PSAR (F-PSAR) is submitted. A site needs to be selected and described as a basis for the assessment of the post-closure safety of the repository.

Technology development needs to have reached to a point where it is possible to show, in a safety assessment, that the repository is safe from a post-closure perspective. A first version of the design premises (see Section 4.2.4) needs to be presented and it must be likely that the technical solution can be delivered and installed in such a way that it can be verified that the requirements are met. For the current repository concept, constructions in the repository sections and technical solutions for backfill are judged to be areas that need developing, since they differ from solutions in the other SKB repositories.

Preliminary acceptance criteria for waste need to be formulated as well as technical solutions for treating and conditioning waste to fulfil the acceptance criteria.

Prior to construction

Prior to start of construction, the F-PSAR will be updated to a PSAR. The PSAR will contain an updated reference inventory, the design of the facility and compliance with the specified requirements.

The work with technology development will have resulted in a basis for design of constructions for the repository. Design can then be carried out in parallel with driving the ramp down to repository depth.

Prior to trial operation

Similarly to SFR, detailed design of the closure components will be executed as a basis for the SAR prior to trial operation. This material must be sufficiently detailed to allow design of the closure. The closure plan is updated based on the detailed descriptions and a decommissioning plan will be prepared.

5.2.3 The Spent Fuel Repository and Clink

Prior to and during construction

Prior to the start of construction of the Spent Fuel Repository and the encapsulation part of Clink, a PSAR will be prepared for each facility. The reporting of post-closure safety for the Spent Fuel Repository will be updated, among other things with parts of the material presented in supplementary material for the application under the Nuclear Activities Act, SSM's viewpoints provided in its review of SKB's applications and the updates in the initial state that ongoing technology development leads to. This reporting will be included in the PSAR.

Primarily more knowledge is required to analyse the uncertainties and judge to what extent these may be reduced. This yields both a more realistic assessment of safety and a basis for optimisation of the repository so that adapted design premises 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 part of Clink is available prior to the start of construction. This means that the detailed design phase (see Chapter 4), should essentially have been completed for all barrier systems, except the parts that require that tests are performed on site. For the Spent Fuel Repository, primarily investigation methods and technology for construction of the repository accesses must be completed. The supporting material is needed for the handling of issues relating to the radiological safety during construction of accesses, the central area and the first deposition area. The material will be presented in Suus (Safety during construction of the final repository, see Section 3.4.6).

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 preparation of the PSAR before the start of construction.

The development of the nuclear fuel measurement is affected by the fact that the technical systems must have essentially passed the detailed design phase for the encapsulation part of Clink. Furthermore, the chosen drying method must be verified and validated and the work to introduce safeguards in the design of the encapsulation part of Clink and the final repository must be completed.

In the same way, methods for welding and inspection of the canister during encapsulation need to be designed and adapted to the prevailing nuclear environment, although implementation work during detailed design and construction phases for Clink remains to be done.

An effective production system for the canister needs to be available at the latest in time for integrated testing of Clink and the Spent Fuel Repository. At present, conditions for common canister production with Posiva are being investigated and if this is done the production system needs to be finished earlier. Technology and methods for production of canisters must be developed prior to commissioning tests, and work on an industrial scale prior to trial operation. The canister design must then be verified against the requirements, manufacturing methods shall be verified, test and inspection processes must be developed and conditions for acceptance must be described.

Build-up of knowledge for questions concerning post-closure safety is primarily focused on SAR; however, timetables may need to be adapted to that which emerges from the ongoing review of SKB's application. Submission of the PSAR and SKB's decision on start of construction, provided that a licence is issued, imply under all circumstances an important cross-check for these issues, since long-term efforts are required for knowledge accumulation, and since SKB at the start of construction must take a crucial investment decision where data for all issues that can affect the future progress of the project needs to be in place.

Start of construction deposition area

Prior to the start of construction of the deposition area in the Spent Fuel Repository, the design specifications for this must be established and thereby design and installation methods for buffer, backfill and plugs need to be ready as well as methods for investigation of repository rock and methodology for excavation of deposition tunnels. Structures and inspection methods that are to be applied must be verified.

Detailed design of the production of buffer and backfill shall be completed as a basis for detailed design of the production building. This means that the following need to be carried out:

- Requirement specifications on material must have been determined.
- A decision on the pressing technology for buffer blocks must be made.
- Manufacturing methods that work on an industrial scale for production of bentonite blocks and bentonite pellets for buffer and backfill must have been determined.
- Inspection methods that work on an industrial scale must have been determined.
- Prerequisites for preparation of a system for quality control and inspections must be established.

In Forsmark, the monitoring of the natural system has continued virtually unchanged after the site investigation and is planned to continue up to the start of construction of the Spent Fuel Repository. Some adaptation has, however, been made and is planned as a result of completed evaluations. Monitoring provides a basis for establishing a reference level that can be used to assess 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, in relation to the site investigation, is predominantly monitoring that will be performed under ground. In the Suus document, which will be completed for PSAR, a plan for how the monitoring of the rock will be carried out during the construction of the Spent Fuel Repository is included. The PSAR gives a plan for how monitoring will continue during the construction of the repository and during the operational phase, as well as of the monitoring planned of the engineered barriers.

Prior to trial operation

In the Spent Fuel Repository and Clink, integration tests and integrated testing are planned which will constitute verification that the activities, construction and deposition, can be carried out in the Spent Fuel Repository so that both safety during operation and post-closure safety is maintained. These tests are carried out in a late stage with the equipment and with participation of the personnel who will operate the facility as a final control that operation can take place in the intended manner.

Integrated testing entails that 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. A system for quality control and inspection of canister manufacturing, production of buffer and backfill components, handling and installation of canisters, buffer and backfill and the rock construction process must be implemented.

Prior to the preparation of SAR, development work will be required in order to be able to handle additional data from a repository extension in the safety assessment. The conclusions from the SR-Site safety assessment and the research questions suggested as well as the viewpoints SSM provides in its review of the applications form an important basis for the identification of the questions concerning post-closure safety where knowledge needs to be deepened for the review in SAR.

Prior to regular operation of the KBS-3 system

Before the system is put into regular operation, the SAR is supplemented with experience from trial operation. As the repository system is commissioned, activities shift to a management phase, where SAR must be kept up-to-date and with periodic overall evaluation of safety and radiation protection every ten years.

After deposition

In Sweden there are no provisions in laws or other statutes requiring, for example, that spent nuclear fuel that has been deposited in a final repository must be possible to retrieve. According to SSM's general advice on the regulations concerning safety in connection with the final 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 a deterioration in the long-term safety of the repository. There may be situations where pre-closure retrieval is necessary.

SKB deems that it is in principle possible both before and after closure to retrieve canisters from the planned Spent Fuel Repository. This has been demonstrated practically in tests in the Äspö HRL, most recently in connection with the mining of the so-called Prototype Repository (Svemar et al. 2016). The execution will be significantly more work and resource-consuming after closure.

The encapsulation part of Clink is designed so that it is 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 submitted as a basis for construction of the Spent Fuel Repository. To allow retrieval some technology development is also needed. It needs to be ensured that the deposition machine can retrieve canisters, also practical methods to handle bentonite blocks that partially have begun to become water-saturated need to be developed.

The understanding of the site as well as quality control and inspection of the process production of the engineered barriers in the repository is essential in order to be able to evaluate the post-closure safety of the Spent Fuel Repository. Monitoring of the evolution of the repository's barriers from deposition and until closure can further enhance this knowledge. It is also an important confidence issue, even if the additional knowledge will only cover a very short time of the evolution of the repository. SKB is planning for such monitoring, but a number of limitations in what is possible need to be considered. Furthermore, a suitable methodology and strategy for monitoring needs to be developed and tested.

Monitoring is primarily not intended to find manufacturing errors or other deviations in the material, equipment or handling. These important tasks are handled within the framework of the quality control programme, see Section 4.3.4.

SKB intends, together with Posiva, to develop methodology for monitoring of the repository. SKB and Posiva are also participating in the EU project Modern2020, where these issues are being studied. Experiments regarding monitoring of engineered barriers will also be carried out in the Finnish facility Onkalo, and at Äspö HRL. In conjunction with these tests, different methods for monitoring will be developed and tested. The goal is to develop methods that can be used to monitor the engineered barriers under repository conditions during many years, or even decades.

Prior to closure

As a basis for the application to seal the Spent Fuel Repository, a revised SAR with an updated assessment of post-closure safety, plus a plan for closure and decommissioning is submitted. It presents the technology and the procedures intended to be used for closure of remaining underground openings (closure of deposition tunnels is carried out during the operating period) and boreholes as well as the measures that are planned for monitoring and control of the repository and the activity at closure. The updated assessment of post-closure safety will be based on the as-built facility and the planned closure measures.

5.3 The low- and intermediate-level waste

This section describes the needs for research and development regarding the low- and intermediate-level waste in order to obtain a deeper process understanding for the waste, including the waste matrix to be disposed of in SFR or SFL. Similarly, handling and conditioning of, in particular, the long-lived low- and intermediate-level waste needs to be developed. In preparation for the decommissioning of the nuclear facilities, waste containers and waste transport containers must also be developed for the long-lived waste. Knowledge of the radionuclide inventory for the different repositories needs to be deepened as also briefly described. The work that is on-going and is planned during this RD&D period is further described in Chapter 6.

5.3.1 Radionuclide inventory

In-depth knowledge of the radionuclide inventory for the low- and intermediate-level waste is needed both prior to the development of the PSAR and before the extension of SFR as well as for the safety evaluation for SFL. The uncertainties regard both predicted waste quantities and content of radionuclides in each waste package. The radionuclide inventory is updated continuously and special effort is made for the so called “hard to measure” nuclides, which in many cases contribute significantly to the long-term risk.

5.3.2 Process understanding

Studies whose results will serve as a basis for PSAR before the extension of SFR are presented below. The results that are relevant for SFL will also be used in the design and safety evaluation of SFL.

Sorption impact

In SFR, sorption of radionuclides on cement components is one of the most important processes for retarding the release of radionuclides. Chemical degradation of organic matter in the waste can generate products that can form complexes with radionuclides and thereby affect the degree of sorption on cement minerals. Similarly, there are organic compounds in the form of additives in the concrete.

In particular, three questions concerning the degradation of organic material will be further studied: i) How degradation products from cellulose affect plutonium sorption to cement. Within SR-PSU it was identified that the amounts of cellulose deposited in 1BMA will give rise to reduced sorption of especially plutonium. ii) Survey of which degradation products are formed from filter aids and how they affect the sorption capacity of cement. iii) How degradation products from organic cement additives affect the sorption of radionuclides by possible complexation. Before the extension of SFR, it is important to identify which cement additives are suitable for construction of the concrete structures.

Gas formation

The low- and intermediate-level waste includes, among other materials, metals and organic matter. In case of corrosion of aluminium and zinc in an oxygen-free environment hydrogen gas is produced, which may damage the concrete structure if the pressure becomes too high. The process needs to be better understood to be able to design the optimal system for gas transport in the facility. By an experimental study where aluminium and zinc is embedded in concrete which is saturated with and exposed to oxygen-free groundwater, SKB is expected to obtain better knowledge of the corrosion rate and thereby also the gas evolution.

Another process that potentially may cause gas formation is methanogenesis by microbial degradation of organic materials, such as cellulose. SFR is designed to create unfavourable conditions for microbial degradation and in SR-PSU requirements on high pH and low organic content materials were formulated. Further studies will be conducted to gain a better understanding of how the future pH development in different parts of the repository may vary and how different pH conditions can affect the gas formation.

Swelling of ion exchange resins

Swelling of ion exchange resins is another process that potentially may affect the integrity of the barriers and thereby also the water flow and radionuclide transport out of the repository. Experiments with bitumen-solidified ion exchange resins will be performed to estimate how long the resaturation process takes and when the measured swelling pressures will occur.

5.3.3 Handling of the low- and intermediate-level waste

Acceptance criteria for the long-lived low- and intermediate-level waste will be defined only when the design of SFL has been determined. Today, there is, however, a need to clarify the planning prerequisites for handling of the waste produced during operation and decommissioning of the nuclear facilities. In the ongoing safety evaluation, requirements on the waste from the proposed repository concept, so-called preliminary waste acceptance criteria, were formulated. The work also includes an analysis of what will be done with the waste that in its current form does not fulfil the acceptance criteria.

Segmented core components from upgrading of the NPPs are today interim-stored in steel tanks at the NPPs. The decommissioning waste from the NPPs will according to the plans also be placed in steel tanks. Future acceptance criteria for the long-lived waste may require that the waste is stabilised before final disposal in SFL. Methods for stabilisation of waste in steel tanks must then be developed. Also, future requirements may be made on conditioning and packaging of waste from AB SVAFO and Studsvik Nuclear AB, with a need for technology development as a result.

As well as fuel, core components are also stored in Clab, mainly BWR control rods. In order to secure the interim storage capacity for spent fuel, different alternatives for the continued interim storage of fuel and core components have been evaluated, see Section 3.4.3. Segmentation of BWR control rods is still regarded as the main alternative before final disposal in SFL, to realise this in-depth technical studies need to be carried out. Development activities related to this will be performed when Clab needs to free storage capacity for fuel or in the connection to the commissioning start of SFL.

Reactor pressure vessels from PWRs will be disposed of in SFL. These may be segmented and handled in smaller parts, or handled whole (with or without internals left in the pressure vessel). Studies for evaluating alternatives will be carried out as a basis for the continued planning of SFL. The whole BWR reactor pressure vessels are intended to be deposited in the extended SFR, and up to the start of the detailed design of the SFR extension, technology development is taking place in order to be able to handle the RPVs. Within this technology development, other large components are also studied to shed light on the advantages and disadvantages of the method with depositing these whole.

5.3.4 Waste packaging and waste transport container

In order to be able to carry out the decommissioning of the nuclear facilities in an optimal manner, development work needs to be carried out concerning waste and waste transport containers for long-lived waste and short-lived decommissioning waste. A new waste transport container for long-lived waste is planned to be certified and commissioned in 2020 (ATB 1T) and further 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.

Continued technology development of waste packages for decommissioning waste, *tetra-moulds* and *double-moulds*, will take place under the RD&D period. The prerequisites need to be established for the detailed design of the extension of SFR.

Technical issues concerning how long-lived waste from AB SVAFO and Studsvik Nuclear AB should be treated and packaged need to be resolved when acceptance criteria for long-lived waste has been established. This work can for instance include development of new types of packages.

5.4 The spent nuclear fuel

This section describes the needs for technology development and research that are needed to obtain better process understanding for the properties of the spent nuclear fuel and the knowledge needed for the development of the technology for handling.

To SAR there are a number of issues related to the assessment of post-closure safety that require further research with the state 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 a basis for system design and detailed design of Clink and completed during construction and commissioning. In Chapter 7 the programme for research and technology development with regard to the spent fuel during this RD&D period is described.

5.4.1 Process understanding

If the integrity of a canister is breached and water enters, the properties of the fuel are crucial for determining if and when radionuclides can be released. The results of the safety assessment SR-Site show that the rate at which radionuclides are released from the fuel's different parts are of crucial importance for the post-closure safety of the Spent Fuel Repository. An in-depth understanding of the mechanism for dissolution of the fuel matrix is required to support the experimental results and thereby reduce the uncertainties in future safety assessments, which was also expressed by SSM both within the framework of the licensing process and the review of the RD&D programme 2013. Dissolution data are needed for new types of doped fuel and for high-burnup fuel. The fraction of radionuclides that is not embedded in the fuel matrix, and can thereby be relatively quickly released, needs to be quantified more accurately as well as the rates of the dissolution of the metal parts of the fuel assemblies and of control rods. There are also some uncertainties concerning speciation and solubilities of released radionuclides.

5.4.2 Handling of the spent nuclear fuel

Non-regular fuels are fuels that substantially differ from the regular spent nuclear fuel. Two important examples of a non-regular fuel are damaged fuel (with several subgroups) and samples and residues from analyses of various types, mainly from Studsvik. These are handled, treated and analysed separately, taking into account the fact that these fuels are present in relatively small quantities. Continued planning and handling of these non-regular fuels will be done during the coming years. Principles for how the small quantities of non-regular fuel should be handled in the assessment of post-closure safety will be presented in the PSAR for the Spent Fuel Repository.

In order to ensure that the fuel will be possible to handle in both Clab and the encapsulation plant, a programme is conducted for inspection of ageing of fuel, where changes in the properties of the fuel during pool storage are being studied. SKB will also follow the state of research regarding the ageing of fuel more generally in order to ensure that the fuel can be handled in the encapsulation plant.

5.4.3 Fuel information, criticality and safeguards

In the coming years further studies will be done on how information on the spent nuclear fuel best should be managed and stored prior to commissioning of the complete KBS-3 system.

As there is a temperature requirement on the bentonite in the final repository, the decay heat from the fuel is an essential design parameter. Work with both calorimetric determination, determination by nuclear measurements (particularly gamma and neutrons) and calculation codes will continue during the period. For various milestones of Clink, the objective is, to provide a basis for the design of the parts where monitoring equipment will be used and to ensure that sufficient knowledge of fuel properties exist in order to carry out analyses.. At the same time, methods will be developed where other essential fuel parameters are determined, such as amount of fissile material, radionuclide inventory, fuel identity, burnup and decay time, reactivity, etc.

The development in recent years of the capacity to assess criticality will be consolidated and made more efficient, meaning for example that a common computational environment with VNF (Vattenfall Nuclear Fuel) will be established, procedures made more efficient and that requirements following from credit burnup being applied are met. The current version of the calculation tool Scale will be updated and validated. Requirements on the criticality analysis from the final repository will be verified, for example regarding the impact of pores in the insert. A strategy will be prepared for fuel bundles that do not meet the requirements on burnup. The work is continuing with analysis of consequences of geometrical changes inside the canister.

Continued work with the methodology and equipment for verification of fuel for safeguard purposes are planned prior to submission of the PSAR for the Spent Fuel Repository and Clink. This will be coordinated with the aforementioned decay heat project. The work is being pursued in cooperation with the IAEA, Euratom and SSM.

5.5 Canister for spent nuclear fuel

This section describes the needs for supplementary research that are needed on the copper canister properties prior to future SAR for the Spent Fuel Repository. Further, the technology development that is needed, for the canister to be manufactured, to be verified against stipulated requirements and to be used in the KBS-3 system, is described. An important milestone is the updating of the production report[5] for canisters that will serve as a basis for the PSAR for the Spent Fuel Repository and the PSAR for Clink. Chapter 8 presents the planned efforts during the RD&D period.

5.5.1 Process understanding

For the assessment of post-closure safety of the Spent Fuel Repository there are issues regarding corrosion and copper creep which require further efforts, primarily until SAR, with the state of knowledge in the PSAR and prior to a decision on start of construction as important cross-check points.

Corrosion

In the long-term, sulphide is the most dominating copper corroding agent in the repository environment. In the SR-Site safety assessment, the dominant contribution to risk was canister failure due to corrosion by sulphide after loss of the buffer. A better understanding of the details of this corrosion process strengthens the scientific basis of the safety assessment. Of crucial importance is also the quantitative understanding of which sulphide concentrations that can exist in the repository environment, both in clay materials and in the groundwater (see further Sections 5.7 and 5.8). Moreover, efforts are required concerning localised corrosion, copper corrosion in pure, oxygen-free water, radiation-induced corrosion and stress corrosion cracking, with a focus on the unsaturated period, as well as on verification of different copper material's (base material, welded, cold-worked etc.) resistance to corrosion.

[5] The production reports for the KBS-3 system describe how the final repository is designed, how the repository and its barriers are built and inspected. The production reports comprise a part of the SAR, and there are production reports for the spent fuel, canister, buffer, backfill, closure and underground openings.

Copper creep

The understanding of creep of copper in case of mechanical loads is incomplete. Among other things, a better understanding is needed of how an addition of phosphorus leads to favourable creep properties. The results are needed to improve creep modelling in the design analysis of the canister and to stipulate requirements on phosphorous contents in the copper material, both for the updating of the initial state of the canister in the PSAR.

5.5.2 Design and manufacturing

When it comes to technology development for the canister, the design phase should essentially be completed to PSAR. This means that the design and defect analyses need to be updated based on updated design premises. For this, an update of the material model for the cast iron insert is for example required. Additional process understanding is needed to determine the requirements on hydrogen, oxygen, sulphur and phosphorus in canister copper and requirements on the copper content in the cast iron insert.

A prerequisite for a proper production system for the canister is that technology and methods for production of canisters are developed, and that they work on an industrial scale. Furthermore, technology and methodology for welding of canisters need to be well developed so that the welding process works on an industrial scale. The equipment that will be used in Clink for sealing of the canisters needs to be adapted to the nuclear environment there.

5.5.3 Inspection and testing

Testing of canister components and welds is done to determine, for example, properties, dimensions and the occurrence of defects while inspections are aimed at verification against the acceptance criteria.

As a prerequisite for future qualifications, SKB must, for the PSAR, present an inspection programme for the canister. An inspection programme describes SKB's and the suppliers' procedures for supervision, inspection by independent bodies (called third-party inspection) and SSM's monitoring by reviews and inspection. This means that the required inspection techniques/processes need to be described and that important acceptance criteria for inspections are available. Regarding non-destructive testing, SKB will need to demonstrate that the canister, including its components and welds, is testable. For this judgement, background data in the form of possible and probable defects, their acceptable sizes and a general description of the testing methods/testing processes that will be applied to ensure that non-acceptable defects can be detected is needed. The overall inspection and testing processes will, however, not be developed for the PSAR for Clink or the Spent Fuel Repository but will rather be finished prior to commissioning tests.

A prerequisite for a qualified production system for the canister is that technology and methods for inspection of canisters are developed so that they work on an industrial scale. The inspection that will be carried out in Clink must also be adapted to the nuclear environment there.

5.6 Cementitious materials

This section describes the research that is needed to obtain better process understanding of the properties of cementitious materials in the waste matrices, barriers and structures in the final repositories. Furthermore, the technology development that is needed for the design of concrete structures, material and production procedures is described.

Before the PSAR prior to the extension of SFR, SKB is planning activities within both these areas. Prior to the safety evaluation for SFL, SKB is mainly planning studies linked to the design of concrete structures, material and technology.

For the detailed design of the Spent Fuel Repository, SKB plans mainly studies linked to the design of low-pH cementitious material for plugs, and for grouting and rock support.

For a detailed programme for the RD&D period with regard to cementitious materials in SFR, SFL and the Spent Fuel Repository, see Chapter 9. The programme linked to cementitious materials in the waste matrix is described in Chapter 6.

5.6.1 Process understanding

During the time period covered by an analysis of post-closure safety, the cementitious material's composition and properties will slowly change. These changes can be caused by chemical processes, such as interactions with groundwater or components 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.

As cementitious materials have a central function in fulfilment of post-closure safety, knowledge and the ability to model the evolution of the material properties over time further need to be strengthened.

SKB plans to continue to coordinate the research that occurs on the long-term properties of cementitious materials for the two repositories, SFL and SFR.

Interaction and degradation of material

By interactions with groundwater and ions dissolved in the groundwater, the cementitious material's composition and structure will change. The scope of this change depends on the composition of the original material and of the groundwater. SKB is planning now to develop its programme for studies of interactions between groundwater and concrete under repository conditions in order to gain a better understanding of what effects these processes may have on the properties of the cementitious materials over the relevant time spans.

In the existing silo in SFR and the planned rock vault for legacy waste, BHA in SFL, contact between cementitious materials and bentonite will occur. When these materials a time after closure become water-saturated, chemical interactions can occur, which leads to ion transport and changes of the material's composition, properties and structure. Studies of interactions between cementitious materials and bentonite are being pursued as an experiment/long-term test at the Äspö HRL.

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

With manufacturing of cementitious materials, in addition to cement, water and aggregate, different types of additives can be used to alter the material's composition to, for example, ensure a good workability of the fresh material. SKB is planning to initiate a feasibility study on how a changed cement composition may affect the material's properties during the periods covered by the assessment of post-closure safety. The results are judged to have to be available prior to the formulation of a research programme for SFL, which will be put together after the safety evaluation is completed.

Mechanical loads and gas transport

When the pore water in a cementitious material freezes, it will expand and the material will be exposed to an internal pressure. If a sufficiently large fraction of the pore water freezes, the internal pressure becomes so great that fractures arise in the material or the material completely falls apart. The temperature at which the degradation takes place is dependent on the pore structure since the freezing temperature of the water is clearly dependent on the size of the pore in which it is enclosed. SKB plans during the RD&D period to carry out additional studies linked to freezing of fresh and aged cementitious materials, mainly concrete, during permafrost-like conditions.

After closure, the concrete structure together with the grout and the waste packages/waste will be subjected to external loads from for example groundwater pressure or rock breakout. Repository components may also be subjected to internal pressures from for example swelling waste or gas-producing processes. As a part of the technology development program preceding the construction of the extension of SFR, a survey is under way of mechanical loads due to swelling waste, for example from corrosion and ion exchange resins, as well as structural mechanical impacts on the concrete barriers. In order to obtain a better system understanding with respect to structures and crack growth in these during construction and operational phases, SKB will conduct fracture risk calculations for concrete structures. The results from the calculations comprise the design premises for construction and premises for operation and maintenance.

Gas produced by degradation of organic waste and by metal corrosion in the repository must be able to be transported through the cementitious materials without properties relevant for the post-closure safety being affected. Research on gas transport through the cementitious materials will be coordinated with the development of material and methods for outward transport of gas from a repository structure.

5.6.2 Construction, manufacture and installation

In preparation for the extension of SFR and prior to the construction of SFL and the Spent Fuel Repository, SKB plans to carry out extensive work linked to the design of concrete structures and materials for the different repositories. A large part of the work must be carried out and concluded within this RD&D period. The following sections give a summary of the planned development work. The full programmes are presented in Chapter 9.

SFR and SFL

Before the construction of the extension of SFR, SKB plans to conduct studies and development work, which includes design of concrete structures, development of material and production methods for the following areas and system/components: Grouting, caissons, management and final disposal of whole reactor pressure vessels plus systems for gas transport. The main focus will, however, be on the development of the structural concrete for the caissons in 2BMA, as well as development and testing of production methods for construction. The results from these studies and development work will be included in the PSAR.

In order to realise the repository design of SFL, with large amounts of concrete backfill, material needs to be developed and methods for installation prepared. Development of these parts comprises an important basis for creating a first layout of the repository. Design of concrete structures, material and technology covers the two waste vaults for legacy-waste and core components in SFL and for grouting of waste packaging. At present, SKB has no development work of a structural concrete for the two waste vaults, BHK and BHA, in SFL. The assessment is that the concrete that is currently being developed for the caissons in 2BMA will be used also in SFL. SKB is waiting therefore with an individual development programme until the development project for 2BMA has been evaluated.

Prior to completion of the safety evaluation for SFL, a study is planned linked to material and methods for engineering of a concrete structure on concrete directly against rock and backfill of a rock vault with concrete. The programme includes a review of existing methods and materials.

The final repository for spent nuclear fuel

Current design premises and requirements on the Spent Fuel Repository assume use of low-pH materials for ensuring that the material leaching products do not adversely affect the bentonite in buffer and backfill. SKB has previously developed this type of material for grouting, rock reinforcement and for plugs for deposition tunnels. However, the long time spans that extend over construction, operation and closure of the Spent Fuel Repository lead to that the availability of the constituent components in the current mix design may change and some products may be cancelled. For this reason it is of importance that the mix is designed in such a way that the material properties will not be completely dependent on a specific product and that constituent components can be replaced with products with similar properties.

Prior to detailed design, SKB plans to conduct studies and development work within two areas: Design of low-pH concrete as a plug for closure of the deposition tunnel and design of low-pH materials for grouting and rock reinforcement.

The focus of these development works is to ensure that the current mix design is made more robust towards changes in the availability of the constituent components, for example cement or additives.

5.7 Buffer, backfill and closure

The main purpose of the clay barriers in the Spent Fuel Repository (buffer and backfill), SFR (silo filling) and SFL (clay barrier in the rock vault for legacy waste) is to restrict water flow around the canister and around the low- and intermediate-level waste. This is achieved by a 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 Spent Fuel Repository, the design of buffer, backfill and closure needs to be further developed so that a sufficient basis can be given for the PSAR and for the continued design of the repository and the production system for bentonite components. This is described in the production reports for the KBS-3 system that serve as a basis for the PSAR. The need for measures for quality assurance during manufacturing, handling and installation need to be further developed. This is done by a product and process mapping and through developing preliminary quality plans. For this to be possible – and in order to be able to establish clear requirements and practically achievable testing methods, research is needed regarding the properties of the buffer material and how it behaves after installation until it becomes water saturated and full swelling pressure develops.

Additional efforts related to SFL are dependent on the outcome of the safety evaluation and will be defined when it is finished. For SFR the state of knowledge is considered essentially satisfactory regarding the silo while some efforts are required for backfilling and closure of boreholes prior to the PSAR and SAR for the extension of SFR.

The programme for the buffer, backfill and closure is presented in Chapter 10.

5.7.1 Process understanding

For the assessment of post-closure safety, there are a number of issues that require further research. This applies to all repositories with bentonite barriers. For the Spent Fuel Repository and the extended SFR, this applies primarily until SAR, with the state of knowledge in the PSAR and prior to a decision on start of construction as important cross-check points. The ongoing safety evaluation for SFL will primarily be based on existing knowledge when it comes to the properties and function of the bentonite, but for future assessments, data need to be strengthened.

Most processes in the bentonite barriers are common between the different facilities and much of the results from the research that is conducted for the Spent Fuel Repository can also be used predominantly for SFL, but also for the silo in SFR.

Homogenisation and water uptake

In all repositories bentonite barriers are installed as components consisting of blocks, and pellets with voids in between. For the function after closure, it is the water-saturated homogenised properties that are crucial. This means that it is important to be able to understand and describe both the homogenisation process itself and the barrier's properties after water saturation. Today, there are deficiencies that need to be rectified in the models that describe the bentonite's mechanical evolution. It is determined to some extent by how water uptake in the clay occurs. It is therefore important to gain a better understanding of water uptake in bentonite, especially for drier conditions. Homogenisation is also important after mass loss due to erosion.

Vaporisation and salt enrichment

The relatively high temperatures in the Spent Fuel Repository together with a slow water supply could lead to vaporisation of water in the buffer near the canister, with salt enrichment as a result. Condensation in the colder part of the buffer will reduce the importance of this process. Further studies to understand the relationship vaporisation/condensation under different conditions are, however, necessary.

Piping/erosion

During the period from the installation of the bentonite barriers up to and including the point at which hydrostatic pressure has been restored, 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 order to evaluate the consequences of erosion, it is important to understand how water is taken up in pellet fills under different conditions. A programme with both experiments and modelling has therefore been initiated. The results will be used to be able to establish better requirements on inflow criteria and the installation sequence. This process is common to all repositories with bentonite barriers.

The hydromechanical properties of the bentonite barriers

Further investigations of the relationships between swelling pressure, hydraulic conductivity and shear strength are required for different bentonite materials in order to provide a fundamental understanding of the connection between bentonite material composition and properties. This is in principle already concluded for silo material in SFR, but for the Spent Fuel Repository and SFL information is needed for both the technical design and for the assessment of post-closure safety.

Microbial sulphate reduction

Microbial reduction of sulphate in the buffer may cause canister corrosion. At sufficient density/sufficient swelling pressure in the bentonite the microbial activity is inhibited to the extent that the sulphate reduction can be neglected. The work of determining at which limits this occurs will continue. A greater understanding of the processes that limit the microbes' activity is also desirable. It is furthermore important to better understand the possible extent of microbial sulphate reduction before the buffer becomes saturated. The results will be used to improve requirement specifications on the barriers and to determine when microbial reduction of sulphate cannot be ruled out in the assessment of post-closure safety.

Colloid release/erosion

For the Spent Fuel Repository efforts are required above all on buffer loss due to colloid release/erosion, also a relevant issue for SFL. The results of such efforts can directly affect the outcome of the analysis of post-closure safety, for example by the fact that the pessimistic assumptions regarding buffer erosion in the safety assessment SR-Site could be relaxed.

Mineral stability

Efforts are also required concerning the long-term stability of the bentonite with regard to temperature, iron content and cement. For SFL, the interaction between bentonite and cement is one of the most important processes, but the information is also needed for the evaluation of the silo in SFR.

5.7.2 Construction, manufacture, installation and inspection

The main activities in the field of construction, manufacture and installation apply to the Spent Fuel Repository. Efforts in the closure and borehole sealing are relevant for both the Spent Fuel Repository and for SFR. Additional efforts for SFL can arise in conjunction with the safety evaluation for the facility.

Buffer and backfill

Based on updated design premises, the designs for buffer and backfill in the Spent Fuel Repository need to be revised (prior to the PSAR). Requirements on installed density are dependent on the material. Since it is not reasonable to assume that only one material will be used during the Spent Fuel Repository's operating period, a process needs to be developed in order to, based on design premises and the properties of the material, describe the design linked to a given material. The design of the backfill requires above all efforts regarding verification of the ability of the backfill to serve as a constraint against upwards swelling buffer.

As a basis for the PSAR, an ongoing evaluation of pressing technology for manufacturing of buffer blocks will be completed and a final choice of pressing technology made. In time for detailed design of the production building for bentonite components, development regarding technology and methods related to production needs to be finished. This means that the following need to be carried out:

- Requirement specifications on material must have been determined.
- The technology for pressing must have been further developed, which for uniaxial pressing entails work to minimise the lubricant and processing.
- Manufacturing methods that work on an industrial scale for production of bentonite blocks and bentonite pellets for buffer and backfill must have been determined.
- Inspection methods that work on an industrial scale must have been determined.
- The principles for quality control and inspection must have been determined.

Technical systems for disposal include, for example, deposition machine, backfill robot and transportation system for buffer and backfilling components. Prototypes or at least schematic solutions are available for the deposition work. The deposition process is intended to be automated and in order to control and monitor such a system an overriding control system is under development. Prior to the PSAR, however, material needs to be submitted from the ongoing development of technical systems for updating of production reports and preparation of system descriptions. The equipment will be further developed to be available when needed for integration tests and commissioning tests.

As a basis for the PSAR, requirements on installation sequence and installation of buffer and backfill components will be updated and clarified. An important question is, for example, which processes occur in the buffer, and what measures need to be taken with respect to these, during the time from the buffer being placed in the deposition hole until the backfill is installed above 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 plug must be designed in detail and verified. To verify that the installation of buffer and backfill works as intended and yields results within acceptable intervals, full-scale tests need to be conducted in underground conditions.

Closure

A closure sequence for the Spent Fuel Repository needs to be developed and requirements established on layout so that the plugs fit in the positions where they are needed.

SR-PSU and subsequent analyses will result in updated requirements on the closure components for SFR. Within completed technology development, a need was identified for further development for design and installation of the seal.

Detailed design of the extension of SFR and better knowledge of the rock properties can also affect the design of the closure components. Based on these data, closure will be revised in order to produce a more adapted closure including other aspects (other environmental impact, flexibility and cost-effectiveness at installation etc.). As a part of the revision the need for verifying tests will be inventoried. Based on the revision the closure plan will be updated prior to the PSAR.

Borehole sealing

Shorter boreholes may need to be sealed prior to the start of construction (Spent Fuel Repository), which makes it necessary to develop requirements and methods for borehole sealing.

In the area around SFR and where construction is planned, there are a number of boreholes that will need to be sealed. The boreholes must be sealed in order not to jeopardise safety after closure but also with regard to safety during construction of the SFR extension. Prior to the PSAR, the boreholes that lie within the extension area will be sealed. Other boreholes located in the area around SFR will be sealed prior to the closure of SFR. This means that materials and methods for borehole sealing need to be devised prior to the PSAR and applied in conjunction with the start of construction of the extension of SFR.

5.8 Rock

Technology development is needed as a basis for the PSAR and Suus for the Spent Fuel Repository and in order to enable the start of construction (approved PSAR, accesses designed in detail). Most of the planned technology development can be done without new research. Further development is judged to be able to lead to more effective selection criteria for deposition holes, which in turn means that fewer holes are rejected unnecessarily.

For the assessment of post-closure safety, there are issues that require further research, primarily in the lead up to SAR, with the state of knowledge in the PSAR and prior to a decision on start of construction as important milestones. The continued work with SFL is based on current knowledge about the barrier function of the rock, and also on the new knowledge resulting from the work on the Spent Fuel Repository and SFR. The ongoing safety evaluation for SFL, however, may point towards areas where further knowledge is needed.

The areas with remaining issues regarding technology development, research and analysis of post-closure safety for the rock are summarised below. The current situation and programme for the RD&D period is presented in Chapter 11.

5.8.1 Process understanding

Modelling of groundwater flow and transport of solutes

DarcyTools is SKB's own calculation tool for hydrogeological simulation. A number of different development initiatives are planned e.g., development in order to be able to utilise data from the detailed characterisation during construction of the Spent Fuel Repository (e.g. new upscaling algorithm and conditional simulation). Some further development of the commercial tools ConnectFlow and MikeShe are also planned, and development efforts on how the hydrogeological modelling tools will be coupled to/integrated with the surface description used in the biosphere modelling.

In recent years, hydrogeological modelling has developed so that geochemical processes and transport processes can now be integrated with the flow modelling. Efforts are required to further develop and test these new tools and extend their application areas (for example microbial processes) for use in site modelling and safety assessment.

Efforts are also required within the field of transport of solutes, above all on matrix diffusion and sorption, in terms of conceptual understanding, reduced uncertainty in transport parameters and further development of modelling tools. Efforts are also needed concerning advective transport, dispersion, electromigration, gas transport and colloidal transport. The efforts can above all lead to less pessimistic assumptions in the assessment of post-closure safety, i.e. determination of site-specific data with a reduced uncertainty interval should be striven for.

Seismic impact on safety

Earthquakes in the vicinity of the repository can induce shear movements along fractures intersecting the 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 making adequate requirements on the canister's resilience to shear load.

To reduce the negative impact of earthquakes on post-closure safety, SKB is developing, within the framework of the detailed investigation programme, methods for identification of critical structures in addition to the method with proxy parameters developed and applied within the framework of SR-Site.

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 exploit the potential for a more efficient use of the repository volume.

The largest remaining uncertainty by far concerns the relationship between the 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.

Characterisation and modelling of the mechanical properties of the rock mass

The estimation of rock stresses within the target volume at Forsmark is 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.

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

Fracture propagation in crystalline, hard rock is dependent on the thermal and hydraulic properties and on the prevailing rock stress conditions. Integrated modelling is therefore required to bound the implications of the induced deformation, both in the near and far-field, as a result of thermally, seismically or glacially induced loads.

The effect of climate on processes in the geosphere

The hydrogeological system will have different impacts on a final repository during the different stages of permafrost growth and thawing; also, these effects will be site-specific and hence need to be analysed in site models. SKB's GAP (Greenland Analogue Project) and GRASP (Greenland Analogue Surface Project) projects have provided knowledge on local catchment properties under permafrost conditions. Freezing of groundwater in colder climate conditions leads to reduced permeability and also causes volume changes which may result in the generation of new fractures, or in fracture propagation of existing fractures. The load from an ice sheet affects the hydrogeological system differently depending on the thermal basal conditions of the ice. Glacial meltwater from warm-based ice sheets can lead to diluted and oxygenated groundwater reaching a repository for spent nuclear fuel.

Models that take into account climate effects on site-specific properties related to hydrogeology, geochemistry and transport of solutes need to be further developed for Forsmark. Furthermore, the impact of climate on permafrost and glaciation processes and properties also need to be developed.

5.8.2 Production, verification and inspection

Methodology for detailed characterisation with associated modelling

The detailed characterisation (including monitoring) and modelling done in conjunction with construction and operation of a final repository, provide step by step information on the properties of the rock and its adequacy for deposition, i.e. how well they satisfy stipulated requirements for post-closure safety. For development of the detailed characterisation programme, methods, instruments and modelling methodology for verifying that stipulated requirements are fulfilled will be developed and described.

Modelling methodology needs to be established for geoscientific modelling on different scales and for different purposes. An important area where further efforts are required, both for future construction and for the assessment of post-closure safety, is development of the methodology for DFN modelling, i.e. modelling of the fracture network. Firstly, the ongoing development of methodology for conditional DFN modelling will be completed; secondly, the effect of different conceptual assumptions that serve as a basis for the DFN modelling will be further investigated. The possibility of including coupled hydromechanical processes in DFN models will also be examined.

Tunnel production

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

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 to create a more production efficient method for excavation in the deposition area such as rock excavation of deposition tunnels and boring of deposition holes, can, however, continue until detailed design of the deposition area begins.

Considering the deposition sequence in the Spent Fuel Repository where several different machines and equipment are used, the tunnel floor needs to be sufficiently level so that the tunnel will be easily accessible and to reduce maintenance and wear of equipment. The reference method today 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 excavation methodology at repository depth in Forsmark.

Prior to detailed design of the Spent Fuel Repository's deposition area, functioning methods and subprocesses 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 the deposition tunnel.
- Rock excavation methods including method for rock reinforcement and grouting determined for the deposition tunnel.
- Method for levelling of floor in the deposition tunnel.
- Method for boring of deposition holes, levelling of base and fabrication of bevel.
- Investigation method for selection and acceptance of deposition holes.

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.

5.9 Surface ecosystems

SKB's research programme for surface ecosystems is primarily to create a basis for calculations of potential radioactive dose to humans and the environment in the assessment of post-closure safety 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 programme for the RD&D period is presented in Chapter 12.

Research issues concerning radionuclide cycling and dose calculations in surface ecosystems for the three different repositories overlap each other. SKB deems 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. For SFL, complementary data for certain radionuclides needs to be produced as a basis for F-PSAR.

The most important remaining issues in surface ecosystems concern i) uptake paths 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 chlorine, molybdenum, gadolinium and nickel).

5.10 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 safety after closure. The current situation and programme for the RD&D period is presented in Chapter 13.

Important parts in the work on climate issues are gaining a better process understanding, validating the modifications of climate models that are used to describe the span of the climate that the repositories may be subjected to during the coming 100 000 to 1 million years. Moreover, other subject areas in SKB's safety assessments will be provided with climatological input data, assumptions and boundary conditions for different types of issues. Overall, the goals are process understanding, preparing climate history, updating climate scenarios and model work.

SKB deems that in the area of climate 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. There are, however, issues with a bearing on all three repositories, which need to be further studied. These issues concern primarily i) age and stability of the bedrock surface in Forsmark (including glacial erosion and denudation), ii) palaeoclimate during the last glacial cycle, iii) climate variations which make up transitions between SKB's climate domains, iv) sea-level variations in the short term and in the long term, including isostasy and shoreline displacement, v) validation of the permafrost model, vi) work with and application of a newly developed description of ice sheet hydrology from completed studies of the Greenland ice sheet (GAP – Greenland Analogue Project) for the hydrogeological modelling, and vii) the earliest possible onset of cold climate, permafrost and ice sheet growth in Scandinavia.

5.11 Decommissioning

Based on previously carried out development work and current decommissioning planning, a number of development activities have been identified to carry out the decommissioning of the Swedish nuclear power plants in a safe and efficient manner. Development areas are largely common for the licensees and relates to waste management, licensing, logistics, resources and coordination rather than to research and technology development, although adjustments of available technologies will be required.

How conventional and radioactive waste is to be managed and disposed of during decommissioning is a prerequisite that has great importance for a decommissioning project. This entails logistic challenges but also means that alternatives for management of the waste need to be established before it's generated. The waste that is produced during decommissioning therefore needs to be characterised in detail as a basis for the planning of disposition pathways and preparation of criteria for sorting.

When it comes to clearance and management of very low-level waste, there need to be several possibilities. An example is the land-fills situated at the nuclear power plants that today are used for operational low-level waste. The material that will be cleared during decommissioning can be managed in several ways, for example by recycling the material as backfill on site or by managing certain waste at conventional waste facilities. In some cases conditional clearance would need to be applied, which means that clearance is associated with restrictions regarding how the material may be handled after it is cleared.

In order to be able to carry out decommissioning and handling of intermediate-level waste in an efficient manner, SKB and the nuclear power companies plan to use larger waste packages referred to as *tetra-mould* and *double-mould*. Before the packages can be manufactured and incorporated, a development project remains, and requirements and handling need to be verified by SKB and the nuclear power companies together.

Managing some radioactive waste in the form of large components can have advantages, among other things from a dose and efficiency perspective. Such management must be adapted so that waste in this form can be transported and disposed of. An example that has been previously studied is the handling of whole BWR reactor pressure vessels. Measures are also needed for managing PWR pressure vessels as whole and other large components.

During the entire life cycle of a nuclear facility, it is of importance, and a requirement, to have records of the waste generated in the facility. During decommissioning, the quantity of materials that will be handled will be large and registration of the waste generated needs to be reconsidered, among other things in order to ensure efficiency and that waste data can be transferred from the licensees to SKB.

Decommissioning of the nuclear facilities where SKB is the licensee holder, and thereby responsible for fulfilling requirements, is fairly far in the future. The need for developments during this RD&D period is limited. The needs that exist however relates to keeping the decommissioning plans up to date and making use during operation of experience and events relevant for decommissioning. The decommissioning plan for Clab will be revised and an update is planned to harmonise with regulations from SSM and follow the common structure developed with the nuclear power companies.

Another aspect that SKB is responsible for is taking into account the set of requirements for decommissioning when new nuclear facilities are constructed, for example in the case of Clink and the new final repositories.

5.12 Other questions

Information preservation and deep boreholes

Questions relating to the preservation of information and knowledge of the final repository and to the evolution of other concepts for final disposal, especially disposal in deep boreholes, have been constantly recurring during all years of the 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, the matter has been continually raised. SKB is therefore moving forward with the work on how to preserve documents, information and knowledge of the final repositories across generations far into the future and continues following developments in the areas of drilling and disposal in deep boreholes, see Part IV.

Future human activity

To predict what humans will do during the periods relevant for final disposal of radioactive waste, thousands or millions of years, is an impossibility. International and national work elucidates the issue of identifying relevant future human activities to include in a safety assessment. Instead of speculating on different types of human intrusion that could occur, IAEA (2011), the NEA (2012) and the ICRP (2013) recommend developing one or more stylised scenarios to demonstrate the robustness of a repository concept.

There is an international consensus that future human intrusion shall be included in the safety evaluation of the final repository. However, there is no consensus on how this can be done. Therefore IAEA in 2012 started the project Hidra (Human Intrusion in Disposal of RAdioactive waste). During Phase 1 of the project, a methodology for the handling of human intrusion in safety evaluations was suggested. SKB participated actively in Phase 1 of Hidra and will also participate actively in Phase 2, where the proposed methodology will be tested and evaluated.

Part II

Waste and final disposal

6 Low- and intermediate-level waste
7 Spent nuclear fuel
8 Canister
9 Cementitious materials
10 Buffer, backfill and closure
11 Rock
12 Surface ecosystems
13 Climate and climate-related processes

6 Low- and intermediate-level waste

This chapter describes the research and technology developments that are being conducted, started or will be concluded during this RD&D period regarding the low- and intermediate level waste. During the period, knowledge of the radionuclide inventory for SFR and SFL will be gathered. Research-oriented studies aim to provide a better process understanding of the waste and its waste matrix for the period after closure. Preliminary acceptance criteria for long-lived waste need to be formulated for processing and conditioning this waste. Transport containers and waste containers need to be further developed for the decommissioning of the nuclear facilities.

6.1 Radionuclide inventory

SKB continuously improve methods for estimation and prognosis of the radionuclide inventory in low- and intermediate level waste. This is done by updates and supplements of information in the database Triumf NG (SKB 2010f) as more detailed information is obtained. In addition, further methodology development for activity determination of difficult-to-measure nuclides is being performed in consultation with the licensees. All methodology for determination of the radionuclide inventory is associated with uncertainties. SKB has initiated efforts to inventory and quantify these uncertainties and to conduct experimental analyses in order to verify the calculation models used.

6.1.1 Reference inventory

Current situation

The compilation of the radionuclide inventory in the application for an extended SFR did not include the estimated radioactivity from decommissioning and dismantling of the facilities on the Studsvik site. The work of preparing an inventory for decommissioning waste is in progress at the respective licensees and is expected to be completed by the end of 2016.

The database of the radionuclide inventory in SFR (Triumf NG) was inventoried with respect to measured values of key nuclides cobalt-60 and cesium-137. A calculation of the total activity supplemented with estimated MDA values (MDA, minimum detectable activity) has been done for cobalt-60 and cesium-137 for each repository part, which also applies for the nuclides whose activity is determined by correlation with key nuclides.

An updated reference inventory has been calculated for long-lived waste from the Swedish nuclear power plants (Herschend 2014), which includes existing waste in interim storage facilities and forecasted waste from future operation and decommissioning of the nuclear power plants. The inventory has been determined from reported waste data and completed decommissioning studies (Anunti et al. 2013, Larsson et al. 2013, Hansson et al. 2013). The activity content has mainly been determined by calculations according to the same methodology as used for the decommissioning studies. An inventory of the control rods that are interim-stored in Clab has been carried out to permit a more detailed determination of their radionuclide contents.

For the legacy waste and waste from continued operation and decommissioning of AB SVAFO's and Studsvik Nuclear AB's facilities, a preliminary inventory has been developed for the safety evaluation for a final repository concept for SFL. The inventory is based on the reported amount of legacy waste, the prognosis for future operational waste and the measurements and calculations of activity made prior to dismantling of the research reactor in Studsvik. However, the inventory at the present time is very uncertain as the documentation concerning a large fraction of the legacy waste is inadequate.

Programme

From the planning of segmentation of reactor internals from the reactors B1 and B2 in Barsebäck and the research reactor in Studsvik it becomes obvious that SKB's existing waste registry needs to be developed to be able to receive and process the information that accompanies waste packages with segmented reactor internals. The activity content in the waste will be based on calculated activity levels for each individual segment in combination with measured dose rate. The waste registry should, therefore, be adjusted to receive information on each individual segment. Furthermore, it should be possible to link the registry to a calculation model which, based on measured dose rate and operating history, can determine the activity content.

SKB will monitor the results of sampling and sample analyses from segmentation of reactor internals at the reactors in Barsebäck and the research reactor in Studsvik in order to verify the calculation methods that are used to estimate induced activity, plus contamination on the primary system surfaces.

The inventory of control rods that are interim-stored in Clab, which has been conducted during 2015, aimed at producing a calculation model that makes it possible to consider the actual operating history of each control rod in the determination of the activity content. The model will be used to provide a more detailed description of the radionuclide inventory in SFL as well as support in the continued handling of the control rods up to deposition in the final repository.

SKB will monitor the planning of the spallation facility ESS to be able to consider the possibility of disposing waste from operation and decommissioning of the facility.

6.1.2 Method development for difficult-to-measure nuclides

Current situation

SKB has for some time calculated the amount of the difficult-to-measure nuclides molybdenum-93, technetium-99, iodine-129 and cesium-135 in operational waste with the aid of a calculation model (Lundgren 2005, 2006). In the inventory compilation prior to submission of the application for an extension of SFR, a number of improvement possibilities were identified in this calculation model, which is why an update was considered desirable.

Since 2015, the methodology for producing a prognosis for the contents of the nuclides molybdenum-93, technetium-99, iodine-129 and cesium-135 in operational waste has changed, as correlation with cobalt-60 and cesium-137 is no longer performed. Instead, the activity is predicted with the aid of the calculation model for difficult-to-measure nuclides (Lundgren 2005, 2006).

SKB has inventoried the content in the database Triumf NG with respect to reported activity from completed system decontaminations for the primary system of the reactors. Difficult-to-measure nuclides whose activity is determined by measurement of process water could be underestimated if water analyses from completed system decontaminations have not supplemented the regular analyses of these nuclides. A supplement of the database has been initiated with respect to reported released transuranic activity for completed system decontaminations, where data was previously missing. For the system decontaminations where water analyses have not included transuranics and other difficult-to-measure nuclides, an estimate of released radioactivity will be calculated based on available measuring data, the quantity of dissolved uranium in the reactor as well as vectors for fuel and activated corrosion products.

The investigations which have been done at KTH regarding the release of carbon-14 during bitumen conditioning of ion exchange resins, which is applied at Forsmarks Kraftgrupp AB (FKA), have been concluded. In addition, carbon-14 in the exhaust air from FKA's waste facility was measured for more than a year, and samples of dried ion exchange resins have been retrieved and analysed from all reactor units. With these studies, it is now established that the drying step removes large quantities of inorganic carbon-14 from ion exchange resins and to some extent also organic carbon-14, and therefore are no additional analyses planned.

The work that was described in the RD&D Programme 2013 concerning the inventory of carbon-14 activity in waste from non-nuclear activities is concluded and information on the activity content is, thereby, available (Johansson 2015).

Programme

Further development of the calculation model for the difficult-to-measure nuclides molybdenum-93, technetium-99, iodine-129 and cesium-135 in operational waste is under way. The work includes addition of additional series of measured data in order to better assess the origin of release of radioactivity and hence also the choice of parameter values. Furthermore, the calculations of the activity concentration in process water, which in today's models are based on reactor type, will be replaced with reactor specific values. In the updated model, it will also be possible to vary the age and burnup of the damaged fuel. The updating of the calculation model will also include a correlation between fuel related nuclides and the quantity of dissolved uranium as well as the amount of fissile material on the core, based on the ratio in each reactor's calculated equilibrium core. In this way, it is possible to abandon the correlation of fission products with the quantity of cesium-137, which is desirable as cesium has a different dispersion pattern in the reactor than many other fuel related nuclides.

Reported amounts of transuranics will be revised and supplemented by additional measured or calculated values to ensure a complete account of transuranic activity in operational waste.

At the compilation of the radionuclide inventory in SFR prior to the PSAR, updated uptake factors for carbon-14 on ion exchange resins will be used (Aronsson et al. 2016, Aronsson 2016a, b).

SKB is participating in an EU project (Cast, Carbon-14 Source Term) that is investigating questions concerning carbon-14, such as release rate of induced carbon-14 and speciation of organic carbon-14. Results from these investigations are expected when the project concludes in 2017.

SKB also conducts methodology development regarding the distribution of the calculated quantity of difficult-to-measure nuclides on the produced waste. The aim is to determine the distribution so that the activity is assigned to the waste which, in terms of origin and time of appearance, is in agreement with the systems and the operating period under which the activity arose. This way, the uncertainty in the distribution of difficult-to-measure nuclides between the different repository parts in SFR is reduced.

In order to improve the estimation of chlorine-36 in neutron-irradiated components, SKB is making efforts to determine the concentration of chlorine-35 in steel before activation. Steel samples (representative for reactor and core-component material) have been sent for analysis with a recently developed method based on a combination of neutron activation and accelerated mass spectrometry (Winkler et al. 2015).

In certain systems where the activity content is built up over a relatively long time, the long-lived nuclides will accumulate, while more short-lived nuclides decrease due to faster decay. Determination of difficult-to-measure nuclides in the waste from these systems cannot be made solely by means of measured short-lived activity and correlation factors since the correlation between nuclides changes over time. SKB thus identifies a need to develop methods for calculation for estimation of difficult-to-measure nuclides in systems with extended activity build-up, such as Clab's interim-storage pools and tanks with untreated ion exchange resins.

6.1.3 Uncertainties in radionuclide inventory

Current situation

At the compilation of the radionuclide inventory for the application for an extended SFR, an inventory of the uncertainties in the different methods for activity estimation was made. Uncertainties concerning the same nuclide was added and resulted in an increase per radionuclide that was evaluated in the assessment of post-closure safety.

To verify the calculation model for difficult-to-measure nuclides, a measurement program has started. Reactor water samples have been gathered during 2014 from the reactors O1 and O3 in Oskarshamn. The samples have been analysed with ICP-MS (Inductively Coupled Plasma Mass Spectrometry) for determining the concentration of nuclides technetium-99 and iodine-129 (Åkerblom 2015).

Programme

SKB intends to develop the handling of uncertainties in the radionuclide inventory, which, among other things, includes identifying distribution functions for different measurement data and to examine error propagation of the uncertainties.

Since the sample preparation and the method have proved to be suitable for measurement of technetium-99 and iodine-129 in reactor water, SKB plans to measure the reactor water samples from several reactors during the coming year in a similar manner. Results from reactor water measurements for each reactor type are desired since they would provide an overview of the calculation model reliability.

SKB, in cooperation with Studsvik Nuclear AB (SNAB), is investigating the possibility to produce a sample preparation method for measurement of molybdenum-93 in reactor water with ICP-MS. The purpose of this development work is to, ultimately, be able to carry out verified measurements also for this nuclide.

From the analyses which will be carried out within the framework of Barsebäck Kraft AB's (BKAB) sampling programme for project Hint and AB SVAFO's sampling programme for decommissioning of the research reactor in Studsvik, it will be possible to gather data for determination of the uncertainty in calculated activity (induced activity, plus contamination on system surfaces in contact with the reactor water). These data can either be used to quantify the uncertainty in calculated activity or serve as a reference data for further development of the calculation methods.

6.2 Acceptance criteria for long-lived waste

There is a need to clarify the planning premises for the handling of the long-lived waste that arises during operation and decommissioning of nuclear facilities. The planning premises constitute support for the nuclear power plants' decommissioning planning and for their handling of, for example, scrapped components to be disposed of in SFL. SKB gradually develops planning premises for the waste, as the planning of SFL progresses. As details of the repository design are finalised, it will be possible to further define the planning premises and to formulate preliminary acceptance criteria for the waste. Acceptance criteria for the waste in SFL are formulated in conjunction with the applications for licences to build and commission SFL.

Current situation

The ongoing safety evaluation for SFL aims to, among other things, formulate requirements on the waste based on the proposed repository concept. Requirements may include limitations in both nuclide and material content, as well as restrictions on in which form the waste may occur. These requirements comprise the basis for the continued development of preliminary acceptance criteria for the long-lived waste.

Programme

After the safety evaluation is completed, an inventory is made of the requirements that are imposed on the waste by the proposed repository concept. These requirements, related to the safety of the repository after closure, together with for example requirements to facilitate safe and effective transport, handling and operation, comprise a basis for formulating preliminary waste acceptance criteria. The work also includes an analysis of what will be done with the waste which in its current form does not conform to the acceptance criteria. Inventory and evaluation of suitable processing and conditioning methods will, if necessary, take place to identify possible measures related to the waste in order for it to fulfil the acceptance criteria. Such measures may for example include sorting or melting of the waste or special requirements on the waste package properties. The work will begin after the safety evaluation is completed.

6.3 Conditioning of long-lived waste

Based on the acceptance criteria for long-lived waste, technical issues on how the waste will be treated and packaged in order to fulfil the acceptance criteria need to be solved. This can for example include development of new types of containers or development of materials and methods for stabilisation of waste with cement. Stabilisation of the waste ensures that the waste does not move during transport and handling and it further aims to reduce the void in the waste package.

6.3.1 Stabilisation of waste in steel tanks

Current situation

Segmented core components from upgrades of the power plants, are today interim-stored in steel tanks at the NPPs. Moreover, additional decommissioning waste from the NPPs is also planned to be placed in steel tanks. This is currently done within the Hint project, where BKAB segments core components and reactor internals and places the segments in steel tanks.

Should the acceptance criteria for the waste require that the segments, which today are interim-stored in steel tanks, are stabilised prior to disposal in SFL, a method and a facility for stabilisation of waste in steel tanks will need to be developed.

Programme

During the latter part of the RD&D period, it may be necessary to commence development of a method and a facility for stabilisation of waste in steel tanks based on preliminary waste acceptance criteria.

6.3.2 Transloading of waste

Waste from AB SVAFO and SNAB exists in a variation of geometries. In order to facilitate safe and efficient handling of this waste, the waste may need to be placed in new waste containers.

Current situation

Waste containers in order to facilitate safe and efficient handling of waste from AB SVAFO and SNAB are developed on a conceptual level within the framework of the SFL concept study (Pettersson 2013). In order to use these containers, a transloading station needs to be developed for packing of moulds, 200-litre drums and 280-litre protective drums in containers.

Programme

During the latter part of the RD&D period, it may be necessary to begin the development of a facility for transloading of waste into new containers. The needs are governed by the preliminary waste acceptance criteria and the planning of SVAFO's facility for conditioning of waste.

6.4 Management of reactor pressure vessels and large components

Development work for disposal of whole BWR pressure vessels has been carried out during 2013–2015, so that the whole chain from dismantling to disposal in SFR is handled in an optimal way.

PWR reactor pressure vessels and internals from Ringhals NPP are planned to be stored on the facility until they are disposed of in SFL. The existing decommissioning plan for the Ågesta reactor assumes that the RPV is segmented in the facility. There may be advantages to avoiding this management and therefore, alternative management need to be analysed.

For more information on the nuclear power companies' planning and execution of decommissioning, see Part III.

Current situation

Several studies (Edelborg 2013, Haglind and Egeltun 2014, De la Gardie and Calderon 2014, Olofsgård and Baczynska 2016) that compare handling of whole and segmented BWR pressure vessels have been conducted as a part of the planning of the extension of SFR. A preliminary waste type description for whole BWR pressure vessels has also been developed (SKB 2014e).

Parts of the studies are also relevant to handling of PWR reactor pressure vessels. Segmentation of PWR reactor pressure vessels in the nuclear power plant or on another site has been studied by SKB together with SNAB and Westinghouse. A comparative analysis between different handling alternatives for PWR reactor pressure vessels is performed jointly by SKB and Vattenfall. The analysis aims to shed light on the entire handling chain, including dismantling, transport and final disposal, with respect to radiation safety, technology, the environment and costs and is expected to be presented in the end of 2016 or the beginning of 2017.

The nuclear power companies have in 2015 reported to SKB what other large components may be useful to adapt the waste system to. This refers mainly to adjustments in the extended SFR. Components described are for example steam generators, pressure containers and reactor pressure vessel lids from PWRs. Based on this material, SKB has initiated studies to specify what consequences this has on the waste system or if it is possible to handle in existing plans.

Programme

The whole BWR reactor pressure vessels are intended to be deposited in the extended SFR, and up to the start of the detailed design of the SFR extension, technology development takes place in order to be able to handle the RPVs. This includes, among other things, how set-up, internal grouting, grouting and closure will be done, see Section 9.2.2.

Within this technology development, other large components are also studied to evaluate the advantages and disadvantages of depositing these whole.

Results in the completed analysis for PWR reactor pressure vessels will be used as supporting information in the decommissioning projects for the reactors in Ringhals and Ågesta, as well as for SKB's planning for the design of SFL and SFR. The final choice of management and handling is also dependent on the development of waste acceptance criteria.

6.5 Waste containers and waste transport containers

In order to be able to carry out the decommissioning of the nuclear facilities optimally according to what is described in Part III, development work needs to be carried out within the area of waste containers and waste transport containers for the low and intermediate-level waste.

6.5.1 Waste containers for waste from AB SVAFO and Studsvik Nuclear AB

Based on the acceptance criteria, technical issues need to be solved concerning how waste from AB SVAFO and Studsvik Nuclear AB, SNAB, will be treated and packaged in order to fulfil the acceptance criteria. This can for example include development of new types of containers.

Current situation

Within the SFL concept study, a basis for waste containers has been formulated for safe and efficient handling of the waste which today is placed in moulds, 200-litre drums and 280-litre protective drums (Pettersson 2013).

AB SVAFO is conducting a feasibility study until 2017 to inventory different methods to retrieve, characterise, treat and condition low and intermediate-level waste. The purpose of the feasibility study is to consider the possibilities of designing and building a facility for handling and conditioning of AB SVAFO's waste prior to deposition in the final repository. At such a facility, the waste could be conditioned to the desired containers. If AB SVAFO chooses to treat and condition their waste, the need for new waste containers like "overpack" will possibly change.

Programme

During the latter part of the RD&D period, development of waste containers may continue based on preliminary waste acceptance criteria.

6.5.2 Waste containers for decommissioning waste

Current situation

During 2014, preparation work was carried out where the needs and limitations were discussed for realising the use of the *tetra-mould* as a waste container for dismantling and demolition of nuclear facilities. Based on the preparation work, SKB decided to develop a design basis for the *tetra-mould* and also develop the *double-mould*, see Figure 6-1, and in addition to modify waste transport containers.

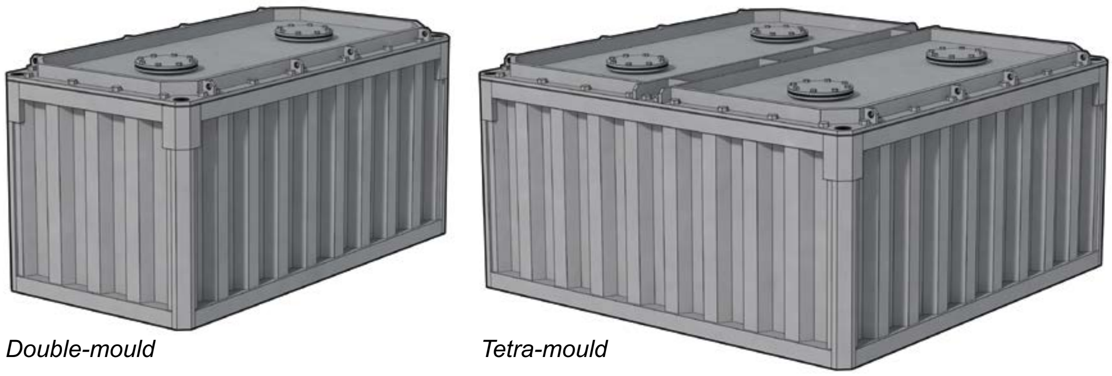


Figure 6-1. *Illustration of the two future waste containers double-mould and tetra-mould.*

The dimensions of the *tetra-mould* and *double-mould* correspond to four or two steel moulds. Steel moulds are a type of waste container used for short-lived intermediate-level waste. The work was done during 2015 and provides clearer conditions for the NPPs' decommissioning planning and the technology development for the extension of SFR.

Programme

SKB, together with the nuclear power companies, has decided that it is justified to progressively continue developing the waste containers *tetra-mould* and *double-mould* to permit a more efficient handling. In order to realise and manufacture the containers, manufacturing data need to be prepared and the degree of detail needs to be developed in certain areas, which has been proved to be necessary in the previous work.

Areas which, among other things, need to be developed are the containers' weight limits. For example, a *double-mould* that may weigh up to 20 tonnes, instead of the so far assumed 10 tonnes, entail that the waste container capacity would be utilised more efficiently. Furthermore, the procedures to lift the waste container need to be reconsidered. Another important requirement is that the container interacts with the caissons in 2BMA in order to be able to withstand the water pressure after closure.

During the coming years, SKB will pursue efforts together with the nuclear power companies so that the data can be used for procurement and fabrication of waste containers. There will then be a possibility for joint procurements depending on the nuclear power companies' needs.

6.5.3 Transport containers for new waste containers

Transport containers for long-lived waste aim to permit safe and effective transport of long-lived waste from the nuclear power plants, SNAB and other interim storage facilities, 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. Waste containers are in turn dependent on acceptance criteria for the waste.

Current situation

A new waste transport container, ATB 1T, for transport of steel tanks is developed in cooperation with American 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 the new waste containers. If new waste containers are developed, the corresponding transport containers will also need to be developed. Pending preliminary acceptance criteria for the long-lived waste, no development of new waste transport containers is conducted.

6.6 Degradation products from organic materials and their interactions with radionuclides

In SFR, sorption of radionuclides on cement minerals is one of the most important processes that delay the release of radionuclides from the repository. The waste contains a certain amount of organic material whose degradation products can interact with radionuclides in the same way as complexing agents. These substances can to a certain degree effect the sorption of some radionucledes. Organic complexing agents can also effect the solubility of radionuclides by making them more soluable and available for transport. Additives in cement may possibly be broken down to organic compounds that could also affect the sorption of radionuclides. Three programmes are planned in order to gain greater knowledge in this area.

6.6.1 Degradation products from cellulose

Current situation

Organic material exists in different waste types in SFR. A large portion of the organic material consists of cellulose which, under alkaline conditions, can be broken down 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 and thereby reduce the accessibility for complexation with radionuclides (Ochs et al. 2014). For the system cement-plutonium-ISA, few and not reliable data are available on how the presence of ISA in different concentrations affects the ability of plutonium to sorb to cement minerals. So far, the sorption reduction factor, which has been attributed to plutonium in the presence of varying concentrations of ISA, has been uncertain and SKB has in SR-PSU chosen a conservative sorption reduction factor for plutonium(III)/plutonium(IV). SKB started a study in 2014 to gain a deeper understanding of the above mentioned system.

Programme

Experimental studies and modelling of the results are under way with the aim to deepen the understanding of the system cement-plutonium(III)/plutonium (IV)-ISA. The ongoing study will likely lead to new, less uncertain, sorption reduction factors for plutonium being implemented in future safety assessments for SFR and SFL.

6.6.2 Degradation products from filter aids

Current situation

An example of an often 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 that can affect the sorption and/or the solubility of radionuclides (Duro et al. 2012). In the most recent completed study, the impact on nickel(II) and europium(III) under repository-like conditions was investigated (Tasdigh 2015). The study showed that degradation products from 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.

Programme

Further studies of filter aids are planned in a project named Cori (Cement Organics Radionuclide Interaction) through the technology platform IGD-TP for closer European cooperation on nuclear waste disposal. SKB is participating in the preparations prior to the application.

6.6.3 Degradation products from cement additives

Current situation

In the SFR extension, 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. These may degrade, under the conditions prevailing in the repository after closure (high pH, low Eh), and form highly soluble organic compounds that can influence the sorption and/or the solubility of radionuclides. Before the start of the construction of the extension of SFR, it is therefore important that SKB has acquired an understanding of the effect of these organic polymers so that SKB can stipulate requirements on which concrete additives the contractors are allowed to use in the concrete to avoid possible impact of the safety functions which apply for the repository after closure. Programme

In 2016, SKB has initiated a cooperation with KTH where the degradation of a proposed additive family (polycarboxylate based superplasticiser) will be studied. The study aims to determine if and in such cases how quickly the polymer degrades and which products are formed. Any influence of degradation products on sorption will be studied.

6.7 Corrosion of aluminium and zinc

Anaerobic corrosion of aluminium and zinc in the waste deposited in SFR can in a short time generate large quantities of hydrogen gas. Provided that this 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. The rate at which hydrogen gas is formed, is determined by how quickly metals corrode in the repository environment. It is therefore important to understand the corrosion processes for aluminium and zinc in order to be able to assign realistic corrosion rates in the post-closure safety assessment of SFR, in order to assess the potential damage mechanisms in the concrete structures on one hand, and to design the gas-evacuation systems on the other.

Current situation

SKB has adopted a pessimistic corrosion rate for aluminium and zinc of 1 mm/yr. This, together with the assumptions that all aluminium and zinc consist of five-millimetre-thick plates which corrode from two sides and that the corrosion begins immediately after closure (SKB 2015b), results in all waste consisting of aluminium and zinc corroding during the first two and a half years after closure. Hydrogen gas development from corrosion of aluminium and zinc in the repository becomes, with these assumptions, intense and of short duration.

The experiment Concrete and Clay, which is being conducted in the Äspö HRL (Mårtensson 2015), has yielded data that provides a better basis for the description of corrosion of aluminium and zinc in the repository environment. The project started in 2010, and even though the main purpose is to qualitatively create an understanding of how degradation products are spread in a cement matrix, and to what extent new minerals are formed in these processes, it has also yielded new information on the degradation rate of the metallic waste.

In 2015, samples containing aluminium and zinc have been retrieved and analysed (Kalinowski 2015). The study showed that the corrosion of aluminium and zinc, during the five years which passed since the experiment was installed, was very low, roughly 50 μm/yr, which should be compared with previous assumptions on 1 mm/yr. It is also possible that a large proportion of the observed corrosion occurred in connection with the fabrication of the samples via a reaction with the wet concrete, the long-term corrosion rate is possibly lower than the calculated mean value for these five years.

Programme

SKB has, together with researchers at KTH, initiated a study of corrosion of aluminium and zinc in concrete exposed to oxygen-free groundwater. During the project, samples will continuously be retrieved and analysed after different exposure times, corrosion products will be characterised spectroscopically and corrosion rates will be determined with gravimetric analysis. The study, which is expected to continue for at least two years, is expected to provide a deeper understanding of how hydrogen-producing corrosion of aluminium and zinc takes place over time in the repository environment. The time for the conclusion of the project will be determined as knowledge of the corrosion process increases.

6.8 Microbial gas production

Current situation

Another process that potentially may cause harmful gas build-up is microbial degradation of organic materials, such as cellulose. By ensuring unfavourable conditions in SFR for microbial degradation, the possible gas formation can be limited. In SR-PSU, requirements are formulated on high pH in the repository, one of the purposes of high pH is precisely to limit microbial activity.

Programme

Further studies will increase the understanding and predict the future pH development in different parts of the repository and also how different pH conditions will affect gas production through methanogenesis.

Within the EU MIND project, where SKB is both coordinator and member of the endusers' review board, remaining questions from SR-PSU, are further investigated. Results from the research will be available, to some extent, prior to the PSAR. Within the project, for example, experimental studies on how pH affects the microbal mediated methane formation. Modelling of microbial processes that are relevant for low and intermediate-level waste is also a part of the project. The public reports which have been produced and will be delivered from the project MIND are described on the project's website, www.mind15.eu. The project will be concluded in 2019 with a synthesis of the research that has been conducted regarding biological degradation of waste and waste forms. Knowledge concerning the quantity of methane gas that potentially might be formed is needed to better understand and model gas transport and uptake processes of gas in the surface system, see Section 12.1.

6.9 Swelling waste – bitumen-solidified ion exchange resins

Swelling waste may affect the integrity of the concrete barrier and thus affect the transport of radionuclides from SFR. It is therefore important to know if and how large the swelling pressure from the waste can be.

Current situation

Swelling waste has been studied mainly during the 1980s and 1990s. Results from these studies have been used for modelling the effects of swelling waste on concrete barriers in the silo and 1BMA (von Schenck and Bultmark 2014). SKB has, after this modelling, performed swelling tests on dried ion exchange resins to improve the assumptions of the swelling pressure that conceivably could arise in waste types F.17 and F.18 (Andersson et al. 2014). The results show that the swelling pressure will not affect the steel mould integrity.

Programme

Swelling of dried ion exchange resins of the type used in FKA has been studied by Andersson et al. (2014). In this study, bitumen was not used as a solidification material, which is the case for the waste containing ion exchange resins from FKA that is disposed of in SFR. Experiments are planned at the Äspö HRL where the possible swelling of a bitumen-solidified ion exchange resin will be studied. In these experiments, the leaching rate of anions and cations from a bituminised waste form will also be investigated. SKB has made a conservative assumption that all radionuclides embedded in bitumen are available for transport as soon as the repository is resaturated with water, as there are no data on leaching rates from such a waste form in the literature. The experiments aim is also to reduce the pessimism in the PSAR prior to the extension of SFR.

7 Spent nuclear fuel

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

The spent fuel generates heat even after it has been removed from the reactor (decay heat). Because of the decay heat, the fuel must be cooled to avoid overheating. The amount of decay heat depends on how long the fuel has decayed and its burnup that is to what extent it has been irradiated on the core.. 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 commisioned. The reason for these changes is to achieve as efficient utilisation of the fuel as possible. The nuclear power companies plan to further increase fuel burnup.

In the planning work it is important to clarify the consequences of higher burnup for all parts of the KBS-3 system. According to the planning premises (see Section 1.1.4), the total amount of spent nuclear fuel to be disposed of will comprise about 6000 canisters. One canister contains about 2 tonnes of fuel. The amount of spent nuclear fuel is given as the quantity of uranium that was originally present in the fuel. In addition to all spent fuel from today's Swedish nuclear power plants, the amount of spent nuclear fuel to be deposited in the Spent Fuel Repository also includes fuel from the Ågesta reactor, fuel residues from testing programmes at Studsvik, and MOX fuel (mixed oxide fuel). These fuel types comprise a very small fraction of the total quantity. 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. In the facility, 23 tonnes of MOX fuel is stored, which has been obtained from Germany in exchange for the fuel that at an early stage of the Swedish nuclear power programme was sent to La Hague in France for reprocessing, where uranium, plutonium and waste products are separated.

Today, SKB approves new fuel types introduced at the Swedish nuclear power plants. The verification process includes all parts of SKB's activities that have a bearing on fuel, such as transportation systems, interim storage, encapsulation plant and final repository. The compatibility of the fuel types with the KBS-3 system is verified.

SKB's programme for handling of fuel comprises several parts, from requirements of information on the properties of the fuel before it is used in the fuel cycle, to formulating a programme for safeguards that is internationally approved. During the coming RD&D period, development work will be carried out relating to decay heat determination and criticality verifications for all facilities and shipments included in the KBS-3 system. Development of safeguards is an area where SKB works in close cooperation with international bodies.

If a canister is breached and water enters, the properties of the fuel are crucial for determining if and when radionuclides might be released. The results of the safety assessment SR-Site (SKB 2011b, Section 13.5.11) show that the rate at which radionuclides are released from the different parts of the fuel is of crucial importance for the post-closure safety of the Spent Fuel Repository. An in-depth understanding of the mechanism for dissolution of the fuel matrix is required to support the experimental results and thereby reduce the uncertainties in future safety assessments, a sentiment which was also expressed by SSM both within the framework of the licensing process and the review of the RD&D programme 2013. There are also some uncertainties concerning speciation and solubilities of released radionuclides. The research programme for fuel dissolution and radionuclide chemistry is described in Section 7.7.

7.1 Non-regular fuels

SKB has decided that non-regular fuel should be analysed and treated separately from regular fuel. Non-regular fuels refer to, for example, fuel from the Ågesta reactor, spent fuel from testing programmes at Studsvik, MOX fuel and damaged fuel. Non-regular fuels are handled in one or more separate processes where special solutions are devised, analysed and carried out.

Current situation

Handling and clearing the Swedish nuclear power plants of damaged fuel has begun in a special project. Some of the damaged fuel has been treated with a specific method, the so-called Studsvik method, and will be transferred into a form that does not require further treatment prior to final disposal. The method involves transporting the damaged fuel rods to a hot cell, where the rods are sawn into one-meter lengths, dried according to international protocol and thereafter encapsulated in special cases, which are gas- and water-tight. Other methods, with the same qualitative final results, will also be used in the project.

Programme

When the nuclear power plants are emptied of damaged fuel, a similar treatment of the damaged untreated fuel in Clab will take place. A detailed plan for this will be established for the later part of the RD&D period (2019–2022). Work is ongoing, together with the nuclear industry, to develop methods that rationally can manage the smaller quantities of damaged fuel, which are believed to be generated after the plants are emptied of existing damaged fuel.

7.2 The ageing of fuel

At Clab a number of individual fuels are continuously inspected to follow changes in the properties of the fuel during interim storage. The ageing of fuel also has importance for the encapsulation plant, as the handling of the fuel in Clink must be able to be carried out in a predetermined way . Therefore, the development of fuels in Clab and findings from international research are followed. When fuels in Clab are handled in pools, verification of the fuel is relatively simple, unlike dry storage in canisters where such inspection is not possible in a simple and continuous manner.

Current situation

Deficiencies that have been handled as a result of the analysis of ageing management are:

- Inspection of fuel.
- Long-term inspection programmes for fuel.
- Automation of fuel verification.

Inspection is performed to ensure that the fuel entering the facility is in manageable condition. Any obstacle for handling is registered in the fuel database.

A long-term inspection programme has been developed to determine which fuels should be monitored further. A wide selection of fuel types has been selected. If unexpected results are obtained through the inspections, these are to be handled with the ageing programme methodology.

To reduce the risk of handling mishaps in connection with inspection, the sequence the handling machine uses during the fuel verification is automated..

Programme

Monitoring of practices around the world is performed to stay updated on the latest findings regarding degradation (ageing) of fuel in a wet storage environment. This includes research results in the area as well as visits and experience exchanges with similar facilities. SKB is participating in different international forums where ageing issues are dealt with, for example IAEA. SKB has joined the OECD-NEA project Scip III (Studsvik Cladding Integrity Project) and intends to participate actively during this period. SKB receives information, via the international nuclear community, on different events where incidents have occurred during fuels handling.

7.3 Decay heat and fuel measurement

At deposition, a number of fuel parameters need to be known. Decay heat and criticality (neutron multiplicity) are fundamental since they are limited by the safety assessment. The radionuclide inventory needs to be known both for the assessment of post-closure safety and for the safeguards. For safeguards, information is needed concerning fuel identity and removed rods.

Current situation

SKB is pursuing an extensive programme for the development of methods for determining the properties of the fuel with sufficient accuracy. This is carried out in broad international collaboration with participants from countries in Europe and Asia and also from the USA. The basis is to utilise both non-destructive and destructive techniques in order to sufficiently understand the fuel, develop suitable measurement methods and determination codes as well as to test them on spent nuclear fuel (destructive techniques are used only to acquire knowledge on fuel and independently link results with non-destructive measurements; the goal is to only use non-destructive technologies in the encapsulation plant). The goal of the work is ultimately to obtain a measurement system that can meet the fuel characterisation. The requirements on such a system, however, depend on how much other knowledge is available on the spent fuel, for example operating histories, which are also studied, see Section 7.4.

In order to ensure that a measurement system that can handle a number of requirements is developed, a number of aims have been established for the systems. The goal is that the measurement system will be permanent, complete (measure all fuel assemblies) and robust. It should also yield unambiguous results, have low uncertainty, and a high through-put. Complexity in the measurement system's build-up and analysis of its signals is, in principle, acceptable as long as the determination yields coherent results and the system has high through-flow capacity.

This differs in an essential manner from existing measurement techniques, which are often devised for safeguard purposes. These are typically mobile, conduct sampling (non-complete), are developed for use in the field and have low through-flow capacity.

Therefore, development of instruments and methods is required. 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 establish the determinations of above all (but not only) decay heat. Furthermore, cooperation will take place with code developers, in order to develop and update these, such as Scale (Origen) from Oak Ridge National Laboratory.

SKB has since the RD&D programme 2013 carried out measurements of 50 fuel assemblies in Clab (25 PWR and 25 BWR fuel, which are called SKB-50), within the framework of a broad international cooperation, where the United States Department of Energy (through for example Los Alamos National Laboratory, Oak Ridge National Laboratory and Lawrence Livermore National Laboratory) and Euratom constitute the foundation. In the cooperation there have also been participants from Japan, South Korea and Belgium as well as Sweden, for example through Uppsala University. Among other things, all 50 fuel assemblies have been measured by gamma spectroscopic methods and some with calorimetric and neutron detection methods. In Favalli et al. (2016) the determination of burnup, initial enrichment and decay time based on gamma measurements have been carried out and developed for SKB-50.

Programme

During this RD&D-period the measurements will be completed with all the planned measurement methods on the 50 assemblies. A number of newly-developed methods will be tested on a part of the fuels in Clab. At the same time data is analysed with the purpose of achieving the above-mentioned goal.

7.4 Fuel information

Current situation

When the KBS-3 system is commissioned, a database with values on all the fuel properties that are important for the fuel handling and for the safety analysis report is needed. As a first step in this work, SKB is conducting a project on fuel information in which the purpose is to clarify which fuel information is needed for Clab, Clink, the transportation system, as well as the Spent Fuel Repository's operation, the safety analysis report and design. The project also includes a status analysis in order to identify and document existing fuel information and where this is stored.

Programme

In the coming years further studies will be 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.

An important task for SKB and the nuclear power plants during the next three years is to prepare a plan to ensure that the fuel information needed for management and final disposal in the KBS-3 system is not lost. This is particularly important for future decommissioning of nuclear facilities.

7.5 Criticality

The overall goal of SKB's programme regarding deepened and broadened criticality analysis is that criticality analyses for all of SKB's present and future facilities should be carried out according to the same principles for burnup credit and BA-credit (BA, burnable absorbers). In the applications for Clink and the Spent Fuel Repository, credit has been taken in the criticality analyses for the decrease in reactivity that occurs when the fuel is irradiated in the reactor and burnup increases (burnup credit). In the case of Clab, burnup credit is not taken today for PWRs or BWRs, but BA credit is taken for BWRs. This means that the fact that all fresh BWR fuel contains a reactivity-reducing substance (BA), mainly gadolinium-155, is taken into account. The BA-insert entails that the fuel has lower reactivity during the first part of its burnup, when it would otherwise be at its highest reactivity. When the fuel has been in the reactor a couple of years, the BA will be burned out and the BA-fuel will have roughly the same reactivity as an equally enriched fuel without BA. Development of the methodology and validation of calculation programs shall comply with modern internationally accepted standards and requirements.

Furthermore, SKB will ensure that it has the necessary competence regarding criticality.

Current situation

During the previous period, an implementation was carried out of the method used by SKB for burnup credit in the applications for the Spent Fuel Repository and which follows a method developed by Oak Ridge National Laboratory in the USA.

On a more detailed level, the following have been done:

- Validation of the calculation tools SKB uses for criticality analysis.
- Description of the methodology and the principles SKB uses for burnup credit.
- Update of the criticality analysis for the encapsulation part of Clink.
- Update of the criticality analysis for the Spent Fuel Repository.
- Ensuring that the competence and the tools needed to perform criticality analyses for Clab, Clink and the Spent Fuel Repository are available within the Vattenfall Group.
- Update of the criticality analyses for Clab with modern calculation tools and for the fuel types that are most reactive today.

Programme

Important developments for this RD&D period are:

- Industrialisation of the development that has taken place during the preceding period, i.e. to develop a common data environment with Vattenfall Nuclear Fuel (VNF), determine procedures for verification reports, determine file structures and name conventions, devise more efficient procedures and ensure that they meet extended requirements as a result of burnup credit.
- Update and validation of new version of the computer code Scale.
- Ensuring that the requirements made on the basis of the criticality analysis, for example concerning cavities in the insert, can be verified by the inspection systems of the final repository system.
- Preparation of a strategy to deal with the fuel bundle that does not meet the canister's burnup requirements.
- Continuation of the work with the consequences of geometry change inside the canister in the long term.
- Continuation of dialogue with SSM concerning acceptance requirements for safety against criticality.

7.6 Safeguards

Safeguards enable regulatory authorities and inspection bodies to ensure that nuclear material is not diverted. SKB's facilities must comply with the requirements that are made on safeguards by both Swedish regulatory authorities and international inspection bodies. This means that there must be an administrative system for accounting for nuclear material and where it is located, plus technical systems for inspection and supervision that nuclear material is not diverted.

In the case of encapsulated fuel, the safeguards system will contain information on the individual canisters' content of nuclear material, which fuel assemblies the canisters contain, when the fuel was encapsulated, transported and arrived at the Spent Fuel Repository, where the canisters are deposited, and the total quantity of nuclear material in the repository. In other words, it must be possible to identify individual canisters and their content.

In the case of new nuclear facilities, safeguards must be taken into account in the design stage so that inspection and supervision are facilitated. An important component of the nuclear safeguards system in the Spent Fuel Repository is being able to verify that the facility has been built in accordance with approved drawings. This is done so that the inspection bodies can ensure that there are no routes out of the facility that have not been indicated and that there are no areas where activities other than those indicated are carried out.

Steps for inspection

The spent fuel's content of nuclear material is dependent on its enrichment, burnup and decay time. The content may be calculated either based on information from the fuel's operating history or based on measurements of gamma and neutron radiation in combination with calorimetric measurements. The available methods can provide sufficiently accurate information on how much nuclear material the fuel contains, in order to satisfy the requirement on safeguards. To guarantee the content of nuclear material in each canister, the fuel will, after it has undergone measurements in the verification position of the encapsulation plant, be positioned so that its identity cannot be mistaken with unverified fuel. After handling, drying and emplacing the fuel in the canister, the identity and position of the fuel is verified by optical equipment. The identity of the canister is verified at the same time so that there can be no doubt about which nuclear fuel has been placed in each canister. This step, also called "rebatching", is critical in the handling and it must therefore be carried out in such a way that no confusions can occur. After that the canister has been closed, it is 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 a seal and other equipment to ensure that the canister that is received in the final repository has not been tampered with and that no nuclear material has been diverted. It is thus essential that there is constantly a continuous knowledge regarding the nuclear material content and its position (continuity of knowledge (CoK)). This is ensured for example through the use of seals (mechanical and electronic).

In the Spent Fuel Repository, seals are inspected. The canister identity is verified in the final repository's receiving station when the canister is transloaded to the deposition machine to make sure it agrees with the consigned canister identity. The final repository's openings are under surveillance to make sure that no nuclear material can be diverted.

Current situation

SKB has initiated and started a project together with SSM, the European Commission/Euratom and the IAEA to introduce safeguards in the design of the encapsulation plant. In this project two main areas in need of technology development have been identified: Verification of the fuel and verification of the canister. SKB has decided to carry out decay heat measurements and fuel verification, and a collaboration with the European Commission/Euratom, United States Department of Energy (U.S. DOE) and others has been initiated and partly carried out to determine whether a common measurement for verification of fuel can be performed.

Programme

There is a need for similar studies for canister verification, and work is being pursued internationally in this area. SKB has initiated a project where the European Commission Joint Research Centre, JRC, in Ispra now tries to develop a technique to tag the identity of the canister on the inside so that it can be identified from the outside with the aid of ultrasonic testing. The requirement is that the tagging must have a minimal impact on the integrity of the canister, it should be authentic, i.e. not feasible to change, permanent and unique.

In cooperation with JRC Ispra, the use of a seal for canister transport will also be developed so that handling, inspection and verification of the seal can take place in a safe, efficient and reliable way.

The requirements on safeguards for the different steps in the handling of spent fuel and sealed canisters are general and comprehensive. The overall requirements come from the IAEA, which sets the framework. The European Commission/ Euratom, also imposes requirements on safeguards according to the treaty Sweden has signed. This means that it is important that SKB's development of methods and equipment for handling, verification and logistics takes place in cooperation with the IAEA and the European Commission/ Euratom. One step in this cooperation is the preparation of a draft BTC (Basic Technical Characteristics) document. The first versions of the draft BTC for the encapsulation plant and for the Spent Fuel Repository have been submitted to the European Commission, Euratom for comments. The updates and development of the draft BTC will take place for a long time to come and the final version will not be adopted until trial operation.

Methodology and verification of safeguards are being planned for the coming RD & D period and will be coordinated with the aforementioned decay heat project.

7.7 Fuel dissolution and radionuclide chemistry

This section describes the research programme for fuel dissolution in a repository environment. 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, such as the iron insert and the metal parts of the fuel, which affects the chemical environment and thereby the speciation of dissolved radionuclides inside the canister. The anoxic corrosion of iron forms hydrogen gas, which has proved to counteract oxidative dissolution of the fuel matrix; this is called the hydrogen effect.

7.7.1 Fuel dissolution

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 can 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 investigate how these parameters affect fuel dissolution, leaching experiments with spent fuel have been performed. The results of these are described below.

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. Since corrosion of control rods in scenarios with early canister failure has proven to provide a non-negligible contribution to the radiological risk in the safety assessment, a study to investigate corrosion of control rods has also been carried out.

Current situation

In order to gain a better understanding of the processes that take place during fuel dissolution, experiments are often performed with synthetised uranium dioxide with additives of elements similar to fission products; this material is called Simfuel. The fission products, present in the spent fuel, affect uranium dioxide chemical properties and its tendency to oxidise. Exactly how this happens and how significant this effect is needs to be further investigated. Therefore, studies aimed at investigating the difference between Simfuel and uranium dioxide (UO_2) have been performed, which show that Simfuel has lower redox reactivity than UO_2 (Sundin et al. 2013, Lousada et al. 2013a). The most important oxidant in a repository environment is expected to be radiolytically produced hydrogen peroxide (H_2O_2). The study has shown that H_2O_2 is, to some extent, decomposed catalytically on the solid uranium dioxide surface and that this is much more efficient on Simfuel compared with pure UO_2. Thus, pure UO_2 is more prone to oxidisation by H_2O_2 than Simfuel (Lousada et al. 2013a). Further studies with different types of metal oxide surfaces have shown that the mechanism for catalytic decomposition of H_2O_2 takes place through formation of OH radicals (hydroxyl radicals). Different oxides show different behaviour depending on the type of metal ion, which also has an effect on the kinetic parameters for segmentation of H_2O_2 (Lousada et al. 2013b).

Catalytic decomposition of H_2O_2 on the surface of spent nuclear fuel is a process that could probably reduce the oxidative effect of radiolysis in the repository environment. In an attempt to further understand and describe the mechanism that lies behind the decomposition of H_2O_2 on the surface of Simfuel, studies have been carried out to shed light on the different steps in this process. An important step relates to the reactions involving the OH radicals connected to the surface. The OH radicals connected to the surface that are formed as an intermediate step when H_2O_2 is fragmented catalytically on metal oxide surfaces can be detected with a method described by Yang and Jonsson (2014). With the aid of this method, an overall reaction mechanism and a kinetic model have been developed that describe the catalytic decomposition of H_2O_2 on ZrO_2 (zirconium oxide) (Yang and Jonsson 2015). Further understanding of the mechanism has emerged through a combination of experiments and calculations, which show that ZrO_2 surface catalyses the production of H_2O_2 from hydrogen (H_2) and oxygen (O_2) dissolved in water. This is followed by a decomposition of H_2O_2 into OH radicals on the ZrO_2 surface (Barreiro Fidalgo et al. 2016).

The surface reactivity and catalytic properties of the fuel are linked to the surface structure and electrochemistry. Results from Raman spectroscopy and XPS (X-ray photo electron spectroscopy) of yttrium and palladium doped uranium dioxide electrodes have revealed the presence of different crystallographic domains. Studies indicate anodic oxidation of non-stoichiometric domains in these materials, which are more electrochemically reactive than Simfuel in solutions with H_2O_2. These more reactive domains are expected to occur predominantly at the grain boundaries in spent fuel (Razdan et al. 2014).

The question of how old UO_2 is affected by alpha radiolysis from UO_2 surface has been shown through a review of existing information on uranium ore in Cigar Lake. In the 1.3 billion year old uranium ore the uranium still remains mainly in reduced form as uraninite. This indicates that no outer oxidants reached the ore through the surrounding clay and that the radiolytic oxidants have not caused a thorough oxidation. With today's knowledge concerning the effect of hydrogen on oxidants, it can be concluded that oxidants were probably neutralised at the surface of the uraninite through a recombination between radiolytically-produced reductants and oxidants (Bruno and Spahiu 2014). Radiolytically-produced oxidants can also be affected by the surrounding material. An experiment performed to investigate the effect of bentonite on oxidative dissolution of UO_2 shows that the bentonite can delay or partially inhibit the release of uranium to the environment, mainly due to the fact that the oxidants are partly consumed by their reaction with the bentonite (Barreiro Fidalgo et al. 2014).

In the EU project REDUPP a test was carried out with three different groundwaters with different salinities and different uranium dioxide samples. Results from the test with freshwater showed, in contrast to previous results, some effect of alpha radiation (Ollila et al. 2013). The effects of the three different groundwaters show some variation, but they all result all in a slightly elevated dissolution rate compared with previous experiments using synthetic groundwater, and all show that dissolution takes place in conjunction with precipitation and/or sorption. It is noteworthy that the lowest dissolution rates were measured for the most saline groundwater, which indicates that the high bromide content does not affect the dissolution rate negatively (Evins et al. 2014).

The EU project "First Nuclides" was conducted for the purpose of investigating the effect of irradiation history and burnup of the fraction of quickly released nuclides. The results showed that after some months the release rate of both iodine and cesium decreased. Certain elements, such as strontium, uranium and technetium, exhibited variation depending on the redox conditions in the experiments (Kienzler and Lemmens 2015). From the new dataset that the project presented, it can be deduced that the linear heat rate of the fuel influences the fission gas release (FGR), and thereby the fraction of quickly released nuclides, more than burnup. Within the project "First Nuclides", the effect of different additives in the fuel and the effect of varying sample preparation were examined. Results were generally as expected, with a fraction of the released iodine and cesium lower than the FGR. The doped fuel (i.e. with added chromium and aluminium) showed a lower release of iodine and cesium than standard fuel (Roth et al. 2013, 2014, Roth 2015).

Studies with a focus on selenium-79 were also carried out within the framework of the project "First Nuclides", and the results show that only a minor portion of the selenium-79 inventory (0.5 to 0.8 percent) is released after one year of leaching. Results from both X-ray spectroscopy and thermodynamics give evidence that a considerable portion of selenium-79 is imperviously bound to the fuel matrix, which can explain why the rapidly released fraction is so low (Curti et al. 2015).

An attempt to investigate the dissolution of high-burnup standard fuel was carried out in Studsvik with American fuel with a burnup of up to about 75 MWd/kgU. Results were difficult to interpret, which was probably due to the impact of colloids. The undesirable colloid formation was probably a result of sample preparation, above all the dry milling procedure. Therefore, a new study has been initiated with a Swedish high-burnup fuel (65 MWd/kgU) where the sample instead consists of fragments that are not milled. The leaching experiment was carried out in the presence of hydrogen in order to simulate the environment in a damaged canister. The initial results are in line with what was expected (Puranen et al. 2016).

The control rods that will be disposed of together with the fuel in the Spent Fuel Repository contain an alloy of silver, indium and cadmium. So far, no relevant leaching data have been available for this material, which therefore had to be handled with a high degree of pessimism in the safety assessment. Now there is available data from a leaching study with both unirradiated and irradiated control rods. The unirradiated rods leached during both oxidising and reducing conditions, which clearly demonstrated the effect of oxidation on all three constituent metals. During initially reducing conditions, no measurable dissolution of silveris observed , and in spite of the water radiolysis that occurred in the test with irradiated control rods, the results indicate that the inherent radiation of the alloy is not sufficient to cause significant dissolution of silver from the material (Roth et al. 2015).

Programme

Mechanistic studies aimed at improving our understanding of radiation-induced fuel dissolution will continue in cooperation with KTH. The continued studies will focus on the processes that take place at the fuel surface. These studies, which will take place within the framework of a doctoral project, are expected to shed light on the impact that fission products in the fuel have on radiation-induced dissolution of spent fuel. Furthermore, the effects of changes of the solid phase and matrix composition on the kinetics and mechanisms of radiation-induced dissolution of spent fuel will be studied within the framework of a doctoral project. In particular, uranium dioxide doped with gadolinium, neodym and yttrium will be studied, but also the effects of varying grain sizes can be included, to shed light on the effects of additives and fission products on uranium dioxide reactivity.

Experiments with spent fuel will continue in cooperation with Studsvik Nuclear AB. There are planned experiments for the next few years with high-burnup fuel in autoclave, for establishing the matrix dissolution rate under reducing conditions. Furthermore, a number of experiments with fuel in tight glass vials are planned. These experiments involve also fuel interaction with iron and iron minerals, and they are planned to last several years.

During this RD&D period, an application for a new EU project will be submitted, where the research programme focuses on studying the effects of chromium and aluminium and possibly other additives on the matrix dissolution. Furthermore, leaching studies are planned of MOX fuel. These studies should be supplemented with experiments with model systems, i.e. synthetic materials which are designed and leached in accordance with the project's strategy, and taking into account coupled modelling studies. Finally, there are studies planned that will be address the issues relating to the IRF, which could not be answered during project "First Nuclides".

7.7.2 Radionuclide speciation and solubilities

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 transport of radionuclides out of a damaged canister.

Current situation

In the EU project Skin, sorption and uptake mechanisms for very low concentrations of radionuclides in mineral were investigated. In summary, the project Skin developed a scientific methodology to quantify the degree of irreversible uptake of radionuclides in the mineral phases after initial surface adsorption (Grambow et al. 2014).

As a part of the Skin project, radium-barium co-precipitation was studied. Studies of how concentration of radium and barium in solution changes as a function of the reaction time reveals that the kinetics of uptake of radium in barite during recrystallisation is a complex process. Overall thermodynamic equilibrium was not reached in these tests, but it is evident that the experimental systems approach equilibrium. Transformation of pure barite into a thermodynamically stable solid solution of radium-barium-sulphate needs 2–3 years at room temperature. The results show that the solid solution radium-barium-sulphate should be treated as a thermodynamic equilibrium system in a long-term, geological perspective (Brandt et al. 2015). Further studies of the mechanism of formation of radium-barium-sulphate from barite clearly show that radium uptake in all studied barite particles is not limited to the surface but includes the whole solid phase (Klinkenberg et al. 2014).

In the EU project Redupp the surfaces of materials with a fluorite structure, i.e. the same crystal structure as UO_2 and spent fuel were studied. The focus of this study was to study the effects of crystal surface structure on non-oxidative dissolution. It came forth that the calculated stability of these surfaces depends on the surface structure (Maldonado et al. 2013); this work allowed further analysis of how, on an atomic level, water sorbs and reacts with a UO_2 surface. The study involved modelling hydroxylation and modification of the UO_2 surface, which occurs for a given set of surfaces and for varying pressures and temperatures (Maldonado et al. 2014).

In order to increase our understanding of how the surfaces of material with fluorite structure affect the dissolution process, leaching experiments with cerium dioxide (CeO_2) and thorium dioxide (ThO_2) were carried out, within the framework of Redupp (Corkhill et al. 2014). The results show that surface reactivity varies and, for polycrystalline material, that it is, above all, the grain boundaries that initially contribute to an increased dissolution rate (Corkhill et al. 2014). Within the project Redupp, studies with a focus on the solubility of ThO_2 were also carried out (Myllykylä et al. 2013). Leaching experiments with ThO_2 demonstrate a significant effect of carbonate concentration in the solution and effect of pH. High-resolution mass spectrometry allowed measurements of very low concentrations of thorium in solution, and the results are useful for the thermodynamic databases used for solubility calculations. Analyses of the leached surfaces indicated that surface initial structure, including the grain boundaries, has some effect on the dissolution (Myllykylä et al. 2013).

Improvement of data included in the thermodynamic databases, which is used for solubility calculations, entails continued experimental work, to obtain more and new data. One such study has been carried out by Zanonato et al. (2014), and is aimed at expanding the data base for uranyl speciation. The effect of temperature on speciation in systems with uranium (VI) hydroxide was studied at 25 °C and 100 °C. Results shows that a significant increase of hydrolysis occurs at higher temperatures, but also that changes occur in the relative amounts of complexes with different charges. The quantity of complexes with a lower charge (+1, −1) increases at the expense of those with higher charge. Temperature is proven to have a significant effect on the formation of hydroxide-peroxide complexes, which are formed at 0 °C but do not occur at 100 °C (Zanonato et al. 2014).

Programme

Remaining questions regarding radium-barium co-precipitation will continue to be studied in cooperation with the German Research Agency Forschungszentrum Jülich.

The possibility of formation of the secondary uranium mineral coffinite has been discussed, and has been studied from different angles during recent years, and a summary and presentation of new data is presented in Guo et al. (2015). The new publications justify efforts in this area, and a study of coffinite formation has therefore been initiated in collaboration with Amphos 21. This work will continue during this RD&D period. Within the framework of this study, new information will be incorporated in databases and solubility calculations, and will be put into context.

Uranium speciation needs to be further studied in view of the new data that suggests the possible existence of calcium-uranyl-carbonate complexes. This is also relevant for other radionuclides. SKB will continue to participate in the work with NEA-TDB (OECD/NEA Thermochemical database) in order to develop and improve the databases used for speciation and solubility calculations.

Parts of a planned EU project will deal with chemistry in a damaged canister, with a focus on the interaction between the incoming groundwater and the corroding canister insert. This will fill knowledge gaps and reduce uncertainties regarding radionuclide speciation and solubility in a repository environment.

8 Canister

The canister consists of a cylindrical copper shell and a load-bearing insert of nodular cast iron, see Figure 8-1. Channel tubes of structural steel for the fuel assemblies have been cast into the insert. Two different designs of the insert have been developed where the size and number of the channels are adapted to fuel assemblies from pressurised water or boiling water reactors (PWR or BWR inserts).

Section 5.5 gave a brief description of SKB's planned development programme for the canister. The issues with the highest priority are: i) research concerning corrosion and creep of copper, ii) development of verification methods for the canister and requirements on material properties and allowed defects, iii) development of manufacturing methods for canister components, iv) inspection and testing of canister components and welds. This chapter provides a more detailed description of SKB's programme in these areas.

8.1 Corrosion

8.1.1 Sulphide corrosion

In the long-term perspective, sulphide is the most important corrosive agent for copper in the repository environment. Canister failure due to corrosion by sulphide after the buffer had eroded away provided the dominant contribution to risk in the SR-Site safety assessment. To strengthen the scientific basis for handling of sulphide corrosion in future safety assessments, a better understanding is needed of the detailed mechanisms in the corrosion process, as well as a better basis for the possible sulphide concentrations in groundwater and bentonite that are used in the safety assessment.

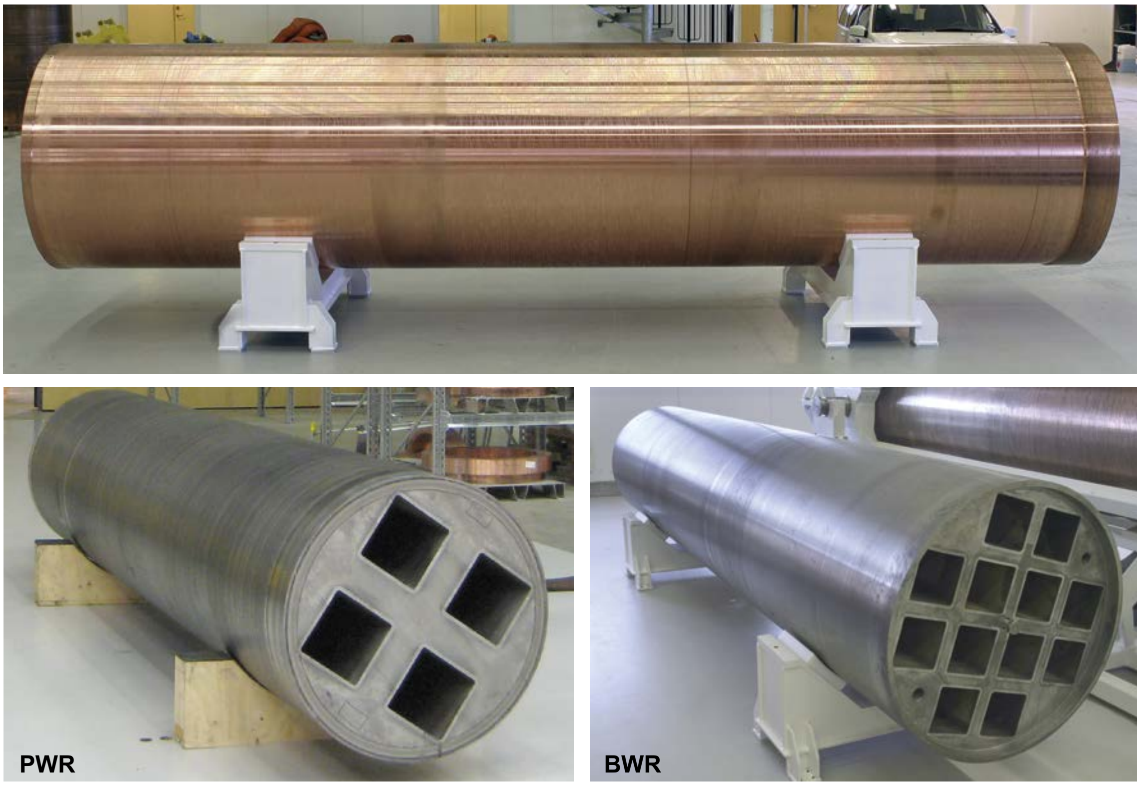


Figure 8-1. *The spent nuclear fuel is encapsulated in copper canisters. Two different inserts of nodular cast iron are used for PWR and BWR fuel.*

Current situation

Formation of copper sulphide film on copper in sulphide solution, and the mechanisms for this, have been studied further by means of electrochemical methods (EIS, electrochemical impedance spectroscopy) and different types of microscopy (SEM, scanning electron microscopy; FIB-SEM, focused ion beam scanning electron microscopy). In particular, the question of what limits film growth has been in focus. The work is being conducted at the University of Western Ontario in Canada. Factors that affect film growth are both the concentrations of sulphide and chloride in solution and the ratio between them, and sulphide flux at the interface between film and solution (Chen et al. 2014a). In order to study film growth at different fluxes of sulphide, stationary and rotating electrodes have been used. At low fluxes (with stationary electrodes) film growth is limited by the diffusion of sulphide in the solution. The same initially applies at higher fluxes (with rotating electrodes that provide convection), but, with the formed sulphide layer, the transport of Cu^+ in the copper sulphide film becomes limiting (Chen et al. 2014b).

From experimental data, a critical flux of sulphide has been estimated, during which corrosion can be considered to be limited by mass transport of sulphide. These critical fluxes have then been compared with the estimated fluxes in the Spent Fuel Repository, with data from SR-Site (King et al. 2014). The estimated sulphide fluxes in the repository are generally several orders of magnitude lower than the critical flux (both for cases with intact and eroded buffer), and the corrosion rate of the canister will be limited by mass transport of sulphide under virtually all conditions. SKB therefore draws the conclusion that sulphide corrosion of copper in the Spent Fuel Repository is limited by mass transport of sulphide, and that this can be derived from the experimental results. This has been presented as a response to a request for supplementary information to SSM for the application for the Spent Fuel Repository (SKB 2015c).

That the film grows in the interface between the film and solution (and thus not between the metallic copper and the film) has been demonstrated in tests with a gold marker on the film (Chen et al. 2012). Under the conditions where growth is limited by the reaction in the interface between solution and copper sulphide film (i.e. by transport of Cu^+ through the film), the copper sulphide film becomes compact and partly passivating, which could give rise to localised corrosion (pitting). This also requires, besides high sulphide flux, high sulphide concentrations, $[SH^-] \geq 5.0 \times 10^{-4}$ M (Martino et al. 2014). Such concentrations are higher than the sulphide concentrations that can be expected in Forsmark.

The passivation of copper in sulphide solution has also been studied by others. Mao et al. (2014) present experimental data interpreted as degradation of a passivating layer, and these data are then used in modelling with a PDM (point defect model). It cannot be ruled out that the measured current is affected by diffusion limitations, and therefore reflects not only passive behaviour of the film. These studies are included in a larger project led by SSM (Macdonald et al. 2014), which so far has mainly yielded data for a future "mixed potential model" for corrosion of copper in the repository. Macdonald et al. (2014) also present modelling of the impact of bentonite on layers of corrosion products, but do not appear to use updated data for bentonite material properties but rather data for concrete, making the conclusions difficult to interpret.

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. That the sulphidation mechanism is of chemical rather than electrochemical character when the oxide film mainly consists of cuprite (Cu_2O) has been demonstrated previously (Smith et al. 2007). Copper oxide layers consisting of Cu(II) oxides also sulphidise, and studies using X-ray spectroscopy have shown that the reaction occurs through a comproportionation of Cu(II) and metallic Cu(0) to Cu(I) which subsequently reacts with sulphide and forms Cu_2S (Kristiansen et al. 2015). The results strengthen the previous conclusion that sulphidation is chemical, and it is, therefore, pessimistic to add these corrosion contributions in the safety assessment (which was done in SR-Site).

Saturated bentonite as a source (via microbial sulphate reduction) and sink for sulphide is being studied in experiments, see Section 10.3.2. Results concerning sulphate reduction show that there is a clear threshold or a narrow range in swelling pressure over which the microbial activity completely vanishes. Handling of formation and availability of gaseous sulphide ($H_2S(g)$) during the unsaturated phase is described in Section 10.1.5.

A literature survey on the solubility of pyrite (FeS_2) has been completed (King 2013). The conclusion regarding pyrite as a source for sulphide in bentonite was that dissolution of pyrite may consume a fraction of the oxygen initially present through oxidative dissolution, but that it is unlikely that pyrite dissolves during the long anoxic time period. This supports the fact that the calculations in SR-Site with dissolution of pyrite during the entire disposal period are pessimistic. However, the contribution to the total corrosion by sulphide was small.

When it comes to sulphide concentrations in groundwater, SKB's studies have continued after the site investigations. The difficulties with sampling and analysing sulphide in representative groundwater from fractures at great depths have been investigated (Rosdahl et al. 2011, Drake et al. 2014, 2015). Studies show that microbial processes in packed-off borehole sections can largely affect local sulphide concentrations when the water in the section is stagnant. SKB's conclusion is, however, that the data (sulphide concentrations) for the safety assessment does not need to be changed in view of these research results, since the affected sulphide concentrations are screened out. New measurements in new, or old, boreholes may of course provide new data.

In summary, the research results have strengthened the conclusions that sulphide corrosion is limited by mass transport, that dissolution of pyrite provides a negligible contribution to sulphide corrosion and that compacted bentonite prevents sulphide production. Similarly, the work has provided a better understanding of what can disturb measurements of sulphide in groundwater. The issue where new results warrant further research is the question of the conditions that can lead to formation of passive films of copper sulphide, which could give rise to localised corrosion.

Programme

Sulphide is the most important corrosive agent for copper in the repository environment, since the thermodynamic driving force for the formation of the corrosion product copper sulphide (Cu_2S) is so strong, and all sulphide reaching the canister will react with the copper. In cooperation with Posiva, several studies focusing on sulphide issues are being carried out.

To strengthen the scientific basis for handling of sulphide corrosion in future safety assessments, a better understanding of the detailed mechanisms in the corrosion process is needed. SKB will therefore continue studies of copper in sulphide solution, above all with electrochemical methods. The work goes essentially in two directions, firstly studies that in more detail explore the potential for occurrence of localised corrosion (through studies of the stability of sulphide film, formation of a passive layer, galvanic coupling and the site of the cathode reaction) and secondly studies of how other ions (chloride, sulphate, carbonate etc.) or bentonite affect the corrosion mechanism.

Also the work with studies of sulphidation of copper oxide film will continue although with a smaller effort. As part of an ongoing PhD project at KTH, DFT calculations (DFT, Density Functional Theory) of the reaction mechanism for sulphidation of oxide film will be carried out, with the main purpose of further improving SKB's understanding of the sulphidation process.

For further studies of microbial sulphate reduction and sulphide concentrations in bentonite, including gaseous sulphide, see Sections 10.1.5, and 10.3.2. The unsaturated period in the buffer is described in Section 10.1.2.

The interactions between microbes and different materials will be studied to clarify the processes in packed-off sections in boreholes. Methods for sampling and analysis of dissolved gases in fractures and rock matrix will be developed. Reduction of Fe(III) and the release of Fe(II) from the rock matrix are important processes that will be studied, since sulphide solubility can be limited by Fe(II) minerals.

For use in future safety assessments, models for calculations of copper corrosion by sulphide will be further developed in cooperation with Posiva. Additional models of the repository's near-field that include sulphide transport, and production and sinks for sulphide, will be developed in parallel.

8.1.2 Localised corrosion

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. The background to this is observations from both large-scale experiments and laboratory studies. Studies of the prerequisites for localised corrosion in sulphide-containing water (reducing conditions) are discussed in Section 8.1.1 on sulphide corrosion.

Current situation

SSM requested, as a supplement to the application for the Spent Fuel Repository, an in-depth analysis of localised corrosion, which SKB provided in a report (King and Lilja 2013). The report contains a literature review of data, and a discussion of groundwater compositions and the conditions required for localised corrosion to occur. The conclusion was that the porewater compositions that are present in the repository favour general corrosion, and do not give rise to any passive film. Passive film is a prerequisite for the occurrence of localised corrosion. The work has also been published in King and Lilja (2014).

The literature-based study of passivity under repository conditions shows that localised corrosion should not occur in the Spent Fuel Repository (King and Lilja 2013, 2014), but there are other results that do not exclude as strongly that conditions that could cause localised corrosion may possibly occur during the unsaturated oxidising period (Kosec et al. 2015). The full-scale copper canisters that were retrieved from the Prototype Repository in the Äspö HRL show a surface morphology that could be interpreted as shallow localised corrosion (Taxén et al. 2012). In this context it should be noted that measurements of corrosion potential in the Prototype Repository just prior to retrieval, as well as spectroscopic analysis of corrosion products, show that the chemical environment has been more or less oxidising throughout the exposure time (Rosborg 2013a, b). To further evaluate the canister surfaces from the Prototype Repository with respect to localised corrosion, 240 observations were made with SEM on different parts of the retrieved canisters (Taxén 2013). These analyses show pit depths of on average 2–3 µm and a maximum measured pit depth of 7 µm; this after about 8 years of exposure in the Äspö HRL. How surface morphology was affected by exposure in the experiment is not possible to determine as the initial state of the canister surfaces had not been characterised. The analysis will, however, benefit future analyses of canisters from the Prototype Repository's inner section.

Some surface roughness was found on U-bend samples from the retrieved MiniCan experiment in the Äspö HRL (Aggarwal et al. 2015). A coarser topology was observed in the samples' bent regions than in the unaffected, which is expected. The surface morphology can, however, be considered to be accommodated within the concept of uneven general corrosion (surface roughness) discussed in King and Lilja (2013, 2014).

Programme

For evaluation of localised corrosion, it is important to describe the water with which the canister will come into contact. For the corrosion assessments, pore water has been considered to be groundwater in equilibrium with bentonite. In order to get a better description of what water chemistry copper in bentonite is exposed to, an experiment has been started at the University of Western Ontario in Canada, with the purpose of measuring potentials on copper in bentonite.

SKB will also further consider the possibility of getting data for assessing localised corrosion through probabilistic assessments.

In order to be able to better characterise the observed surface morphology of the retrieved canisters from the Prototype Repository (Taxén 2013), SKB intends to investigate the surface of canister copper which has only been exposed to the atmospheric corrosion that occurs under normal humidity but has not been exposed in any corrosion experiment. Such an analysis should give a reasonable idea of the topological initial state of the surface of the canisters in the Prototype Repository.

8.1.3 Copper corrosion in pure, oxygen-free water

In March 2015, SKB submitted a comprehensive report to SSM on copper corrosion in pure, oxygen-free water (Hedin et al. 2015). SKB has further provided information to SSM in a letter in April 2016 (SKB 2016b) on the development of the issue after the comprehensive report. The reports are based on several extensive studies supported by SKB. The conclusion is that there is no 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.

Current situation

In the comprehensive assessment submitted by SKB (Hedin et al. 2015) and in the letter of April 2016 (SKB 2016b), references are made to most of the work conducted by SKB on the issue of corrosion of copper in pure, oxygen-free water, and they are not repeated here. In addition, material has been published in scientific journals and SKB reports, mainly from experimental and theoretical studies of the Cu-O-H system and from studies of surface reactions between water and copper and copper oxide respectively:

- Theoretical studies of solvent effects on both ideal copper surfaces and nanoparticles of copper have shown that the qualitative picture from the process in the gas phase remains, meaning that dissociation of water with subsequent hydrogen formation does not occur to a greater extent in solution than in gas phase (Lousada et al. 2015, Halldin-Stenlid et al. 2014, 2016). The conclusion from these studies is therefore, as before (Hedin et al. 2015), that surface reactivity is not sufficient to explain the hydrogen gas evolution that has been observed in some experiments with copper in oxygen-free water.
- Concluding documentation from the work with the search for a new stable phase in the Cu-O-H system (Li et al. 2015, Soroka et al. 2016).

The two supplementary memos mentioned in the above letter (SKB 2016b) regarding further analyses of copper in the water of some of the test tubes at Micans (Blom and Pedersen 2016) as well as outgassing measurements of canister copper exposed to water at 70 °C for about three years at Micans (Berastegui et al. 2016), have been completed and sent to SSM. SKB and the research group at Uppsala University continue to work with scientific papers on the SKB-supported studies of copper corrosion in oxygen-free water. Early parts of the work at Uppsala University have also been published in Boman et al. (2014).

A publication that describes how the issue of corrosion of copper in pure, oxygen-free water has been handled in the licensing process with a reference group has been published (Andersson 2013).

The letter (SKB 2016b) also describes the discussion, initiated by SKB, which has been published in the journals Corrosion Science and Journal of the Electrochemical Society in view of published articles there. In short, SKB has expressed its opinion on, among other things, the reporting of the experimental conditions in the experiments, conclusions regarding corrosion and corrosion mechanisms, the use of references that do not support the claims they are said to support, as well as the incomplete agreement between model calculations and thermodynamic data.

Programme

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. The experiments that formed the basis for statements of the contrary have been repeated in a project supported by SKB, under more controlled conditions, without corrosion being detected. An alternative, comparatively simple method has been developed with support from SKB and no corrosion has been found with this method either. SKB-supported theoretical studies have not resulted in any previously unknown stable phases in the Cu-O-H system being identified. SKB therefore plans no further extensive studies in the area. Some minor, supplementary studies are however planned during the RD&D period.

Further quantum chemical and experimental studies will be carried out within the PhD project at KTH in order to better understand the reactivity at the interface between copper and water. Among other things, studies are in progress of the water-splitting and hydrogen-forming reaction on the surface of Cu_2O, which is a more realistic model of the surface of a copper sample. The effects of different types of defects on the Cu surface are also being explored.

The methodology with electrochemical studies (impedance spectroscopy, voltammetry etc.) will be further developed, in particular with the aim of evaluating kinetic data for copper in water, but also for investigations of the influence of chloride or other ions on corrosion mechanisms.

8.1.4 Radiation-induced corrosion

Radiation-induced corrosion of the copper canister occurs due to the radiolysis products formed when water on the outside of the canister absorbs gamma radiation from the fuel in the canister. Particularly during the first 300 years in the Spent Fuel Repository, the dose rate on the canister surface will generate radiolysis of water. In the SR-Site safety assessment, a total corrosion depth of about 14 µm due to this corrosion process was calculated pessimistically.

Current situation

The effects of gamma radiation on corrosion of copper have been studied further in a PhD project at KTH (Björkbacka et al. 2013, Björkbacka 2015). Experiments have been conducted with copper cubes in pure water in a nitrogen atmosphere at radiation dose rates between 0.1 and 1 kGy/h. It should be noted that 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, but that exposure times in the experiments have been adapted so that the total dose nevertheless is of the same order of magnitude as in the Spent Fuel Repository (Björkbacka 2015).

The surfaces and the aqueous solution have been investigated by means of various spectroscopic methods. The initial studies show higher corrosion for irradiated specimens than for un-irradiated ones, which is expected. Corrosion appears both as formation of Cu_2O and as local cavities with a depth of about one micrometre. Recent 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.

In the Canadian nuclear waste programme, corrosion experiments under irradiation are being performed. Ibrahim et al. (2015) used a radiation dose of 0.35 Gy/h, which is the same order of magnitude as the initial radiation dose on the outside of the KBS-3 canister at deposition. The experiments were performed in air, at 70–85 °C and with different humidity, which can represent conditions during the saturation phase. The formed corrosion products (a double layer of Cu_2O and CuO) are the same as for corrosion in an aqueous solution in contact with air. Condensation of water vapour gives a lateral spreading of the corrosion products. At higher humidity (and after a long time at lower humidity), however, localised corrosion attacks (shallow pits) were observed. Irradiation led to a faster initial coverage of the surface with corrosion products, but not to an increased total corrosion. Experiments were also performed at a 10 000 times higher radiation dose (3 kGy/h), with increased corrosion (film thickness in the order of < 1 µm) as a result (Ibrahim 2015).

Experiments at higher doses may be useful for studying mechanisms, but direct extrapolation to lower doses for a long time from single experiments cannot be done without detailed understanding of the mechanisms.

Programme

In order to be able to better assess how oxide films affect radiation-induced corrosion in the groundwater environment in the Spent Fuel Repository, the work at KTH will continue with a study of the reaction mechanism for this corrosion process. SKB will also continue to follow the work in the Canadian programme.

8.1.5 Stress corrosion cracking

SKB submitted a description of the current situation regarding stress corrosion cracking in a supplement to SSM in February 2014 (SKB 2015c), and this description and the references there are not repeated here.

In order for stress corrosion cracking to occur, a sensitive material is required in combination with tensile stresses and aggressive ions (in the case of copper, the ions nitrite, ammonium or acetate). 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 and acetate. Since the necessary ions are lacking 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 can also occur in the presence of sulphide has been discussed particularly since 2008 when a Japanese research group (Taniguchi and Kawasaki 2008) presented results that indicated such a process. Recent studies have, however, not provided results in the same direction.

Current situation

SKB summarised in the supplement that one single study showed what was interpreted as stress corrosion cracking in a sulphidic, oxygen-gas free environment, and then at a sulphide concentration of 10^{-2} M, but that the results have not been possible to verify in two following-up SKB studies. Furthermore, it was noted that the highest sulphide concentration in Forsmark is 1.2×10^{-4} M, i.e. almost two orders of magnitude lower than in the experiments where the original observation was made. During the initial oxidising period, one of the prerequisites for stress corrosion cracking is not present (the aggressive ions). It was therefore concluded in the supplement that SKB continues to judge that stress corrosion cracking cannot threaten the integrity of the canisters in a KBS-3 repository in Forsmark.

In the retrieved MiniCan experiment from 2011, there were a number of samples to evaluate stress corrosion cracking, which was also described in the supplement (SKB 2015c). There were both bent copper samples to investigate fracture initiation and pre-cracked samples to investigate crack growth. Samples have been further studied metallographically with optical microscopy and SEM (Aggarwal et al. 2015). No fracture initiation has been observed in either of the two bent samples (Aggarwal et al. 2015, Smart et al. 2014). Nor has any crack growth occurred in the two pre-cracked samples, which however were incorrectly installed; no external load was imposed so only residual stresses in the material after cold work may have been present.

Programme

The research programme for stress corrosion cracking on copper is continuing with additional efforts, focused on sulphide-containing water, although according to SKB there is a lack of both a well-documented mechanism and clear experimental results for stress corrosion cracking under reducing conditions. An important component in the analyses is the prerequisites for formation of a passive film of copper sulphide, see Section 8.1.1.

SKB will continue the work at Swerea KIMAB to, if possible, explain the differences in results from the different experimental studies, and is investigating whether differences in specimen design may be a factor. There are also identified differences in surface treatment and the time in sulphide environment between the different experiments. Reporting of results is expected in 2016.

In the autumn of 2015, two additional test packages from the MiniCan experiment at the Äspö HRL were retrieved. Analyses are in progress during 2016 and include investigation of stress corrosion cracking in the canister material, as well as bent and pre-cracked samples of copper, in a similar manner as for the previously retrieved canister 3.

Even if SKB is confident that the state of knowledge regarding mechanisms for stress corrosion cracking are essentially sufficient for the assessment of post-closure safety, efforts will be made to calculate the consequences of any stress corrosion cracking.

8.1.6 Verification of different copper materials for corrosion sensitivity

Prior to the PSAR for the Spent Fuel Repository it must be ensured that the copper material in all parts of the finished, sealed canister is sufficiently corrosion-resistant.

Current situation

The development of electrochemical methods to investigate the difference in the corrosion susceptibility of different copper materials (welded, cold-worked etc.) that was announced in the RD&D Programme 2013 has continued. A description of the method and the first results with a comparison of copper materials with different phosphorus content has been published (Taxén and Sparr 2014). The results show that all the tested copper grades have lower potentials (are more noble) than a high-purity copper. This is, however, of no importance for a canister that has the same composition of copper everywhere, and for which the corrosion is limited by the availability of corrodants (oxygen and sulphide).

Programme

The method will be used, and possibly refined, to also study cold-worked materials (ongoing work that is nearly finished) and welded material.

8.2 Creep of copper

Creep of copper can be divided into several specific issues and activities:

- Mechanistic understanding of the influence of phosphorus in order to show that copper ductility is maintained even in a long-term perspective, i.e. even at very low strain rates.
- Creep testing in laboratory environment, with evaluation of results with respect to obtained ductility in specimens and the stress state's influence on the ductility.
- Quantification of the maximum permanent deformation of the copper shell that can be obtained in the final repository and with which stress state (degree of tri-axiality). This serves as a basis for the formulation of requirements on copper ductility.
- Assessment of fulfilment of requirements: Based on mechanical testing and understanding of long-term properties of phosphorus, an evaluation is made of whether the copper meets the requirements on ductility needed for maintaining the canister integrity for 100 000 years.

SKB has made a summary of the work and the current state of knowledge for creep of copper on several occasions, most recently as a letter to SSM regarding the application for the Spent Fuel Repository in April 2016 (SKB 2016a).

Regarding the importance of creep of copper for the post-closure safety of the repository, it is also of great importance to investigate the extent to which load cases that are not deformation controlled can actually be expected to occur in the final repository. Deformation-controlled cases can be handled with elasto-plastic models while non-deformation-controlled cases require a creep model for handling, see further the reporting in the last paragraph in the programme part of Section 8.2.2.

8.2.1 Impact of phosphorus

Current situation

The work of studying the effect of phosphorus on creep properties has essentially continued according to the described plan in SKB (2014f).

With DFT calculations (DFT, Density Functional Theory), studies have been performed to investigate how point defects (vacancies and foreign elements) interact with larger defects (defects in atomic planes, so-called stacking faults and grain boundaries). Li and Korzhavyi (2015) show that in terms of energy it is more favourable for phosphorus to be in a substitutional position (a copper atom has been replaced with a phosphorus atom) than interstitially (the phosphorus atom is positioned between copper atoms). The stacking fault energy in copper decreases when point defects are present, and the effect of phosphorus is larger than for vacancies, H, O and OH. Decreasing stacking fault energy implies a tendency to segregation. Further studies (Li 2015) of impurities of *3sp* elements (magnesium, aluminium, silicon, phosphorus and sulphur) and their impact on how a grain boundary is held together, show that the impurities in general improve the strength of the grain boundaries with decreasing electronegativity (i.e. from sulphur to magnesium). For phosphorus however, the stacking fault energy and the tendency of phosphorus to spread evenly in the grain boundary has a major impact, and overall the calculations show that phosphorus will counteract the formation of cavities in the grain boundaries.

The thermodynamic calculations of the stability of copper oxides have been verified by experiments with oxidation of copper with water vapour. The same study has showed the occurrence of copper oxides rich in phosphorus. The preliminary conclusion in the study was that copper phosphates are more stable than oxides at low oxygen gas pressure. In Cu-OFP (oxygen-free phosphorus-doped copper), oxygen will therefore predominantly be in phosphates and not in the form of oxides.

A study at Chalmers University of Technology (Thuvander 2015) on behalf of SSM investigated the distribution of phosphorus in copper inside the grains with APT (Atom Probe Tomography) and in the grain boundaries with TEM-EDS (Transmission Electron Microscopy – Energy Dispersive Spectroscopy). Results could not detect any extensive segregation of phosphorus to grain boundaries, since the APT measurements showed roughly the same content of phosphorus inside the grains as the total content, and no phosphorus could be detected at the grain boundaries with TEM-EDS.

Studies with TOF-SIMS (Time-of-Flight – Secondary Ion Mass Spectroscopy) have been performed, and the preliminary results show also here that no clear segregation to grain boundaries occurred, since no difference in phosphorus content can be measured. However, it cannot be ruled out that small quantities of phosphorus (for example in monolayer levels) accumulate in or near the grain boundaries, which is difficult to detect.

Modelling work concerning the effect of phosphorus on creep properties has continued. Hypotheses regarding differences in sizes of grain-boundary sliding with and without phosphorus have not been confirmed. However, the proposed mechanism that phosphorus affects the formation of cavities is still valid. The mechanism is considered probable in the sense that higher stress is needed for cavities to form if phosphorus is present, which in turn depends on a higher diffusion of phosphorus in grain boundaries in Cu-OFP than in Cu-OF (oxygen-free copper).

Programme

To support modelling of creep, and especially the long extrapolation that is needed, it is central to understand the impact of phosphorus on the creep properties in detail. SKB will continue in the same way as previously stated, with a combination of studies with different aims and scales, from atomic scale to specimen scale.

The calculations of interactions between atoms and dislocations and defects (with DFT) are focused primarily on including more extended defects such as grain boundaries, as well as studies of diffusion. Different calculation techniques will be used to investigate the stability of the results. The thermodynamic studies intend to investigate the impact of other impurity elements present in the copper and in the first phase investigate the stability of phases consisting of phosphorus and these elements.

Further experiments will be carried out to try to find phosphorus in copper and study how it is distributed, but it has been difficult to find techniques that are useful and sufficiently sensitive. Primarily, the work will be focused on studying the grain boundaries.

More indirect methods will be used instead to understand the effect of phosphorus on the copper material, and in particular its creep properties, see Section 8.2.2. In addition, creep testing is also planned to some extent, e.g. to better estimate the impact of different impurity elements and phosphorous contents.

8.2.2 Deformation and failure

Current situation

SKB has in Raiko et al. (2010) stipulated preliminary requirements on copper ductility, which means that the ductility must be at least 160 percent , which is obtained in a round tensile specimen if the fracture surface is at most 20 percent of the original cross-sectional area. This requirement applies to both elasto-plastic tensile testing and creep testing. It is further stipulated that the maximum permissible strain is 80 percent in the copper shell. A multiaxial stress state is thus not included explicitly, but implicitly, since fulfilment of requirements was derived from round tensile specimens with or without creep. Raiko et al. (2010) also state that the maximum true plastic strain is 20–30 percent at most in the copper shell. It was also further stated that the initial gap between the canister components (insert and copper shell) will disappear due to creap by external pressure. The appendix to Andersson-Östling and Sandström (2009) describes the area reduction obtained for a large number of specimens, and typically, the area reduction is about 80–90 percent, which is equivalent to about 160–230 percent strain for OFP copper that is "as manufactured " or welded with friction stir welding in the normal way.

Thereby it was concluded that the integrity of the copper shell will be maintained with respect to creep for 100 000 years. As parts of the supplements to the application for the Spent Fuel Repository, further studies were carried out with respect to isostatic load cases over 100 000 years (Hernelind 2015) and uneven load cases (Hernelind 2014). The largest true plastic strain in the copper shell was then calculated to 40–60 percent. Stress states and principal directions of stress were also provided. Unosson (2014) described copper's elongation at fracture as a function of the degree of tri-axiality at elasto-plastic testing.

After the supplement to the license application for the Spent Fuel Repository submitted to SSM in February 2014 (Sandström 2014a), some studies and results have been added, which are described in the text below. SKB's descriptions of creep ductility in copper with and without phosphorus have been evaluated on behalf of SSM (Pettersson 2012, 2016) and thereby criticised, above all for how an incomplete understanding of the impact of phosphorus makes it difficult to extrapolate creep properties over long periods of time. SKB has now been able to more clearly resolve both the issue of grain boundary sliding and the occurrence of creep cracks.

Phosphorus has previously been assumed to reduce grain boundary sliding and cavity formation in copper, which would thereby provide an explanation for the higher creep ductility compared with copper without phosphorus. Differences in grain boundary sliding could not, however, be determined, neither in the previous study by Pettersson (2010), nor in the recently published SKB studies (Wu et al. 2015, Sandström et al. 2016). However, the continued modelling work still assumes that there is a difference in the formation of cavities in copper with and without phosphorus. A new model assumes that the cavities arise due to the formation of chains of dislocations (stacking) in the grain boundaries. Due to the fast diffusion of phosphorus in grain boundaries, a substantially higher stress is required to form cavities in copper with phosphorus than without. This makes it possible to better explain the measured differences in creep ductility between the material types (Sandström et al. 2016, Sandström 2016).

In a compilation of how cold work and notches affect creep behaviour (Wu and Sandström 2015), crack growth was found at 125 °C for CT specimens (CT, compact tension) with reduced central cross-section, by side grooves. For specimens without side grooves no significant crack initiation was found. The conclusion of this study was that a high degree of multi-axiality and temperatures of at least 125 °C are required for crack initiation. Previously tested CT specimens have then been investigated with a scanning electron microscope to see whether any crack growth could be observed. Preliminary results show that crack growth could be found in tests performed at 175 °C, but with an appearance typical of ductile behaviour. No signs of intergranular creep cracks were found.

In a PhD project at the Department of Material Technology at KTH, dislocation dynamics is used to study how dislocation density changes at loading. Hosseinzadeh Delandar (2015) simulates copper in the form of a single crystal, and the results show that an inhomogeneous structure will occur at plastic deformation. The effect increases with strain rate.

Programme

Both as a step in understanding the role of phosphorus on a microscopic level and better for documenting the differences in creep properties, creep testing of copper without phosphorus (Cu-OF) will be performed. This means that more comparable data will be obtained than the few older tests that exist today. This means testing at primarily room temperature and up to 125 °C, but higher temperatures will also be considered. New creep rigs with step motors will be used, which means that there will be no need for reloading of the lever arm. The loading strain and creep strain are both included in the evaluation, and any annealing or cold work of the material is documented.

A smaller effort will be made to expand the creep models to include also tertiary creep, in addition to the primary and secondary creep which is already covered. Potentially, some creep testing will also be needed for this.

The PhD study on dislocation dynamics at KTH continues with studies of creep deformation, how dislocations interact with other defects such as dissolved impurities, precipitates, cavities etc., as well as how deformation hardening on a macro-scale can be linked to the development of microstructure.

Furthermore, an in-depth evaluation of previously conducted testing with respect to plastic strain and stress state in different creep specimens with arbitrary geometry is planned.

In parallel with the work of developing the creep model, efforts are made to identify the load cases that require a creep model. The load cases that are deformation controlled can be solved elasto-plastically and require no creep model. It is therefore very important to correctly describe the load cases that are relevant for the final repository. The shear load case is governed by displacement, and hence it is possible to use conventional elasto-plastic constitutive models to describe the deformation of copper for this case. The same applies for cases caused by uneven swelling; the load is imposed due to the swelling, is governed by displacement and an upper limit for canister deformation can be estimated. The isostatic load case is force driven due to the external pressure on the canister surface. The direction of the load is, however, such that the copper shell is unloaded against the insert and a steady-state relation arises due to the geometric constraints imposed by the design of the KBS-3 canister. Studies have shown that this position remains virtually unchanged at very large external loads. The copper is deformed so that a uniform stress state arises, and thereafter the deformation ceases. Thus, for these cases it has been determined in engineering terms how much deformation and strain that can arise in the copper shell after deposition. Further work will be performed to identify whether any load cases must be regarded as force driven with respect to the copper shell. If so, the case must be solved with a constitutive calculation model that includes creep. Thereafter, the requirements on the copper material's ductility can be revised and a new assessment of fulfilment of requirements can be made.

8.3 Design

8.3.1 Design analysis

Current situation

The canister design has been verified to withstand mechanical loads in the design analysis report (Raiko et al. 2010) and requirements have been specified on the canister's mechanical properties and permissible defect size. In addition, an assessment of fulfilment of requirements was made in the production report (SKB 2010c). Within the framework of supplements regarding the mechanical integrity of the canister, SKB has added experimental data, technical justifications and consequence analyses to the application (SKB 2014g).

Requirements on the canister's mechanical properties and permissible defect size are formulated deterministically based on design premises, results from manufacturing and testing of canister components, and material standards. In order to evaluate canister robustness, SKB has also performed probabilistic analyses of both the isostatic load case and the shear load case.

The acceptable size of defects varies in different parts of the inserts cross-section, but otherwise the set of requirements regarding defects and material properties is uniform for the whole insert. Despite the fact that mechanical loads vary between parts of the canister in the final repository, the requirements on the canister have not been adapted for this to any large extent. Furthermore, manufacturing processes result in both a systematic and a stochastic variation of properties and defects. To adapt the requirements to changes in design premises, variations in mechanical loads, and the systematic variation of properties within components, SKB will expand the set of requirements.

Preliminary results indicate that the requirement on elongation at fracture in nodular cast iron can be relaxed. The set of requirements with respect to idealised defects in nodular cast iron is being studied further, and also here preliminary results indicate that these requirements can be relaxed, in particular for the central parts of the insert. SKB is further investigating with which confidence the requirements must be satisfied and how this should be verified in production.

Programme

SKB will update the design analysis report (Raiko et al. 2010) for the PSAR so that the extensive information that has been produced to respond to requests for supplements is incorporated and clearly compiled. Furthermore, new verifying analyses and damage tolerance analyses are made of both the isostatic load case and the shear load case where the new design premises on isostatic load and the properties of the buffer are considered. Since the manufacturing process for copper lids has been updated to include annealing, the permissible handling loads for the copper canister are updated where material data for soft-annealed copper and maximum permissible handling temperature is assigned.

The work of updating Raiko et al. (2010) has also had as an objective that by more detailed analyses, if possible, to relax the set of requirements with regard to material properties and permissible defect sizes and also make the presentation of the stipulated requirements clearer. Special emphasis has been placed on explaining why parts of the canister's components do not require inspection.

The shear load case is in many respects the design basis for the requirements on the canister. Therefore, the shear load case is given particular attention to obtain as well-adapted requirements on the canister as possible. Among other things, lower bentonite stiffness and the depth/length relation of defects are considered. Sensitivity analyses of the consequences for canister integrity if very large defects in the insert nevertheless would occur are also performed.

It is further investigated through more accurate analyses if it is possible to relax the requirements on elongation at fracture and yield strength in the insert. The overall goal is to not specify higher than necessary requirements on the insert, which results in different requirements with respect to defect sizes in both the insert cross-sections and in the axial direction. Sensitivity analyses investigate and justify why certain parts of the insert such as the bottom , the top and the part between the channel tubes can be excluded from inspection. As an example, Figure 8-2 shows a calculation case where a canister with very large defects is only marginally affected by the shear load case. SKB is investigating whether requirements on elongation at fracture can be relaxed in a limited area in the centre of the inserts, due to the low elastic strains.

For the isostatic load case, the combination of the most pessimistic requirements is considered with respect to yield strength, edge distance and acceptable defects for BWR and PWR inserts, as well as axial crack-like defects at the channel tube corner and the impact of residual stresses.

Work is in progress with respect to relaxing the acceptable defect sizes over the insert cross-section as well as requirements on inspection in general. This is particularly done for crack-like defects, with the assumption of a depth/length relation based on analysis of real defects in the demonstration series (I53–I57), with an extent in tangential and radial direction which starts from the outer corners of the channel tubes. Apart from this, the consequences of shearing of a canister whose insert is assumed to have very large initial fractures between the channel tubes are analysed. Requirements are formulated for constituent steel parts including the steel lid, as well as a study of the possibility of using the ASME standard (American Society of Mechanical Engineers) for the insert. Furthermore, it is investigated if the requirements regarding elongation at fracture in the insert and other material requirements on the insert can be relaxed. Special emphasis is placed on better substantiating the constitutive modelling of nodular cast iron, even for very large deformations and material failure. In addition, a programme is carried out to further explain the use of fracture toughness obtained at 2 mm stable crack growth in a specimen as well as to investigate how fracture toughness and ductility are affected by pre-compression of the nodular cast iron. The permissible handling load for the insert is also being studied.

For the copper shell, new studies are performed concerning the maximum residual deformation that the copper shell may be subjected to. In these studies, both elasto-plastic material models and material models that contain creep are used, see Section 8.2. 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 excentrically mounted insert is studied. Thereafter, the requirements on the copper material's ductility can be revised and a new assessment of fulfilment of requirements be carried out.

Permissible handling loads are analysed as well as the permissible size of handling defects in the copper shell.

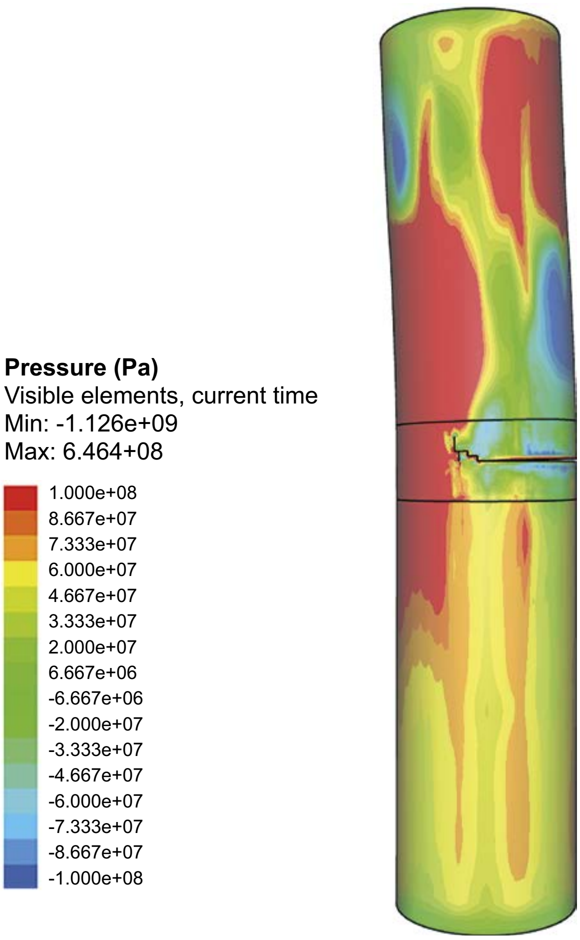


Figure 8-2. *Areas with compressive stress (positive values red, yellow and green areas) and areas with tensile stress (negative values green and blue areas) in an insert that initially had very large defects before the canister was exposed to the shear load case. Crack propagation ends when the crack tip reaches the compressive area that is formed in the insert due to the reinforcing function of the channel tubes (Unosson 2016).*

8.3.2 The role of hydrogen in copper

Canister copper has a hydrogen content of about 0.5 weight ppm. A better understanding is needed of how hydrogen is absorbed in the material and how it is released, for example during heating, in order to interpret experiments on corrosion in oxygen-free water and to evaluate whether, and if so how, hydrogen affects the material properties in copper.

Current situation

In a supplement to SSM in September 2014 (SKB 2015c, Question 9), SKB provided information on hydrogen in copper. As a supporting document a report was submitted (Sandström 2014b), which presents SKB's studies of hydrogen charging of copper and hydrogen impact on material properties. Descriptions and references in these reports are not repeated here.

In a study (Ganchenkova et al. 2014) within the Finnish Research Programme on Nuclear Waste Management (KYT), DFT calculations have been used for studying how hydrogen and other impurities affect the formation of small clusters of vacancies in copper. The results show that hydrogen stabilises the formation of otherwise unfavourable divacancies and favours nucleation of clusters of vacancies. In such a cluster, hydrogen will primarily be positioned on the surface of the cluster and prevents the cluster from collapsing. If the cluster becomes large enough, hydrogen molecules can form, which, however, is not a driving force for forming larger pores. Conclusions from the work provide further understanding of the driving forces for the impact of hydrogen on the copper material, but do not contradict observations made for hydrogen charging (or attempts to) of copper.

Hydrogen content in copper exposed to gamma radiation in aqueous solution has been studied with TDS (thermal desorption spectroscopy) by Lousada et al. (2016). Radiation doses of up to 69 kGy have been used, which is of the same order of magnitude as the doses received by the surface of the canister in the Spent Fuel Repository. The used dose rate (0.135 Gy/s) is, however, about 1 000 times higher than in the repository. The hydrogen content after irradiation is given as up to 0.4 weight ppm after correction for blank samples that have been in water the same time, but not irradiated. Background concentration is, however, not given explicitly. The authors summarise that the uptake of hydrogen probably depends on several factors, where the main factor is gamma radiolysis of the water closest to the canister, but also the formation of defects in the copper as well as changes in the shallowest layers of copper atoms. With so few details regarding conditions in the experiment and the fact that only thin foils were studied, SKB cannot in a meaningful way draw any conclusions on the importance of these results for the copper canister in the Spent Fuel Repository.

Programme

The previously started studies with creep testing during simultaneous hydrogen charging (cathodic electrolytic charging) are expected to be finalised in 2016.

So far studies of hydrogen charging of copper have shown that it requires much more extreme conditions (high radiation, electrolytic charging etc.) than those in the Spent Fuel Repository in order for hydrogen to get into the copper, and thereby possibly affect the material properties. New efforts will, however, be made to investigate whether, and if so how, hydrogen generated by corrosion processes or radiolysis could affect the mechanical properties in general and the creep properties in particular.

To reduce the risk of hydrogen embrittlement, work is being pursued to reduce the occurrence of oxide particles in the weld, see Section 8.4.3.

8.3.3 Requirements on maximum copper content in nodular cast iron

Gamma and neutron radiation can affect the material properties of the nodular cast iron. If copper particles are precipitated it may cause deterioration of the mechanical properties of the material. The extent of this process needs to be studied so that well-founded requirements can be set on the maximum permissible copper content in iron materials in the PSAR for the Spent Fuel Repository. Previously, the question has been raised if so called LBP (late-blooming phases) can form as an effect of the irradiation in the repository.

Current situation

In a supplement to SSM in September 2014 (SKB 2015c, Question 10), SKB submitted a report of how irradiation is expected to affect the material properties of the nodular cast iron insert. Descriptions and references in the report are not repeated here.

In an experimental study at the Department for Reactor Physics at KTH (in cooperation with Shimane University, Japan), nodular cast iron specimens were irradiated with electron radiation to determine the threshold energy at which clustering of copper (aggregation of copper atoms that can provide particles) can occur. For evaluation of the effects in the Spent Fuel Repository, an updated analysis of the radiation levels in nodular cast iron (Toijer 2014) will be used. This shows that the previously used models overestimated the effects of gamma radiation. The preliminary results from the study at the Department or Reactor Physics at KTH show altogether extremely small effects of radiation on the tested nodular cast iron from inserts from SKB.

A feasibility study to investigate whether LBP can occur in the nodular cast iron is under completion, where thermodynamic calculations of phase diagrams (Calphad, CALculations of PHase Diagrams) are used together with quantum mechanical calculations for analysis of interactions between dissolved atoms and vacancies in the copper lattice. The preliminary conclusions are that precipitation of LBP is thermodynamically possible for the "G-phase " (a phase rich in nickel, manganese and silicon), M_2P- and M_3P-phosphides (phases where M stands for chromium, iron, or nickel, and P stands for phosphorus) and a sulphide phase called MnS. That the phases are thermodynamically stable does not, however, necessarily mean that they will precipitate.

Programme

In order to ensure that the requirements on the maximum content of copper are appropriate, SKB is planning further work to study the precipitation of copper particles and subsequent embrittlement of the nodular cast iron. The possible impact of radiation on the channel tubes of steel will also be taken into account.

The preliminary and qualitative conclusions regarding LBP will be followed up by further calculations using ab initio and Calphad calculations in combination, to better estimate the probability of precipitation in the nodular cast iron in the canister. Verifying experiments may also be needed.

8.4 Manufacturing

Production of canister components is carried out under defined manufacturing requirements. The manufacturing requirements are formulated so that the acceptable values of the design parameters in the canister's reference design are met. This means that the requirements are at least as strict as in the reference design, but in many cases stricter. The reference design applies independently of the chosen production method. The manufacturing requirements can, however, vary to ensure low rest risk that tested and controlled design parameters will not deviate from the specified acceptable values anywhere in the component.

This section presents the current situation and programme for manufacturing of the different canister components, the seal weld and deposition.

8.4.1 Copper components

Manufacturing of copper components can be divided into extrusion of tubes, pierce and draw processing of tubes and forging of lids and bottoms. Common for the manufacturing is that it is supposed to deliver products that with high reliability meet the specified requirements regarding material properties, dimensions and defects.

Current situation

SKB has in the production report (SKB 2010c) and associated background reports explained the outcome of the demonstration series of extruded pipes and forged copper lids with respect to meeting requirements on geometric dimensions and material properties. In the supplement to the application (SKB 2014h), SKB has further explained the outcome of the chemical composition of pipe ingots and the possibility of controlling the ductility in forged copper lids by heat treating these after forging. Furthermore, any possible defects in the copper components have been analysed and classified. Acceptance criteria for defects with the aid of FE simulations (FE, finite element) of extrusion and forging processes have been developed.

Programme

For the copper components, SKB plans to further investigate the requirement on the average grain size in the copper components, which may lead to a need for formulating an additional requirement on sound attenuation in order to ensure the reliability of ultrasonic testing. SKB intends to explore and clarify acceptance requirements with regard to material composition (for example maximum oxygen content in the copper shell) and supplement and justify requirements for defects in the copper components (Jonsson and Rydén 2014). SKB plans to validate the FE calculations underlying the calculated allowed defects through suitable experiments. SKB plans to study and improve the strategy for how quality assurance of phosphorus content and other specifications of the chemical composition in the copper components will be carried out, as well as for how to obtain a more even grain size. SKB also intends to develop the forging process for lids and bottoms to reduce the spread in ductility and grain size in the finished product and minimise the risk of forging laps.

In addition, SKB is considering to study, and if possible select, the pierce and draw processing as an alternative reference method for manufacturing of copper tubes with integral bottom. SKB and Posiva have over a long time jointly conducted development of the pierce and draw processing process. It is now possible to produce copper materials that meet the production requirement on elongation at fracture of at least 40 percent and the reference design requirement on an average grain size of at most 800 µm, but not always the requirement on an average grain size of at most 360 µm. The average grain size is assessed according to ASTM E112, which is an American standard for grain size assessment. More work remains to develop the process and verify that an even grain size can be obtained also in the integral bottom of the tube, but also to determine what defects can occur during pierce and draw processing, and acceptance criteria for these defects. In addition, SKB intends to determine how qualification of copper production can be carried out, also with respect to pierce and draw processing.

8.4.2 Canister insert

In order for the canister to guarantee radiation safety during transportation, deposition and final disposal, SKB needs to show that the mechanical strength is sufficient in the entire insert and that there are no unacceptable deviations in the insert.

Current situation

SSM has in its review of the RD&D Programme 2013 expressed that SKB needs to evaluate the significance of randomness in the casting processes and its impact on the mechanical properties and that SKB should develop the PWR inserts to the equivalent level as the BWR inserts (SSM 2014a, p 92). SKB has observed that the mechanical properties vary between and within the nodular cast iron inserts, which can be linked to the material's microstructure and possible defects such as pores and oxide inclusions. The cause can be the distribution of graphite in the nodular cast iron and phenomena such as solidification shrinkage of metals and the varying solubility of gases during solidification. SKB has developed a constitutive model for the impact of the stress state on nodular cast iron's behaviour during deformation with respect to graphite (Dahlberg et al. 2014). SKB has carried out a design analysis for BWRs presented in Raiko et al. (2010), where the calculation assumptions are summarised. The requirements on the BWR insert listed in Table 8-1 in Raiko et al. (2010) are derived from this design analysis. In Section 3.3.2 there is a detailed derivation of the requirement on elongation at fracture. For PWRs, the corresponding analyses are made in Dillström et al. (2014, Appendix 6). Dillström (2014, Appendix 7) summarises the requirements on the mechanical properties of the nodular cast iron with respect to the reference design of the canister.

Requirements on the reference design have thus been derived from the damage tolerance analyses for BWR and PWR inserts. The required material properties are as follows (SKB 2014h):

- yield strength,
- ultimate strength,
- elongation at fracture,
- fracture toughness.

SKB has in (SKB 2010c) with associated background material presented the outcome of the demonstration series of BWR (I53–I57) with respect to material properties and geometric dimensions. Within the framework of the work that has been pursued to answer requests for supplements to the application, a data analysis has been carried out (Shipsha 2013) for the outcome with respect to mechanical testing of the three most recently fabricated PWR inserts IP23–IP25, along with a clarification of the manufacturing requirements on the material properties of nodular cast iron, and an evaluation of how quality assurance of these properties will take place.

SKB casted a BWR insert in May 2015 with the goal of introducing the process improvements launched during the development of PWR inserts in the years 2008–2012, and subsequently studying variations in material properties within the insert. To support the statistical analysis, the number of tensile samples was increased axially and radially compared with previous assessments. Figure 8-3 shows that several of the samples can be considered to be duplicates depending on the symmetry at solidification.

Individual values and average values for elongation at fracture are presented in Table 8-1. The table shows systematic variations between samples vertically as was already known, but also systematic variations between sample positions radially. The difference between samples extracted in the boundary and the centre is significant. In the centre the strain is expected to be lower at prescribed loads of the canister, and it is therefore investigated if the requirements on elongation at fracture can also be relaxed at the centre.

Table 8-1. Values for elongation at fracture for a BWR insert that was cast in May 2015.

Height	1	6	10	15	2	5	11	14	3	4	12	13	Average
Bottom	23.2	22.4	22.0	22.7	20.5	20.9	19.7	20.7	22.8	21.3	20.9	22.9	21.7
25 %	22.5	22.7	23.7	17.5	19.5	20.8	20.4	20.0	21.8	21.1	21.8	11.4	20.3
Middle	18.9	18.5	22.8	14.7	19.8	17.3	20.2	18.8	22.4	16.3	14.0	13.8	18.1
75 %	13.0	12.0	13.8	15.3	14.1	16.4	19.4	19.2	11.0	8.2	9.4	10.6	13.5
Top	16.6	12.2	16.8	14.0	17.5	14.5	14.3	15.2	10.4	8.8	12.8	9.0	13.5
Average	**18.3**				**18.5**				**15.5**				**17.4**

Programme

Fulfilment of requirements will be assessed (Odeh and Owen 1980, Table 7) by comparing average values, requirements and standard deviation with tabulated values for the fraction of measurements within a unilateral confidence interval. An analysis of variance of the mechanical values will be carried out.

SKB will further develop the manufacturing technology and clarify the requirements on the steel parts of the insert. For the nodular cast iron, how microstructure with respect to perlite content, nodule size etc. should be covered by manufacturing requirements is investigated. SKB intends to continue its work with formulating the manufacturing requirements and further developing SKB's strategy to verify the manufacturing requirements for the initial state of the inserts. SKB plans to update the set of requirements for non-destructive testing of the insert (Källbom et al. 2014) and link the defects more clearly to the final selected methods for manufacturing. In order to refine the casting process and reduce the variations in mechanical properties, SKB has commissioned research at the Division of Casting of Metals at KTH to explain how microstructure and defect distribution are affected by chemical composition and convection during casting. In particular, the convection for bottom and top pouring in relation to the inoculation procedure is being studied. SKB continues research at KTH's Department of Solid Mechanics on the impact of graphite on nodular cast iron's properties and behaviour and is further developing constitutive models for the nodular cast iron. These models will be used to simulate the mechanical load of the cast iron insert with FE programs like Abaqus. SKB also intends to determine how qualification of insert manufacturing can be carried out.

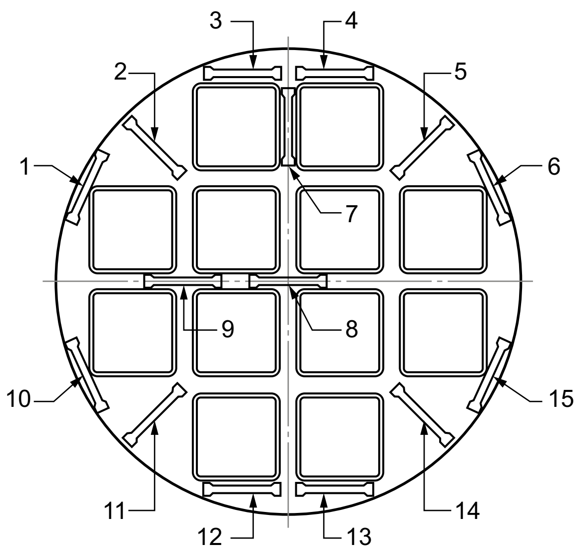


Figure 8-3. *Sample extraction in the insert cross section.*

8.4.3 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.

FSW joins materials in the solid state, which differentiates the process from classical fusion welding. By letting a probe rotate in the material, friction heat is generated. The increasing temperature makes the material softer so that, if the temperature is sufficiently high, it starts to move with the probe. 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 2010c).

Current situation

After the RD&D Programme 2013, development has focused on both industrialisation and automatisation of the welding process, as well as investigation of oxide bands in the weld metal. The work is being conducted together with Posiva since 2014.

During the period, the work with oxide bands has had the purpose of describing and explaining the formation of oxide particles in the weld. Studies and modelling of copper oxidation kinetics have been carried out (Björck and Elger 2013). These models have been applied to the welding process for a better understanding and for calculation of maximum oxygen concentrations in the gas shield (Björck 2015). A Masters' degree project (Pehkonen 2014) resulted in a proposal for a new design of the gas shield based on the experiences of the gas shields presented to SSM in 2014 (SKB 2014h).

When it comes to automatisation, the cascade controller (which alters the rotational speed of the welding tool) has been improved so that the tool temperature around the whole joint line varies by only ± 5 °C. The controller has also been developed so that it will function even if the tool temperature signal vanishes.

Since the RD&D Programme 2013, the radial extent of the joint line hooking has been evaluated with regard to the variables that can have an impact: the length of the probe, weld depth and the probe's relative height position to the joint line. The results show that with a probe length of 51 mm and a centred height position (tolerance 2 mm), the joint line hooking can be limited to below 2 mm.

In order to reduce the oxide bands in the welded metal, a new gas shield for the Canister Laboratory weld has been built according to Pehkonen (2014). This provides a complete gas shield around the whole joint line throughout the welding sequence. The evaluation of the gas shield was done by three 360 degree welds. The first sequence had shielding gas, argon, on the outside of the joint line; the other had shielding gas, nitrogen, also on the inside. The third sequence had the same configuration as the second but with 2 atomic per cent of hydrogen mixed in the shielding gases. Results were evaluated by measuring the oxygen content in the weld, performing heat treatment in the hydrogen atmosphere according to ASTM B577 and metallographic investigation. The evaluation showed that the highest oxygen content was below 3 weight ppm in all welds, which is below the specification for the base material of 5 weight ppm. Heat treatment in the hydrogen atmosphere was compared according to the scale from ASTM F68. All welds were found to have a level 1/C, which is the lowest level. None of the three welds exhibited signs of hydrogen embrittlement. A report with the above results will be published in 2016. The report discusses and substantiates the cleaning method for the joint surfaces.

In order to further reduce joint line hooking, since the RD&D Programme 2013 an additional controller has been developed and commissioned, a so-called depth controller which alters the force with which the welding tool presses against the canister to control the weld depth. The depth controller also has the purpose of minimising the formation of flash which can disturb the process, for example in the overlap sequence.

Preliminary results show that the weld depth can be controlled well with the current settings. Additional experiments must however be carried out to investigate how for example different manufacturing methods for lids and tubes affect the weld depth and/or the settings for the depth controller.

Programme

SKB and Posiva will continue to deepen the understanding of the welding process stability generally through systematic studies with respect to disturbances in the process and incoming components such as manufacturing tolerances between lid and tube and when the joint surfaces are not completely in contact with each other.

8.4.4 Inspection and testing

Current situation

SKB has developed a combined set of requirements, including acceptance criteria for the canister and its components. SSM has requested information on the impact on testing methodology of properties and variations of defects in order for SKB to ensure that the canister components can be inspected (SSM 2014b).

Within the framework of SKB's supplement to the application in the licensing process, SKB reported the current status for non-destructive testing of the canister's components and welds. The reporting includes integrated complete description (Ronneteg and Grybäck 2015) of a developed preliminary set of requirements for non-destructive testing together with a description of the inspection technique developed with associated analyses in the form of both simulations and reliability studies. Results are also presented from initial analyses of inspectability for the canister components with ultrasound. Furthermore, the reporting covers a proposal for the strategy which the process for qualification of non-destructive testing of the canister should have (Ronneteg and Grybäck 2014).

As a result of the reliability of ultrasonic inspection of the insert's central parts being questioned in SSM's preliminary review report from 2015 (SSM 2015), a study has been initiated with the aim of applying radiographic inspection techniques to these areas. Preliminary results of this study indicate good opportunities to obtain sufficient reliability for inspection of these areas, especially considering that the acceptance requirements for these areas can probably be relaxed, see Section 8.3.

Programme

SKB plans to conduct studies and development regarding the inspection programme for the canister. As a basis 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. Other guidelines and standards (IAEA etc.) can 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 a product shall be obtained for example. Within the system it must then also be studied how the inspection programme activities in general should be reported and documented, since this is a part of quality assurance.

SKB intends to proceed with finding detailed acceptance criteria for non-destructive testing through further analyses of possible and likely defects and link these to the calculations made regarding handling of the canister in fabrication and deposition, and the canister's long-term 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 developing supplementary ultrasonic technique for inspection of the insert's volume and evaluation of alternative techniques for surface inspection of the insert. In addition to the developed testing method, the possibility to apply radiographic inspection methods for inspection of the insert's central parts is also evaluated. In addition, work has been initiated with a focus on development of eddy-current technique for surface inspection of the copper components.

Due to the fact that radiographic inspection has proved to be necessary in order to ensure reliable inspection of FSW welds, whether alternative detector concepts can provide better detection capacity is evaluated.

Figure 8-4. *Ultrasonic technique is one of several techniques for non-destructive testing.*

The development of nondestructive testing will also be more specifically focused on technique for sizing linked to the defects that are expected to occur in the canister's components and welds and the acceptance criteria that are formulated. In addition to the inspection technique developed for final inspection, a review is planned of the inspection currently carried out of starting material such as copper ingots. Based on this review, possible technique development will be initiated.

To substantiate the developed technique for non-destructive testing, further studies are conducted of possible variations (regarding material, defects and geometry) in the canister's components and welds and how they can affect the canister's inspectability. Based on these studies, possible requirements on for example sound attenuation and surface finish will be defined. To further substantiate the inspection technique, work continues on ultrasonic simulation of both the inspection technique and its response linked to both artificial and real defects.

For other inspections of the canister, for example regarding dimensions, a revision of tolerances will be carried out. Based on this revision, possible development of technique for verification of these requirements will be initiated.

In the encapsulation plant, the weld quality will be inspected by ultrasonic and radiographic inspection, but since oxides cannot be detected with these methods, the FSW welding process needs to be qualified and criteria will be formulated on the maximum permissible oxygen content in the gas shield.

9 Cementitious materials

In SFR large amounts of cementitious material are found in waste matrices, engineered barriers and structures. The concept for SFL that is currently being safety evaluated also includes large amounts of cementitious materials. Finally, the Spent Fuel Repository will also contain a certain amount of cementitious materials, mainly in the form of low-pH concrete in plugs, grouting and rock support.

Chapter 5 briefly presents SKB's planned development programme linked to the design of repository structures, material development and production procedures for construction and closure. Further, a general description of the programme for research on processes of importance for the post-closure safety of the systems was given. This chapter provides a more detailed description of SKB's programme in these areas. The programme for research to gain a better understanding of processes is presented as a whole (Section 9.1), while the programme for design, material and production methods is described separately for the three repositories, SFR (Section 9.2), SFL (Section 9.3) and the Spent Fuel Repository (Section 9.4).

9.1 Cementitious materials – development after closure

This section describes the scientific research SKB plans to conduct to gain a better understanding of how the function of cementitious materials in the repository environment changes during the time periods that the safety assessments cover.

9.1.1 Groundwater impact

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 alters the concrete's pore structure, which affects the hydraulic and mechanical properties of the material. Mass transport properties in the concrete matrix also change.

Current situation

SKB has in the preceding RD&D period worked to deepen process understanding of degradation of concrete under repository conditions, both by means of modelling and through experimental studies.

For SR-PSU, models have been developed with a refined geometric representation of repository parts and engineered barriers. These have been used to calculate groundwater flows through SFR (Abarca et al. 2013) and, linked to this, concrete degradation (Höglund 2014). The impact of fractures in the barriers has also been evaluated. This work led to the ability to evaluate the concrete barriers' hydraulic, chemical and mass-transport properties as a function of time.

A development project has been carried out in order to obtain a calculation tool which improves the prospects for simulation of coupled processes (Nardi et al. 2014). This is relevant for modelling of the interaction between concrete and groundwater, concrete and waste and concrete and other materials.

In order to gain a better understanding of how cementitious materials are affected by their interactions with groundwater and solutes in the groundwater over very long times, SKB is participating in the project Longterm Cement Studies, LCS. The project is being led by Nagra in cooperation with a number of international organisations and includes both new experiments and extensive modelling work of previous completed studies and experiments. Through the combination of experiments and modelling work, a better understanding of the relevant processes and their rates can be obtained and modelling tools improved.

SKB has during the period 2010–2015 financed a PhD project at Chalmers University of Technology. The main purpose of the project has been to study experimentally the chemical, physical and mechanical properties of artificially aged cement based materials. Within the project an electrochemical leaching method was developed with which samples of aged concrete, relevant for the timescales appropriate for the assessment of post-closure safety, could be manufactured in a reasonable timescale (Babaahmadi 2015). The properties of the samples were then studied with a large number of methods, with a focus on chemical composition, material structure and mechanical and physical properties.

Programme

SKB plans to develop its programme for studies of interactions between groundwater and concrete under repository conditions. The focus will be on modelling of coupled processes. Methods and calculation tools developed during the preceding RD&D period will now be applied.

The properties of the concrete backfill in the rock vault for core components in SFL after closure will be evaluated through a number of modelling activities. A first phase focuses on hydrochemical modelling of concrete leaching caused by incoming groundwater. In a second phase model development is planned in order to include mechanical degradation processes. The purpose is to describe the concrete's physical and chemical properties over time for further analyses of post-closure safety in the safety evaluation for SFL.

Modelling studies have also been initiated to gain a better understanding of the relationship between the timescale for accelerated leaching experiments (Babaahmadi 2015) and the natural leaching process under repository conditions. The work aims further to correlate modelled changes in the concrete's microstructure, caused by leaching processes, to macroscopic properties of aged concrete samples, which for example describe the material strength.

9.1.2 Modelling of gas transport

The gas that is formed by corrosion of metals and microbial activity during the time after closure can build up an internal pressure if a well-functioning system for gas transport is not available. The pressure can affect the structural integrity of the concrete structures. In order to avoid detrimental loads on the barriers, gas therefore needs to be able to be transported out of the waste domain.

Current situation

With simplified models for metal corrosion and gas transport, an analysis of how gas production may affect the structural integrity of concrete barriers has been conducted (Eriksson et al. 2015). According to the analysis, it is the corrosion of aluminium and zinc in the waste that causes 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).

Programme

New analyses will be carried out of how gas flow through the grout-embedded waste in SFR affects the barriers. Gas transport through the concrete structure in the 2BMA repository part will be modelled with a two-phase flow model. Development of technical solutions for gas transport through the concrete barriers in SFR is handled within the relevant technology development programme, see Section 9.2.3.

Within the framework of the safety evaluation for SFL, modelling studies have been started to improve process understanding of transport of gas through the system of concrete and granite. An initial phase comprises modelling of two-phase flow through the barriers in the waste vault for core components.

In the safety evaluation for SFL, further research needs pertaining to the outward transport of gas from SFL will be identified. The programme will be described there.

9.1.3 Impact of degradation of organic waste

Organic materials that degrade in a cement matrix can affect the properties of the cementitious materials. This can be reflected in a change in the composition of the pore water, but also in changes in the composition of cement minerals. In addition, the organic degradation products affect the concrete's ability to limit releases of radionuclides, see Section 6.6.

Current situation

How degradation products from different types of material, representative for low- and intermediate-level waste, interact with cement minerals and how this affects the concrete is being studied, among other things, in the Concrete and Clay project, which has been pursued by SKB in the Äspö HRL since 2010 (Mårtensson 2015). One of the main goals of the project is to qualitatively develop an understanding of how degradation products spread in a cement matrix, and to what extent new minerals are formed in these processes.

During 2014 samples were retrieved including steel containers with groundwater from repository depth, organic matter representative of low- and intermediate-level waste as well as a small quantity of crushed cement paste (Wold 2014). Analysis of the samples showed that degradation of organic waste has been very limited and no or only very low amounts of degradation products could be identified.

Programme

SKB plans to retrieve and analyse additional samples from the Concrete and Clay project during this RD&D period. This is in order to study degradation of organic material and interaction of degradation products with the cement matrix. Primarily samples similar to those analysed by Wold (2014) will be retrieved, see Figure 9-1 left-hand picture. If the degradation products can be identified, concrete cylinders with material representative of low- and intermediate-level waste will also be retrieved for analysis, see Figure 9-1 right-hand picture. A first investigation of steel containers with organic material is planned at the end of 2016.

9.1.4 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. For example, aluminium's ability to be included in the cement gel (C-S-H gel) and form calcium aluminium silicate hydrate (CASH) can be mentioned here (L'Hôpital et al. 2015). In addition, the mechanical pressure that occurs if voluminous corrosion products congregate on or around metal surfaces can lead to fracturing in the matrix.

Current situation

In the Concrete and Clay project which was mentioned in Section 9.1.3, also studies of corrosion of metallic material in a cementitious material and how the corrosion products interact with the cement matrix are being carried out. During 2015 a specimen (Figure 9-2 right-hand picture) containing metallic material samples representative of low- and intermediate-level waste materials has been retrieved and analysed (Kalinowski 2015).


Figure 9-1. *Steel containers containing material samples, groundwater from the Äspö HRL and crushed cement (left-hand picture) and a concrete cylinder with material samples (right-hand picture).*

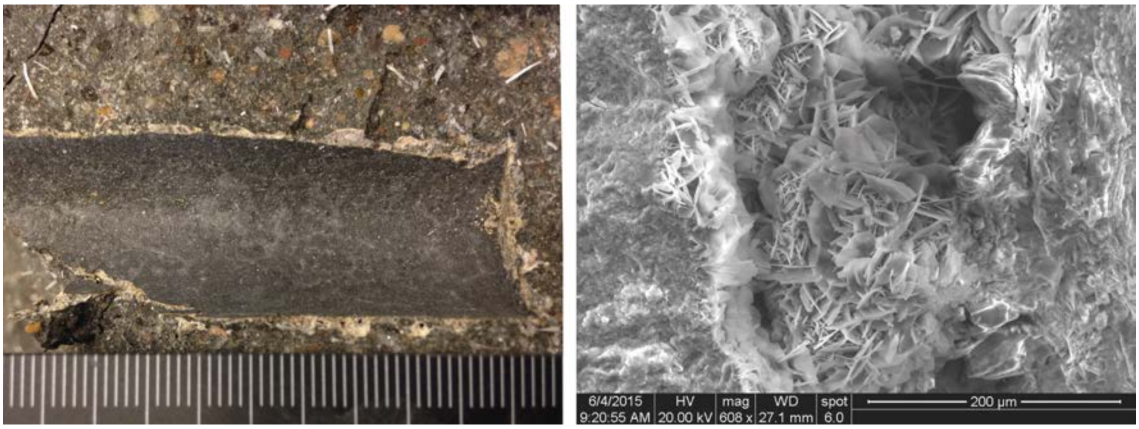


Figure 9-2. *Aluminium bar embedded in concrete with a clear white transformation zone in the interface (left-hand picture) and a pore in the interface which is partially filled with secondary minerals (Kalinowski 2015).*

The study showed that the corrosion products of iron, steel, aluminium and zinc during the five years in which the samples were deposited in boreholes in the Äspö HRL have a very limited spread in the cement matrix. For aluminium and zinc, elevated concentrations could be detected up to on average 300 µm and 1 200 µm from the interface and a crust of corrosion products was also visible to the naked eye, see Figure 9-2 left-hand picture. No elevated concentrations could, however, be detected around the steel samples. For aluminium, pores in the interface between metal samples and cement matrix were also noted. In all probability, these formed in connection with the fabrication of the samples when hydrogen gas evolved from a reaction between the still wet concrete and the aluminium sample. The analysis showed that these pores are partially filled with secondary minerals, Figure 9-2 right-hand picture. X-ray diffraction (XRD) showed the presence of mainly metal hydroxides and for the aluminium sample also calcium aluminosilicates. For further reading concerning corrosion rates for aluminium and zinc, see Section 6.7.

Programme

In the Concrete and Clay project, SKB intends, in around 2020, five years after the retrieval described above, to retrieve and analyse an additional specimen of concrete with metallic material similar to that discussed above. Through this analysis, a second point on the line of how corrosion products spread in the cement matrix over long periods of time is obtained, which could lead to a better understanding of this process. See also Section 6.7.

9.1.5 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 a time after closure become water-saturated, the chemical interactions can lead to changes of the cement material's composition, properties and structure. This process can also include degradation products from the waste.

The influence of the leachate from cementitious materials on the properties of the bentonite in the Spent Fuel Repository has been studied previously and further studies are planned, see Section 10.3.4.

Current situation

SKB has, in a number of previous works, studied interactions between bentonite and cementitious materials and their impact on post-closure safety 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.

SKB also deposited a large number of samples within the Concrete and Clay project in the Äspö HRL in 2014 for studies of interactions between different types of bentonite and cement paste consisting of pure Portland cement as well as of low-pH paste, see Figure 9-3. A major fraction of cement and concrete specimens also contained powder of different metals and metal salts, representative of low- and intermediate-level waste, but also samples without such materials were used (Mårtensson 2015).

During 2015, SKB also initiated work within the international research project Cebama (www.cebama.eu) within which interactions between different types of cementitious materials and adjoining materials such as clays or bedrock materials of various types are being studied. The project, in which around twenty different research institutes and ten different waste organisations are active, will continue until 2019.

Programme

The first retrieval of the experiment, which is focused on studies of interactions between cementitious materials and bentonite within Concrete and Clay, will occur at the earliest around 2019. The focus will be analysis of ion transport and mineral alterations in the interfaces between cement and bentonite as well as dispersion of degradation products from the materials that are mixed into the small specimens. Samples are then planned to be retrieved and analysed periodically and the project is expected to be concluded in 2025 at the earliest with retrieval and analysis of the last samples.

9.1.6 Impact of mineral additives

In the manufacturing of cementitious materials, in addition to cement, water and aggregate, different types of mineral additives can also be used. These materials – for example silica, finely ground limestone or fly ash – can be added either in cement production or in mixing of the cementitious material. The purpose of these additives can be to reduce the environmental impact of the material as well as to control the properties of fresh and hardened material, but there may also be economic reasons. Regardless of the reason for the use of additives, they will not only affect the composition and properties of the fresh and hardened material, but can also affect how its properties change after closure.

Figure 9-3. *Installation of a package with 30 bentonite blocks. Each block also contains four smaller material samples.*

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 Degerhamn Anläggningscement. This is a pure so-called Portland cement without additives, corresponding to cement type CEM I, which was used in the construction of the concrete structures in the existing repository parts in SFR.

As the use of different mineral additives increases in both manufacturing of cement and mixing of concrete, the need to gain a better understanding of how the altered composition affects the cementitious materials' long-term properties also increases.

Programme

During this RD&D period, SKB plans to initiate modelling studies to evaluate properties after closure of cementitious materials with altered binder composition. These studies will provide a better understanding of how the use of future materials can affect the safety of the repositories after closure.

9.1.7 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 can become so great that fractures arise or the material completely falls apart. The temperature at which this occurs is mainly dependent on the pore structure and of 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).

In Emborg et al. (2007) and Thorsell (2013) it is shown 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 concrete that freezes in a volume that does not permit volume expansion corresponding to an environment that can be expected to prevail in conjunction with a permafrost (Figure 9-4) 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 times that are relevant in the deep 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 for the first permafrost, probably will not correspond to the properties that the investigated materials exhibit in the studies referenced above.

Within the PhD project funded by SKB at Chalmers University of Technology, certain preliminary and non-complete studies have been carried out of the freezing properties of concrete that has been leached in order to represent aged concrete. The studies showed that the amount of freezable water was higher in the leached material than in the reference material. Moreover, the studies showed that the impact of leaching was stronger for the capillary pores than the gel pores (Babaahmadi 2015).

Programme

SKB plans to carry out in 2016 a directed study of the freezing properties of concrete leached to different calcium silicon ratios with the method developed by Babaahmadi (2015). By comparing the samples' porosity with the porosity estimated in the modelling work that SKB has carried out through the years, see for example Höglund (2014), an idea of the samples' artificial age can be obtained and a link to SKB's expected climate evolution can be made.

Figure 9-4. Experimental set-up used by Pålbrink and Rydman (2013). The strong steel container contains a water-saturated gravel bed in which a concrete cylinder has been installed. The steel cylinder corresponds here to the bedrock while the gravel bed and concrete cylinder correspond to the backfill and concrete structure in 1BMA. Note the cylinder lid with inlets for cords from temperature gauges attached to the sample.

9.1.8 Internal and external loads

The concrete structures will during construction and operation and in conjunction with closure and backfill be exposed to loads which could lead to the occurrence of fractures. Apart from this, the concrete structures are likely to be exposed to loads during the period after closure caused by for example gas production, swelling waste or solid corrosion products on the waste packages which put pressure on the grout. To estimate the integrity of concrete structures during the operating period, the risk of fractures needs to be analysed and calculated.

Current situation

SKB has previously studied the strength of the concrete structures in the repository. The silo outer walls have proved to be resilient to both internal and external loads (von Schenck and Bultmark 2014).

Programme

In order to gain a better understanding of how changes in the concrete affect the concrete structures during construction and operation, SKB will conduct cracking risk calculations and investigate limits for fracturing and crack growth in concrete structures. The analyses provides an opportunity to compare calculated limit load cases with other loads that may occur as a result of events such as rock breakout or gas pressure from within.

9.2 Design of concrete structures and materials in SFR

This section presents SKB's programme for design of concrete structures, material development and production methodology for SFR.

9.2.1 Waste vault for intermediate-level waste

The waste in 2BMA will be embedded in a cementitious grout. The purpose of this grouting is for example to stabilise the waste and create a support for the concrete structures in conjunction with backfill and resaturation of the repository. Moreover, the grout limits the water flow through the waste domain and contributes to the high pH being maintained over a long time, at the same time as it permits transport of the gases that can be formed by degradation of the waste in the repository.

The concrete structures in 2BMA, according to the reference design in the application for the extension of SFR, will consist of independent caissons, see Figure 9-5. These will be built of unreinforced concrete by slip-form construction. With this as a basis, a programme for design of material and technology for construction has been formulated. As a part of the further development and refinement of the reference design, detailed design analyses, load calculations and verifying tests will be performed, see also Section 9.1.8.

Current situation

During 2014/2015 grout for 2BMA was developed. The requirements that apply today for the silo grout were used as a starting point with the addition that organic additives, for example cellulose, may not be used. The development work showed that it was possible to produce a stable cementitious grout that met requirements on its properties by using a bentonite slurry as stabiliser (Lagerlund 2015b).

During 2015, a plan for further development and detailing of concrete structures for 2BMA has been developed. The plan is comprised of a stepwise verification and validation, where materials, design and production methodology are discussed.

Since early 2015 a programme for development of a structural concrete for the caissons in 2BMA has been operated. The starting point in the design of the concrete is that the aggregate will consist of 100 percent crushed material, which is fabricated of excavated rock spoil from the extension of SFR, and that a minimum of additives will be used. As a basis for the development work, SKB has investigated the composition and properties of rock spoil from blasting of the current facility (Lagerblad et al. 2015). Based on this investigation, suitable quarries with similar materials in the vicinity of SFR have been identified. These are used now in the development of the structural concrete for 2BMA. The work is being conducted at present with the objective to identify a concrete recipe that fulfils the formal requirements on workability and strength. Moreover, great importance is placed on the recipe being stable against changes in the availability of and properties of the concrete's components that can be expected over the time period when the caissons will be built.

During 2015 studies and planning for demonstration tests began for the engineered barrier described in the application for the extension of SFR. This is one step in the verification and validation of the concrete structure and production methodology for 2BMA. The studies show that demonstration tests should be performed and that a suitable site for casting is the Äspö HRL.

Programme

SKB intends to restart development work on a cementitious grout for 2BMA, with a focus on optimisation of the properties of the grout during upscaling.

The programme for the further material development of structural concrete to the caissons in 2BMA includes over the coming years the following steps:

- Continued development of material in the concrete laboratory.
- Verifying upscaling tests on a concrete station.
- Test casting of a representative section of a caisson under repository-like conditions in Äspö HRL.

The two first steps will be carried out partly in parallel to permit quick feedback if the results in upscaling experiments at the concrete station deviate considerably from those obtained during development in the concrete laboratory.

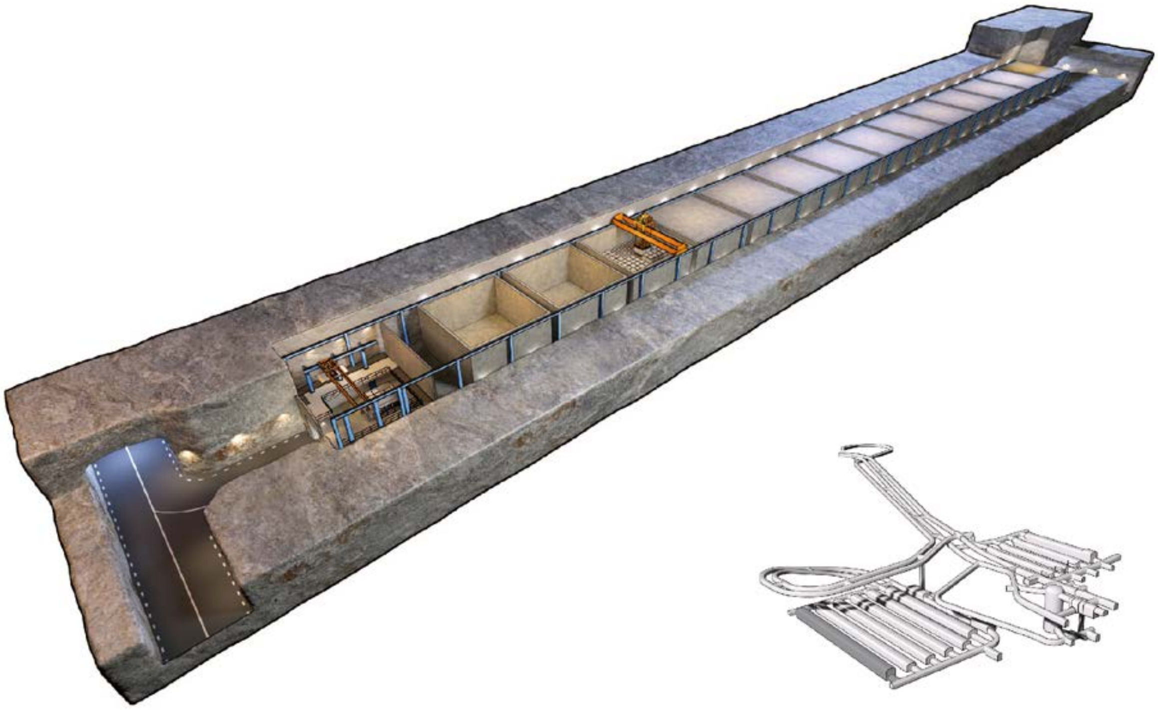


Figure 9-5. 2BMA – rock vault for intermediate-level waste in the extended part of SFR.

When a final recipe has been designed and verified at a concrete station, test casting is performed of a representative section of a concrete caisson in the Äspö HRL in order to verify the recipe developed. In the casting these components are instrumented in such a way that it will be possible to follow up changes in dimension, temperature and stress state.

The possibility of fabricating repository structures in 2BMA in several steps/cast stages and using joint seals to seal the joints between the different parts will also be investigated. In addition, the effects of the properties of the foundation on the occurrence of fractures in the concrete structure are being studied.

Test casting of the representative section is currently planned for the end of 2016 but the time point is dependent on the progress made in the material development programme.

SKB's programme for further work with the development of the concrete structures and the production method for 2BMA includes over the coming years the following activities:

- Analysis, calculations and design of the structure details (such as the casting joints and the impact of external load).
- Further development of production methodology for construction.
- Verifying tests of design and production technology.
- Preparation of operational and maintenance programme.

The results from the development of the concrete structure for 2BMA give an in-depth knowledge concerning the concrete caissons and barrier system in 2BMA and are judged to be available for the detailed design of the extension of SFR.

9.2.2 Waste vault for reactor pressure vessels

In the waste vault for reactor pressure vessels, BRT, nine whole BWR pressure vessels will be placed, see Figure 9-6. In conjunction with closure, the RPVs will be embedded in a cementitious material that constitutes a barrier after closure. The reason for this is for example to create a passive environment which limits corrosion on the outside of the RPV. In addition, grouting contributes with sorption capacity for radionuclides in the cement matrix.

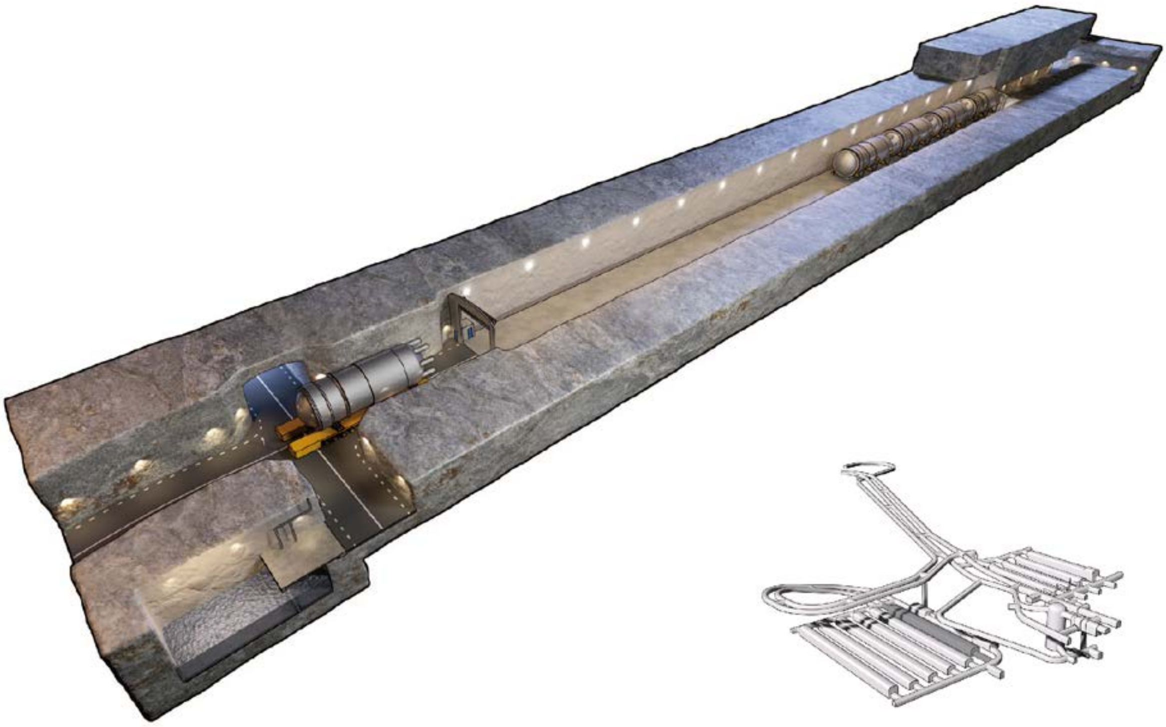


Figure 9-6. BRT – waste vault for reactor pressure vessels in the extended part of SFR.

Before the RPVs are embedded, they will be partially or completely filled with a cementitious material, referred to here as internal grouting. This is carried out partly to bind any loose contamination, partly to contribute to the sorption of radionuclides. Moreover, internal grouting creates a passivating environment in the RPV and thereby reduces the corrosion rate on the inside of the RPV. Finally, an internal grouting prevents the RPV from floating up during the subsequent grouting and prevents any future collapse of the tank.

Current situation

In SKB's Closure plan for SFR (Luterkort et al. 2014) grouting of the RPVs is described in general terms. According to the plan, concrete walls will be erected between the RPVs and the rock vault walls. Prior to closure of the repository the volume between these walls and the RPVs is filled with a grouting material. The Closure plan includes no detailed plan for how internal grouting will be carried out or any description of what types of material should be used.

Programme

SKB intends to carry out a research and investigation programme for development of material and production methodology for internal grouting in RPVs which comprises the following three main areas:

- Studying and clarifying how the RPVs will be prepared so that internal grouting can be carried out in a safe manner without risk for injury to individuals or dose exposure.
- Studying and clarifying the set of requirements for internal grouting with regard to material properties, how this is linked to the choice of method and how large a portion of the RPV must be filled so that the desired function is obtained.
- Development of material and production methodology for internal grouting.

SKB also intends to carry out a research and investigation programme for development of material and production methodology for grouting of the RPVs which comprises the following three main areas:

- Studying and clarifying the set of requirements for grouting regarding the choice of material properties and methodology.

• Production methodology for construction. The work aims at developing a method for grouting that fulfils the set of requirements and whose execution takes place in a safe manner without risk of injury to individuals or dose exposure.

• Development of material and production methodology for grouting.

Results from these studies are judged to need to be available at the latest in conjunction with the updated safety analysis report prior to trial operation of the SFR extension.

9.2.3 System for gas transport

During the long time periods considered in a safety assessment, waste, waste containers and any reinforcement will degrade with expected gas formation as a result. To facilitate the gas to be transported through the different repository parts, a system for gas transport is needed. A technical design for such a system has been proposed for the silo but not yet tested and is lacking entirely for 1BMA and 2BMA.

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 materials will be cast on top of the silo, see Figure 9-7 (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 materials (Moreno et al. 2001, Höglund and Bengtsson 1991). Höglund and Bengtsson (1991) show that the probability of precipitation of secondary minerals in the sand filled pipes is very low and that their function would not be threatened during the first 5 000 years after closure. Further studies concerning their function over a longer time frame have not been performed.

The Closure plan for SFR (Luterkort et al. 2014) includes no detailed technical solution for how systems for gas transport in 1BMA and 2BMA will be designed or which types of materials should be used.

Programme

SKB will update the analysis of the long-term function of the planned system for gas transport in the silo. The basis for these studies is the relevant gas formation rates from degradation of organic waste and corrosion of metallic material occurring in the silo as well as information on mineral alterations in the system components.

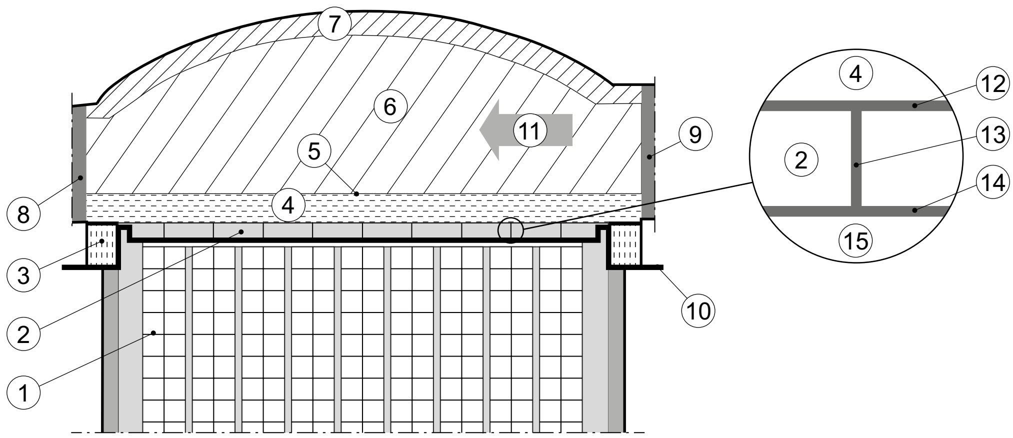


Figure 9-7. *Backfill material in the silo top. 1) Embedded waste. 2) Reinforced concrete, one metre. 3) Compacted filling of bentonite mixture 30/70 bentonite/sand. 4) Compacted filling of bentonite mixture 10/90 bentonite/sand, 1.5 metres. 5) Unreinforced concrete slab. 6) Compacted filling of friction material. 7) Cement-stabilised sand. 8) Confining wall of concrete. 9) Confining wall of concrete. 10) Boundary between works belonging to grouting and backfilling. 11) Work direction for backfilling with material. 12) Sand layer, 100 millimetres. 13) Gas evacuation pipes with diameter 0.1 metre. 14) Sand layer 50 millimetres. 15) Grout (permeable). (Luterkort et al. 2014.)*

The programme for the assessment of the long-term function of the system, which is not yet designed in its entirety, needs to include analyses of the systems assumed in the current closure plan (Luterkort et al. 2014), but alternative systems for gas transport may also need to be considered.

During this RD&D period, development work will be carried out for the design of a system for gas transport in 1BMA and 2BMA. Calculations of gas/water flow through the structure as a function of the amount of waste and gas-producing processes need to be carried out as a basis for this development work, see further Section 9.1.2.

An analysed system for gas transport in 1BMA and 2BMA is judged to need to be available at the latest in conjunction with the updated safety analysis report prior to construction of the SFR extension.

9.3 Design of concrete structures and materials for SFL

9.3.1 Rock vault for core components

In the rock vault for core components, BHK, the post-closure safety of the repository part is based on the waste being surrounded on all sides by a thick layer of concrete, see Section 2.1.2. In order to ensure post-closure safety for the whole time period covered by a safety assessment, materials and methodology for installation of materials must be chosen carefully.

Current situation

At present, SKB is conducting a study concerning the choice of material and methodology for foundation engineering of concrete structures and backfill of BHK. A number of different methods are being considered and relevant aspects related to safety after closure, technical feasibility and costs are being highlighted.

At present, SKB has no development work of concrete for the concrete structure in BHK. The assessment is that the concrete that is currently being developed for the caissons in 2BMA will also be used in SFL. SKB is waiting therefore with an individual development programme until the development project for 2BMA has been evaluated.

Programme

SKB intends during the period for SFL's safety evaluation to finish the investigation concerning the choice of material and methodology for foundation engineering of the concrete structure and backfill of the rock vault with concrete.

In addition, SKB is also planning during this RD&D period to start an investigation concerning the effect of the properties of the backfill material on water flows and radionuclide transport in BHK. The results from this study will serve as a basis for an assessment of whether previously developed cementitious materials can satisfy the requirements for BHK.

Only when the conclusions from these studies have been carefully evaluated does SKB plan to – if the necessity is shown to exist – formulate a programme for experimental development of materials and techniques for foundation engineering of the concrete structure and backfill of BHK with a cementitious material.

9.3.2 Grouting of waste packages

In conjunction with closure of the two rock vaults in SFL, the waste will be embedded in a cementitious grout. The purpose of this grouting is to stabilise the waste and create a support for the concrete structure in conjunction with backfill and resaturation of the repository and to restrict advective flow of water around the waste packages.

Current situation

During 2012–2014 SKB carried out a number of studies and development works on grout for 1BMA and 2BMA (Lagerlund et al. 2014, Lagerlund 2014, 2015a, b) which will serve as material for development of a grout for BHK and BHA.

Programme

In conjunction with the safety evaluation for SFL, SKB plans to evaluate previous development work on grout for repository parts in SFR. The purpose is to clarify whether the set of requirements on the grout for BHA and BHK can be met by any of the existing grouts.

9.4 Design of concrete structures and materials for the Spent Fuel Repository

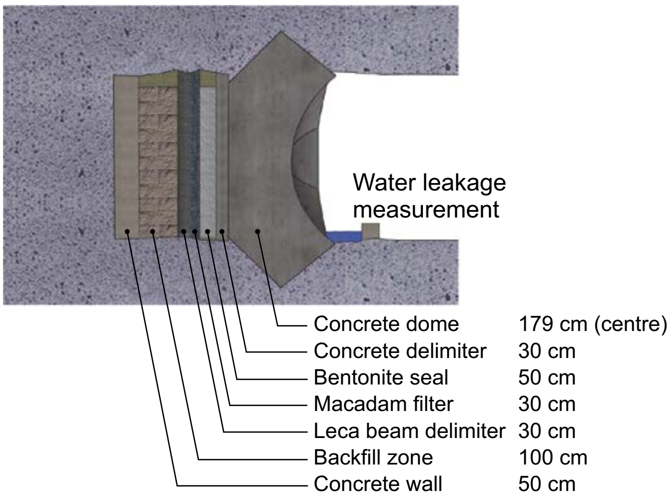
9.4.1 Plugs for deposition tunnels

Current situation

The development and verification of plugs for closure of the deposition tunnels in the Spent Fuel Repository has been carried out with analytical and numerical calculations, laboratory experiments, scale tests and by 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).

Important conclusions from the construction phase of the full-scale test were that it was possible to construct the plug system in an appropriate manner and that it was possible to build an unreinforced concrete dome of the low-pH concrete mix design called B200 that meets performance requirements. A graphical illustration of the test and a picture of the installed plug are shown in Figure 9-8.

At the end of September 2014, the measured leakage past the plug was 44 mL/min (corresponding to 2.6 litres per hour). The leakage declined exponentially after it arose and has now, after two years, decreased to 18 mL/min. The test also shows that the water pressure that acts on the concrete dome gradually declines despite the fact that the water pressure on the upstream side of the bentonite seal is constant. This fact shows together with sensor data for relative humidity and total pressure that the bentonite seal becomes more and more homogenised and successively become more water tight.



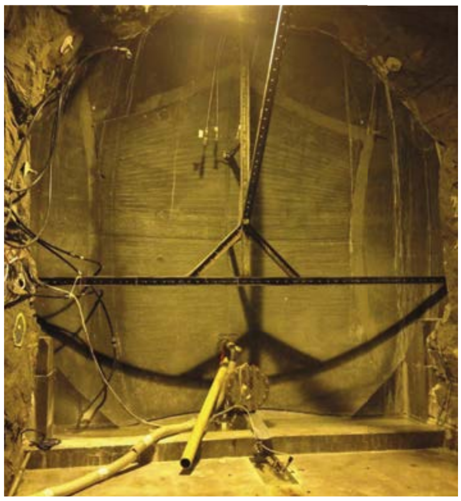


***Figure 9-8.** Schematic section and picture of the installed plug.*

Programme

The plug that is installed in the Äspö HRL, after a completed monitoring period of three years, will be excavated and its function evaluated. In conjunction with excavation, the interfaces between rock, concrete and bentonite will be investigated. Material sampling and measurement of the plug system components will then also be carried out to verify calculation models and measurement data. The results of the evaluation will serve as a basis for further development work.

From a technical point of view, it was difficult to force the unreinforced concrete dome to completely detach from the rock in connection with cooling prior to contact grouting. This was caused partly by adhesion between low-pH concrete and the smooth rock surface being very good, and partly by the shrinkage of the concrete being lower than expected. Additional calculations and tests may be needed in the detailed design phase to verify these sequences.

Further studies are being conducted in order to expand the recipe for low-pH concrete to a concept where the goal is to abandon the requirement that a specific named product must be used and instead state clear requirements on the fundamental properties of constituent products. This will lead to less testing and verification being needed when additives etc. are removed from the market or when a specified type of cement is no longer available. Work is also being pursued for development and delivery of the requirements made on the layout of a concrete station, based on the requirements from fabrication of low-pH materials and storage and preparation of their constituents. Finally a study is also being conducted with a focus on developing industrially adapted methods for fabrication and inspection of cementitious low-pH materials.

9.4.2 Low-pH cement materials for grouting and rock support

Current situation

Current design premises and requirements on the Spent Fuel Repository assume use of low-pH cement materials for grouting and rock reinforcement. SKB has previously commissioned development of a low-pH cement material for grouting (Bodén and Sievänen 2005) and rock support (Bodén and Pettersson 2011). Construction, operation and closure of the Spent Fuel Repository covers long periods of time, which means that the availability of components in the current mix design will change considerably and some products will be completely omitted over time. For this reason it is of importance that the recipe be designed in such a way that the properties of the materials will not be completely dependent on a specific product and that constituent components can be replaced with products with similar properties.

By expanding the mix design for low-pH cement materials, materials can be made more robust to future changes in availability. This is done now in such a way that instead of a specific recipe with stipulated product names, a concept with clear requirements for the basic properties of constituent components is stated. This also means that the need for testing and verification is significantly reduced when constituent components that are no longer on the market are replaced with similar products.

Programme

During this RD&D period, an update will be carried out of the recipes for low-pH injection grout for rock sealing, low-pH materials for shotcrete and rock bolt grout which are robust to changes in the availability of constituent components. In addition, a production-adapted process will be developed for use of low-pH materials for construction and operation of the Spent Fuel Repository in Forsmark. Furthermore, inspection methods and maintenance needs during the operating period, will be developed.

10 Buffer, backfill and closure

The main purpose of the clay barriers, i.e. buffer and backfill in the Spent Fuel Repository, the silo fill in SFR and the clay barrier in the rock vault for legacy waste in SFL (BHA), is to restrict water flow around the canister and the waste packages. This is achieved by a 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 is one of the engineered barriers in the Spent Fuel Repository and consists of the material installed in boreholes and in all underground openings except deposition tunnels. In the Spent Fuel Repository, the closure and plugs shall maintain the principle of multiple barriers through preventing the formation of conductive water paths between the repository area and the ground surface and preventing the backfill from expanding out of the deposition tunnels. The closure should also keep in place the closure in underground openings located adjacent and below. In the upper part of the ramp and shafts, the closure shall considerably impede unintentional intrusion into the repository. The installed closure's hydraulic conductivity is the property that has the greatest importance for its barrier function.

10.1 Evolution of the bentonite material after installation until saturation

In order to more extensively understand and describe the bentonite material's evolution until saturation, efforts are required with respect to piping erosion, swelling, homogenisation of blocks, pellets and cavities, water vapour circulation and microbial sulphide formation under unsaturated conditions.

10.1.1 Piping/erosion

A hydraulic problem during the operating phase concerns piping and related erosion effects in the buffer and the backfill. The inflow of water to the deposition holes that is required for wetting of the buffer will take place mainly through fractures in the surrounding rock. If the inflow is localised to fractures that provide water at a more rapid rate than that 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. The continued evolution is then determined by the bentonite swelling rate, the flow rate through the buffer and the buffer's erosion rate.

Current situation

Results from earlier studies have been published in Börgesson et al. (2015a). The goal has been to understand and develop models for critical processes that occur in an early phase after the installation of buffer and canister, such as piping, erosion, water uptake in pellet-filled gaps and early water absorption as well as to create opportunities for stipulating requirements on the tightness of the end plugs in the deposition tunnels.

The studies have only treated processes that take place during a period until the water flow has stagnated and full water pressure has formed against the plug. The following processes in the bentonite have been studied:

- Erosion.
- Piping.
- Water flow in pellet-filled gaps.
- Ability to stop piping.

- Water absorption by the bentonite blocks.
- Formation of water- or gel-filled pockets in pellet-filled gaps.
- Outflow of bentonite gel.
- Self-sealing of fractures with erosion water.
- Buffer upswelling before backfilling.
- Self-healing of erosion pipes.

The analyses have led to a number of preliminary conclusions regarding possible damage that can affect the buffer and backfill:

- Erosion follows the empirical expression that was used in the SR-Site safety assessment.
- Piping and associated erosion arises and is maintained until the water pressure gradient is above the plug instead of the backfill and the flow rate out from the backfill through the plug is lower than 10^{-4} L/min. The value of this flow rate is a result of process studies showing that the flow channels self-heal and the piping and erosion cease at lower flow rates.
- Self-healing of fractures in the plug or rock cannot be expected to occur under all circumstances, which means that it cannot be credited when post-closure safety is assessed. This is a pessimistic assumption based on the fact that it has not been possible to demonstrate that this function is always active.
- Erosion channels of limited radial extent (1–2 cm) will self-heal to such an extent that they do not have a significant influence on the hydraulic properties of the bentonite when stagnant water pressure conditions are achieved.
- Water- or gel-filled pockets may form at low inflow rates (See Figure 10-1).

Piping in the silo in SFR has been analysed in Börgesson et al. (2015b). If the entire inflow to the silo is pessimistically assumed to come from one location and the erosion model from SR-Site is used, the mass loss from erosion will be 5–500 kg. The wide range is because it cannot be predicted whether the channel will be horizontal or vertical. With the assumption that the entire mass loss occurs locally, the resulting swelling pressure in the inflow point will be low, but most of the bentonite in the silo will, however, be completely unaffected. The conclusion is therefore that piping and erosion is not a problem in the silo.

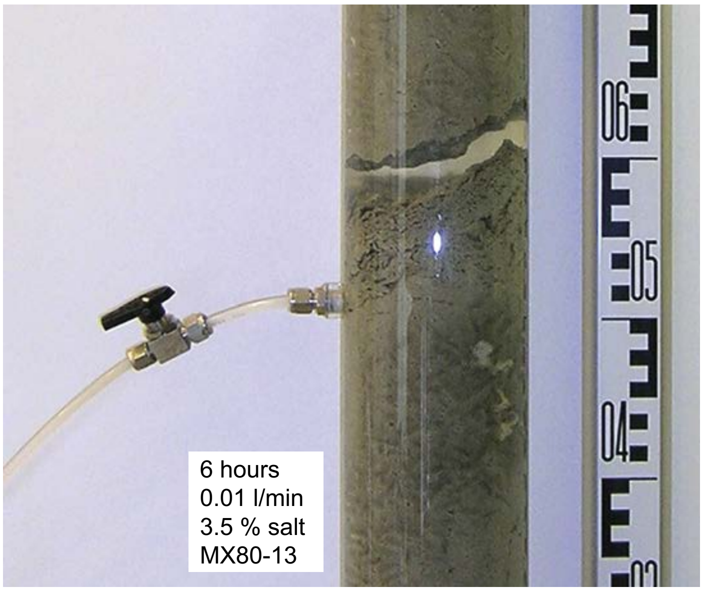


***Figure 10-1.** Examples of formation of a pocket in an experiment with a pellet tube from project Åskar (Börgesson et al. 2015a).*

Programme

There is a need to understand and model water transport in pellet fills, partly in order to describe water uptake correctly, and partly to plan the measures that are needed for water handling during installation of the backfill in the tunnels in the Spent Fuel Repository. Today there are no models that can be used in simulations of such water transport in the large range of water inflows that are expected in a final repository. A set of experiments where different inflow rates and temperature gradients are applied will be conducted. These experiments will be used both to test existing models and to develop and verify new models.

There are preliminary observations that indicate that piping in the buffer can be bounded or even stopped with a flange in the wall of the deposition hole. SKB intends to carry out additional studies to see whether the problem can be limited by relatively simple means.

10.1.2 Water uptake in the buffer

When the buffer blocks and the pellet fill have been installed in a deposition hole in the Spent Fuel Repository, the buffer will absorb water from the surrounding rock. During the saturation phase, the buffer will develop a swelling pressure that exerts a mechanical force on the rock, the canister and the backfill. Water transport in the unsaturated buffer is a complicated process that is dependent on, among other things, temperature, density, montmorillonite content and water ratio in the different parts of the buffer. The most important driving force for achieving water saturation is the relative humidity in the buffer, which can be regarded as a negative capillary pressure in the buffer pores leading to water uptake from the rock. The hydraulic conditions in the rock surrounding the deposition hole determine the course of the saturation process. With an unlimited supply of water, full water saturation will be achieved between canister and rock within a few years.

Current situation

SKB has had further developed the description of the resaturation process in the buffer in the rock in Forsmark, in response to SSM's requirements on supplements to the licence application for the Spent Fuel Repository (SKB 2013c). The main issues treated were:

- Conceptual model uncertainty in connection with calculations of resaturation and homogenisation.
- Analysis and discussion of possible distributions of resaturation times for the buffer considering local hydrogeological conditions at repository depth in Forsmark.
- Security importance of the degree of saturation and the possible impact of heterogeneous conditions in the buffer.
- Thermal impact on the bentonite's mechanical and hydraulic material properties.
- The possibility to, with adjustments of the degree of saturation or artificial supply of water, affect/shorten the saturation process.

In the supporting material for SR-Site (Åkesson et al. 2010), 10^{-13} m/s was assigned as a "typical" conductivity in the rock matrix. With this value, according to Figure 10-2, 65 percent of the deposition holes in Forsmark would be saturated in ~2 000 years and most of the others in the range of 300–2 000 years. Only a small fraction are saturated in less than 100 years.

Water uptake in the outer section of the Prototype Repository in the Äspö HRL has been modelled within the framework of the international cooperation for comparison of calculation tools (Task Force EBS). The modelling funded by SKB is presented in Svemar et al. (2016). The main purposes of the modelling were:

- To predict the hydraulic state in the outer section of the Prototype Repository at opening and retrieval.
- To describe the THM processes (thermo-hydro-mechanical processes) during operation of the Prototype Repository.

Secondary goals were:

- To increase the understanding of the modelling of wetting of the buffer and backfilling through the rock.
- To increase the understanding of the significance of different types of water transport, local or distributed, for wetting.
- To validate models and finite element solvers.
- To increase the understanding of THM processes in the system.

Figure 10-3 shows an example of results from modelling of a case with the assumption of high local flows and low flow in the rock matrix and different initial conditions in the buffer. Results from the retrieval showed that the buffer rings were completely water-saturated and only the buffer directly below and on top of the canister was partly unsaturated. This shows that the combination of assumptions on an initially homogenised buffer together with a low matrix flow is wrong. However, both cases of an "installed buffer", i.e. a representation with blocks, rings, pellets and gaps and both cases with high flow in the rock matrix, gave a good representation of the final water saturation. This shows that it is necessary to verify both models and assumptions regarding initial state against relevant experiments.

Water uptake in the silo in SFR has been modelled (Börgesson et al. 2015b) in order to estimate the time from closure to full water saturation in the silo repository, describe the saturation process in the system and study how variations in the system representation alter the saturation process.

For the base case, the silo repository was calculated to be fully saturated in the range between 13 and 53 years. The largest uncertainty in the estimated time comes from how the rock is represented with respect to using undrained/drained or dry conditions. The properties of the silo content and top fill also had a significant effect on the total saturation range. The conclusion is therefore that the bentonite in the silo will saturate relatively quickly when the drainage is shut off.

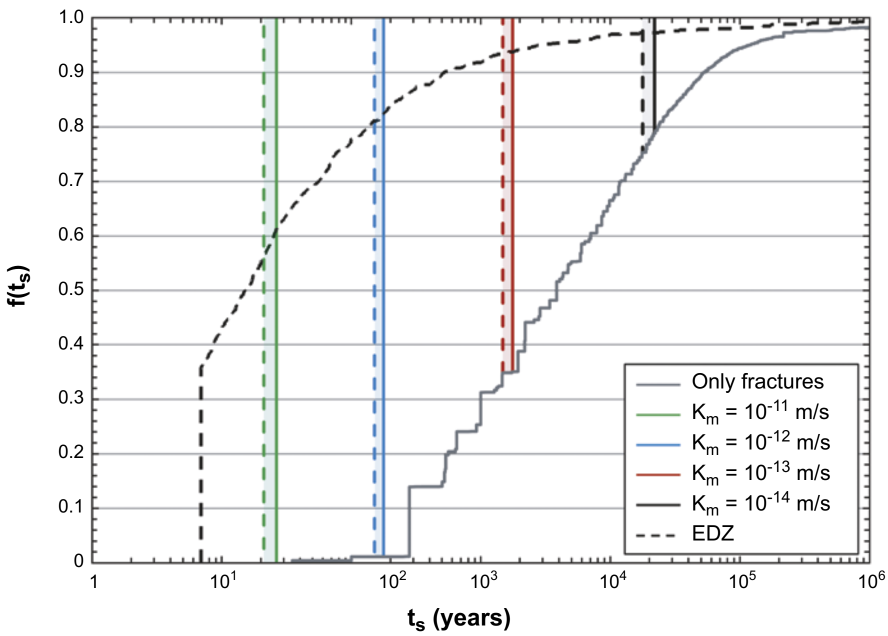


Figure 10-2. *The solid grey line shows the cumulative distribution of water saturation times for the buffer in Forsmark with the assumption that the rock matrix is tight. The coloured lines show the range when all deposition holes are fully saturated for different assumptions on the hydraulic conductivity in the rock matrix. The dashed line shows the distribution of water saturation times for a case with a completely hypothetical highly conductive EDZ in the deposition tunnel (SKB 2013c).*

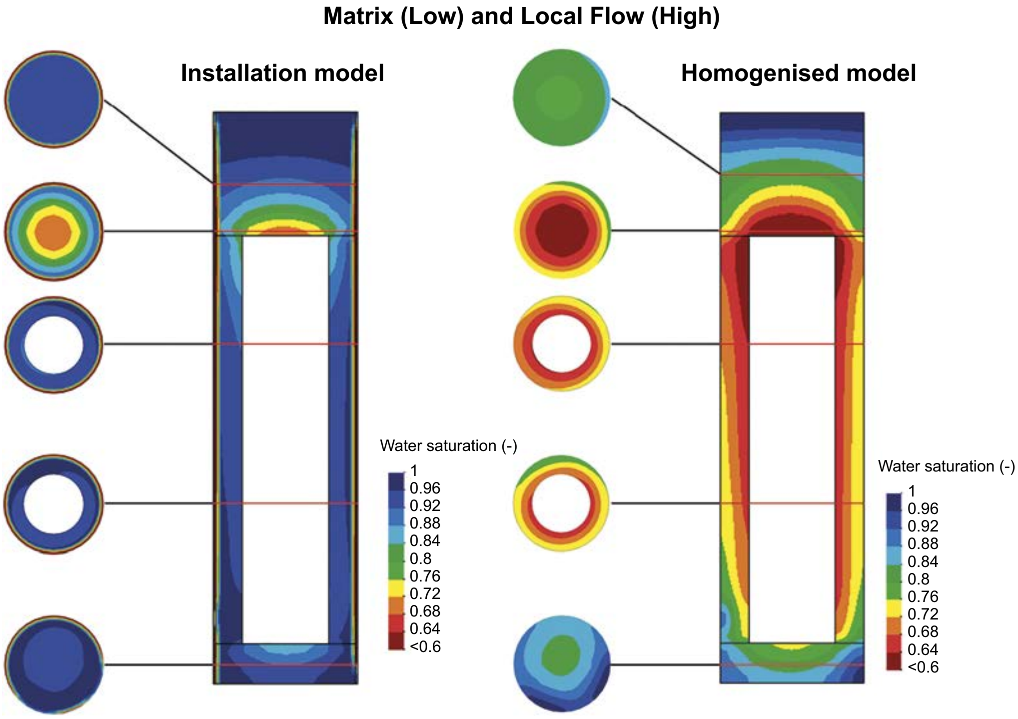


Figure 10-3. *Final water saturation in the models with low matrix flow and high local flows. The buffer is far from fully saturated, in particular in the homogenised model. Furthermore, the saturation was strongly non-axisymmetric, due to the local inflow (Svemar et al. 2016).*

Programme

The work of verifying and updating the models for water saturation of the buffer continues. Most of this work is being done within Task Force EBS. An important issue is to be able to handle a slow water uptake, either in a discrete fracture or through the whole rock matrix and determine how it will affect the saturation process and water distribution in buffer and backfill.

Figure 10-2 and SKB (2013c) show that the hydraulic conductivity of the intact rock matrix is of crucial importance for the buffer's water saturation time in a repository in Forsmark. Uncertainties in data and possible variability in the rock will have a great impact on the distribution of water saturation times. It is therefore necessary to develop statistical material for the hydraulic conductivity of the rock matrix. During the detailed characterisation phase, sufficient data for the hydraulic conductivity of the rock matrix will therefore be collected by performing laboratory measurements on samples from drill cores. Furthermore, the possibility of performing different types of in situ tests will be evaluated.

The water supply has implications for how bentonite blocks and pellets homogenise. This determines the final state for both buffer and backfill in the Spent Fuel Repository and for the bentonite barrier in BHA in SFL. This is therefore an important parameter in the studies of homogenisation, which are described in Section 10.1.3.

The long times for resaturation of buffer and backfill are not expected to have any major impact on the performance of the barriers after saturation. It may, however, be important to understand how the chemical environment is affected during the unsaturated period. The intention is to commence these studies in a laboratory environment, with similar equipment to that used for water vapour circulation (see Section 10.1.4). This may possibly be combined with geochemical modelling of the gas phase composition.

10.1.3 Swelling, homogenisation and self-healing

Water uptake after deposition will lead to swelling in the buffer and backfill in the Spent Fuel Repository, which are inhomogeneous at emplacement. This causes all gaps in the buffer, between rock and buffer and between 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 residual inhomogeneity is of importance for the design premises and the configuration (pellets and blocks) with which the buffer is deposited. 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 the pellet fill and for healing of erosion channels, but swelling pressures and compression properties are also important for e.g. buffer upswelling and impact on the plug. The ability of the backfill to prevent buffer upswelling is discussed in Section 10.4.2.

The same issues are relevant in BHA in SFL, where the large volume of bentonite can lead to further questions concerning upscaling of laboratory tests to a real scale.

Current situation

SKB is pursuing a laboratory programme to study swelling and homogenisation. The main purpose of the programme is to provide results that can be used for modelling of some well-defined performance tests to improve the models or to determine mechanical parameters for the thermo-hydro-mechanical modelling of buffer construction. Results from the tests will be reported continuously (Dueck et al. 2014). The test series include radial and axial swelling, axial swelling in long tubes and self-healing tests. An example of results from a swelling test is shown in Figures 10-4 and 10-5. In the test a 40-millimetre-high bentonite sample was placed in a test cell that was 50 millimetres high. Water was supplied and the bentonite could swell upwards in the axial direction. Figure 10-5 shows the axial swelling pressure, as well as the radial swelling pressure on different heights from the bottom of the sample. The pressure increases rapidly in the lower part of the sample (red and blue), but then decreases when the bentonite begins to expand. Initially, there is a water-filled void in the upper part and thus no swelling pressure, but when the bentonite expands, the pressure increases. Pressure equilibrium is, however, not reached during the time of the test and probably not in a longer-term perspective either. This example demonstrates that inhomogeneous barriers must be handled both in design and in the assessment of post-closure safety. The hope is, however, that it will be possible to show that the given design provides residual inhomogeneities so limited that the assumption of a completely homogeneous barrier can be defended.

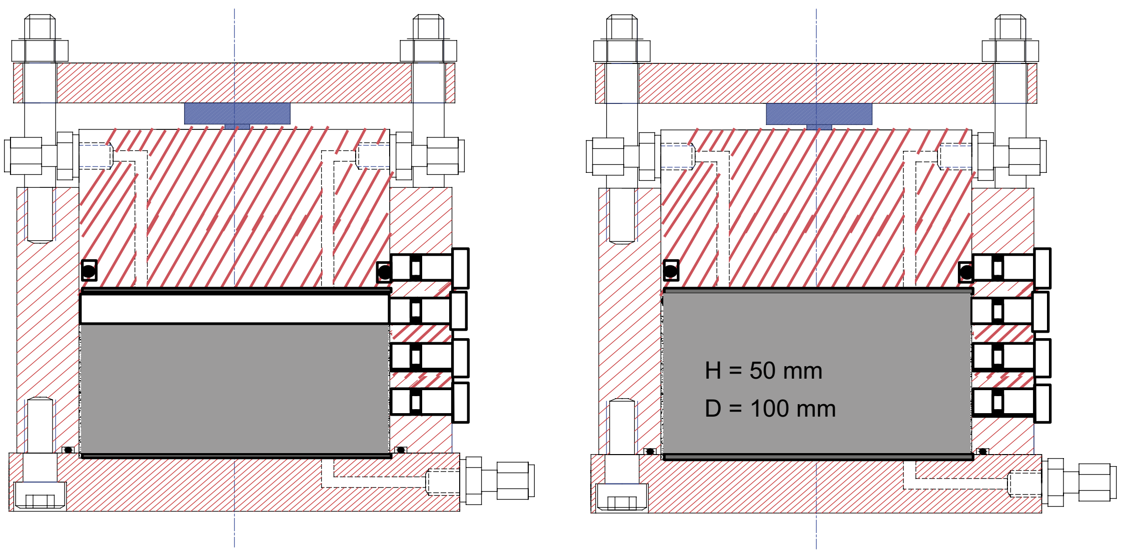


***Figure 10-4.** The experimental set-up used for axial swelling tests (HR-A). Water is supplied only from a filter placed above the sample (Dueck et al. 2014).*

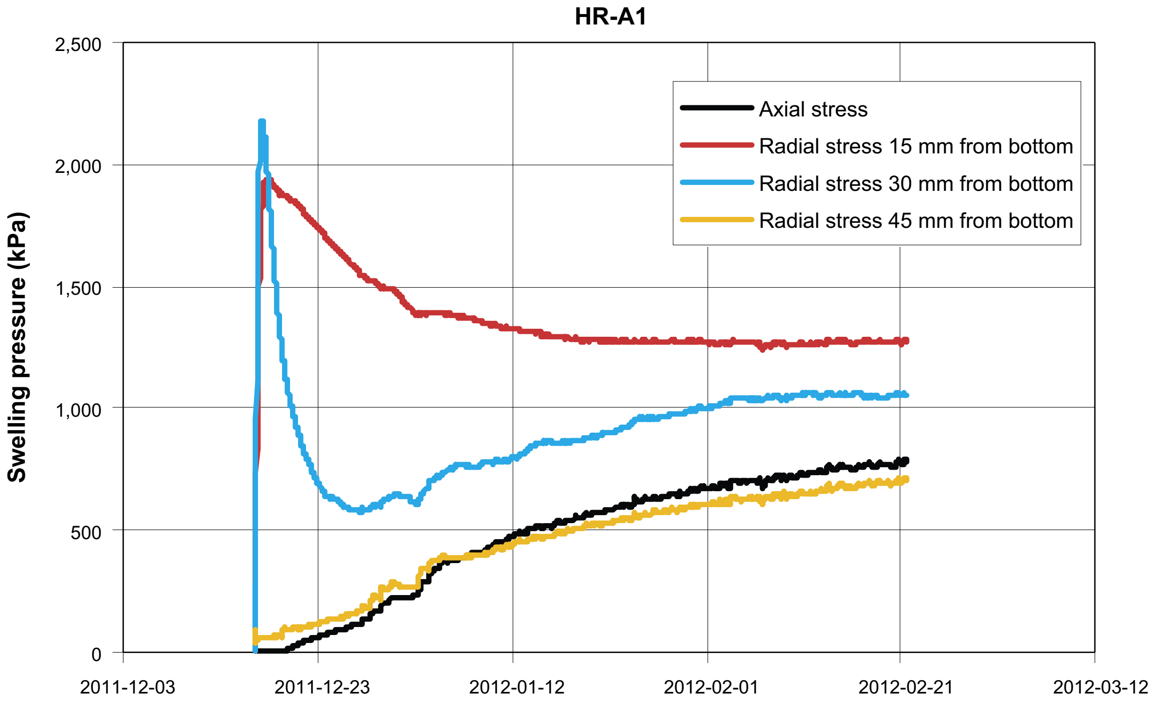


Figure 10-5. *Temporal evolution of swelling pressure from sample HR-A1. In this test the original bentonite was 40 millimetres high with a dry density of 1 666 kg/m and swelled out to 50 mm (Dueck et al. 2014).*

Modelling of a part of the tests is being conducted within Task Force EBS. The results show that there are difficulties in modelling the expansion of bentonite. There are deficiencies in both our fundamental understanding and in the numerical solutions.

Programme

The programme of laboratory studies will continue to shed light on swelling and homogenisation in the case of slow water supply. The modelling in Task Force EBS will also continue. There, one of the goals is to model self-healing in a test case where a piece of bentonite has been sawn off from a block (Figure 10-6). This case has proved to be difficult to handle, especially numerically, and further work is needed.

The question of how the installed bentonite barrier is homogenised is common for all waste programmes where bentonite is used in buffer, backfill or plugs. In all cases blocks and/or pellets are installed together with voids that are necessary for installation. In order for the barrier to obtain the desired properties, the homogenisation must be sufficiently efficient. In order to demonstrate this, both a fundamental understanding and predictive models for the process are needed. There is therefore an initiative for a new EU project which will focus on homogenisation of bentonite (Beacon). The hope is that the project can start in 2017.

10.1.4 Water vapour circulation

Questions have arisen as to whether water from rock fractures can be vaporised against the canister and be transported out into the backfill, thereby causing salt enrichment against the canister which, if extensive, could possibly cause corrosion (the sauna effect, see Figure 10-7). 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, and an experimental programme to verify this is under way.

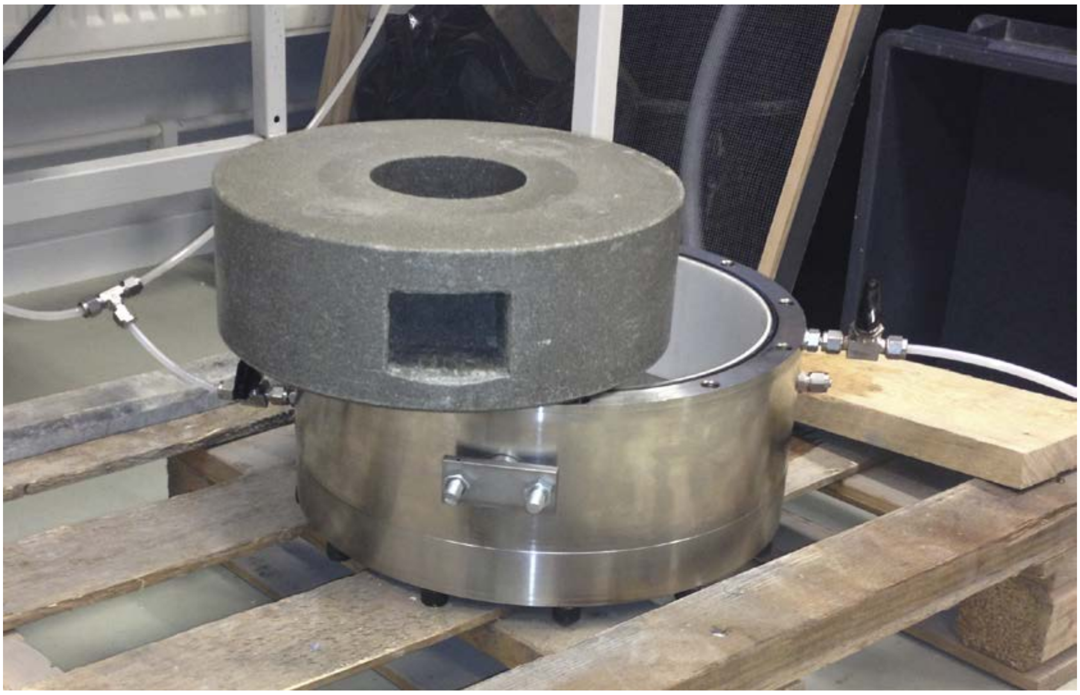


Figure 10-6. Sawn-out hole in the side of a bentonite block. This experiment is used as a modelling task in Task Force EBS.

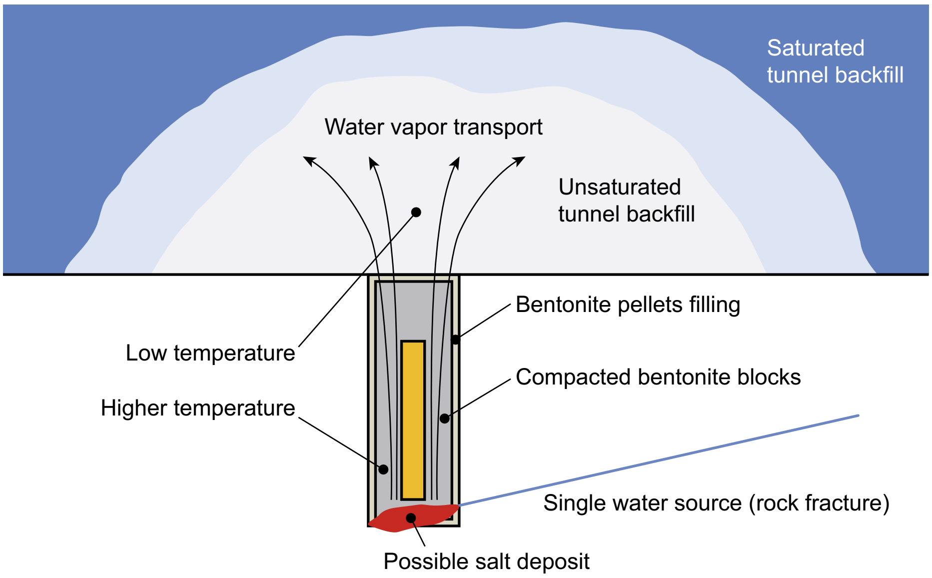


Figure 10-7. Schematic figure of the so-called sauna effect where salt is enriched around the canister when water vapour is transported out of the deposition hole (Birgersson and Goudarzi 2016).

Current situation

A set of tests have been conducted where transport of water vapour in a gap between bentonite blocks and a heater (in the form of a copper tube) has been studied (Figure 10-8). The purpose of these tests is to better understand how water vapour interacts with unsaturated compacted bentonite blocks. An understanding of such processes is essential in order to evaluate the sauna effect. The purpose is to demonstrate that the formed vapour stays in the deposition hole and cannot escape, which could entail an enrichment of salt at the canister.

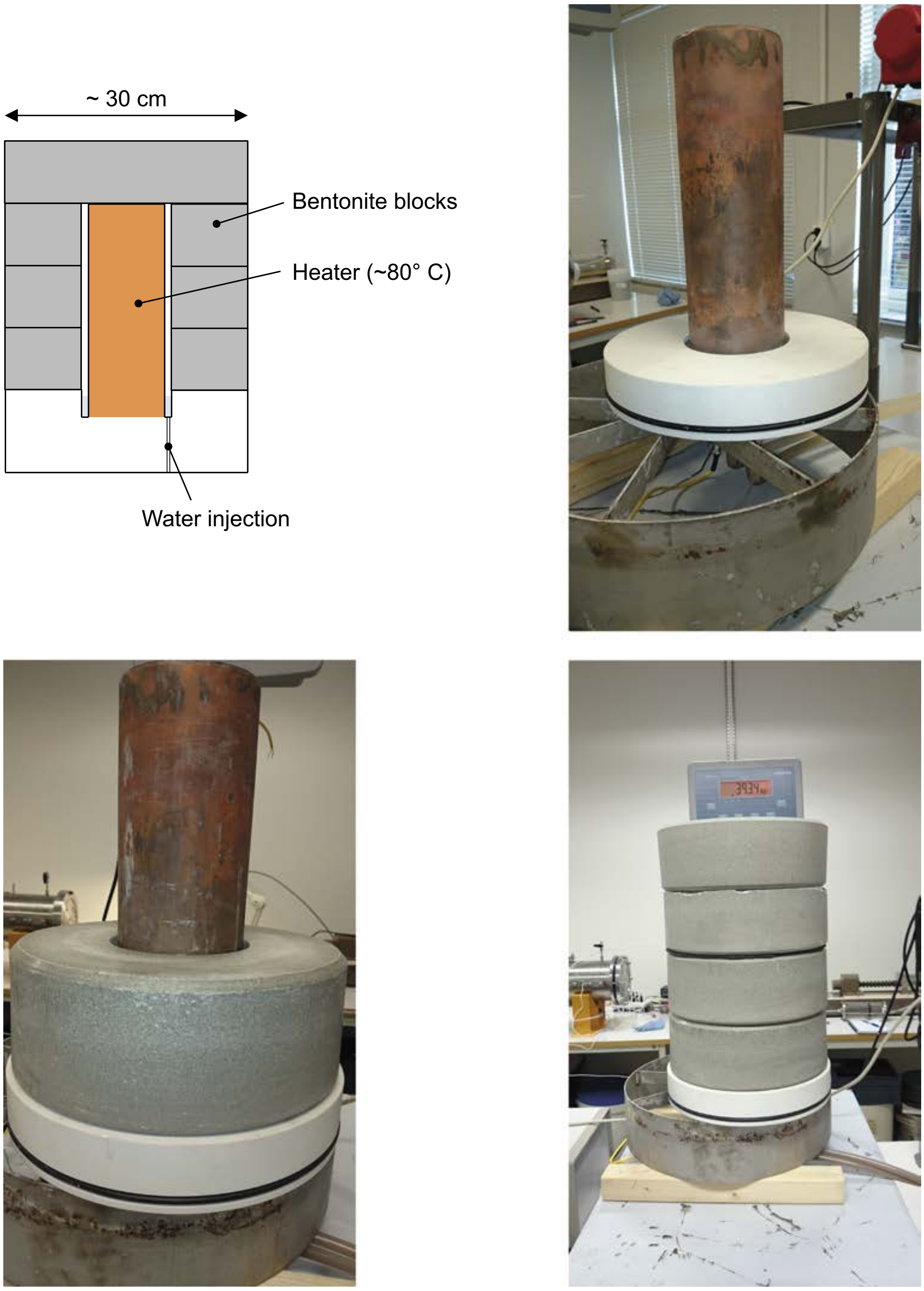


Figure 10-8. Upper left: Schematic illustration of the experimental set-up. Upper right: The heater (copper tube) and the bottom plate. Bottom left: One bentonite ring installed. Bottom right: Three bentonite rings and the massive upper block installed. The entire set-up is mounted on a scale (Birgersson and Goudarzi 2016).

The results show that water vapour can be transported relatively unhindered in the gap between heater and bentonite blocks in those cases where the uppermost block (the lid) was not in place. This is despite the fact that the blocks had relatively low initial water content and a large affinity for absorbing water. The observation is, however, in agreement with results from similar experiments where pellets were used instead of bentonite blocks (Birgersson and Goudarzi 2016).

The experiments also show that the bentonite blocks absorb water by local condensation and not directly from the vapour phase. When the uppermost block (the lid) is in place, water condenses, more or less randomly at different locations in the experiment. These condensation zones then act as a sink and continue to absorb more water. A consequence of the formation of condensation zones is that the bentonite blocks have a tendency to fracture (Figure 10-9).

Figure 10-9. *Example of a block with a forming condensation zone and the fractures caused by it.*

Tests with the uppermost block (the lid) in place show that local condensation effectively hinders water vapour from being transported out of the buffer system. This suggests that vapour transport out of the deposition hole and into the backfill will not occur.

Programme

Results presented in Birgersson and Goudarzi (2016), show that local condensation zones will prevent water vapour from being transported out of a deposition hole. The results also show that water vapour can be transported rapidly in gaps, which in turn means that limited salt enrichment on the canister cannot currently be ruled out. The programme with "sauna tests" will therefore continue and the focus will be on studying salt enrichment and determining what the consequences are when water is supplied through a bottom block of bentonite, instead of a tube, as was the case in the previous tests.

10.1.5 Microbial sulphide formation

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 over 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 in the gas phase is rapid, so in practice there are no transport limitations in an unsaturated repository. The key factor is then how much corrodants are available in the gas phase in tunnels and deposition holes. The amount of oxygen can easily be estimated with the available void volume. Although it is not likely, it cannot be completely ruled out that hydrogen sulphide could be formed by the dissolution of sulphide minerals or microbial sulphate reduction in the unsaturated buffer and backfill. In the buffer this is handled pessimistically by the assumption of mass balance and a requirement on 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. During saturated conditions, the diffusion resistance in the backfill and in the buffer will strongly limit 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 1 percent by weight of organic carbon. This requirement is independent of the shape of the organic carbon.

Current situation

During the last research period, a series of experiments were conducted to investigate whether there is a "water saturation window" in the buffer where microbial activity would be more favoured than in the unsaturated buffer blocks or the saturated buffer. Unfortunately, it was found that the experimental technique was insufficient and no conclusions could be drawn from the experiments. The issue of a water saturation window is, however, not critical since it assumes that sulphate and nutrients are added to the buffer, and if water saturation is slow the water supply will also be slow.

In order to get an idea of the contents of organic materials that can occur in different bentonites and in their form, SKB sent samples to a commercial laboratory for analysis. A summary of the results can be found in Table 10-1. The measurements show that most of the studied materials have very low organic material contents. Due to the low contents it is very difficult to determine the type of organic material. However, it seems as if the type is dependent on the bentonite's origin and age. The detailed analyses suggest that the organic material in Asha clay consists largely of lignite from wood, whereas the origin in the clays Ibeco and GMZ can be algae and degraded algae.

Table 10-1. Total organic carbon and vitrinite reflectance of five different bentonite samples

Material	Total of organic carbon (LECO) (weight-%)	Vitrinite reflectance Mean values (% RO)
MX-80 (USA)	0.13	No grains could be measured
Asha (India)	0.06	0.65
Ibeco Deponit CA-N (Greece)	0.04	0.54
GMZ (China)	0.04	0.52
Calcigel (Germany)	0.05	No grains could be measured

Programme

In the work with the design premises, it has become clear that there must be a requirement on the quantity/concentration of hydrogen sulphide generated by the installed backfill in order to minimise the risk of canister corrosion from gaseous hydrogen sulphide during the unsaturated period. In order to show that this requirement can be met, measurements of hydrogen sulphide in equilibrium with bentonite will be carried out.

SKB intends to send additional bentonite samples for analysis of their organic content.

10.2 The bentonite material properties in the saturated state

The main function of the clay barrier is to limit the water flow around the canister in the Spent Fuel Repository, to restrict water flow around the waste packages in the silo in SFR and in BHA in SFL, as well as to limit 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 is also supposed to hold the canister in place in the deposition hole, dampen the shear movements of the rock and retain its properties during the period being analysed. In addition, the buffer is supposed to limit microbial activity on the canister surface and filter colloidal particles. The buffer must not significantly degrade the function of the other barriers.

The important properties are then hydraulic conductivity, swelling pressure and shear strength. Previously, SKB has tried to relate these properties to the bentonite material's dry density and montmorillonite content. This has been found to be a good relation for a given bentonite material, but is not entirely appropriate for a comparison between different materials. There are other parameters of bentonite that also affect the hydromechanical properties. The new strategy is therefore to prepare unique relationships between hydraulic conductivity, swelling pressure and shear strength as a function of dry density for all bentonite materials that are available for use in final disposal.

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 above all of importance for the buffer in the Spent Fuel Repository.

10.2.1 Material composition

Current situation

A material laboratory has been set up for advanced analysis of bentonite materials on Äspö. One of its purposes is, by analysing samples from the large bentonite shipments which are ordered in SKB's different projects, to learn more about the methods and how the quality of shipments varies. In order to get data that is as representative as possible, improved sampling methodology is needed compared to what was used previously. One method that has been tested is sampling spears and one of the conclusions is that the spear's diameter and the size of the bentonite granules are of great importance for the sampling in purely practical terms. Powder X-ray diffraction (XRD), X-ray fluorescence spectroscopy (XRF) and cation exchange capacity (CEC) have proved to be valuable methods for studying the bentonite clay's variation in composition with respect to crystalline phases, elemental composition and montmorillonite content. In the experiment Alternative Buffer Materials (ABM), a large number of bentonites are analysed by a number of different international teams which acts as a type of benchmark test for each laboratory.

Programme

Continued work will be carried out to gain a better knowledge concerning sampling, analysis and how well the indirect material parameters explored are linked to the more difficult-to-analyse critical properties such as swelling pressure and hydraulic conductivity.

10.2.2 Swelling pressure and hydraulic conductivity

Current situation

In the material laboratory built on Äspö (see Section 10.2.1), nine measurement cells are installed (Figure 10-10) that with large capacity can measure swelling pressure and hydraulic conductivity. The goal is to further improve knowledge concerning different bentonite materials and how they can vary within the same shipment or between different shipments. The measurements take a long time and are rather complex, so the quality control must be supplemented with other methods, that apart from being faster are also closer to those used today by bentonite suppliers. The method developed measures how swelling pressure varies with the density. Measurements are made with both deionised water and with 1 molar of calcium chloride solution to see the maximum and minimum swelling of the clay. Parallel measurements are made in Finland in cooperation with Posiva to ensure the reproducibility of the method and to obtain more statistics on how the materials vary.

Samples have been taken of the bentonite in the silo in SFR and the hydromechanical properties have been studied (Johannesson et al. 2015). Examples of results are measurements of swelling pressure that can be found in Figure 10-11. In general, the study showed that the silo material resembles MX80 in most respects. Swelling pressure is slightly lower and the hydraulic conductivity is slightly higher for the same dry density, but the effect of ion exchange to calcium is also lower, which is an advantage in the silo environment.

Programme

Continued measurements are planned to gain better knowledge of the measurement method, how the properties of different bentonites vary and how they vary within one and the same shipment. Swelling pressure and hydraulic conductivity are measured in an integrated way in the same test. There are indications that imposed external water pressures in some cases can increase the swelling pressure slightly, this will be further studied.

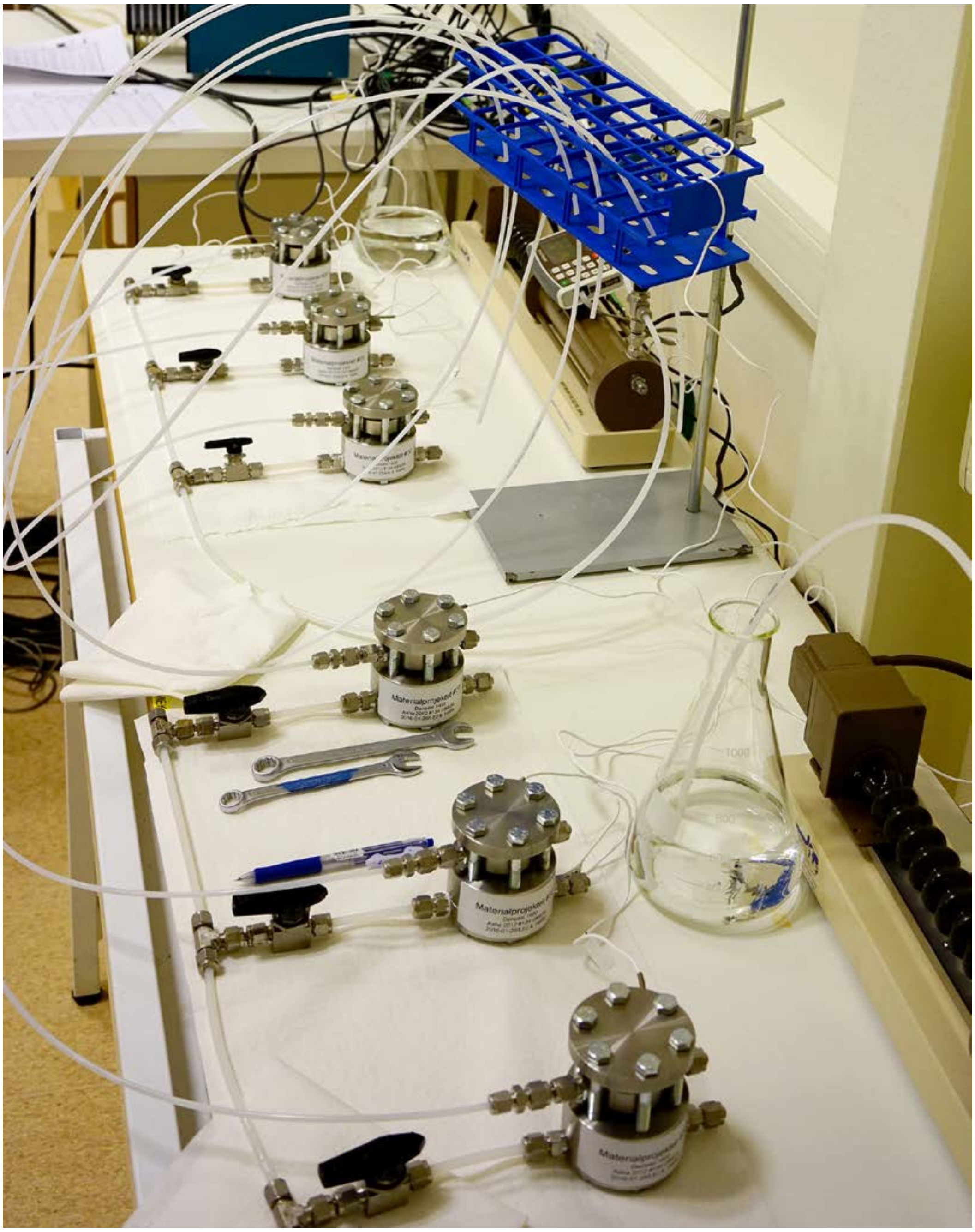


Figure 10-10. *Measurement of swelling pressure and hydraulic conductivity in the material laboratory on Äspö.*

10.2.3 Shear strength

Current situation

Bentonite shear strength is an important parameter for designing the capacity of the canister and for the evaluation of the shear case in the assessment of post-closure safety. The shear strength increases with density and swelling pressure, but the measurements that have been carried out indicate that there are also other parameters that have an influence. Since SKB has defined a technical design requirement for shear strength, it is necessary to test all relevant buffer materials. Development and testing of a standard test method is under way. The main principle is to monitor the uniaxial compressive strength at a given deformation rate and for a dry density corresponding to a swelling pressure of 10 MPa for the material in question. Measurements are conducted for both the material as it is at delivery and a completely calcium-ion-exchanged material.

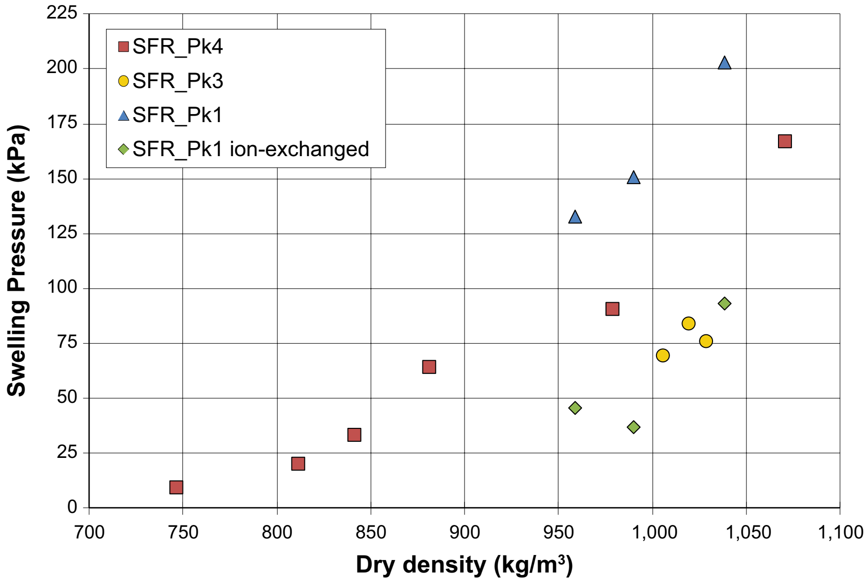


Figure 10-11. *Swelling pressure as a function of dry density of the bentonite material in the silo in SFR. The red points are measured with typical water from SFR (salinity 0.61 percent), the blue are measured in deionised water and the green are measured after a complete calcium ion exchange in a 0.1 molar calcium chloride solution (Johannesson et al. 2015).*

Programme

The limited data that are available indicate that there are larger differences in strength between different materials than previously expected. This means that strength will be an important parameter for the choice of buffer material. The general view from the 1990s onwards has been that an ion exchange to calcium increases the strength. The latest measurements indicate that this might not be the case but that it is how the ion exchange takes place that affects the strength, not the ion exchange itself.

These factors make further measurements of strength necessary, partly to gain further understanding of the process, and partly to obtain more data. It is also important to determine how preparation of the samples affects the results.

10.3 Evolution of the bentonite material after water saturation

When the bentonite has reached water saturation and the desired properties have been achieved, it is important that these properties are maintained during the time the repository in question is expected to function.

For a better understanding of the evolution of the bentonite material after saturation, efforts are required above all on buffer loss due to colloid release/erosion and on microbial sulphide formation and sulphide transport in buffer and backfill. Efforts are also required concerning the long-term stability with regard to temperature, iron content and cement (SFL). Smaller efforts are required on the interaction of bentonite/copper and gas transport. Experiments need to be conducted to verify adequate handling of diffusion in bentonite and transfer of solutes between bentonite and groundwater in the analysis of post-closure safety.

10.3.1 Buffer loss due to colloid release/erosion

The uptake of water and the resulting swelling of the bentonite buffer in the Spent Fuel Repository are hindered by the walls of the deposition hole, and thereby a swelling pressure develops in the bentonite. If fractures intersect the deposition hole, there are no rigid swelling restrictions at 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 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.

In the post-closure safety assessment, SR-Site, colloid release and erosion could not be ruled out and the calculated mass loss was so great in a few deposition holes that advective conditions could not be ruled out.

Current situation

The main part of the research and technology development on bentonite erosion the past period (2012–2016) has been carried out within the EU project BELBAR.

For the erosion process the following was specifically studied:

1. mechanisms of colloid release,
2. the role of divalent cations (clay composition),
3. the effect of mixed monovalent/divalent systems (groundwater chemistry),
4. the dependencies between groundwater velocity and erosion rate,
5. the effect of fracture geometry on mass loss of clay,
6. maximum loss rate of clay.

The composition of both the clay and the groundwater are crucial for the erosion process. Erosion can only occur in groundwater with a composition that is below a certain ionic strength. The highest measured erosion rates always occur in systems with pure sodium montmorillonite and deionised water. The erosion rate decreases with increasing ionic strength. Pure calcium montmorillonite does not erode, but only a small amount of sodium in the system is enough for erosion to start. Natural Febex bentonite, for example, erodes despite the fact that it only contains ~25 percent of sodium in ion exchange position. In horizontal fractures, erosion takes place through dispersive release of particles from the swelled front, whereas in inclined fractures it mainly takes place through structural collapse and sedimentation. In the conducted experiments, the mass loss has been considerably higher in inclined fractures than in vertical fractures. The flow rate in the fractures appears to be of secondary importance for erosion, in particular in comparison with the geochemical conditions (Figure 10-12). In very dilute systems (deionised water), there is a correlation between the flow and erosion, but with increasing ionic strength it becomes very unclear. In vertical fractures, there is no correlation between mass loss and flow.

In SR-Site, extrusion and erosion of bentonite in a fracture intersecting a deposition hole was modelled with an analytical expression based on results from a dynamic expansion model. This work also found that sophisticated finite element methods could not solve the details in a very sharp front in the border region of the expanding gel. A two-region model, which eliminates certain problems in the previous work, has been developed and tested. The model divides the expanding clay in a fracture into two regions: one region where the clay behaves as a Bingham fluid that cannot flow but can expand through the repulsive forces between the bentonite particles, and one border region surrounding the former, where the gel/sol formed can flow and carry away the bentonite particles that diffuse into this region. The two-region model yields results that are more exact than those previously given by the fully-coupled dynamic expansion model thanks to a much higher dissolution in the region where erosion loss occurs.

Sample calculations show that there are two great differences between the previous results and those obtained through the new two-region model. The latter predicts considerably less loss of bentonite to the flowing water. This in turn allows the bentonite to expand much further into the fracture, since a smaller quantity is lost to the water. The loss of bentonite from the deposition hole is calculated, however, as the sum of the expanding quantity and the eroding quantity. It is not possible to make a direct comparison of mass loss between the old and the new model, since the former only calculated the eroded quantity. For water velocities of 10^{-6} m/s and higher, however, the calculated loss is about a factor of two higher with the new model, whereas lower velocities give a larger difference. The new model does not currently handle the effects of gravitation or the impact from minerals other than bentonite.

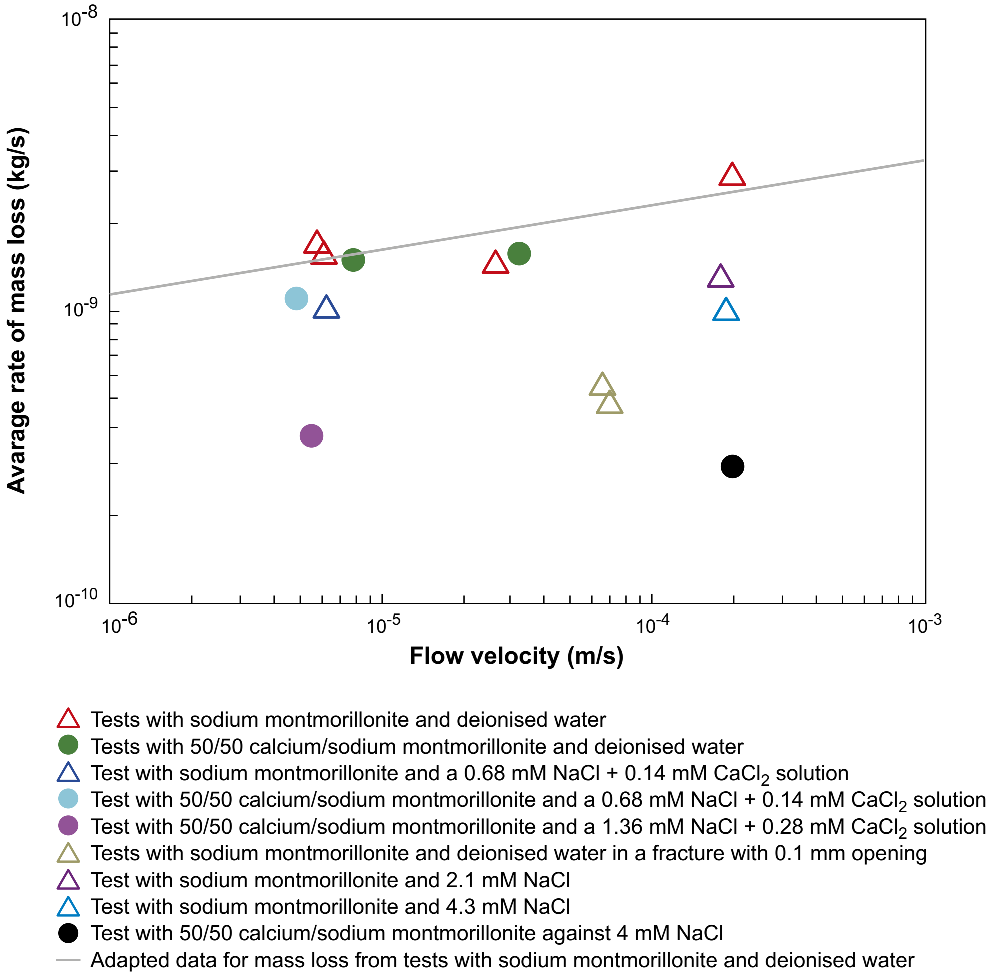


Figure 10-12. *Experimentally observed mass losses in a horizontal artificial fracture with 1 mm aperture expressed as a function of flow rate. The line corresponds to adapted data for mass loss from tests with sodium montmorillonite and deionised water.*

There is currently no quantitative model that can handle mass loss caused by gravitation since it has not been possible to quantify any of the most important coupled mechanisms for the purpose of developing a predictive model for loss of bentonite. Instead of modelling the transport rate of the released bentonite agglomerates in the fractures, an upper limit for the loss of bentonite has been derived, based on the fact that the loss of buffer cannot be larger than the amount that can be transported away in fractures. The loss of bentonite from a deposition hole due to bentonite agglomerates being pulled by gravitation is thus limited by the degree of agglomerate transport in fractures.

Programme

An observation from BELBAR is that the measured erosion in experiments in different laboratories can vary considerably even if the experimental conditions have been relatively similar. The causes of this are unclear and have not been evaluated within the framework of the project. The important work of analysing, evaluating and interpreting the extensive experimental data therefore remains and will need to be carried out during the coming period. Without clear-cut interpreted experimental data, it is difficult to verify the models that will be used in future assessments of post-closure safety.

Experiments with mass loss in artificial vertical fractures were conducted within BELBAR and showed that the loss due to gravitation was larger than previously expected. In a vertical or inclined fracture, mass loss does not mainly occur as a release of colloids, but rather as sedimentation due to reduced tensile strength of the expanding gel. The data from experiments in inclined fractures are still too limited to draw any conclusions on the importance of the process for post-closure safety. It will therefore be necessary to pursue a relatively extensive experimental programme.

The model used for buffer loss in SR-Site will be further developed. The further development described under current situation will be added to a quantitative model that can be used in assessments of post-closure safety. The new two-region model will also need to be tested against the experimental data collected in BELBAR. Furthermore, the simple, bounding model for the effect of gravitation/sedimentation needs to be tested against the new data produced in the experimental programme.

An additional effect that has not been studied is erosion in a situation when mass loss has been so great that advective conditions prevail in the deposition hole. This can be described with a given concentration of colloids and the model used in radionuclide transport for transfer of solutes from a deposition hole to groundwater in a surrounding, flowing fracture. There are, however, no experimental data to support this handling of the effect. Some simple experiments will therefore be called for.

The intention is to carry out the entire programme for buffer loss due to colloid release/erosion in cooperation with Posiva.

10.3.2 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 can give rise to the formation of sulphide under certain conditions. Bentonite dry density or swelling pressure has a great impact on the microbial activity. Sulphide is basically only a problem for the copper canisters in the Spent Fuel Repository, which means that these processes are not relevant for SFR and SFL.

Current situation

Bentonite/water mixtures with different percent by weight of bentonite clay (w/w) have been analysed with respect to dissolved sulphide. The wet chemical method used in the quantification of sulphide is an SKB method (SKB MD 452.011 Sulphide) based on the Swedish standard SIS 028115 (SIS 1976).

In the initial tests, 1, 5 and 10 percent by weight of unprepared Ibeco bentonite (Ibeco RWC-BF 2011) were allowed to equilibrate with boiled and degassed tap water and water from the Äspö HRL in Winkler bottles over five days. After separation of the liquid phase to new bottles and preservation with sodium hydroxide and zinc acetate, an analysis was performed with the above-mentioned method. The measured sulphide concentration was below the detection limit (0.006 mg/L) of the SKB method.

In order to investigate the impact of swelling pressure on the activity of sulphate-reducing bacteria, special test equipment has been developed (Figure 10-13). The principle of the equipment is that a bentonite sample is charged with a cocktail of bacteria, pressed to a given density and presaturated with water. The sample is then moved to the test cell where it is provided with a copper plate underneath and a solution of sulphate, with added radioactive sulphur on top, where also a nutrient is injected. After a given time, the activity of microbially-formed sulphide can be measured on the copper plate.

Figure 10-14 shows the measured activity on the copper plate in a test with Calcigel bentonite with a saturated density of 1 850 and 1 900 kg/m (corresponding to a dry density of ~1 310 and ~1 380 kg/m). The measured swelling pressures for the two samples were ~400 and ~750 kPa. The results show that microbial activity disappears above a saturated density somewhere in the range 1 850–1 900 kg/m^3.

Programme

As all observations indicate that the bentonite absorbs or transforms sulphides dissolved in water, no new experiments will be conducted to measure the equilibrium concentration of sulphide to bentonite without added sulphides. New experiments will be carried out to quantify the absorption and transformation capacity of different bentonites, and further study the mechanism for how this occurs.

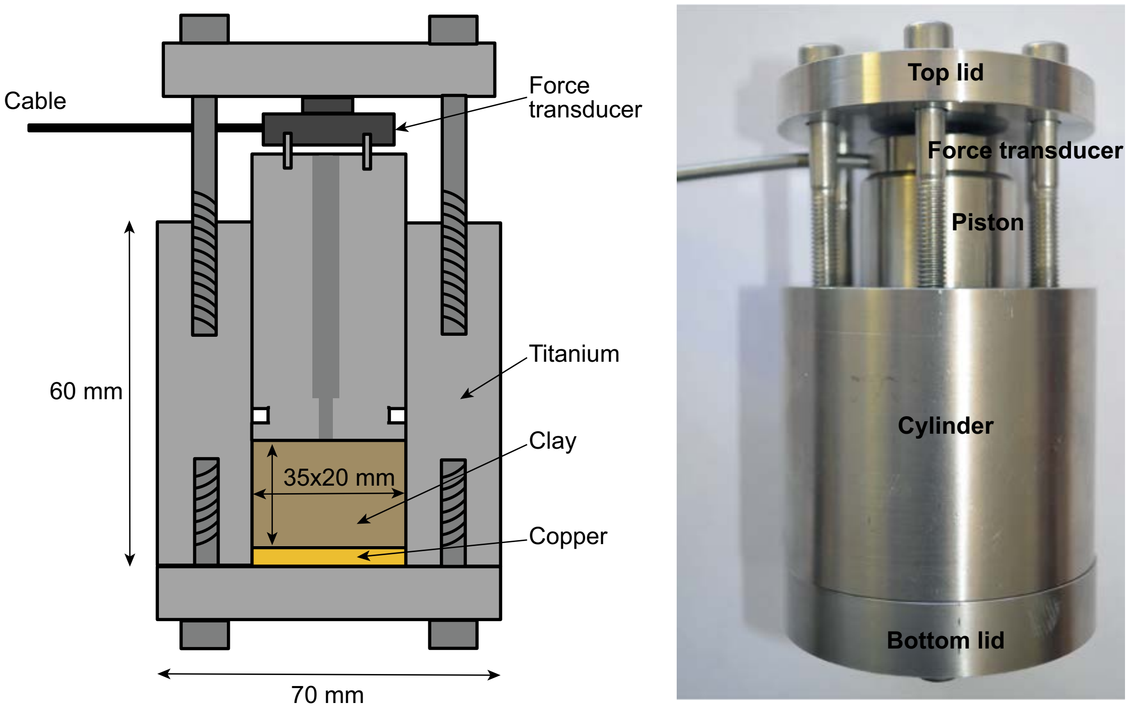


***Figure 10-13.** Left: A schematic cross-section of a test cell. Right: A composite test cell. From Bengtsson et al. (2015).*

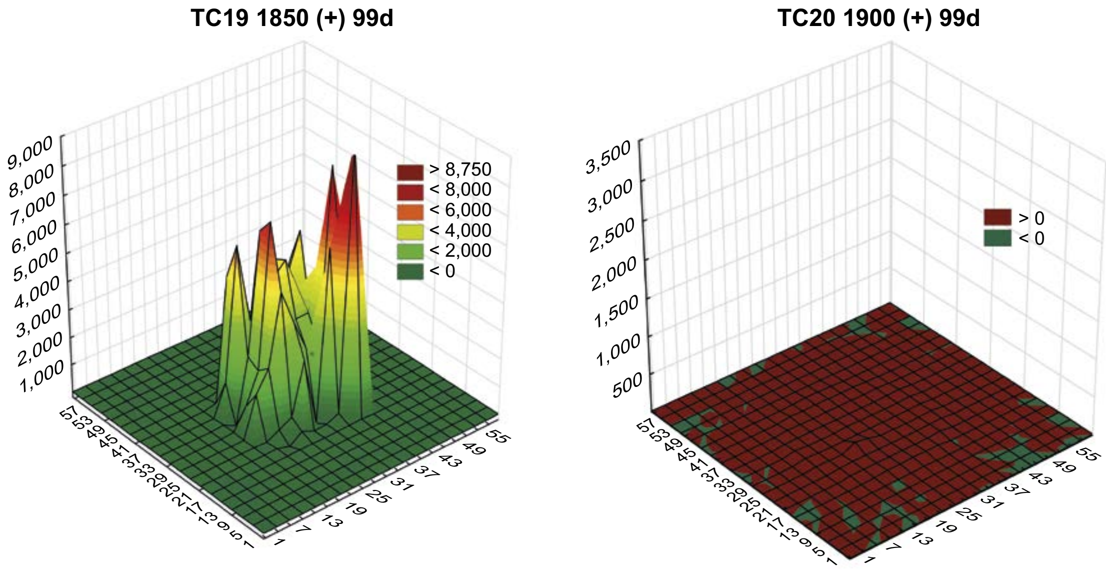


***Figure 10-14.** Measured activity in two samples with Calcigel with a saturated density of 1 850 and 1 900 kg/m³.*

Results from the studies of limitation of microbial activity in compacted bentonite indicate that there is a clear limit over which no activity occurs. For a given material, this limit can be expressed in dry density. This is very helpful both for the definition of a safety function and for a technical design requirement. The data available today, however, indicate that not only density and/or swelling pressure control the microbial activity but the type of clay can also be of great importance. This shows that the understanding of the governing parameters in the process is currently incomplete and, for this reason, all requirements need to be material-specific. The plan is therefore to continue the experiments with new materials, partly to try to find out what it is determining the microbial activity, partly to determine what materials are more suitable or less suitable as buffer materials, but also as backfill materials. These results are of great importance for the future design of buffer and backfill and assessments of post-closure safety. So far, experiments have been carried out at a single laboratory, so SKB also plans an independent verification of results.

10.3.3 Self-healing of bentonite

In the event of mass loss of bentonite from a barrier, for example due to erosion, it is important to be able to understand how the barrier self-heals. The programme for this is described in Section 10.1.3.

10.3.4 Long-term stability of bentonite

Bentonite has been selected as a barrier material, since it is expected to be long-term stable in the environments that exist in the different repositories. Bentonite can be stable for hundreds of millions of years in its formation environment, but changes in the environment can lead to a relatively rapid alteration of the mineral structure. The factors that primarily determine the stability are temperature, availability of potassium and pH. The redox conditions can also be important. Potassium concentrations in Swedish groundwaters are generally low, but there may be relatively large quantities of potassium in the rock. An increased temperature is expected during a relatively brief period in the buffer in the Spent Fuel Repository, while the backfill and the bentonite barriers in SFR and SFL are never exposed to an elevated temperature. In the Spent Fuel Repository, the interaction between high pH and bentonite is avoided with a requirement on low-pH materials, whereas this process is very real in both SFR and BHA in SFL.

Bentonite can be broken down by ionising radiation, but at the relatively low radiation levels that are expected in all repositories, this process is negligible.

Current situation

The Prototype Repository is a full-scale field experiment in the Äspö HRL that simulates the interaction between the canister, the buffer and the backfill in a final repository. The experiment includes six deposition holes distributed over an inner section (four holes) and an outer section (two holes). Between 2010 and 2011, the outer section was excavated after eight years of heating at about 100 °C at the canister surface (Svemar et al. 2016). Chemical investigations of the bentonite from the two deposition holes gave no indications of transformation of the montmorillonite mineral. Hydromechanical investigations showed no major differences in swelling pressure between the original clay and the retrieved clay. Trimmed (sawn-out) samples had slightly lower hydraulic conductivity compared with the original clay, especially at high densities, but there was no connection to the position of the clay (temperature) in the experiment. The clay from the experiment was slightly more brittle compared with the reference clay. The quantity of magnesium oxide increased slightly closer to the heater; which phase this was bound in could not be determined. Mössbauer spectroscopy and X-ray absorption spectroscopy (XANES) showed an increase of the iron(II)/iron(III) ratio in the bentonite while chemistry data showed an unaffected total level of iron. It is already relatively well known that iron(III) in montmorillonite can be reduced to iron(II) under certain conditions; however, the reductant in the prototype experiment has not yet been identified. Artefacts from the electrical heater cannot, however, be ruled out.

In the ABM test, which is a field experiment on a smaller scale in the Äspö HRL, three test packages were installed in 2006 and a further three in 2012. The first experiment was excavated in 2009 and the second in 2013. The ABM experiments are complex since they include a large number of different bentonites in the same experiment as well as a corroding iron heater, which makes the analysis work extensive with respect to ion exchange reactions, precipitation of salts, transformations/new formation, redox reactions and formation of corrosion products. In the second ABM experiment (ABM2), an increase of magnesium oxide in some samples has been linked to a new formation of what seems to be a trioctahedral smectite, probably ferrosaponite (Svensson 2015). The newly formed smectite was found in small amounts in the innermost millimetre from the corroding iron heater, and it is currently not possible to say if it is a transformation product of montmorillonite or if the mineral has formed by transformation of other phases like for example volcanic glass, exchangeable magnesium from the montmorillonite and iron from the corroding heater. Saponite and ferrosaponite are, like montmorillonite, also swelling clay minerals, but are expected to have somewhat different properties. Indications of trioctahedral phases were also observed in the first ABM package and in the TBT experiment (Temperature Buffer Test) (Svensson and Hansen 2013). The formation of trioctahedral phases appears to be linked to iron corrosion, as formation of trioctahedral smectite has not been observed in experiments with copper.

There are microbes in nature (iron-reducing bacteria, IRB) that can reduce trivalent iron as a part of their metabolism. IRB were also studied when the outer section of the Prototype Repository was excavated, where they were observed sporadically, especially near the contact with the rock (Svemar et al. 2016). There are indications that these microbes can in certain cases reduce the structural iron in montmorillonite and thereby also increase the probability of illitisation (Kim et al. 2004).

Programme

In order to study montmorillonite transformation as a function of temperature, samples from ABM2 will be further studied and an extensive compilation work will be carried out for already performed analyses of this experiment.

The purpose of the ongoing LOT experiment (Long Term test of bentonite) at the Äspö HRL is to identify and quantify mineralogical changes in the bentonite as a result of exposure to a repository-like environment. Furthermore, related processes in the bentonite with respect to copper corrosion, diffusion of cations, and survival and activity of bacteria are being investigated. The intention is to retrieve and investigate one more bentonite package from LOT during this RD&D period.

In BHA in SFL, the interaction between high pH and bentonite is an important process. There are data showing that the dissolution rate of montmorillonite at high pH values is strongly temperature-dependent (Rozalen et al. 2009). Considering that the temperature in SFL for long periods of time can be expected to lie somewhere in the range 0–15 °C, it is important to understand how the interaction works at these temperatures. This will also be applicable on the silo bentonite in SFR. Batch experiments with non-compacted bentonite and large amounts of water have been initiated at the Äspö HRL to investigate transformation reactions. The systems investigated are alkaline environments to imitate cement-bentonite interaction, as well as experiments with added iron reducing microbes to study whether they can affect the bentonite in a growth medium. In the alkaline system, the reaction between dissolved sodium hydroxide and bentonite has been studied at 8, 25 and 80 °C. The purpose is to eventually verify the temperature dependency on the transformation reaction previously studied by for example Rozalen et al. (2009), and to expand so that low temperature data is also included. Preliminary results support the model that states that temperature has a very great impact on the transformation rate of montmorillonite. The first experiments with added iron-reducing microbes have preliminarily not been able to demonstrate any changes of the montmorillonite or any other crystalline phase in the bentonite. SKB is also working on developing a quantitative calculation model where these data can be taken into account.

Experiments with bentonite in a growth medium and added iron-reducing microbes have been initiated. Preliminary results have not been able to demonstrate any changes of the montmorillonite, but these studies will continue, albeit on a small scale. In a first step, it is investigated whether the microbes can in any way transform the montmorillonite to another mineral, and if this is the case whether they can do this even at higher density and/or without added nutrients.

10.4 Barrier design

The supporting material presented in the application described conceptual designs of buffer, backfill (including arched plug) and closure. Since then, technology development has been conducted to perform system design for buffer, backfill and plugs in deposition tunnels. Based on updated design premises, the designs for buffer and backfill need to be revised.

A feasibility study concerning the SFL facility design has been carried out and indicates the need for targeted development activities regarding technical solutions for design and construction of the repository structures in the rock vaults.

10.4.1 The buffer in the Spent Fuel Repository

Current situation

The updated design premises mean that the design of the buffer in the Spent Fuel Repository and the verification that requirements are met need to be updated. How the installation method for buffer and backfill is designed also affects the density of the installed buffer and thereby fulfilment of requirements for swelling pressure, hydraulic conductivity and shear strength. The buffer material is a natural material and its properties vary within and between different deposits. The process for acquisition and changing materials and adapting the design of the buffer needs to be described. This work is under way in preparation for the PSAR.

Johannesson et al. (2014) presented a laboratory experiment where early THM processes in the buffer were studied. Figure 10-15 illustrates the volume the test was designed to simulate. In the test, two buffer rings were exposed to the thermal gradient that is expected during the installation phase in the Spent Fuel Repository and the buffer protection function was simulated. During the test, cracks were formed in the blocks and water condensed and accumulated in the bottom of the test setup. The blocks were stable and no break-out of buffer block pieces could be observed. Figure 10-15 shows a picture from the excavation of the experiment. The conclusion from the test was that the installation method was not as robust as previously assumed and the installation method needs to be further developed.

In Eriksson (2016), alternative installation methods for the buffer were studied. One of these methods implies that the buffer is installed under controlled relative humidity and temperature in the gap between buffer and rock. Another method implies that the blocks are coated with different substances to protect them against high relative humidity in the surrounding air. The recommendation in the report was to not continue the work on developing these alternatives, since they were not judged to have sufficient potential.

Based on the knowledge of the early THM processes, which was built up during the work, two installation methods were identified to have good possibilities to evolve into a robust overall solution:

- In relatively dry deposition holes, buffer blocks and pellets are installed at the same time. The low water inflow from the rock until the buffer is supported by the backfill is absorbed by the buffer. This leads to some heaving of the buffer stack but it is acceptable.
- For deposition holes with slightly higher water inflow, a further developed buffer protection is used. The latter installation method has practical disadvantages like water pumps etc. but is only expected to be needed for a small fraction of the deposition holes in Forsmark.

These installation methods are developed at present by SKB.

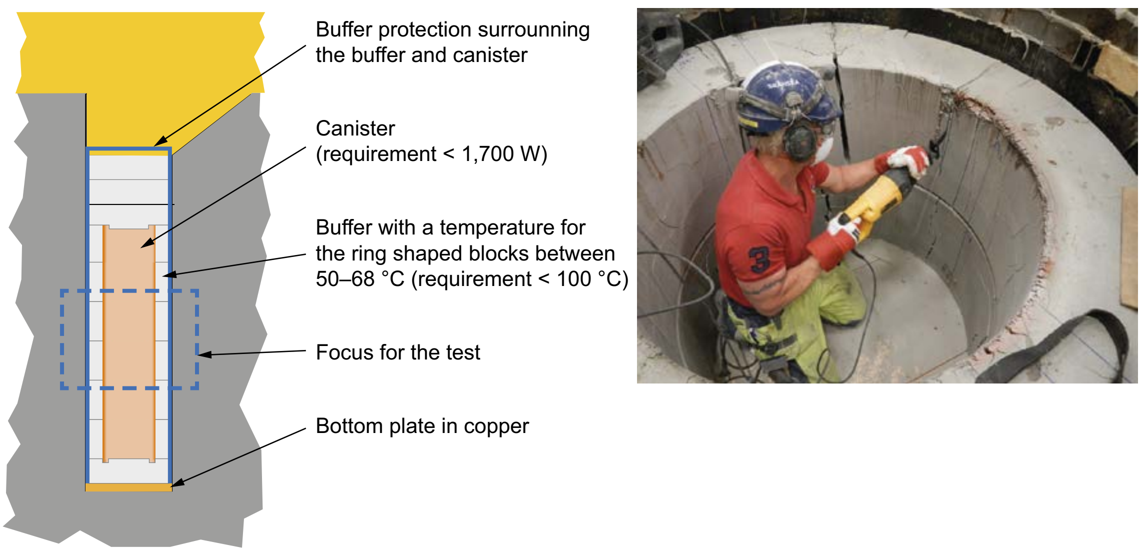


Figure 10-15. *Volume that the test of early THM processes was designed to simulate (left), and a photo from the excavation of the experiment (right).*

Programme

As described under current situation, the work of updating the design and requirements on installation methods for the buffer is under way. This includes two full-scale installation experiments at the Äspö HRL. According to the current installation sequence, the time from installation of the buffer until the backfill is in place over the buffer can be up to 3 months. The buffer expands during this time period due to early THM processes in the buffer. The size of this expansion, heaving, needs to be known to better assess the density and the design of the buffer. Heaving varies for the two installation methods and for different levels of water inflow. As a part of building up this understanding and correctly being able to predict heaving, models of the buffer's early THM evolution are being developed. These models will be verified by comparing their results with the results from experiments on different scales. The results will be used to define for which types of water inflows (size and distribution) simultaneous installation of buffer blocks and pellets can be used.

In order to update the design and verify compliance with requirements, data are needed regarding the relationship between density and swelling pressure, hydraulic conductivity and shear strength, which is described in Sections 10.2.2 and 10.2.3.

In order to design the buffer so that the requirements are met, knowledge is also needed on how the density of blocks and pellets varies, how carefully they can be placed in the deposition hole and how the geometry of the deposition holes is expected to vary. Knowledge on measurement uncertainties for all measurement methods used in the entire chain for quality inspection is also needed. Prior to the PSAR, variations and uncertainties will be determined or estimated and a balanced design of the buffer will be developed. Thereafter, further optimisation of the buffer is done to achieve an installation process as robust and cost-effective as possible.

10.4.2 Backfill in the Spent Fuel Repository

Current situation

Based on the technical design requirements that were used in the application, the design of the backfill in the Spent Fuel Repository has been further developed and detailed (Arvidsson et al. 2015). An evaluation is currently under way to evaluate whether this design fulfils the updated requirements suggested. In the same way as for the buffer, this means evaluating results from laboratory tests where the relationship between density and swelling pressure as well as hydraulic conductivity is determined. Calculations on how the backfill restricts the upward swelling of the buffer are described in Börgesson and Hernelind (2014a). The case with an unsaturated backfill and a swelling buffer gives the greatest upswelling. This case will therefore control the design requirements for the backfill. Modelling shows that the stiffness of the pellets and the dimensions of the pellet fills are most important for the upswelling as long as the backfill blocks are intact.

Within the framework of further development of the backfill, a full-scale installation experiment has been conducted at the Äspö HRL. In conjunction with this, the backfill's ability to counteract the upwards swelling of the buffer was tested. The upwards swelling was simulated with servo-controlled jacks after which force and deformations were measured. These results are used for improving the modelling of backfill deformation at the upward swelling of the buffer.

Programme

The models for how the backfill counteracts the upward swelling of the buffer will be updated based on results from tests in the laboratory and at full scale. The models will then be used to show that the requirements on the buffer's swelling pressure and hydraulic conductivity are fulfilled. The models will also be used for updating the design of the backfill. Specifications for blocks and pellets that guarantee that overall requirements are fulfilled will be studied and defined. This mainly concerns the geometry and stiffness of pellet fills and the pressure/shear strength of the backfill blocks.

In order to show that the backfill complies with the updated technical design requirements, results from laboratory determinations of hydraulic conductivity and swelling pressure will be used. There is a need to make enough measurements to create a reliable statistical material for describing the relationship between density and swelling pressure as well as hydraulic conductivity for potential backfill materials. A basic supporting material will be prepared for the PSAR. It will then be supplemented with additional data during this RD&D-period.

Based on the progressively improved material, the design will be optimised to achieve a design as robust and cost-effective as possible.

10.4.3 Design of clay barriers in SFL

Current situation

A feasibility study concerning the design of the SFL facility has been carried out. The feasibility study indicates that the development of technical solutions for design and construction of the repository structures in the rock vaults requires targeted research efforts, specific for SFL. Methods for backfilling and backfill materials for the waste vaults are also judged to require specific development efforts. Backfill solutions for deposition tunnels developed for the Spent Fuel Repository form the basis for development, but adjustments to SFL-specific conditions, for example design of operational structures, are needed.

Programme

During the period an inventory is made of different methods for backfilling the waste vault for legacy waste in SFL with bentonite. The studies include evaluation of different technical solutions with respect to volumes, technical feasibility, expected results (e.g. impact on swelling pressure) and effects of relevant processes. These initial studies are meant to provide material for the next step in the technology development, which will begin after the safety evaluation results have been presented.

10.5 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 so that the technical design requirements are fulfilled, it is important to have good control of the entire chain from mining of bentonite to finished installation of buffer and backfill. Work is under way to study the whole process from material purchase to finished installation in order to describe each step in the process as well as the quality checks that need to be carried out in order to finally ensure the quality of the installed barriers (fulfilment of requirements). 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.

10.5.1 Material supply and quality assurance of bentonite materials

Current situation

SKB has, within the framework of technology development projects for buffer and backfill, worked with process surveys, bentonite and supplier strategy and preliminary quality plans.

In conjunction with the large-scale installation test of backfill that has been carried out at the Äspö HRL, a number of potential backfill materials have been characterised as well as a larger quantity of bentonite from Ashapura used in the test.

The investigations of the backfill materials can be divided into three parts (Sandén et al. 2014):

1. **Acceptance inspection:** Tests have been conducted on a large number of samples taken from the different big bags the material was delivered in. The tests included among other things determination of water content, swelling index and CEC. The inspection of Asha 2012 also included the determination of liquid limit and grain size distribution.

2. **Hydromechanical tests:** These tests included determination of swelling pressure and hydraulic conductivity, strength of compacted samples, compressibility of water-saturated backfill and an investigation of how compacted clay samples are affected by exposure to high relative humidity.
3. **Chemical and mineralogical investigations:** This part included XRD analysis, determination of exchangeable cations and chemical analysis of the materials.

Results from the investigations showed that one of the candidate materials did not meet the requirements on hydraulic conductivity (the clay Ibeco 2011) and that other materials did not meet the requirements on granule size distribution for procurement (clay from Ashapura).

A similar characterisation of the material for the buffer was made as a part of KBS-3H Multi-Purpose Test (Johannesson 2014).

Programme

Based on the updated design premises for buffer and backfill, the requirement on installed density is dependent on the material. It is not reasonable to assume that only one material will be used during the operating period of the Spent Fuel Repository. Therefore, a process is needed based on design premises and material properties to adapt the design of buffer or backfill to the current material. Based on requirements and design premises, design for buffer and backfill and an overall material and supplier strategy, a preliminary quality plan will be prepared. It will, besides a process survey, include sampling strategy and sampling programme with the methods that SKB intends to use for quality control. There is a need to make enough measurements to create a reliable statistical material for describing the relationship between density and swelling pressure as well as hydraulic conductivity for potential backfill and buffer materials. A basic supporting material will be prepared for the PSAR. It will then be supplemented further during the RD&D period.

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. This is done to build up an understanding of the homogeneity of the materials on different scales and thereby be able to adapt 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.

A strategy for how processed the procured bentonite materials shall be needs to be developed. This influences the design of the facility in the reception harbour. If inhomogeneous materials are procured, the facility and quality system must be adapted to be able to homogenise materials on a large scale. The decision will be made on practical and economic bases.

10.5.2 Manufacturing of buffer components

Current situation

Since the RD&D Programme 2013, approximately 100 blocks have been produced by uniaxial pressing for various experiments, see Figure 10-16. The results of manufacturing and machining of 43 blocks are described in Johannesson (2014). A limitation of the existing presses is that the height of the blocks cannot be more than 500 millimetres, in the current reference design the buffer blocks are 800 millimetres high. A material model for block pressing has been developed and the input parameters have been calibrated and verified against real block pressing. Completed modelling suggests that also blocks of height 800 millimetres can be manufactured with sufficient quality (Börgesson and Hernelind 2014b).

The technology for compacting blocks by uniaxial pressing has been further developed and a new type of mould with a divisible lining has been designed and tested on a small scale. The impact of the lubricant at pressing has been studied and it is likely that blocks can be pressed without lubricant. The new design of the mould together with the elimination of lubricant would entail great advantages due to the possibility of producing completely cylindrical blocks. Machining of the blocks could thereby almost entirely be avoided.

Figure 10-16. *Uniaxial press for fabrication of buffer blocks.*

A description of the production process for buffer components has been developed and is presented in Eriksson (2014). The process description is based on previously performed experiments and tests.

Programme

Pressing of a full-scale block without lubricant has been carried out and will be evaluated. If the quality of the manufactured block is acceptable, the next step in the development is to design and test a mould in full scale.

So far, all buffer block pressing has been done with the material MX-80. SKB also intends to show that the pressing of blocks is possible with other materials and will therefore compact buffer blocks with various materials during the coming years.

Although uniaxial pressing is the method focused on so far, SKB also follows Posiva's development work for isostatic pressing. Work has been carried out to compare the quality of manufactured blocks with the two methods. Uniaxial and isostatic pressing will be evaluated together with Posiva.

10.5.3 Manufacturing of backfill components

Current situation

For the large-scale installation test with backfill performed in the Äspö HRL, 400 tonnes of bentonite blocks with the dimensions 500 × 571 × 400 millimetres were manufactured. The blocks were manufactured by Höganäs Bjuf AB. In order to handle these heavy blocks, some changes in the production line were required. These changes, to permit efficient handling of many large bentonite blocks, means that the manufacturing can be compared to the industrial manufacturing that can be expected in the future. The manufacturing generated 1 820 acceptable blocks while 110 blocks were discarded because they did not meet the geometry requirements.

The block press and associated equipment worked as intended and no problems caused by the equipment were noted. However, some problems related to the bentonite material occurred as the material contained some stones that got stuck in the mould filling equipment and caused overheating of equipment and production disturbances. The bentonite material was bought in "big bags" and there were some variations in granule size between the bags resulting in large variations in block height. Since it was important to keep the block dimensions within the acceptable range in order to conduct the installation test, a decision was made to prioritise block height. As a result of this, the blocks had lower densities than expected.

Tests have shown that the material properties are important to achieve blocks with high strength and smooth fracture-free surfaces. The most important properties are water ratio, granule size distribution and mineralogical composition.

For further description of the conducted tests, results and conclusions we refer to Arvidsson et al. (2015).

SKB has purchased a pellet press for the Äspö HRL where extruded pellets can be manufactured for various tests, see Figure 10-17. The equipment can produce 700–800 kg/hour with a bulk density of 1 000–1 050 kg/m^3. Experience from this pellet manufacturing will constitute an important basis for the coming design of the production building in Forsmark.

Programme

SKB believes that the technology and methodology for manufacturing backfill blocks and pellets exist and that it is verified that available standard equipment works for this purpose. Continued efforts will be focused on material parameters relevant for block pressing. This is in order to gain a better understanding of how different parameters affect the quality of the blocks and to improve requirement specifications for the material prior to block pressing.

Some efforts are also needed regarding control and inspection of the process, above all how the inspection of water ratio at mixing will be carried out in order to provide direct feedback on whether the water ratio of the bentonite material is within the acceptable range for block and pellet manufacturing.

Block pressing and pellet manufacturing will be needed for various tests at the Äspö HRL. Data and experience will be collected on these occasions to further improve the knowledge and understanding of the process, but also for expanding the statistical material.

10.6 Deposition and installation of buffer and backfill

The strategy for equipment/technical systems is to use wherever possible available standard equipment. If there is no standard equipment, the possibility to make a modified standard is studied and the final 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.

Figure 10-17. *Pellet press at the Äspö HRL.*

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, see Figure 10-18. Using modules has several advantages such as:

- Faster error handling.
- Easier to maintain service expertise.
- Simpler stocking of spare parts.
- Lower development costs.

Figure 10-18 shows that transport equipment, buffer handling equipment and backfilling 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 where 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 backfilling works as intended before detailed design is made of the special equipment needed in the Spent Fuel Repository.

10.6.1 Deposition in the Spent Fuel Repository

In the Spent Fuel Repository, equipment for processes such as deposition of canisters is required. In most of these processes there are individual steps that must be carried out in a controlled and safe manner and with high repeatability, which requires some automated functions.

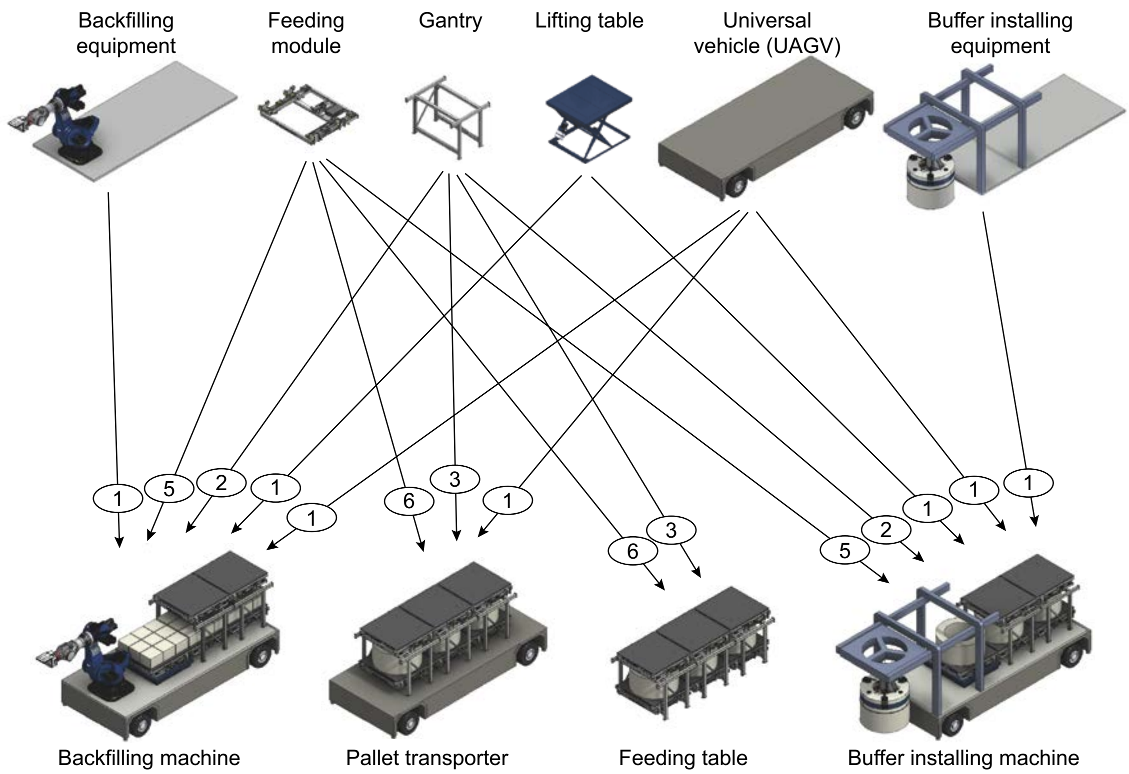


Figure 10-18. *Strategy for modularisation.*

Current situation

To control and monitor automated functions and processes in industrial applications, in most cases, some kind of supervisory system is used. The supervisory system gives an operator the opportunity to get an overview of what happens in the process. In order for these processes to be visualised in the system, the equipment or the 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 has been designed to manage all the functions on a prototype of the deposition machine.

To permit monitoring and control of the processes, the following functions were developed and handled in the demonstration system:

- Wireless communication interface.
- Area inspection (work area for machine without driver).
- Interface to the production control system (the system can receive production orders and create a driving order for a machine).
- Route inspection including route data (positions for deposition holes and other objects).
- Traffic control.
- Data storage.
- Operator interface.
- Alarm management.

In 2015, testing of this demonstration system was completed with the aid of the prototype machine for deposition, Magne. The testing previously performed with the deposition machine at the Äspö HRL shows that the stipulated requirements on reliability, availability and repeatability at automated operation can be met. The emphasis of these tests was therefore to investigate whether it is possible to safely start and stop a process over a wireless network. The results of testing campaigns show that the method is applicable and also possible to implement on a deposition machine, which is driven 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 that are needed to control the processes.

During the coming period, further development will also be made of the deposition machine's mechanical components such as grapple, hoist and radiation-shielding tube.

The need for further technology development for retrieval of canisters, buffer and part of the backfill must be clarified and begin during the RD&D period.

10.6.2 Buffer

Current situation

Requirements on the installation sequence and installation of buffer components needs to be updated and clarified. An important question which was described in Section 10.4.1 is, for example, which processes occur in the buffer, and what measures need to be taken with respect to these, during the time from the buffer being placed in the deposition hole until the backfill is installed above the deposition hole.

There is a prototype for a buffer lifting tool which has been tested for some time. Since the RD&D Programme 2013, prototype equipment for installation of pellets in deposition holes has been developed and tested at the Äspö HRL.

Programme

As already mentioned in Section 10.4.1, two installation methods have been identified depending on the situation in the deposition holes. This results in two different installation sequences and different requirements on the installation of the buffer.

Buffer protection needs to be further developed for the case with "wet" deposition holes.

To verify that the installation of the buffer works as intended and yields results within acceptable intervals, full-scale tests need to be conducted in underground conditions.

10.6.3 Backfill

Current situation

Equipment for installation of backfill has been developed and tested and the tests were concluded with the aforementioned installation test at the Äspö HRL, see Figure 10-19. A total of 12 metres of backfill was installed in a tunnel at a depth of 450 metres with same dimensions as a future deposition tunnel. Installation was carried out with prototype equipment and some temporary solutions including manual handling that will not be used in the Spent Fuel Repository, which means that conclusions regarding capacity cannot be drawn. The assessment is nevertheless that the proposed installation concept is suitable for backfilling. The test showed that it is possible to install blocks and pellets so that the installed density will be sufficient for the backfill to meet the requirements (Arvidsson et al. 2015). However, in order to achieve an automated process that meets the requirements on installation time, further development is needed.

An important issue for backfill installation is knowledge about the water distribution in the rock and information on the size of inflows of water to the deposition tunnels, which is obtained from the detailed characterisation, see Section 11.1.1. Depending on the location and size of the inflows, problems can arise with instability in the installed block stack and erosion of backfill material. There are three strategies for handling water inflow:

1. Stop inflowing water.
2. Drain inflowing water.
3. Storing of inflowing water.

Figure 10-19. *Block installation at 450 metres depth in the Äspö HRL.*

Work has been carried out to identify suitable methods for water handling during the installation phase. These are described in Sandén and Börgesson (2014).

Programme

Further development will take place to ensure that the installation can be carried out with the capacity and precision required. Equipment for installation of pellets is under development and will be tested and integrated with equipment for block installation.

Regarding water management at backfill installation, work will continue to develop and test suitable methods. Potential methods may also need to be tested in full scale. Cooperation with Posiva is under way and will continue on these issues.

10.7 Borehole sealing

10.7.1 The Spent Fuel Repository

There are a large number of investigation boreholes adjacent to the area where the surface facility for the Spent Fuel Repository is proposed and in the rock geometry where the underground facility is proposed. A number of the boreholes must be sealed before the start of construction. Other investigation boreholes that are open today need to be sealed during later stages of the construction process, since they can be utilised for monitoring during the construction phase.

The aim of sealing is to restore the hydraulic properties of the bedrock so that the boreholes do not provide flow paths for groundwater and thereby do not contribute to increased radionuclide transport to the ground surface (from a canister if it is damaged) or transport of undesirable substances from the surface environment to the repository.

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 open boreholes without seals also appear to have a very limited impact on the groundwater flow in the Spent Fuel Repository.

Subsequent sensitivity analyses (Luterkort et al. 2012) also show that a hydraulic conductivity in sealing materials of up to 10^{-6} m/s does not result in increased radionuclide transport to the ground surface. This has led SKB to change the requirement on the tightness of the borehole seal and to reconsider the borehole sealing method previously developed.

Programme

Based on the new requirement level, SKB has initiated a programme for borehole sealing with the purpose to develop and demonstrate a robust method for borehole sealing adapted to the requirements for the Spent Fuel Repository by:

- Inventorying existing boreholes and preparing a preliminary plan for how each borehole shall be sealed based on where there are zones with low and high hydraulic conductivity for the single boreholes.
- Developing a method description for the execution.
- Carrying out verifying laboratory and field experiments.
- Preparing quality and inspection programmes.

Based on the results of this programme, SKB will be prepared to seal the first short boreholes in Forsmark that require sealing before the start of construction. Development activities will, as far as possible, be coordinated with borehole sealing in SFR, see Section 10.7.2.

10.7.2 Final repository for short-lived radioactive waste

In the area where the extension of SFR will be sited, there are a number of boreholes which must be sealed prior to the construction phase. The borehole seal is of importance in the short term during construction but also in the long term after closure of SFR. Sealing of boreholes should be carried out with proven methodology and technology. SKB intends to conduct additional studies to further develop the methodology and technology for borehole sealing based on previous technology development.

Current situation

Modelling and calculations have shown that the tightness of the borehole seal has a limited importance for groundwater flow in the rock vaults. This has led to changes in design basis assumptions, which affects the technology and methodology for borehole sealing. Reasons for sealing of boreholes that lie within the area of the extension before the construction phase:

- The rock will be deformed around new tunnels and rock vaults, which can lead to borehole geometries being disturbed, which can impede sealing of borehole parts.
- The rock will be grouted to reduce the water inflow in conjunction with the extension. There is a risk that the grout penetrates into open boreholes which can interfere with a later sealing.
- Intersection of boreholes that is connected to water-conducting fractures in conjunction with tunnelling, can make the excavation of tunnels and construction of rock vaults more difficult.
- Blast force can propagate quickly through the borehole and, in the worst case, let casings installed in the upper part of the boreholes loose.

After concluded operation, the boreholes at SFR will be sealed permanently.

Programme

Technology development within the following areas for borehole sealing will be carried out:

- Method for borehole sealing and detailed design of the constituent components.
- Experimental studies for constituent components and large-scale field experiments.

10.8 Closure

10.8.1 Closure of the Spent Fuel Repository

Current situation

A simplified design of the closure has been proposed based on performed 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.

10.8.2 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. Within conducted technology development, a need was identified for further development of design and installation of the closure components. Detailed design of the extension of SFR and better knowledge of the 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 developed a coordinated closure plan for SFR (Luterkort et al. 2014) with respect to constituent closure components and installation and sequence. Closure is described on the conceptual level and prior to the PSAR the knowledge concerning materials, design and installation will be deepened.

Programme

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

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

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

11 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 must be isolated. For this to be achieved, access to methods for investigating and characterising the rock at a sufficiently detailed level is needed. Furthermore, the final repositories' underground openings need to be designed in such a way that the long-term stable conditions are not jeopardised, and in order to be able to evaluate post-closure safety, understanding of processes that alter the mechanical and chemical conditions in and around the repository is also needed. A large part of the technology development and research issues related to the rock and the final repositories' underground openings are common for the three different repositories.

Underground openings consist of the rock cavities and residual material from construction in the rock that are required for the final repositories' underground parts. In the design of the underground openings the following need to be taken into consideration:

- The openings' spatial geometry and location.
- The rock surrounding the openings and affected by construction.
- Engineered materials for sealing and rock reinforcement, and residual materials from performance of activities in the final repository facility which, at deposition, backfilling or closure, remains in and on the rock that surrounds the openings.

For the utilisation of the rock as a barrier for the Spent Fuel Repository, it is primarily the placement of the deposition areas and deposition holes with respect to the thermal, hydrogeological, mechanical and chemical properties of the rock that are important.

The application for the Spent Fuel Repository includes the production report for the rock (SKB 2010b) as a reference. Prior to the PSAR, this report will be updated with regard to the development that has taken place since the application was written. The production report gives an account of the underground construction that makes up the reference design and whose safety will be analysed, during operation and post closure. The report also describes the current design with respect to the design premises that affect the repository's ability to contain, prevent or retard the dispersion of radioactive elements. Furthermore, planned production of the underground openings, quality control and inspection are described, as well as expected outcome yielding essential information in order to be able to describe the initial state of the repository.

An update of methods for tunnel production is planned and the technology development projects must therefore produce deliverables as basis for technology decisions . The need for quality assurance measures during manufacturing, handling and installation needs to be further developed by carrying out a product and process survey and developing preliminary quality plans. Based on the product and process surveys, the need for qualification (see Section 4.2.5) can be estimated.

For the assessment of post-closure safety, there are a number of issues that require further research. These issues concern mainly the Spent Fuel Repository and SFL, and are largely common for the two repositories. Since the planned construction of SFL will take place much later than for the Spent Fuel Repository, these issues need primarily to be resolved before the SAR for the Spent Fuel Repository, with the state of knowledge in the PSAR and prior to a decision on start of construction as important cross-check points.

11.1 Detailed characterisation

Detailed characterisation refers to the investigations (including monitoring) and modelling carried out in conjunction with the final repositories' construction and operation, and that step for step provides information on the properties of the rock and material for assessment of compliance. Detailed characterisation is carried out in an integrated fashion with rock works during each facility's construction and, for the Spent Fuel Repository, during the subsequent extensions of the deposition areas during the operating phase.

11.1.1 Methodology for detailed characterisation

Since the framework programme for detailed characterisation during the construction and operation of the Spent Fuel Repository (SKB 2010e) was published, the work with the reference design of the facility in Forsmark has continued, along with the work of updating the design premises regarding post-closure safety. For development of the detailed characterisation programme, methods, instruments and modelling methodology for verifying that stipulated requirements are fulfilled will be developed and described.

Detailed characterisations have several purposes. They should provide the information needed for progress in the rock works and contribute to greater knowledge concerning the 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 gradually extended, an important task will be to adapt the facility to the prevailing rock conditions so that the requirements linked to the repository's nuclear safety (design premises) will be met. Collected data and updated models of the bedrock provide furthermore the accumulated body of data for updated assessments of post-closure safety. Detailed characterisations will also cover the information needs for the environment control and monitoring.

Current situation

The detailed characterisation programme is being updated based on the technology development that SKB has so far conducted within investigation and rock construction technologies. This includes for example experience from the expansion of the Äspö HRL which was carried out 2012–2013 (Johansson et al. 2015c) and experience from Posiva's investigations during construction of rock facilities in Olkiluoto (Posiva 2012, McEwen et al. 2012). The detailed characterisation programme describes investigations and modelling with associated quality assurance. It is on a general level and aims to provide an overall picture of the investigations that are needed during the different phases of the Spent Fuel Repository's construction and operation.

Programme

Prior to the PSAR and detailed design, an operative investigation programme will be developed, where the investigations in connection with the different phases are described in more detail. During this RD&D period, further development of methods, tools and operative programme is planned for detailed characterisation with associated modelling prior to start of construction. The focus of the programmes is in the Spent Fuel Repository, but the developed methods will also be used in the extension of SFR. Tools refers here to measurement instruments and data systems including calculation tools and software used for visualisation and modelling, including associated methodology and strategy for its application. Strategies and methodology for verification of requirements on post-closure safety (design premises) such as control of the excavation damaged zone (EDZ), quantification of permissible occurrence of spalling, and hydraulic criteria for deposition holes will be devised and tested. The in situ experiments for identification and characterisation of critical structures that have been carried out at the Äspö HRL constitute a practical application, see Section 11.1.2.

The development work is focused on the tools that will be used during construction of accesses and the central area in the Spent Fuel Repository, and development for deposition areas that is expected to take a long time and therefore needs to be started early. Other development for detailed characterisation in deposition areas will mainly continue during the latter part of this RD&D-period.

11.1.2 Critical structures

The concept of critical structures and volumes entails geological structures or rock volumes with properties that can have a negative impact on the post closure safety of a KBS-3 repository. Critical structures/volumes are divided into three classes with respect to their influence on the repository area layout (Munier and Mattila 2015). Class 1 includes structures/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 have created borders for the repository area. Class 2 includes structures/volumes with properties such that they can be accepted between different deposition areas but not within deposition tunnels, and which thereby steers the layout of the repository. Class 3 includes structures/volumes with properties such that they cannot be accepted to intersect deposition holes, and which thereby steers the location and acceptance of deposition hole positions.

Methods need to be further developed, both for identification of critical structures of class 3, and in order to be able to select deposition hole positions with hydraulically suitable properties. This means, among other things, that the methodology for model-based acceptance of deposition holes, on the basis of hydraulic tests in pilot holes and verifying tests in deposition holes (see Section 11.1.3), must be well tested and functioning prior to detailed design of the deposition areas. This applies also to the methods for measuring small water flows to deposition tunnels and deposition holes, as well as evaluation and modelling techniques.

Current situation

Investigations for identifying and characterising critical structures have been carried out with the purpose of testing, evaluating and refining investigation and modelling methods, in two nearby parallel tunnels and two cored boreholes in connection with the new underground parts in the Äspö HRL. The work that has now been compiled and will be published shortly, included as a first step measurements in the tunnel system using radar, seismic and resistivity methods. The results of these measurements were integrated, along with the results of hydraulic interference tests in pilot boreholes for the tunnels, with the established structural model of the rock volume in question. Tunnel mapping yielded data on fractures or fracture zones that can be traced around the full tunnel perimeter and that are defined as the full perimeter intersection, FPI. In a subsequent test campaign similar investigations were carried out in the two cored boreholes. Figure 11-1 shows the possible extent of three potentially critical structures, based on interpretation of the completed measurements.

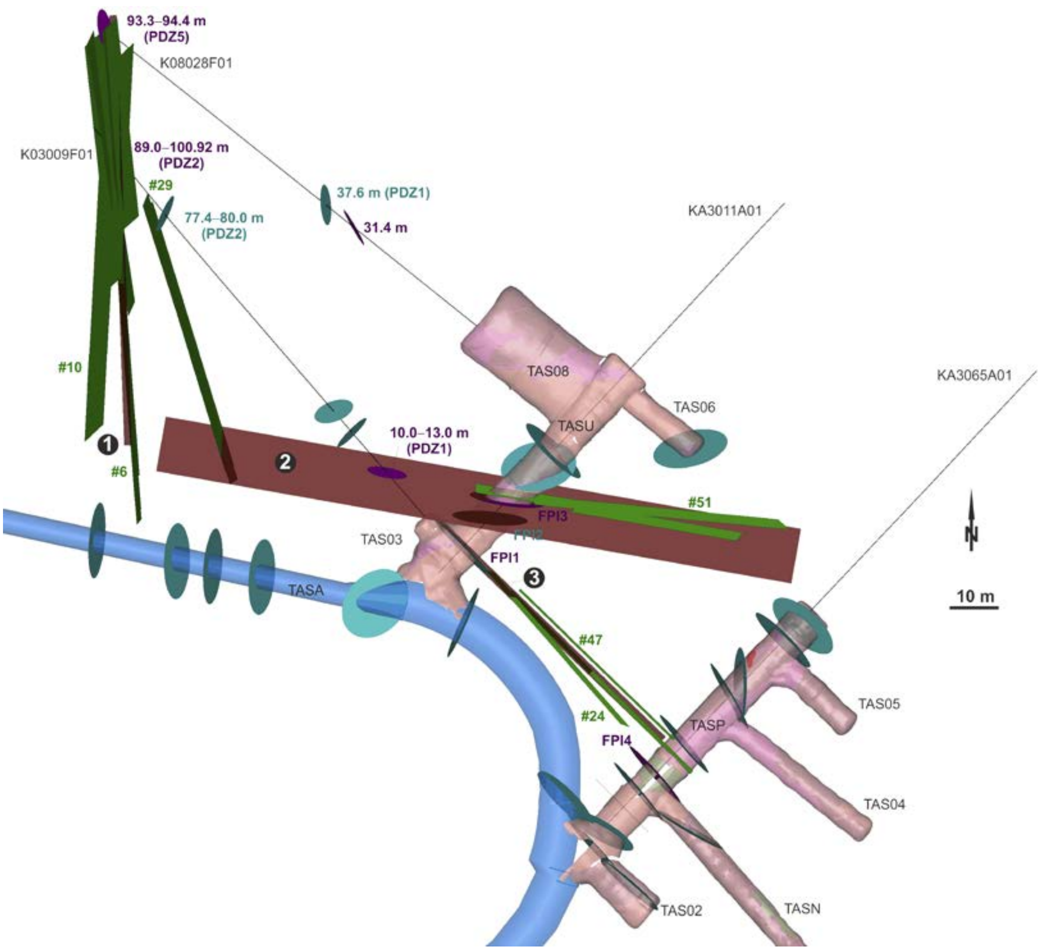


Figure 11-1. Top view over the new underground parts of the Äspö HRL (TASP-TASU-TASA) that beyond nearby boreholes (black lines) shows the possible extent of three potentially critical structures (red sheets numbers 1–3 in solid circles) together with the seismic reflectors (green, #6, #10, #24, #29, #47 and #51) and geological observations (violet disks) that comprise a basis for interpretation. Also other fracture zones that can be traced around the full tunnel perimeter (larger blue discs in tunnels, FPI) and a number of selected structures for possible detection with cross-hole/tunnel geophysics in the two boreholes K03009F01 and K08028F01 (smaller blue discs in boreholes, PDZ = possible fracture zone) have been included as a reference.

Programme

Site-specific conditions determine how the methodology for selection and verification of deposition positions will be applied in the Spent Fuel Repository. Continued development to determine methodology is therefore planned in conjunction with the integrated tests of excavation and deposition that are planned in the Äspö HRL, in Forsmark and in cooperation with Posiva in the future extension of the repository area in Olkiluoto. Further studies for identification and characterisation of critical structures will be focused on the structures' size and on the underlying properties that can serve as indicators of size, plus hydraulic and mechanical properties of the structures, in order to better assess the risk for actual movement in a given structure in connection with an earthquake, see also Sections 11.11 and 11.12. The ultimate goal is that the identification of critical structures near deposition positions to a larger extent shall be based on actual properties and to a lesser extent on the FPI criterion (Munier 2006).

11.1.3 Modelling methodology within detailed characterisations

Design, production, site understanding and post-closure safety define the varying needs and requirements that site descriptive modelling meets during repository construction and operation. Design and production have a need for continuous forecasts on the tunnel scale, which means that modified and in some cases new modelling methodology needs to be developed. Methodology needs to take into account an early and close co-interpretation and integration of geological and hydrogeological information for identification and description of critical structures, see Section 11.1.2, and the hydraulically connected fracture network. The latter make up the geometric basis for continued site descriptive modelling on the facility part scale (e.g. individual deposition area) or the facility scale (the entire repository with its accesses). Other disciplines; hydrogeochemistry, rock mechanics, thermal properties, transport properties and surface systems continuously integrate 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 the geometric modelling in the form of single-hole interpretation and single-tunnel interpretation, which constitute fundamental building blocks in the continued integrated deterministic modelling in 3D. The latter is linked to the stochastic description of fractures (DFN), see Section 11.3, in the rock mass between the deterministically modelled deformation zones. Some initially stochastically described fractures/structures are expected to gradually obtain such informational underpinnings that they can be (re-)interpreted as deterministic structures, mainly in the description of the deposition areas on a tunnel scale.

A transient and disturbed environment during construction and operation of the repository is an important starting point for further development of modelling methodology for all disciplines. Effective surveillance and measured changes related to an established reference level comprise an important prerequisite for monitoring of effects of the gradual extension of the repository and for calibration of quantitative models.

Current situation

There currently exists methodology reports for geological and hydrogeochemical modelling during detailed characterisation in the concept. Work is under way with the corresponding documents for other disciplines.

Programme

Continued development of discipline-specific and integrated modelling methodology for detailed characterisation is planned to take place during this RD&D period. The application and testing of the methodology that exists in the concept is planned in conjunction with the 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.

11.2 Tunnel production

The different parts of the underground facilities must comply with different requirements and design premises depending on their function during operation and after closure. Rock vaults in SFR will be operated with conventional technology, which means that no particular technology development is required. For excavation in accesses (ramps and shafts) and the central area, no further technology development is required; methods for careful drill-and-blast have been developed. The special requirements and design premises that apply for deposition tunnels and deposition holes in the Spent Fuel Repository differ from the premises that apply to other construction objects. This entails challenges, for both the technical execution of the deposition holes and for the investigation methods that will be used for the selection of the location and how best to verify that the criteria have been fulfilled. A large part of the work within technology development for tunnel production concerns how to identify known and proven techniques and materials that are used in the industry and also meet the requirements and design premises for the Spent Fuel Repository.

11.2.1 Grouting

Sealing of underground openings in the Spent Fuel Repository meets greater challenges than normal infrastructure projects. In order of the grouting not to have a negative impact on the rock as a barrier, there are more strict requirements on, which grouting materials that can be accepted, how the grouting fan is designed and that the grouting is performed with a limited and very controlled spreading. The high groundwater pressure also requires another grouting design in order to be able to prevent erosion of grout, and to prevent hydraulic widening of fractures.

Current situation

The Observational Method is applied as a design method when conditions are uncertain or when the geotechnical risk is large. During the expansion of the Äspö HRL, the Observational Method was applied to manage the grouting work and to test the possibility of using the Äspö HRL's Hydro Monitoring System (HMS) as an observation object, in parallel with the observation object inflows (Olofsson et al. 2014). This gave an opportunity to use qualified monitoring of groundwater pressure levels with a system similar to that in Forsmark.

Programme

Experience from previous completed grouting (Funehag and Emmelin 2011, Johansson et al. 2015a, Funehag 2016) is analysed and evaluated and turned into updated method descriptions for grouting with low-pH materials, adapted for the different engineering geological type domains that can be expected to occur during the excavation of underground openings in the Spent Fuel Repository. The work also includes devising a strategy for inspection and verification of grouting results.

11.2.2 Tunnel excavation for deposition tunnels

Current situation

Considering the deposition sequence in the Spent Fuel Repository, where several different machines and equipment are used, the tunnel floor needs to be sufficiently level in order to be easily accessible and also to reduce maintenance and wear of the machines. The method for excavation of deposition tunnels today is drill-and-blast, but alternative methods are being studied in order to achieve a more level tunnel floor with less excavation-damaged zone, for example by mechanical excavation and wire sawing.

Studies have been conducted to increase the understanding of the extent and properties of the excavation damaged zone (EDZ) around a tunnel, and to develop methods for characterisation of the damage zone (Ericsson et al. 2015). In these studies, follow-up of the EDZ has been carried out in tunnels through geological characterisation, measurement with high-frequency ground-penetrating radar (GPR), and by means of injection tests in short cored boreholes in the tunnel floor.

Programme

During the RD&D period, production methods for safe and efficient tunnel production, adapted to the conditions at Forsmark, will be further developed. Method descriptions for excavation of deposition tunnels will be developed, which ensures contour control, among other things level floor and a minimal EDZ.

During the period 2018–2020 an integration test for tunnelling is planned. The integration test includes excavating and investigating a deposition tunnel with preceding pilot drilling and borehole investigations, drilling and investigating deposition holes, establishing plug locations and removing rock for the plug.

11.2.3 Boring of deposition holes

Current situation

SKB has followed Posiva's development work for deposition holes, where Posiva has drilled ten vertical experimental holes in Onkalo, distributed between two demonstration tunnels (Railo et al. 2015, 2016). The deposition holes were drilled with a prototype machine of type Rhino HSP500 manufactured for the purpose. SKB has previously drilled around 15 deposition holes in the Äspö HRL with a different machine.

Programme

Completed drilling in Onkalo will be evaluated, and the method will be improved to make boring of deposition holes more production effective. The method will be tested and subsequently verified in conjunction with the integration test for tunnel excavation (see Section 11.2.2).

11.3 Modelling of discrete fracture networks

Current situation

During the preceding RD&D period, conditioned DFN models have been developed. The purpose of this methodology is to increase determinism in the stochastic DFN models, mainly in the vicinity of the tunnels and deposition holes where different types of data can be measured. An increased determinism is expected to lead to a lower uncertainty in the entities calculated in groundwater flow models based on the underlying conditioned DFN models, and that also are propagated to different analyses of post-closure safety. Furthermore, a larger degree of determinism means that the models are anticipated to also be used to test different criteria for rejecting/accepting deposition hole positions. Methodology for conditioning has been tested with synthetic data from simplified sites. The initial results from the tests are promising and have been presented in Selroos et al. (2015).

MoFrac is a simulation tool for DFN modelling that was originally developed by Srivastava (2002). The programme has a rule based fracture generator that is mainly based on the geostatistical relationships that can be derived from site investigations above and below ground. The induced fractures in the network are strictly conditioned to measured structures such as lineaments, fracture traces and bore hole intercepts, and the programme supports the generation of 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 use areas mainly within mining and the nuclear waste industry, as well as more efficient application of DFN models during underground construction. The project has in a first phase updated source code from Fortran-77 to Python, and developed a first version of a user interface (GUI). MoFrac has been tested against both artificial data and data from the Äspö HRL.

In a joint project between Itasca, Université de Rennes and SKB, an alternative method, UFM (nearly Universal Fracture Model), for generation of DFN models, has been produced (Davy et al. 2010, 2013). 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, with a constant of proportionality, C, to fracture size, l, raised to a constant, b, according to: $\frac{dl}{dt} = Cl^b$.

- A hierarchical retardation process in which large fractures retard or hinder 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 that appears to be able to account for most of the data from the site investigations at scales relevant for the assessments of post-closure safety. The methodology also offers a generic tool that based on simple rules controls initiation, growth and termination of fractures which can be conditioned on observations (see further under "Programme" below).

Furthermore, a project about the effects of geometric uncertainty in data from boreholes is ongoing. Initial results show how different measurement uncertainties affect the total uncertainty on the geometry of a borehole (Stigsson and Munier 2013, Stigsson 2016).

Programme

The effect of fundamental assumptions in the current DFN methodology will be studied, and a DFN methodology will be developed including conditioning on the geometric and hydraulic data obtained when tunnelling and drilling of deposition holes begin. Pilot holes for tunnels and deposition holes are also expected to provide valuable information that can be used for conditioning. During the RD&D period, the methodology will be tested on further synthetic datasets and on real data from the Äspö HRL and/or Onkalo. This part is being carried out in collaboration with Posiva. Furthermore, a study of whether conditioning can utilise data also from tracer tests and on data from geophysical measurements (including electrical resistivity) performed in pilot holes for deposition holes is planned.

During the RD&D period, a methodology document will be developed that describes the next generation of DFN models (i.e. conceptual assumptions and data needs). This also includes application of the aforementioned methodology for conditioning developed in SKB's different modelling tools. Within the framework of the development of the methodology document, a number of conceptual questions linked to the DFN modelling will be explored; these include for example the relationship between intensity and size of the water conducting fractures, possible heterogeneous distribution of fracture intensity, the hydraulic properties of deformation zones and the possibility of applying a DFN concept for description of these zones and uncertainties in the relationship between aperture and transmissivity.

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) and (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 established. The developed methodology for estimating the total uncertainty in the orientation can also be applied for example to fracture orientations mapped in tunnels and deposition holes.

Fractures are assumed to be planar in the use of DFN models in SKB's hydrogeological modelling. To address this in subsequent transport modelling, scaling factors are occasionally applied to approximate the effect of the channelling that arises due to internal aperture variability. With the development of high performance computing (HPC), fracture networks can today be simulated with internal variability. Several different aspects can be explored, for example the validity of the simplified approach that scaling factors lead to, 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 is planned where the calculation tool DFNworks (Hyman et al. 2015) will be used to throw light on these issues.

Two initiatives are planned to gain a greater understanding of alternative DFN conceptualisations. One is based on the software package MoFrac and the second concerns application and further development of the concept UFM where an alternative method for DFN generation, based on mechanical principles is used.

For the next phase a further development of the user interface in MoFrac and additional tests with data from Olkiluoto (Finland) and the Glencore mine (Sudbury, Canada) are planned.

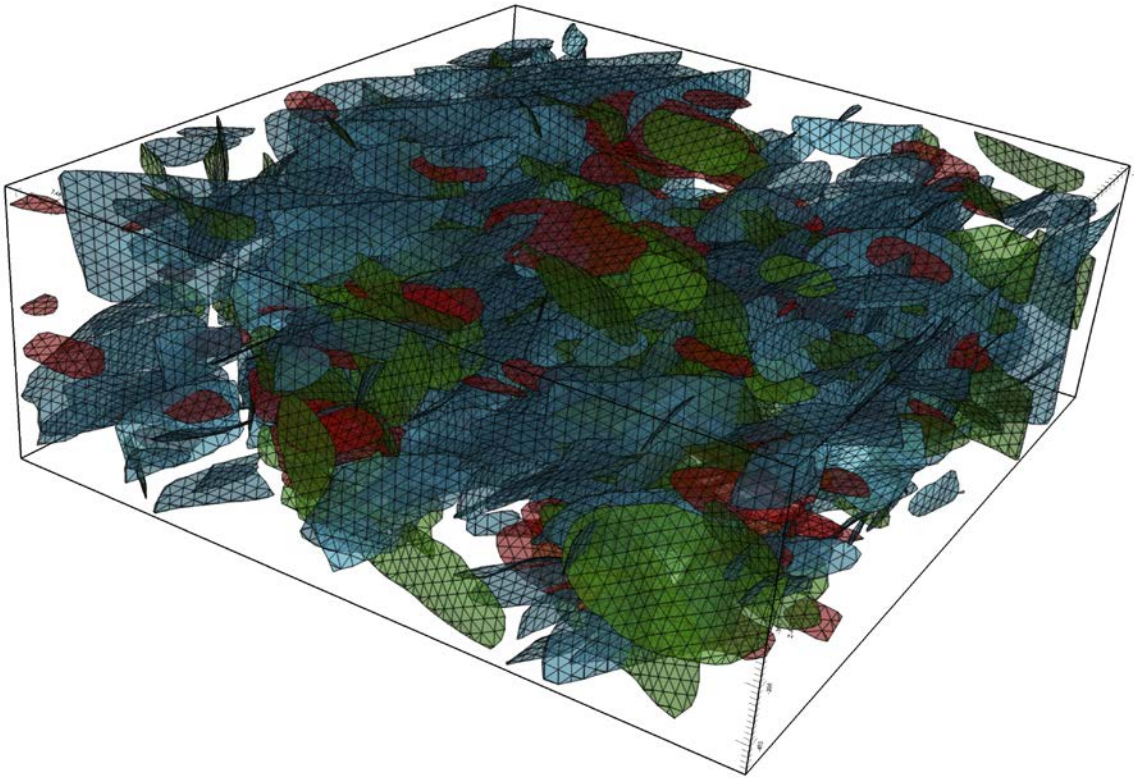


Figure 11-2. *Example of a fracture network generated with MoFrac.*

A weakness in the previously reported UFM methodology is the generalisations and simplifications initially assumed for the fracturing process. In particular, this applies to the initiation process which so far has been assumed to be uniform in 3D, the retardation process which has been assumed to depend only on relative fracture sizes, and the growth process which so far has been assumed to be isotropic, i.e. independent of fracture orientation. This means that correlations and anisotropy usually observed in the rock, such as clustering, intersection and dominant directions have not yet been addressed.

SKB therefore intends to refine the UFM methodology through the introduction of alternative initiation processes, which at present are uniform in space, and by the introduction of correlations to fracture orientations and thereby indirectly to paleostress fields. Furthermore, the possibility to define a degree of saturation, i.e. a state where fracturing essentially ceases, will be investigated based on global energy consumption in the fracturing process. Furthermore, development of a new method to continuously nest different scales with the purpose of more efficient generation of large models is planned.

SKB also plans to develop DFN models of UFM type for hydrogeological analyses. Generic results (Maillot 2015) show that flow characteristics in a DFN model of UFM type have lower effective permeability but more channelling on the network scale than the types of models which SKB previously has used in hydrogeological applications. The purpose of the work is to apply the UFM concept on site data from Forsmark, and in groundwater flow simulations to calculate some of the entities that are produced in safety assessment applications. This permits a comparison between the type of DFN model used in SR-Site and SR-PSU with an alternative conceptual model.

11.4 Hydrochemistry, and transport modelling

Current situation

SKB's capacity of hydrochemical modelling has been developed to include geochemical and transport processes coupled to 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 obtained by developing an interface for communication with the modelling tool PFLOTRAN, which is designed from scratch for HPC (High Performance Computing) machines. This developed coupled modelling tool is denoted iDP (Molinero et al. 2016). Openly available code libraries from PHREEQC have been included for this purpose into ConnectFlow (Joyce et al. 2015). Both calculation tools have for example been used to study the influence of leachates from cement on a repository environment.

The modelled groundwater composition can in turn be used as input data to transport modelling tools, for example MARFA, that have been developed in order to be able to apply a dynamic K_d concept where K_d is a function of mineral and groundwater composition in time, and space (Trinchero et al. 2016). Traditionally, K_d is a constant distribution coefficient that indicates how much of a substance is sorbed and in solution. Traditional transport modelling has also been developed, both regarding modelling tools, and methods for obtaining input data for transport modelling. In a PhD project, mathematical solutions for diffusion into stagnant zones followed by matrix diffusion have been developed (Mahmoudzadeh 2016). A PhD thesis has been published within the field of colloid properties and with couplings to colloidal transport (Norrfors Knapp 2015).

The SKB Task Force on modelling 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 to the field test BRIE (Bentonite Rock Interaction Experiment) performed in the Äspö HRL. In BRIE the hydraulic interactions between the rock and the bentonite has been studied in two downscaled deposition holes of 30 centimetres in diameter. In Task 8 the bentonite saturation process is in focus (see for example Dessirier 2016). Furthermore, a new modelling task, i.e. Task 9, has begun. It focuses on modelling of tests that investigate the transport properties of the rock such as matrix diffusion and sorption. The experiments that comprise the basis for the modelling exercise are REPRO (Aalto et al. 2009), which is being conducted in Onkalo in Finland and LTDE-SD (Nilsson et al. 2010), which was carried out in the Äspö HRL.

Programme

The development of the hydrochemical calculation tools iDP (DarcyTools – PFLOTRAN) and ConnectFlow – PHREEQC will continue in order to be able to apply the tools on other problems where reactive transport has a significant impact, for example the influence of leachates from low-pH cement or penetration of glacial water, on the repository environment. Validation tests are also planned, where both tools will be applied to suitable problems of varying degrees of difficulty.

In the transport modelling, further work will be carried out to take into account the variation in time, and space of mineral, and groundwater composition. The underlying simplifications of the dynamic K_d concept will be further analysed in order to fully defend the methodology. Furthermore, developments where the hydrogeochemical calculation tool (DarcyTools or ConnectFlow) and the transport model (MARFA) are coupled together, in a more effective way are planned.

Furthermore, when it comes to transport of solutes, efforts will be made within the fields of matrix diffusion, and sorption, with regard to conceptual understanding, reduced uncertainty in transport parameters and further development of modelling tools (for example MARFA). Efforts are also required concerning advective transport, dispersion, electromigration, gas transport, X-ray microtomography and colloidal transport to gain a better understanding of transport processes and provide input to the calculation tools. In order to reduce the uncertainty in transport parameters, new measurements should be carried out under relevant and well-defined conditions and on site-specific material. Sensitivity studies of transport parameters should also be carried out to investigate the influence of the variability of the rock, for example calcite occurrence on fracture surfaces and its impact on matrix diffusion.

In an ongoing modelling attempt, matrix diffusion is viewed as advective transport in microfracture networks. This modelling is coupled to experimental results, and uses data from fe.g. X-ray microtomography. The development of MARFA, to handle a dynamic K_d concept, will continue and be linked to available sorption data. Experiments are planned to obtain transport properties of the rock in terms of effective diffusivities and sorption data on larger pieces of rock. In order to be able to perform these experiments during a reasonable amount of time, SKB will use methods based on electromigration. The electrical methods will be further developed and tested to confirm measurement results both on lab and field scale. The knowledge that is obtained within electromigration can also be used to study the effects of earth and stray currents on transport of solutes and corrosion. The methods can also provide insight in the variability of the chemical composition of the porewater, which in turn can provide a better understanding of matrix diffusion. The state of knowledge in modelling of gas transport, both when it comes to dissolved gas in groundwater and as two-phase flow, should be improved prior to future assessments of post-closure safety. Within a PhD project related to the development of the channel network model Chan3D, e.g. the functionality of radial diffusion from channels into the rock matrix will be added and the impact of this will be studied.

Within the SKB Task Force on modelling groundwater flow and transport of solutes, Task 8 will be reported by the modellers, and evaluated by an external reviewer. The modelling within Task 9 will continue as described under the section "Current situation" above.

11.5 Link between near-surface and deep groundwater

In order to understand the hydrogeological processes in the rock, a good understanding of the near-surface groundwater systems is also required. Traditionally, different modelling tools have been used for the surface and deeper lying groundwater systems, but with increased computational power and developed modelling methodologies, the coupled system can better be described.

Current situation

In the modelling of SFR in the SR-PSU safety assessment (Öhman et al. 2014), a description of the near-surface groundwater system was included in a model of the deep groundwater system in a better way than previously. Specifically, a dynamic regolith model for layer geometries, i.e. soil geometries that changed over time, and with calibrated properties, was used. Furthermore, the course of lakes and streams and their levels were included dynamically in time for the different time steps. For the different time steps, preprocessing was done of topography data so that all local low-lying points were removed and replaced by their corresponding water levels so that more realistic pressures were achieved. The surface hydrological system was thereby better represented overall than in previous models.

Furthermore, it can be noted that also in the MikeShe model for the surface system used in the modelling of SFR, a more extensive analysis was carried out of the deep groundwater system; for example, transport from the repository to the surface was studied (Werner et al. 2013).

Programme

A strategy will be developed for how, and on which level, surface hydrological processes can be incorporated in the models for the deep groundwater. The advantages of including a better surface-hydrological understanding are that the models attain a more realistic description and property assigning of the surface layers, and also that the models produce more realistic transport paths for solutes in these surface layers. The strategy developed is based on a procedure where the surface hydrology and the near-surface groundwater systems are first conceptualised, calibrated and modelled in a specific calculation tool (model) for surface hydrology, for example MikeShe, so that measured data (time series of pressure, runoff etc.) can be reproduced. The conceptualisation of the surface system at the relevant complexity level is then implemented in one of the calculation tools for the deep groundwater system (DarcyTools or ConnectFlow). Tests are being carried out to verify that these models can reproduce measured data. A first test of this strategy is planned in the ongoing safety evaluation of SFL.

As concluded above, a better surface-hydrological description is expected to also provide a better understanding of transport in the near-surface system when particle tracking is carried out in the flow models. The particle tracking in groundwater flow models provides discharge points for the biosphere models, whereas radionuclide transport models provide nuclide fluxes. The coupling between geosphere and biosphere model (dose model) is often implemented (for example in SR-Site) by using a conversion factor that represents both a spatial and temporal peak value. This is thereby a highly pessimistic approach which presumably will greatly overestimate dose effects. Furthermore, the dose effects are overestimated due to the fact that dispersion and dilution in the geosphere are neglected in the models that are traditionally used in the assessments of post-closure safety. The effects of these simplifications, and alternative approaches to modelling, will be investigated further. In the ongoing safety evaluation of SFL, the specific effect of dispersion and dilution in the geosphere will be investigated.

11.6 Development of hydrogeological calculation tools

SKB mainly uses the calculation tools ConnectFlow, DarcyTools and MikeShe for hydrogeological modelling. For modelling of waters courses and their interaction with groundwater e.g. Mike11 is used, a one dimensional tool for channel flow that is directly linked to MikeShe.

The calculation tools are used for different purposes but are also partially overlapping. ConnectFlow and DarcyTools focus on the deep groundwater, but also handle the near-surface groundwater system, whereas MikeShe focuses on the near-surface groundwater system and surface hydrology (linked to atmospheric processes), but also handles the deep system on a simplified level (see also Section 11.5). The calculation tools are functional for their respective usages, but all need to be constantly maintained and further developed to remain up-to-date.

Programme

Initially it is noted that the primary development of hydrogeological models required in future stages of SKB's programme is linked to the development of a DFN methodology and is described in Section 11.3. Other development work within hydrogeological modelling (described in this section) has slightly lower priority and should be seen as an effort to maintain and develop the functionality of the calculation tools in line with the progress of scientific research.

For ConnectFlow, mainly efforts to develop the DFN module are planned. One effort aims at including functionality so that the aperture of fractures (and thereby their transmissivity) can depend on the current stress field. Whether the simplest way to calculate the stresses is in ConnectFlow or in a free-standing calculation tool needs to be investigated. In a first step, a simple form of one way hydromechanical coupling (i.e. stresses affect aperture/transmissivity and thereby the flow) is planned.

The DFN module is also planned to be expanded with functionality so that it includes particle transport in density-driven flow, and matrix diffusion. Furthermore, the possibility of implementing a dynamic K_d concept directly in ConnectFlow will be investigated. This would mean that external software for dynamically calculated K_d values (see Section 11.4) does not need to be used; rather that K_d values for sorption would be calculated directly in the groundwater flow model based on local chemical and mineralogical conditions. Specifically, the chemical variation in the matrix could then be fully handled; in simplified concepts such as that implemented in MARFA (see Section 11.5 above), it is assumed that a chemical change in the matrix takes place at the same time as it occurs in the fracture, i.e. the effect of the chemical gradient in the matrix is neglected.

DarcyTools has not previously been used for a detailed analysis of a KBS-3 repository, i.e. calculation routines for all quantitative entities that are propagated to calculations of long-term safety after closure have not been developed. Development is planned to calculate in a consistent manner the quantities required, and also to implement functionality in order to support the different geometric and hydraulic acceptance criteria for choice of deposition holes that are produced in other parts of SKB's programme (see Section 11.1).

The up-scaling methodology of fracture networks to continuum properties that is used today in DarcyTools is associated with certain limitations. In particular, connectivity properties and direction dependency suffer unless the numerical resolution is sufficiently high. Development is planned so that the continuum representation describes in a better way the underlying fracture network with respect to connectivity and direction.

It must also be investigated how the methodology for conditioning produced within the development of DFN (see Section 11.3) is best incorporated in 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 the developed geometric-hydraulic conditioning is difficult in practice in the current version of DarcyTools. Two alternative approaches are possible. One is that conditioning takes place in another tool, and that the conditioned fracture network is then read into DarcyTools for further upscaling. The second approach is based on the geometric information at single fracture level being saved internally in DarcyTools so that it is available even after upscaling. It remains to be resolved how the iterative conditioning process, where changes in fracture locations are required, in practice shall be carried out.

MikeShe has been developed so that the hydraulic properties can change as a function of time to simulate freezing and thawing processes in permafrost environments (Johansson 2016), see also Section 11.8. The functionality for time-varying properties makes it possible to analyse how changed land use and soil frost processes affect the water flows and transport processes. However, this functionality should be refined to simplify the assignment of parameters.

The use of MikeShe for water balance calculations for smaller catchment areas will be developed so that local geology and smaller subareas can be analysed consistently with the dose analyses made in the biosphere assessment.

11.7 The impact of the ice load on the flow and transport properties of the rock

The ice load from an ice sheet affects the hydrogeological system differently depending on the thermal basal conditions of the ice. At depth it is mainly important whether the ice sheet is cold-based or warm-based. If the ice sheet is cold-based, it represents a large "mechanical" load that can lead to compression of the pore system of the rock. 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 a similar manner 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

Unless the mechanical load and the hydraulic pressure interact, a unilateral increase of the hydrostatic pressure, especially near the surface, can for example 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 the properties of the rock mass, a worst-case scenario has been postulated based on a lower limit of normal stiffness of the fractures, 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 implies that it would be possible to bound the consequences of the glacial cycle on the THM behaviour of the rock mass and thus also on its transport properties in a more realistic way.

Programme

Three-dimensional THM models on different scales will be established and the uncertainty range will be studied. The problem is extremely site-specific and models that take into account the impact of ice loads and possible thermal basal conditions in a future ice sheet in Forsmark will therefore be established. The development is initially of a research character; operational, site-specific models can be available at the earliest prior to an application for the operating licence.

11.8 Effect of freezing on the flow and transport properties of the rock

Large temperature decreases at ground surface can cause ground freezing and if this freezing lasts more than two years in a row permafrost arises by definition. Permafrost in soil and rock leads to reduced permeability and causes volume changes, which can in turn lead to new formation, widening and/or propagation of existing fractures. How the hydrogeological system is affected during different stages of permafrost growth or melting is site specific and needs to be analysed in the site models.

Current situation

The understanding of the influence of freezing and thawing processes on hydrology in periglacial areas is limited, and there is a general lack of data in order to be able to conceptualise the hydrological periglacial system (Vaugan et al. 2013, Bring et al. 2016). Within the framework of the GRASP (Greenland analogue surface project) hydrology in permafrost environments (periglacial environments) has therefore been studied on catchment scale.

Field investigations, together 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 carried out in a small catchment on western Greenland in the vicinity of Kangerlussuaq. In areas with permafrost two groundwater systems occur, 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 and in this layer a shallow groundwater flow occurs during the summer months. The investigations within the GRASP have focused on processes in the active layer, since active layer processes have large influence 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 rock have been measured within the framework of the GAP (see Section 13.5).

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 11-3.

During 2015 and 2016, six scientific publications have been published within the framework of the GRASP (Johansson et al. 2015a, b, Johansson 2016, Rydberg et al. 2016, Petrone et al. 2016, Lindborg et al. 2016).

Programme

The focus of the programme is to develop the understanding of spatial and transient changes of rock and soil properties caused by permafrost and glaciations, and its influence on groundwater flow and solute transport. Water flow and transport through the frozen ground needs to be further investigated and in situ experiments are therefore planned at the GRASP site. Field experiments in combination with modelling where the 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 GRASP 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 four years.

The increased conceptual understanding achieved as a result of the GRASP project will be applied to Forsmark to gain a better understanding of how the hydrology can be affected during future periglacial conditions. Within this framework modelling studies for surface hydrology in Forsmark will be conducted, where new data and knowledge from Greenland is used.

11.9 Handling of glacial cycle in hydrochemical and transport modelling

Current situation

The ongoing development in hydrogeological modelling makes it possible for geochemical and transport processes (Section 11.4) to be coupled to the hydrogeological modelling tools.

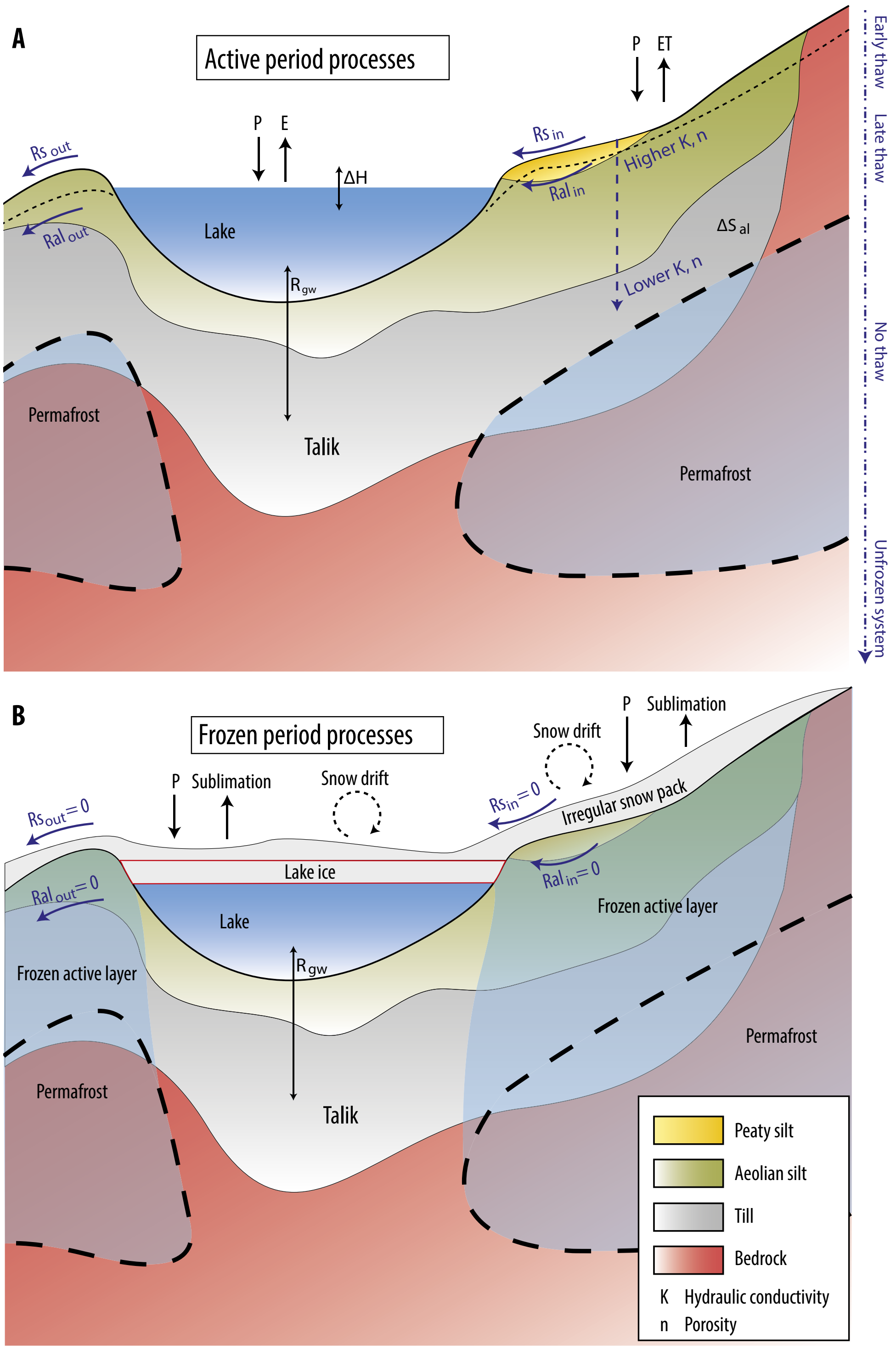


Figure 11-3. *Conceptual model of the hydrology in the periglacial catchment area investigated by SKB on Greenland (Johansson et al. 2015b). Two models have been developed, one for the active and one for the frozen period. Rs – surface runoff, Ral – groundwater flow in the active layer, Rgw – groundwater exchange between lake and talik, P – precipitation, E – evaporation, ET – evapotranspiration, ΔH – lake level variation.*

How the hydrogeological system is affected during different stages of permafrost growth or melting is site specific and needs to be analyzed in the site models. The GAP (Section 13.5) and GRASP (Section 11.8) projects have led to an understanding of how hydrological processes in local catchment areas behave under permafrost conditions.

Programme

Within the programme, further developments of modelling tools that couple hydrogeological and hydrogeochemical processes, (DarcyTools–PFLOTRAN and ConnectFlow–PHREEQC, see Section 11.4) will be tested, and used to study the effects of glacial melt water on the Spent Fuel Repository, e.g. by including penetration of oxygenated groundwater and the redox reactions that can arise. An additional prioritised effort is to use the coupled modelling tools to analyse a glacial cycle. Temperate and glacial climate conditions are included here as well as conditions with permafrost in order to determine if, and how today's geochemical and hydrogeological conditions reoccur after a glaciation. For example, the new knowledge will be applied within site-specific, and safety related modelling of Forsmark.

11.10 Effect of freezing on the mechanical properties of the rock

Current situation

During the past 15 years, SKB has participated in the research project DECOVALEX (Chan et al. 2005) and also contributed with supervision in an ongoing PhD programme together with Chalmers University of Technology. The project has been partially focused on different aspects of permafrost growth and freezing of groundwater in the bedrock, where for example volume changes could lead to new fracture formation 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 capabilities has been reported by Selvadurai et al. (2014). However, there are still unresolved questions and uncertainties that need to be addressed in order to handle this issue for future milestones, for example the PSAR and SAR for the Spent Fuel Repository and SFR.

Programme

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 can cause new flow paths in the rock nearest 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 that incorporate both DFN descriptions and mechanical properties that are relevant for the different repositories will be established.

11.11 Seismic impact on post-closure safety

Under the assumptions made in SR-Site, the effect of earthquakes on post-closure safety constitutes a significant contribution to risk for the Spent Fuel Repository (SKB 2011b). The risk contribution is strongly linked to the frequency-magnitude connection suggested for short- (Bödvarsson 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 with regards to the earthquake activity in conjunction with deglaciation of an ice sheet. An underestimation of the seismic activity entails an underestimation of long-term risk, while an overestimation implies an over-dimensioning of the final repository.

Methodology for modelling of earthquakes has been developed over a long time (Fälth and Hökmark 2006, Fälth et al. 2007, 2008), and has after extensive tests (Itasca 2013, Fälth 2014, Fälth et al. 2015) reached a high level of maturity. However, a larger number of calculation cases are required 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 that indirectly concern earthquake modelling are presented in Section 11.12.

Studies that are planned in the present research programme aim at gaining a better understanding of the glacially induced earthquakes and their couplings with present-day seismicity, with the goal of reducing uncertainties through in-depth understanding and thereby increasing confidence in assessments of post-closure safety.

11.11.1 Seismic monitoring

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 7 000 earthquakes (Figure 11-4) with magnitudes between approximately −2 and 5.3. The last large expansion of the network, in southwestern Sweden, was instrumented in 2008, and since then SNSN has relatively good coverage in the seismically most active areas of the country. Currently, there are 65 permanent stations installed.

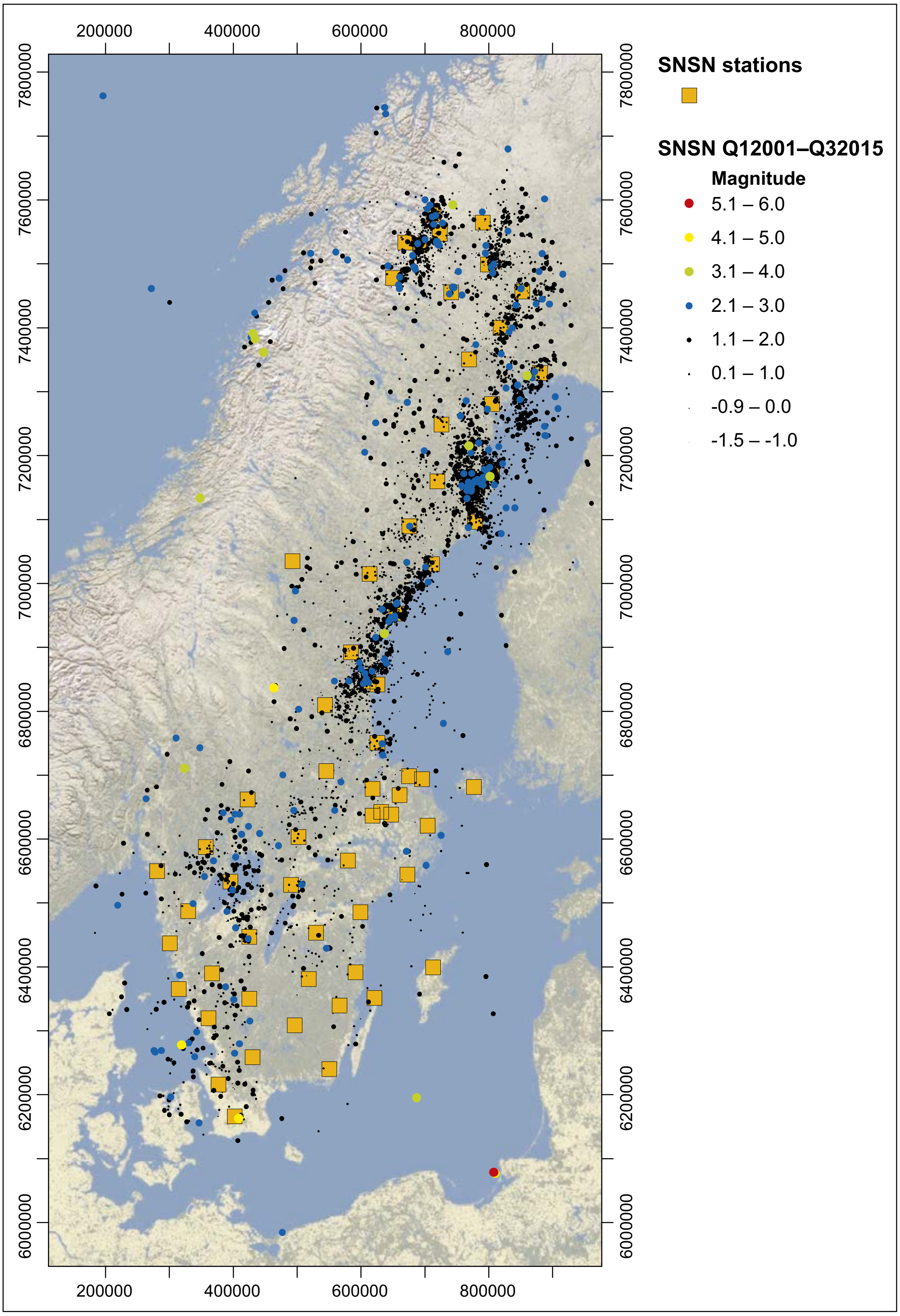


Figure 11-4. *Earthquakes recorded by the Swedish National Seismic Network (SNSN 2015) during the years 2001–2015 (Mw – moment magnitude). Data from SNSN (Bödvarsson 2012).*

Since 2008, SNSN has continuously gathered data in real time from all stations, which means that the quantity of data for analysis has increased significantly compared with previous years when only data segments from detected events were collected. The new, refined seismic network 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.

Programme

A tomographic analysis (see for example Tryggvason and Linde 2006), which yields a three-dimensional velocity model, is under way. The results will be used for relocalisation of earthquakes. Linked to this, a so-called "multi-event analysis" of earthquake localisations is also under way, which considerably improves precision of location determinations and calculated magnitudes. Relocalisations will serve as a basis for new calculations of focal mechanisms and stress inversion which, among other things, will be used to calculate the stress field at great depth.

11.11.2 Investigations of glacially induced faults

Current situation

Burträsk is the most seismically active area in Sweden (Figure 11-5c). Several years ago, Uppsala University densified the seismic stations in the Burträsk area to obtain more exact data (Lund et al. 2015). Results of the densification have been very successful, since it has been shown that the earthquakes cluster along the fault scarp which was first identified by Lagerbäck and Sundh (2008) and later detailed by Mikko et al. (2015). Reflection seismology has previously shown a well-defined fault plane down to about three kilometres' depth 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.

A system of possible, glacially induced faults has been identified in Lillsjöhögen, Jämtland, by Mikko et al. (2015). The area has been subject to geological investigations but existing information is insufficient to be able to confirm or dismiss lineaments as glacially induced faults. The seismic activity in the area is low to very low; however, the few registrations of SNSN (2015) indicate a diffuse band of earthquakes parallel to the system of lineaments in an otherwise low seismic area. This indicates that the structures represent perhaps only a small part of a much greater structure, which, among other things, affects estimations of the magnitude of future earthquake.

Registrations by SNSN (2015) show a very distinct, north-east, trending cluster of earthquakes located north-west of Iggesund (Figure 11-5 b). As a clear correlation between recent earthquakes and glacially induced faults has been detected for most of the faults registered so far (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 that these indications be explored more closely.

Programme

For the area north-west of Iggesund (Figure 11-5 b) a densification of the seismic stations is planned with temporary stations to increase precision in the localisations and thereby allow an identification of the structure that is currently seismically active. Furthermore, a detailed reinterpretation of the existing geological and geophysical information is planned. A small amount of field work (mapping) may be required to confirm the interpretations.

The other recently identified or updated indications of glacially induced earthquakes (Figure 11-6) will be investigated more closely to ascertain whether or not they really represent paleoseismic events. For these, presently smaller seismically active structures, SKB is planning a regional, supplementary analysis to the analyses recently performed by Smith et al. (2014) and Mikko et al. (2015).

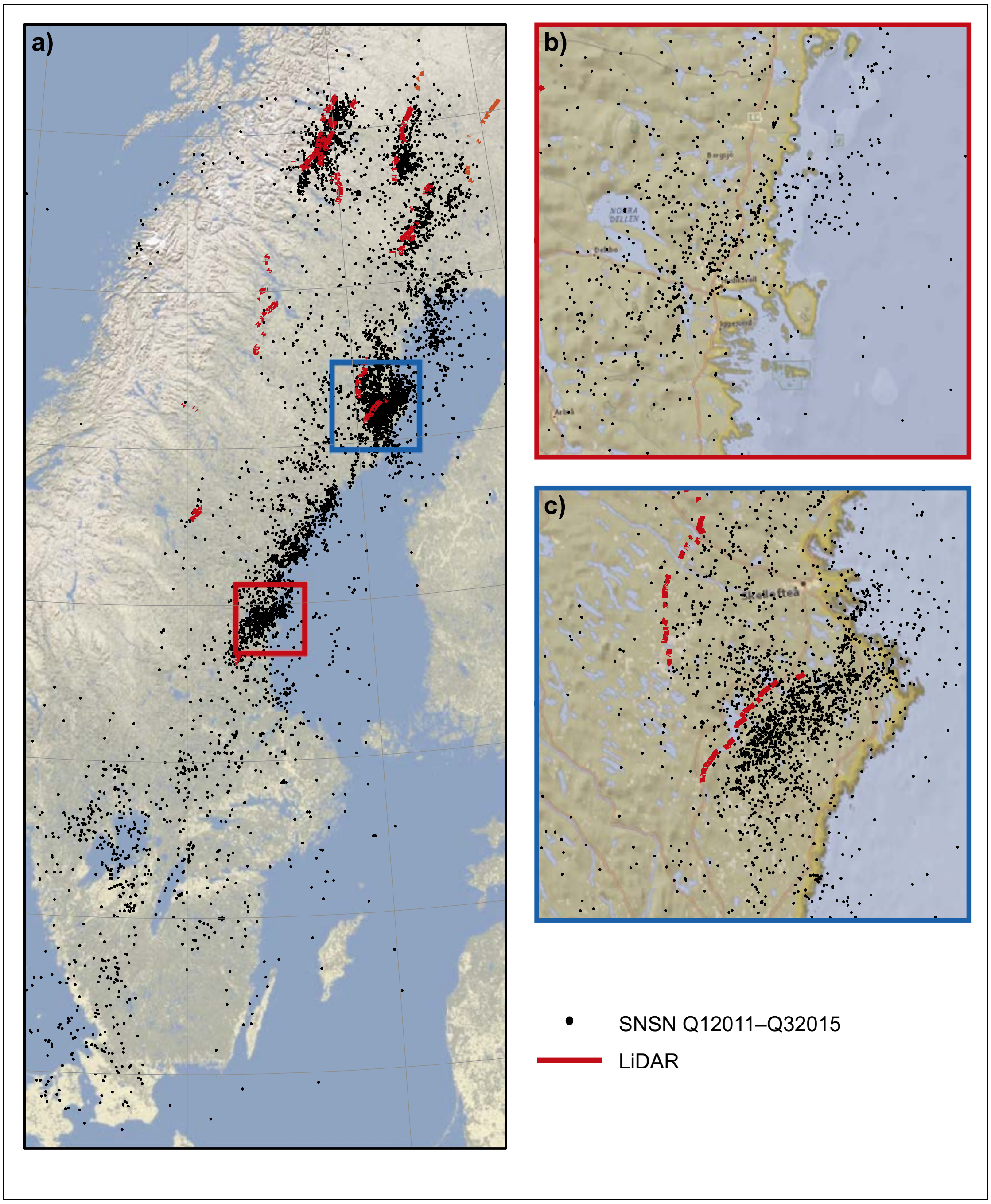


Figure 11-5. *a) Cluster of earthquakes recorded by SNSN (2015) in b) north-west of Iggesund and c) in connection with the Burträsk fault. Fault scarps are identified by Mikko et al. (2015) on the basis of the new national elevation database, NNH, (Lantmäteriet 2015) and of Lagerbäck and Sundh (2008) with the aid of mainly photogrammetry and mapping.*

In a first step, SGU's existing geophysical data is compiled on key areas previously identified by Mikko et al. (2015). These data include aerial surveys, mainly magnetic field, VLF (very low frequency) and natural gamma radiation, as well as surface based techniques, mainly magnetic field, gravitation, radar, resistivity and VLF. Some processing and remodelling of data may be required in order to analyse the specific issue of glacially induced faults. In addition to the possibility to coarsely determine the potential fault geometries, the kinematics of the structures can also be addressed in favourable cases. Modelling of regional data can identify gaps in knowledge and serve as a basis for more detailed studies.

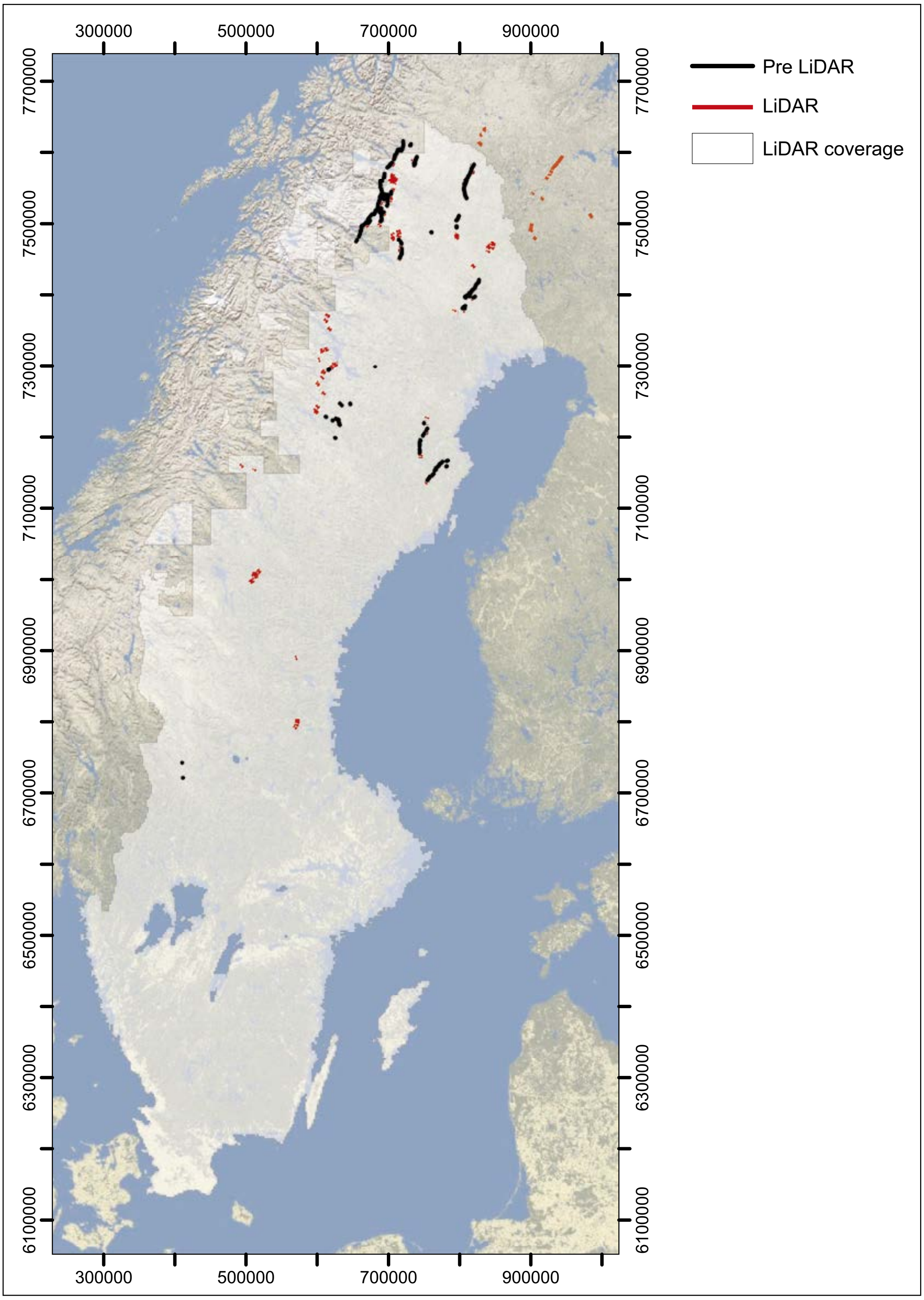


***Figure 11-6.** Identified topographic lineaments that comprise possible glacially induced fault scarps. Black lines represent older interpretations based on mainly aerial image interpretation while the red lines show lineaments identified by recently performed measurements by Lantmäteriet (2015) (Swedish National Land Survey). The grey boxes represent the coverage (2015) of the high-resolution Lidar.*

For areas where more detailed investigations are required, SKB intends to supplement the regional data with point surveys of key locations. Besides geological investigations, the sites will be investigated with a combination of the following methods: terrestrial RMT (magnetotelluric), ERT (electrical resistivity tomography), radar, magnetic fields, dense gravitational measurements supplemented with a number of temporary seismic stations.

11.11.3 Investigation of possible tsunami

A prominent erosion discontinuity in glaciolacustrine clay covered by coarse sand and gravel along the coast of Uppland has been described. Lagerbäck et al. (2005) propose that this was caused by strong, near-coastal sea currents during ongoing regression of sea level while Mörner et al. (2000) argue that it was caused by a tsunami triggered by a nearby, glacially induced earthquake.

Programme

To determine which of the above mentioned hypotheses are the most probable, and thereby provide a deeper understanding of Swedish paleoseismicity, SKB intends primarily to investigate the stratigraphy and sedimentology of the discontinuity and the overlying layers in a number of trenches. SKB intends mainly to ascertain flow directions and shear stress conditions when the coarsest material was deposited. Secondarily, lakes situated above the highest coastline will be examined with intention of searching for evidence of tsunami deposits. The study consists mainly of analyses of sediment cores from lake bottoms. If possible, carbon-14 dating will be linked to geological observations.

11.11.4 Modelling of seismic impact on the final repository

Current situation

A methodology for modelling of earthquakes and their effects on a final repository was defined in Fälth and Hökmark (2006) and applied after further refinement (Fälth et al. 2007, 2008) in SR-Site (Fälth et al. 2010). This development has in part occurred in the form of a PhD project with interim results presented in Fälth (2015). In addition to these efforts, development of methodology and calculation tools has been carried out in cooperation with Posiva, using both rock stress conditions at the end of the latest glaciation (Fälth and Hökmark 2011, 2012) and in the present day (Fälth and Hökmark 2015).

The models that have been analysed so far have intentionally been based on what is judged to be pessimistic or very pessimistic conditions. With increased process understanding, substantiated by observations, more realistic assumptions can greatly reduce the estimate of induced shear, and thereby of long-term risk, but also provide an opportunity to optimise the repository layout. For example, a more realistic assumption of a non-flat fracture plane can greatly reduce induced shear as found by Lönnqvist and Hökmark (2015). On the other hand, the variation of strength over the fault plane would imply that certain parts undergo a more dramatic fracture process, resulting in more powerful seismic waves and ultimately larger induced displacement.

Programme

Besides variation of properties over, primarily, the fault surface and, secondly, the surfaces of the target fractures, the following development efforts are planned when it comes to modelling:

- Improved implementation of fault edges. At the present time the edges of fracture zones lead to unrealistic, local, stress concentrations which result in large induced shear displacements on the nearest situated target fractures.
- Implementation of plastic to semi-plastic deformation along the fault plane. This may govern the fracture propagation process and, as a consequence thereof, the induced shear of the target fractures. Ideally, a so-called “Damage zone” is also included (Choi et al. 2016) in 3DEC or in separate studies using PFC3D. Hybrid models of FLAC3D and PFC3D will be considered after systematic evaluation.
- Implementation of alternative fracture propagation algorithms (rupture algorithms). Currently the kinematics of the fault is controlled in 3DEC models through a parametric reduction of strength until rupture occurs. It is, however now standard in various modern calculation tools to let the rupture propagate spontaneously. The rupture process is then controlled by the properties and stress of the zone to a greater extent than with the approach SKB has so far used. It is common to let the strength be determined by a so-called “slip-weakening-law” (SW) or “rate-and-state-law” (RS). SKB intends to primarily develop spontaneous rupture with SW, since it is less complex to implement. The study includes systematically evaluating the effect of different locations of

the fault's hypocentre, in order to exclude underestimation of induced shear, and systematically examine the effect of the shear velocity of the fault which currently is by default assumed to be 70 percent of the shear wave rate.

- In-depth analyses of fault interaction, or more precisely how an earthquake in one zone can transition to an earthquake in another zone and lead to a greater total effect on target fractures.
- In-depth analyses of the effect of splays from faults.
- Fuller propagation of uncertainties. SKB does not at present have a sufficient number of modelling cases to cover a pessimistic range of outcome. For example, there is a need for greater spatial variability in the input rock stresses (magnitude, direction), spatial variability in the properties of the rock mass (for example fracture intensity), spatial variability in shear and bulk moduli and the temporal variability that follows from the decaying of the thermal load. In this study, calculation cases taking into account alternative glacial load evolutions in addition to those made for the reference glaciation are included. As a part of this study, stresses on/around the fault are being studied, above all how stress variations affect failure extent and stress alteration around the fault.
- Stick-slip. The target fractures have so far been assumed to have static Mohr-Coulomb strength properties (i.e. independent of shear displacements). SKB also intends to model cases where the target fractures are allowed to lose some of their strength in connection with displacement (i.e. to model so-called "stick-slip behaviour").
- "Damage zone". SKB intends to model cases where the target fracture partially lies in the damage zone or touches the source fault both with static Mohr-Coulomb properties and with target fractures whose strength decreases in parallel with that of the source fault. Ideally, this knowledge could be used in order to optimise the repository by using rock volumes which in the current layout are occupied by respect distances.

11.12 The mechanical properties of the 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 with regard to its inherent deformation zones and fractures on different scales. Empirically based methods for characterisation of the mechanical properties of the rock mass are very effective for design and construction, but analysis of post-closure safety requires a deeper and more fundamental understanding and characterisation of the coupled THM processes which affect the behaviour.

The numerical tools for modelling a synthetic rock mass have continuously been developed and in recent years undergone a very fast evolution. 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

Discontinuum methods based on the discrete element method (DEM) have been developed in the software UDEC (Itasca 2014a) and 3DEC (Itasca 2013) for characterisation of the mechanical (M) and hydromechanical (HM) properties of the rock mass. These tools have been used by SKB as a complement to the traditional, empirically based, 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 2014b) have previously shown the ability to mimic fundamental and more subtle aspects of fracturing and fracture propagation in rock (Potyondy and Cundall 2004). This type of software has also recently been developed to be able to include discrete fracture networks and has thereby permitted more complex representations of the rock mass (Mas Ivars et al. 2011). The development of the software 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. Modelling of SRM is, however, still limited by numerical calculation capability, which leads to limitations in terms of model size (the scale of the problem), degree of detail and complexity. Despite this, SRM models have decisively contributed to the generation of DFN based analytical equations that can describe the properties of the rock mass. A strategy in which the analytical and numerical models are used in combination has been produced, and Darcel et al. (2015) have recently applied this strategy on behalf of Posiva with the aim of defining an analytical method to predict effective elastic properties at different scales (up to 100 metres).

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 the stiffness of the rock mass, as well as how strength is affected on different scales. SKB has, together with Posiva and NWMO, participated in the Post project (Fracture parameterisation for repository design and post-closure analysis), which aims to learn more about how a fracture's shear properties can be scaled up. The project will be reported in 2016. The effect of large-scale undulation on the fracture's shear mechanical behaviour is also being studied as a part of PhD studies at Chalmers University of Technology, supported by SKB (Lönnqvist and Hökmark 2015). In another PhD project which was recently concluded, the effects of fracture geometry on hydromechanical properties of crystalline bedrock were studied (Thörn 2015).

The effect of dilatation decreases, with high normal stress; however there are still uncertainties in how much this affects the 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). Esaki et al. (1999), on the other hand, observed transmissivity increases of between one and two orders of magnitude in tests performed under high normal stresses on artificially created granite fractures. However, 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 a repeated shear in the opposite direction, gave insignificant transmissivity effects for shear displacements up to 10 millimetres. In the current understanding of the THM aspects in Forsmark and Laxemar (Hökmark et al. 2010), it is assumed that high normal stresses suppress transmissivity effects.

The uncertainties that still exist concerning HM properties of the fractures thus mainly concern the scale effect and changes in transmissivity as a function of shear and normal load.

Programme

Efforts are planned to develop a "state-of-the-art" methodology for calculation of the hydromechanical properties of the rock mass. The methodology should include the following:

- The scale effect. Effective HM properties on different scales based on DFN (PFC, 3DEC, analytic approaches) need to be established for the initial state.
- Properties of and rules for setting of DFN properties, for example the transmissivity of fractures, need to be analysed and developed so that a relevant initial state can be obtained (see also Section 11.3). How this may be conceptualised in large-scale models (for example DarcyTools, 3DEC) needs to be studied.
- 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.

In order to improve the hydromechanical understanding of fractures, efforts are planned to clarify the relationship between aperture and transmissivity. The hydraulic aperture, and thereby transmissivity, is a function of mechanical aperture, its standard deviation, and 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 analogous 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 effect on flow due to shearing will also be carried out.

The following mechanical parameters of importance need to be determined or estimated:

- The effect of the normal stress on the normal stiffness, shear stiffness, friction angle, cohesion and dilatancy angle.
- The shear displacement where dilatancy starts and ceases.

- The effect of fracture minerals and fracture infilling thickness on mechanical and hydraulic properties of fractures.
- Scale effect on shear strength, normal stiffness, shear stiffness and dilatancy angle, as a function of fracture roughness, undulation, contact area and mineral filling.

11.13 Induced deformation in the rock mass caused by thermal, seismic or glacial load

Current situation

SSM considered that certain conceptual issues from the review of RD&D Programme 2007 had not been taken into account sufficiently in RD&D programmes in 2010 and 2013. SSM 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 where SSM particularly wanted further analyses were reactivation of deformation zones and fracture opening/closure due to the 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 influence of an ice sheet on fracture opening/closure, fracture propagation and short-circuiting of the fracture network between nearby deposition holes (Min et al. 2005, 2015, Back and Stephansson 2012). Regarding technical modelling-related aspects, SSM indicated that when it comes to rock mechanics and coupled processes, SKB should strive in its modelling to quantify the influence of realistic fracture geometries on states and processes relevant to safety.

Regarding the 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 testing of creep in rock samples, numerical model analyses of the effect of reduced fracture toughness due to stress corrosion on the strength of the rock, evidence from plate tectonic processes and observations of rock stresses in quarries. The article concludes that a stress threshold (i.e. a deviatoric stress that can be sustained indefinitely) exists for crystalline rock types (40–60 percent of uniaxial compressive strength). Furthermore, it was concluded that an 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 means that a linear extrapolation to an ultimate zero strength is unwarranted (Potyondy 2007).

SKB has since the RD&D Programme 2007 investigated very long-term processes for rock strength evolution. However, there is 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, which SSM also pointed out in its review of the RD&D Programme 2013. In this context the influence of temperature 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 phases. In such studies, there will be a need to take into account stress corrosion cracking (SCC) in all load cases (tension, shearing, tearing) since fracture growth takes place not only under tension, but also under shearing or tearing at high confining pressure (Backers 2005, Backers and Stephansson 2012).

Programme

The following development regarding modelling capacity is planned:

- Fracture propagation. Development of analytical and numerical methodology for studying how fractures grow together on both small and large scales. A programme with a limited number of laboratory tests is required for this.
- Transmissivity changes in the rock mass around deposition tunnels and deposition holes due to stress changes caused by seismic, thermal and glacial load.
- Spalling in deposition holes and tunnels as an effect of stress changes caused by seismic, thermal and glacial load.

11.14 Rock stress in Forsmark

Current situation

The rock stress model for Forsmark contains large uncertainties, and there is a great need for efforts to try to reduce these uncertainties. In its review of the RD&D Programme 2010, SSM pointed out that SKB should follow up the 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).

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 fracture 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 a number of pilot holes with a diameter of 120 millimetres in a tunnel section and then overcored with a drill bit of diameter 200 millimetres. Inverse modelling of the in situ stress field is based on strain data collected from the LVDT sensors, which measure in four different directions, a detailed laser-scanned model of the tunnel section where the measurement has been performed, and the modulus of elasticity measured on solid cores from the pilot holes. Verification measurements with an LVDT cell in the Äspö HRL at the 450-metre level, where the stress field is well-characterised, show that the results of the measurements with the LVDT cell agree with previous results from overcoring measurements as well as previous results and stress model (Christiansson and Janson 2003).

Programme

During the RD&D period, development of a rock stress model in 3D based on the current structural geology model (Stephens and Simeonov 2015) is planned. This rock stress model will serve as a basis for the updated calculations of the risk of spalling and stability within the different repository volumes, and give boundary conditions for further modelling of induced fracture displacements (seismic and thermal load). The plan is then that the model will be updated continuously in agreement with updates of the structural geology model and when new rock stress data (direct and indirect) are available. The main purpose of the model is to improve the estimation of the spatial variability of the shape and orientation of the stress ellipsoid in high resolution (30–100 metres blocks).

12 Surface ecosystems

SKB's research programme for surface ecosystems primarily intends to create a basis for calculations of potential radioactive dose to humans and the environment in the assessment of post-closure safety for the different repositories. The programme also provides a basis for environmental monitoring, assessments of any environmental changes and assessment of safety in existing and planned facilities. The programme also contributes to a 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. Presentation of results in scientific publications and at international meetings is an important part in ensuring the quality of the work. Furthermore, modelling tools and databases are continuously maintained.

The current research issues for the three different repositories in the area of surface ecosystems are largely overlapping. SKB deems that there are no remaining critical research issues in the area that must be resolved prior to the PSAR for the Spent Fuel Repository. When it comes to an extended SFR, the review may give rise to issues that must be resolved prior to the PSAR for SFR. For SFL, there are several issues that must be resolved before an application is submitted.

There are thus a number of issues that require further research, either as a result of the regulatory authorities' comments on the review of the submitted applications, or because SKB has deemed it necessary in order to reduce uncertainties in future safety assessments. The most important remaining issues in surface ecosystems are found within four different areas: 1) uptake paths and uptake mechanisms for various organisms; 2) temporal and spatial heterogeneity in the landscape; 3) transport and accumulation processes; 4) radiological, biological and chemical properties of certain substances that potentially can provide a great risk contribution.

An overview of SKB's work in this field in recent years can be found in a special issue of Ambio (Kautsky et al. 2013). For the most recently completed safety assessment, SR-PSU, SKB has published reports which describe assumptions in the modelling of the 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). The following is a short description of the work that forms the basis for further studies, and the planned programme for this RD&D-period. A complete account of produced results can be found in the reports from the most recent safety assessments.

12.1 Uptake paths and uptake mechanisms for radionuclides in various organisms

Current situation

Issues concerning uptake paths and mechanisms for radionuclides include both organisms and the aquatic and terrestrial ecosystems they are part of. Dose calculations traditionally use concentration ratios (CR), which describe the concentration of a substance in the organism compared with the concentration in food or a surrounding medium (water, soil or sediment). Uncertainties associated with the CR values are, however, large, and SKB has therefore continously worked on developing alternative methods to estimate radionuclide uptake in organisms (see for example Kumblad and Kautsky 2004, Konovalenko 2012). In the review of the RD&D Programme 2013, SSM encouraged this work.

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 main uptake in the food web is in the form of plant uptake, which is linked to the uptake of water and nutrient salts. In SKB's previous work, important uptake mechanisms for aquatic and terrestrial systems have been described in the books on ecosystems (Andersson 2010, Aquilonius 2010, Löfgren 2010). The basis for these descriptions is based largely on the site investigations, and work is under way to design a future monitoring programme that will provide material for necessary supplements.

A new PhD thesis presents the work of describing radionuclide uptake in aquatic food webs (Konovalenko 2014, Konovalenko et al. 2014, 2016). The PhD thesis does not, however, include running waters. Research indicates that certain radionuclides can accumulate in the bottom of running waters (Lidman et al. 2011, 2012, 2016) and SKB does not rule out that running waters in some scenarios may have a greater importance for radionuclide cycling in surface ecosystems than previously assumed.

Mechanisms for uptake of radionuclides in terrestrial ecosystems have previously been studied with the Coup and Tracey models (Gärdenäs et al. 2009). In a new PhD thesis, the Tracey model has been further developed. Among other things, the agricultural plant uptake of wet deposition via the leaves has been studied and the results have been included in the model (Bengtsson 2013). Work remains, however, on considering the effects of different uptake pathways and of how absorbed radionuclides are allocated in the plant, in the simplified models used in the safety assessment.

During the work with the safety assessment for SFR, factors such as accumulation 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 has been updated in accordance with this, and a new description for dispersion in near-surface atmosphere layers has been implemented (Saetre et al. 2013). In the safety assessment, all carbon-14 is assumed to be 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 carbon-14 could also reach the biosphere in the form of methane in anoxic environments.

SKB has initiated studies to describe the cycling of methane in natural ecosystems (Natchimuthu et al. 2015). Methane reaching the surface ecosystems from the rock can be oxidised by micro-organisms and then be converted to biomass or carbon dioxide (Figure 12-1). Methane 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. Methane that contains carbon-14 from radioactive waste can thus either reach the atmosphere as methane or carbon dioxide, or be taken up in the food web via methane-oxidising microorganisms or via plant uptake of carbon dioxide formed by methane oxidation.

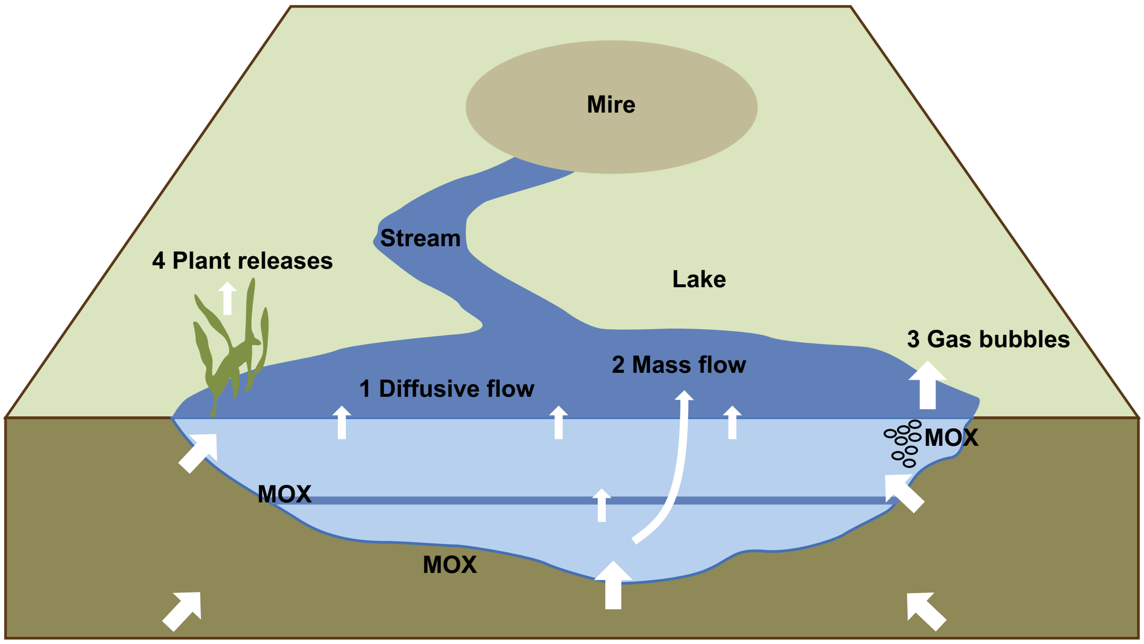


Figure 12-1. *Various processes that affect radionuclide discharge and transformation of methane in surface ecosystems. Methane can be released to the atmosphere by diffusion (1), water transport (2), gas formation (3) or plant emission (4). Methane can also be converted to organic matter or carbon dioxide by methane oxidising microbes (called MOX in the figure).*

Transport and uptake of gas can be of importance also for certain substances that may volatilised, for example iodine and chlorine (Hardacre and Heal 2013), which was also pointed out by SSM in the review of the RD&D Programme 2013. Chlorine in nature is not inert (which is often assumed), but can be transformed to organic compounds with other uptake and storage pathways than inorganic chlorine. The residence time for chlorine in terrestrial systems is surprisingly long, and there are a number of different pools of chlorine that behave in completely different ways (Bastviken et al. 2013). The new picture that is emerging with respect to chlorine turnover in nature is based on studies in surface environments; there are no corresponding studies in aquatic systems.

The assessment of doses to non-human biota that was carried out in SR-Site has been supplemented by Jaeschke et al. (2013), and SSM have responded positively to the additional work. The methodology has subsequently been used in SR-PSU, integrated with transport and accumulation calculations. However, SSM has requested explanations for the differences in results for certain nuclides between the tools Erica and RESRAD (Stark 2015), and, internationally, the need to be able to extrapolate effects of radiation from the individual level to the population and ecosystem level has attracted attention (Bradshaw et al. 2014, Bréchignac et al. 2016). SKB notes that models that describe uptake and accumulation in food webs and ecosystems provide an opportunity for independent comparisons with both the Erica and Resrad tool. This type of model also gives an opportunity to scale up calculated doses to the population and ecosystem level.

Recently, the EU network STAR presented several different methods to estimate concentration ratios (CR), which to a larger extent than previously are based on biological and ecological mechanisms (see summary in Beresford et al. (2016)). Some of these methods are, by extension, interesting to use together with SKB's ecosystem models in the modelling of radionuclide transport, to thereby reduce the uncertainties in the radionuclide uptake in organisms.

Programme

SKB will continue its long-term efforts to replace or supplement certain important concentration ratios for organisms with mechanistic models. For existing models a validation with field data (for example nutrients and other stable substances) is planned. Furthermore, a comparison is planned of the results from the CR based methods used thus far with results from allometric methods (Beresford et al. 2016), in order to draw conclusions regarding uncertainties in different uptake models.

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 lakes, differences when it comes to hydrology, potential for chemical precipitation of different substances (see Section 12.4), and biological uptake mechanisms can provide other prerequisites for accumulation of radionuclides and dose. In parallel, work with development of the terrestrial ecosystem model will continue, with a focus on root uptake in plants and links between radionuclide uptake and water transport or primary production.

The importance of gas transport, for example of methane and carbon dioxide, for the cycling of carbon-14 in water, land and atmosphere will be investigated, along with uptake processes for gas via primary production and microbial metabolism. In order to assess the extent to which methane from deep soil layers is diluted by superficially produced methane, and how large a fraction is oxidised or released to the atmosphere, methane flow in natural ecosystems will be monitored by newly-developed measurement methods.

There are knowledge gaps today when it comes to the properties of chlorine in aquatic environments (wetlands, lakes and running waters), despite the fact that these environments are important recipients of potential releases of chlorine-36. It is possible that large quantities of chlorine may be bound in organic matter in surface sediment and then taken up in the food web, with long residence times and accumulation in organisms as a result. Alternatively, chlorine cycling in these systems is controlled mainly by water velocities and water flux. In order to better model the transport and accumulation of chlorine-36 in aquatic environments, investigations are planned with the purpose of linking new measurements of organic bound chlorine (see Section 12.4) 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 the volatilisation of chlorine as described in Section 12.4.

12.2 Temporal and spatial heterogeneity of the landscape

Current situation

One of the most important factors for calculating dose after release of radionuclides is the variation of the landscape in space and in time. Depending on the type of ecosystem in which a release occurs, the properties of the ecosystem and the point in time at which the release occurs, the calculated dose can be affected by several orders of magnitude.

As a basis for safety assessments in Forsmark and Laxemar-Simpevarp, SKB has developed a historical description that describes the processes that have determined the development of the landscape at the sites to date (Söderbäck 2008). The historical description has been combined with an understanding of the present-day state and function of the landscape (SKB 2008, 2009b), and together these descriptions constitute the basis for describing a probable development of the landscape under different assumptions regarding future climate and shoreline displacement.

The development of the landscape in Forsmark is determined mainly by climate variation and shoreline displacement (see Chapter 13). On top of these large-scale and slow changes, the landscape is affected by the deposition and reworking of sediment, which entails that lakes become shallower and infilled (Brydsten and Strömgren 2010). With the ecosystems succession the chemical and physical properties of the soil and water change, along with the species composition of plants and animals.

Hydrological discharge areas for deep groundwater are located at topographical low-lying points in the landscape, for example around lakes, streams and wetlands (see Section 12.3) The potential for transport and accumulation of radionuclides in these areas is determined partly by the properties and size of the discharge area and partly by the topography and properties in the local and regional catchment area. In many cases, the large-scale evolution of the landscape also defines the time periods when deep groundwater can flow up to the surface of a given site.

There is thus a natural covariation between size and properties of the discharge areas that may receive releases, which is 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 releases to the object are possible. Variations in local topography, the sequence of soil layers and ecosystem succession can also be expected to create heterogeneity within a discharge area, which was pointed out by SSM in reviews of SR-Site and the RD&D Programme 2013.

In order to investigate how small-scale variation in topography and soil layer thickness influence 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). The studies include the main discharge area for SFR, and a soil profile in the Krycklan area in Västerbotten. Since tools have recently been developed that link water flows to sorption and precipitation (for example via PHREEQC) in spatially distributed models, there are good prospects to study how for example flow paths, layer thickness and redox reaction zones affect accumulation, and how dissolved organic matter influences the transport of various substances (see further Section 12.4). On a larger scale this is linked to the MARFA model (see further Section 11.4).

The PhD study in the Krycklan catchment, previously funded by SKB, has continued in the form of a postdoctoral appointment. Several studies show that the landscape large-scale mosaic of forest and wetland has a large impact on mass-transport of various substances, and that gradients in thin layers may have a strong influence on both the forms of occurrence and accumulation of various substances (Lidman et al. 2013, 2014, 2016, Köhler et al. 2014).

The formation, size and transport properties of wetlands are important variables in the modelling of dose. A review of SGU's 80-year-old peat archive from northern Uppland has been used to illustrate the potential for accumulation of peat and the effects of cultivation of peat land in the Forsmark area (Sohlenius et al. 2013a.). Aside from the occurrence of peat, the thickness and properties of other deposits also affect both groundwater movement and sorption of various substances. SKB has therefore developed an improved soil depth model for the Forsmark area in conjunction with the safety assessment SR-PSU (Sohlenius et al. 2013b).

Figure 12-2. The flat landscape in the Forsmark area with a mosaic of wetlands and lakes cut off from the sea during the past decades. Water areas will, with the ongoing land uplift, within a relatively near future be adequate as agricultural land and they are important areas for a potential exposure of radionuclides from a final repository.

In order to understand the impact of the landscape on groundwater flow and solute transport in a cold-climate domain, an area on Greenland in front of the ice sheet margin has been studied within the GRASP project (see Chapter 13). Among other things, the hydrological interaction between the active layer, lake and talik was studied, as well as the importance of deposition of windblown material for the formation of lake sediments and their chemical properties (Johansson et al. 2015a, b, Rydberg 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. A working group within MODARIA has worked to achieve a consensus when it comes to the description of future climate evolution (Becker et al. 2014), and they are expected to report their results within the coming year. The project is important in order to establish on an international level a common view of climate evolution and how it might affect surface ecosystems.

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 Strömgren 2010, 2013, Lindborg et al. 2013), but the tools and methods need to be updated for future safety assessments, both in order to meet future needs and to secure competence in the area. SKB is aware of the problem of lack of transparency in dose calculations when they are driven by complex support models (Walke et al. 2015), which SSM has pointed out. However, experience from SR-PSU shows that it is difficult to foresee and simplify processes that are dependent on landscape evolution before the dynamics of release are known. SKB therefore intends to primarily work with simplified models in connection with the interpretation of results (Kautsky et al. 2016).

Programme

In conjunction with the review of SR-Site, the spatial heterogeneity of the landscape and alternative delimitations of biosphere objects have been discussed, and SKB has submitted supplementary information to SSM in response to review comments. There is still a need to shed light on the importance of different assumptions concerning the spatial variation of a release for the calculated dose to the most exposed group. SKB therefore plans to continue the work with the COMSOL tool, which will provide an opportunity to resolve or illustrate several issues concerning the spatial resolution. The work will be supplemented with sensitivity analyses in Ecolego, where current descriptions of wetland development and the extent of soil layers in Forsmark are utilised. SKB also plans to reconsider how the temporal and spatial delimitation of the most contaminated areas affects the exposure of non-human organisms.

Knowledge from the GRASP project and from Krycklan will be applied to the Forsmark area to illustrate 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 in BIOPROTA and within a potentially continued IAEA programme, MODARIA II.

12.3 Transport and accumulation processes

Transport processes refer here mainly to abiotic transport taking place with water, particles and, to some extent, gas. Accumulation processes here refers to processes in loose deposits (for example sorption), but not uptake of radionuclides in organisms (see Section 12.1).

Transport modelling is a central component in both dose modelling and ecosystem modelling. Hydrology affects both the areas which may be contaminated by a release of radionuclides and the amounts (and concentrations) of nuclides that reach these areas, whereas sorption will determine how much can be accumulated in the soil layers or be associated to particles. The uncertainties in sorption (K_d) are often very large, affecting the dose calculations.

Current situation

In the reporting of results of SR-PSU, there is a description of how the current understanding of transport and accumulation processes has been applied in the most recent safety assessment (Saetre et al. 2013). Modelling of solute transport in the rock and in the surface systems is mainly carried out within the different safety assessments, but development of suitable models and their application in safety assessments are dependent on results and conclusions from a number of research and development projects.

All hydrological and meteorological data gathered within the framework of the GRASP project have been published in the form of a data article (Johansson et al. 2015a). It presents time series for meteorology, ground temperature, surface water and groundwater levels, as well as surface water flows in the investigated area on Greenland, together with the hydraulic parameters for the saturated and unsaturated groundwater flow. The hydrological data set constitutes the basis for a modelling study (Johansson et al. 2015b) where conceptual and numerical modelling have been carried out in order to describe hydrological flows and processes in the investigated area, and to quantify water balance in the catchment area.

In SSM's review of SR-Site, it is noted that the degree of the discretisation in the soil layer model used in transport calculations may affect the distribution of decay progeny in relation to their parent nuclides. SKB has investigated the phenomenon and has partly reproduced the reviewers' results by increasing the degree of detail in the transport model that was used in SR-Site. SKB's calculations confirm that decay of radium-226 for example gives a redistribution of radionuclides in the soil profile, and that under certain circumstances it might affect the concentrations in the environment. Additional studies are required to determine in which situations this phenomenon occurs.

A report by Nyberg et al. (2011) indicates that there could be areas in Forsmark where gas can flow up from the bottom sediments ("pockmarks"). In the referral of the RD&D Programme 2013, SGU thought that such areas should be investigated with more detail. In 2013, SKB made an additional mapping of bottom substrates in the area between Norra Piren and the Biotest basin. The field investigation included mapping with the aid of a side scan sonar and verification with video recording and SCUBA diving. During the investigation, no craters were observed that could be interpreted as anything other than natural depressions in the bottom, indicating that no pockmarks are present in the area (Wallin et al. 2016).

Besides the COMSOL tool providing an opportunity to refine temporal and spatial resolution (see Section 12.3), it is also a complement to the modelling carried out with MikeShe. COMSOL makes it possible to study in detail mechanisms driven by physical, chemical and biological processes, similar to that which can be done with the CoupModel (Gärdenäs et al. 2009), at the same time as it is possible to use the landscape as a driving factor for water flows. An ongoing project is studying the possibility to calculate K_d values that vary over time and space for some well-known elements with COMSOL. The space- and time-dissolved K_d values can then be used in the dose modelling to calculate how accumulation of different substances varies in time and space, which may provide opportunities for reducing the uncertainties in the dose calculations.

On a landscape level, it has proved possible to quantitatively predict the effect that wetlands have on surface water chemistry and thus also on the flow of different elements through the landscape (Lidman et al. 2014). This corroborates the relevance of SKB's conceptual model of landscape development, which is based on identifying the succession of different landscape types in time and space. Wetlands also influence the water chemistry and mass flow in the landscape by reducing weathering of mineral soils, and they contribute to the sequestration of substances in peat (Lidman et al. 2011) (see also Section 12.4). The size of streams can also affect the water chemistry, as the contribution of deep groundwater increases downstream in a larger catchment area. The phenomenon, which has been detected the signature of uranium isotopes ($^{234}U/^{238}U$) and oxygen isotopes ($\delta^{18}O$), can be expected to affect both the outflow of radionuclides to the surface and the hydrochemical environment (Lidman et al. 2016).

To calculate the effect of nitrogen dispersion and nitrogen leaching during construction and operation of the repositories in Forsmark, the oceanographic models have been updated and implemented with the knowledge produced in the 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 calculations of doses from the operation of nuclear facilities, see Sundell-Bergman et al. 2015), will provide valuable insights regarding the importance of water transport for dispersion and cycling of substances in the sea area outside Forsmark.

Programme

When it comes to hydrology and transport, several projects that also touch upon the surface systems are described in Chapter 11. During the next few years, newfound knowledge from the GRASP project on hydrology and transport in a colder climate will be transferred to the Forsmark area.

An important question pointed out by SSM in the review of SR-Site is how to transform a detailed hydrological model to parameters that describe hydrological flows, which can then be used in a simple box model in the safety assessment to describe element transport. Alternatives to the current methodology will be investigated, with the objective of developing a methodology that describes the transport of water and solutes during longer time periods and in an accurate, simple and comprehensible manner.

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. These processes will be studied in detail in the COMSOL tool. 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. A well-described profile in the Krycklan area will be used to evaluate how well a spatially distributed model can recreate known leaching and accumulation patterns. In the long term the intention is to construct a more generic model that can be applied to quantify the importance of these processes in the different catchment areas in Forsmark.

A revision of the implementation of the uranium decay chain in t surface ecosystems model will be made during the coming period, as requested by SSM. The revision is planned to include transport of and exposure to radon, and aims at shedding light on the consequences of different transport rates for the calculated dose.

12.4 Radiological, biological and chemical properties of potentially important elements

In addition to the uptake of elements in various organisms and food chains (see Section 12.1) and transport and accumulation of elements in the regolith (see Section 12.3), elements also have different chemical and radiological properties. These properties, in collaboration with the chemical environment, affect both the mobility and radiotoxicity of the different elements. 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 surface ecosystems in Forsmark (Tröjbom et al. 2013). The values are based primarily on measurements from the site and secondarily on available international data. During the period 2009–2015, SKB participated in the two IAEA programmes EmrasII and MODARIA. Within these programmes, aggregated CR and K_d values were compiled, and they will be published in a forthcoming report. There is, however, a need to supplement the compilation with additional data for certain elements that may be important for the different final repositories and where there are knowledge gaps at present (for example beryllium, europium, radium, molybdenum, nickel, chlorine, gadolinium, of which several will be present in the future waste from the spallation facility, ESS, currently being built in Lund). The data may in part be taken from existing site investigation data and simulations of CR or K_d, but in the case of certain substances supplementary measurements are also needed.

The chemical environment of an element is crucial for its chemical speciation, which determines the element's properties in this environment. The chemical environment can be characterised with pH, redox, and concentrations of main components (i.e. dominant anions and cations) and the amount of dissolved organic matter. Through a number of different processes, such as land uplift, weathering and ecosystem succession, the chemical environment at a site can change over time. In Forsmark for example calcite and marine ions are expected to leach out with time, at the same time as the amount of organic matter is enriched. Hence, the conditions for immobilisation and mobility of various elements in the landscape can change substantially compared with present-day conditions (Tröjbom and Grolander 2010, Sohlenius et al. 2013b). In several studies, SKB investigated how different factors in the landscape affect radionuclide mobility.

In Forsmark, it has been observed that a considerable portion of the surface water and the near-surface groundwater in the site investigation areas are close to the thermodynamic equilibrium with the mineral baryte ($BaSO_4$), which thereby controls the mobility of barium (Jaremalm et al. 2013). Co-precipitation of radium with barium probably limits the mobility for radium in these areas, and it can explain why these two elements sometimes exhibit considerably higher K_d values than comparable elements such as calcium and strontium in the site-specific K_d measurements (Sheppard et al. 2011). The leaching of calcite in the area will likely displace these equilibria in the future, meaning that other K_d values may be needed to represent the immobilisation of radium.

On a landscape level, it has proved possible, based on fundamental chemical properties of the elements, to quantitatively predict the effect of peat-forming wetlands on surface water chemistry and the flow of various elements through the landscape (Lidman et al. 2014). The occurrence of wetlands and other organic soils is also 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 they often occur in the aqueous phase bound to either colloidal iron or DOC. Discharge of deep groundwater also has a significant effect on the aquatic environment (see Section 12.3). The deep groundwater affects both pH and DOC concentrations, and thereby also the mobility for many radionuclides (Lidman et al. 2016).

Programme

Chloride, which has previously been regarded as the dominant form of chlorine in nature, has proven to be reactive, and the quantity of organic chlorine is much higher in many environments than the quantity of mobile chloride (see Section 12.1). SKB is therefore planning 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, studies are planned to estimate the cycling of organic chlorine in sediments and water, and to describe the total flows of volatile organic chlorine compounds from different environments.

In addition to estimating uptake and transport processes for methane and carbon dioxide (see Section 13.1), measurements will be carried out to determine contents and the amount of stable isotopes of these elements. The measurements will be supplemented with new methods that can measure much more effectively than before flows of methane and carbon dioxide (e.g. with the aid of sensor networks and a hyperspectral camera for methane).

SKB's chemistry database from the site investigations and from the monitoring programme will be assessed to interpret patterns with regard to mobility and uptake of nickel, molybdenum and gadolinium, which are potentially important nuclides in SFR and SFL. A joint project with Posiva is planned to make a common evaluation of chemistry data, where the two companies' extensive databases with rich background information will be co-interpreted.

A programme is 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 the future. Among other things, the effects of the gradual leaching of calcite in the soil profile will be investigated aside from other factors that form the chemical environment. This is in line with SSM's recommendations in the review of the RD&D Programme 2013.

Additional valuable data from the Krycklan catchment provides an opportunity to study leaching and enrichment in a zone close to a stream. 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.

13 Climate and climate-related processes

Section 5.10 describes a number of general questions within SKB's climate work that require further research, either because they have emerged in the comments from the regulatory authorities in the review of the submitted applications, or because SKB itself has judged it necessary in order to reduce uncertainties in ongoing and upcoming safety assessments. These remaining issues are: i) age and long-term stability of the bedrock surface in Forsmark, including glacial erosion and denudation; ii) climate change: SKB's reference glaciation based on the last glacial cycle; iii) climate change: transitions between different climate domains; iv) climate change: the earliest possible onset of cold climate, permafrost and ice sheet growth in Scandinavia; v) sea-level variations in the near-future and in the long-term, isostasy and shoreline displacement; vi) validation of the permafrost model; and vii) supplement and application of the newly developed description of ice sheet hydrology from the Greenland Analogue Project (GAP).

Most of these issues concern all three repositories: the Spent Fuel Repository, SFR and SFL. This chapter describes more thoroughly the current situation and planned activities within the above areas.

13.1 Age and long-term stability of the rock surface in Forsmark

Current situation

The current state of knowledge suggests that the total amount of denudation, including glacial erosion, in the Forsmark area has not generally been extensive to date, and that it will continue to be limited also in the future (Olvmo 2010). A primary reason for this conclusion is the very flat 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 level. There are therefore questions remaining concerning the extent of the glacial erosion and denudation when studying the area in Forsmark in more detail, and also concerning what could happen in the future with nearby areas where greater glacial erosion has been documented (Olvmo 2010). A question that is linked to this is whether glacial tectonics have previously occurred in the Forsmark area, where large parts of the uppermost part of the bedrock are removed by ice sheets, and, if it has occurred in the past, whether this could occur again during future glaciations.

In the comments on the RD&D Programme 2013, 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 carry out the planned studies to clarify the uncertainty in the estimated values of glacial erosion and evaluation of the permafrost model reliability". SSM's review of SR-Site also expressed a desire for further investigation and justification of SKB's standpoint that future denudation and glacial erosion at Forsmark will be limited.

Programme

In view of the above points, a study was initiated in 2015 concerning the age and stability of the bedrock surface in the Forsmark area. Major questions included in the study are: i) have glacial tectonics occurred at Forsmark? ii) how extensive has denudation (erosion+weathering) and the glacial erosion been at Forsmark during previous glacial periods, and during the entire period since the sub-Cambrian peneplain formed, when studying the area in detail? Based on this, how great denudation/erosion do we expect over the coming 100 000 and 1 million years in and around the repositories? iii) Ccan the documented area with more glacial erosion southeast of Forsmark grow towards the site for the Spent Fuel Repository, and if so, at what rate?

The age of selected bedrock surfaces in the Forsmark area will be determined by sampling and subsequent cosmogenic dating, i.e. with the aid of cosmogenic nuclides (beryllium-10, aluminium-26 and neon-21) in the upper metres of bedrock. The ages provide information on how long the bedrock surfaces have been exposed to cosmogenic radiation, i.e. how long they have been exposed and comprised part of the ground surface. The results also provide a quantification of the magnitude of the glacial erosion that occurred in different parts of the landscape. The methodology is complex and requires accurate preparations regarding the choice of sampling sites, as well as accurate assumptions in the assessment of the measured isotopic concentrations. The results will, together with other results from the study, be used for the analysis of the long-term stability of the bedrock surface in Forsmark.

Mapping of glacial land forms for identification of glacial erosion patterns in the Forsmark area will be carried out, both by remote sensing and through field studies. A geomorphologic analysis looking at how much the ground surface in different parts of the area has been reduced by glacial erosion from the original reference surface, which here consists of the sub-Cambrian peneplain, will be carried out (Figure 13-1).

Lateral growth rate for the area south-east of the Forsmark region with more glacial erosion will also be studied. Southeast of Forsmark is an area with stronger observed glacial erosion. The study aims to assess the potential lateral rate of glacial erosion in these glacially-induced erosion valleys, i.e. how quickly this area potentially could grow towards the repository area in the future. The study includes detailed mapping and analysis of glacial bedrock forms and sediments, as well as remote sensing and field checks.

The results from the studies are expected to provide a detailed picture of the extent of denudation, including glacial erosion, which has occurred in the Forsmark area to date, and thereby also provide material for an extended analysis of the future denudation evolution at the site.

13.2 Climate variations

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 natural climate archives and climate models to construct the climate scenarios.

13.2.1 The climate in SKB's reference glaciation

SKB uses a reference glaciation in order to exemplify the development of climate, ice sheet, permafrost, sea level, and denudation during a typical Late Quaternary glacial cycle. In SKB's safety assessments, the effect of the reference glaciation on hydrogeology, geochemistry, geosphere, landscape evolution, repository barriers and, finally, the function and safety of the repository after closure is analysed. The reference glaciation also constitutes a scientific starting point for building and 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 glaciation consists of a future scenario where conditions during the last glacial cycle (the Weichselian glaciation and Holocene interglacial) are repeated for the coming 100 000 years.

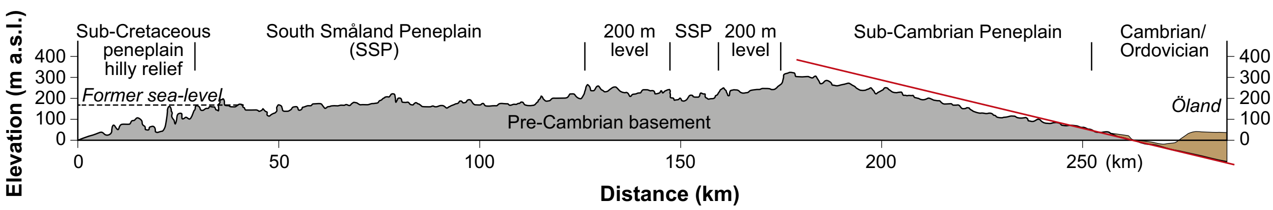


Figure 13-1. The question of how much the sub-Cambrian peneplain has been lowered by denudation is important not only for estimating the amount of glacial erosion but also for interpreting the inventory of cosmogenic isotopes in the bedrock in terms of erosion rate. The figure shows a topographic profile from Småland (modified from Lidmar-Bergström et al. 2013), with the sub-Cambrian peneplain in red. The topographic and geological setting is similar for Forsmark.

Current situation

During the preceding RD&D period, the work with reconstructions of the climate during different periods of the Weichselian, the Holocene and the former interglacial Eemian, continued through analyses of the lake sediments from Sokli in northern Finland. The studies contribute important information on how quickly the climate and vegetation in Scandinavia can switch during different phases of a glacial cycle, and with quantification of how large climate variations can be in the form of temperature and in some cases precipitation. During the preceding RD&D period, the studies at Sokli resulted in seven scientific articles, and a PhD dissertation at Stockholm University (Shala 2014):

- **The last glacial-interglacial cycle:** A description and to some extent re-evaluation of the last glacial-interglacial cycle based on long-term climate proxy archives from central and northern Europe (Helmens 2014).
- **The preceding Eemian interglacial:** Observations of fast dynamic climate variations during the preceding interglacial, the Eemian, with severe cooling of the climate in the middle of the warm period optimum (Helmens et al. 2015).
- **The current Holocene interglacial:** The reconstruction of the Holocene climate conditions in northern Scandinavia from lacustrine sediments (Shala 2014, Shala et al. 2014a, b), with the same methodology previously used in the study of the Early Weichselian and Mid-Weichselian periods. The results have also shown an unexpected early beginning of the Holocene warm optimum in northern Europe (Väliranta et al. 2015).
- **Method development:** Method development studies regarding quantitative reconstructions of palaeoclimate from palaeo fossils in lacustrine sediments have also been carried out (Salonen et al. 2013, Engels et al. 2014).

The results from the studies above have been used directly in the descriptions and justifications of climate scenarios for the safety assessment work, most recently in the safety assessment for an extended SFR (SKB 2014b, 2015b, Kautsky et al. 2016). The studies also provide a better understanding of SKB's reference evolution and methodology for handling long-term climate variations.

Programme

Since the results to date show that variability in climate and environment during the last glacial cycle were considerably larger than previously thought, quantification of this variability will continue to be carried out during this RD&D period. These palaeoclimatic studies will also be concluded and summarised during the period.

The ongoing PhD project studying the climate during the Eemian interglacial based on data from Sokli will be completed according to plan. The planned time for the completion of the PhD project is 2018.

SKB will also continue ongoing studies of palaeoclimate based on measured temperatures in boreholes in Forsmark, Laxemar and at Lake Vättern.

13.2.2 Climate change: transitions between climate domains

Current situation

In the RD&D programme 2013, a planned study aimed at detailed investigations of transitions between climate domains (in contrast to the more static climate model studies SKB previously conducted for the glacial, periglacial and temperate climate domains) was announced. This study has now been going on for three years and has included a PhD project at Stockholm University that was concluded during the spring of 2016. The study uses both geological archives and climate modelling to provide more complete information and examples of what the climate may be like at the transitions between the different climate domains.

The study focuses on the last deglaciation, a period which spans a range from full glacial conditions to warmer temperate climate conditions during the Holocene, via cold periods such as the Younger Dryas. The results from the parts of the study that mainly use proxy data from geological archives indicate very fast and strong changes in climate at the start of the cold Younger Dryas, with distinctive temporal and spatially consistent patterns in climate change (Muschitiello et al. 2015, Muschitiello and Wohlfarth 2015). The study also presents the first robust evidence on the link between the impact from continental meltwater and fast changes in the atmosphere-ocean circulation in the North Atlantic (Muschitiello et al. 2015b, 2016).

The results from parts of the study that mainly use proxy data from geological archives have been supplemented with simulations of the climate during the warm Bølling period (about 13 000 years ago), and the cold Younger Dryas period (around 12 000 years ago).

Programme

Climate simulation of the transition from the warm Bølling period to the colder Younger Dryas will be analysed and compared with proxy data from geological archives during this RD&D period. A description of the whole study of transitions between different climate domains is also planned, where the results from climate modelling and the studies of geological climate archives are combined and put in relation to other new literature within the field.

13.2.3 Onset of future cold climate, permafrost and ice sheet growth

Current situation

The timing of the next glaciation and associated periods of cold climate is of great interest in the assessment of post-closure safety for SKB's final repositories (SFR, SFL and the Spent Fuel Repository). A compilation of the state of knowledge on this issue was therefore made during the preceding RD&D period (Näslund and Brandefelt 2014). The results from seven published studies where the climate is simulated over the coming 100 000–200 000 years confirm that there is great uncertainty in the timing of the next glaciation. The onset of colder climate and glaciation depends in these models partly on the known future variation in insolation and partly on assumptions regarding the concentration of greenhouse gases that will prevail in the atmosphere long into the future. In the model studies that have been compiled, glaciation takes place during future periods with low insolation at high latitudes, around 0, 50 000 and 100 000 years in the future (Figure 13-2). Using today's high concentration of carbon dioxide in the atmosphere (~400 ppm), it is only the latter two periods (50 000 and 100 000 years in the future) that results in glaciation, since it is not until then when the concentration of carbon dioxide in the atmosphere might have fallen enough for glaciation to be initiated. These conclusions are also supported by a new modelling study (Ganopolski et al. 2015), which proposes that a new glaciation would not start for 50 000 years even without human climate impact, and that moderate carbon emissions will postpone the next glaciation so that it will begin after 100 000 years or later.

A few studies of the timing and size of future glaciations over even longer time perspectives, up to one million years in the future, have also been made (Huybrechts 2010). This timescale is relevant in the analysis of post-closure safety for the Spent Fuel Repository and SFL. The results show that repeated glaciations are to be expected during this long time period, but the uncertainties in the results are very large.

After a request for supplementary information by SSM, SKB has conducted a study during the current RD&D period regarding the maximum ice sheet thickness over Forsmark. The question is of interest for the integrity of the copper canisters under high isostatic load during periods of ice sheet coverage. The study contains climate and ice sheet simulations of one of the largest known glaciations from geological information, the Saalian glaciation. The study focuses on exploring uncertainties in simulated climate and ice sheet thicknesses for the peak of this glaciation, 140 000 years ago, by statistical analysis of large amounts of univariate and multivariate sensitivity simulations. The maximum ice thickness for Forsmark during this glaciation has in these studies been estimated to around 3 500–4 000 metres (Colleoni et al. 2014, Quiquet et al. 2016). Parts of the study are also published in Colleoni et al. (2016) and Wekerle et al. (2016). SKB's response to SSM states, using also results from Huybrechts (2010) that a glaciation with a maximum ice thickness similar to the one reconstructed over Forsmark for the Saale period by Colleoni et al. (2014) is unlikely during the coming 100 000 years, and possibly also during the coming 1 000 000 years. The uncertainty in the estimate of the timing of a future period with maximum ice thickness is, however, still large. The results regarding maximum ice thickness are used in SKB's work with the design premises regarding isostatic load for the canisters in the Spent Fuel Repository.

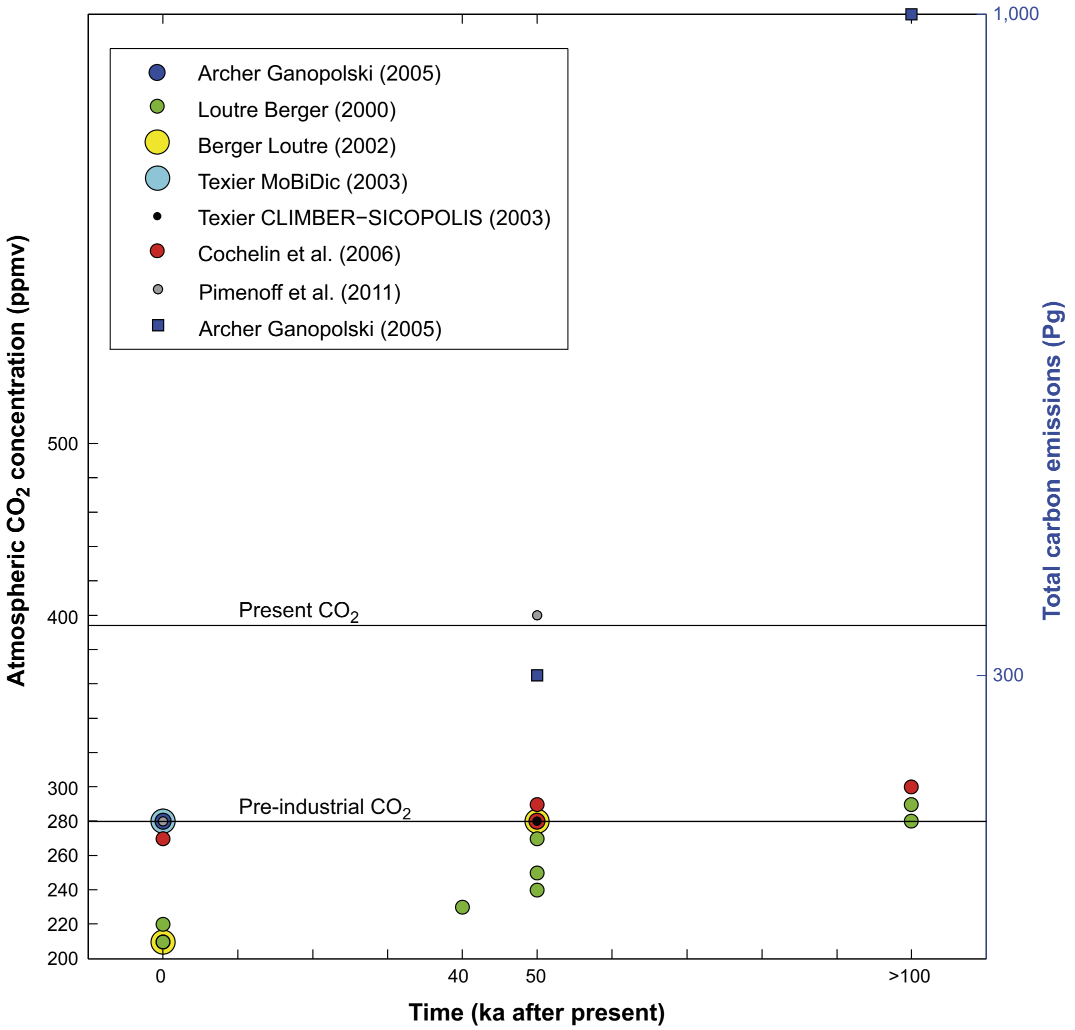


Figure 13-2. *Approximate time for the next glaciation as a function of the carbon dioxide concentration in the atmosphere (circles) or total carbon emissions (squares) from the seven studies that were included in the compilation by Näslund and Brandefelt (2014).*

As a comment to the RD&D Programme 2013, SSM writes: "Considering the limited repository depth for SFR, issues linked to freezing of cement/concrete should be considered to be of importance for post-closure safety. SSM believes that the results reported so far are ambiguous, which motivates further efforts."

Within the SR-PSU safety assessment, a detailed study was conducted of the earliest expected timing of permafrost growth in the Forsmark area (Brandefelt et al. 2013). A main question was when permafrost, and associated freezing of the relatively shallow SFR repository, could occur at the earliest in Forsmark in the future. This was investigated by modelling of the future climate, modelling of permafrost, and palaeoclimatological studies of how the climate has varied during selected portions of the Weichselian glaciation. The results show that, with pessimistic assumptions of greenhouse gas concentration in the atmosphere, it cannot be ruled out that permafrost could form in Forsmark during the coming two periods with low insolation to the Earth, which occurs around 17 000 and 54 000 years in the future (Brandefelt et al. 2013, SKB 2014b). For the first of these periods, freezing of the concrete structures in SFR can be ruled out, also with very pessimistic assumptions, while it cannot be ruled out for the latter period (SKB 2015b). The risk of permafrost forming in Forsmark earlier than 17 000 years in the future was judged to be very small. In SR-PSU, the above-mentioned results were used to construct a dedicated climate case, which describes the earliest possible future period with permafrost in Forsmark (the *Early periglacial climate case*).

Programme

SKB plan, together with Posiva, to carry out an improved model analysis, compared with Huybrechts (2010), of the extent and times for glaciations during the coming one million years. Huybrechts (2010) notes that the study contains very large uncertainties. The climate and ice sheet models that exist are in rapid development, and have been so in recent years. The uncertainties in Huybrechts (2010) could be reduced by simulations with new and improved climate and ice sheet models, as well as with modified methodology. This is expected to provide more reliable and useful results in the assessment of repository evolution on the timescale of a million years (Spent Fuel Repository and SFL).

SKB will also in the safety assessment work apply the knowledge produced by the wider research community concerning future climate evolution, with a special focus on the earliest possible onset of cold climate, permafrost and ice sheet growth in Scandinavia.

13.3 Sea-level variations, isostasy and shoreline displacement

Current situation

The current state of knowledge concerning sea-level variations of relevance for Forsmark is described in the SR-PSU Climate report (SKB 2014b, Sections 3.3, 4.1.3 and 4.3.3). This report also describes the state of knowledge for isostasy and shoreline displacement (SKB 2014b, Section 2.2). The report compiles data on possible relative sea-level variations, including their uncertainties, for the period up until year 2100 and on time-scales of 10 000 and 100 000 years.

The relative sea level at Forsmark, and its variation over time, consists of the net result of isostatic and eustatic changes. The present isostatic uplift in Forsmark is 8.4 mm/yr, and for e.g. the period up until year 2100 it will compensate for a large part of the eustatic sea level rise.

Sea-level variations are in SKB's work divided into: i) slow, persistent, and (in a relatively near future) non-reversible processes; and ii) fast, short-duration and reversible processes (SKB 2014b). The former consists of the global sea level rise (and its components from melting ice sheet and glaciers, and thermal expansion of ocean water), and the latter of the temporary sea level rise that occurs in for example during heavy storms. Figure 13-3 summarises all these components for Forsmark for the years 2000 and 2100. The figure summarises data from different sources, among other things estimates on global sea level rise made by IPCC (2013) and results published later. The figure includes a cumulative "worst case" scenario where maximum effects on global/regional and local scale are summed up. In this scenario, the total, pessimistically calculated, maximum relative sea level rise in Forsmark year 2100 is +3.3 metres during heavy storms (Figure 13-3). After the storm event, the relative sea level would return to the average level of year 2100.

In relation to the construction and operation of the repositories in Forsmark, it is very important to note that the slow permanent sea level rise, which has the largest uncertainty, is observable over time. This means that during the coming decades it will be possible to observe how it develops and to take further actions with respect to design and construction, if needed.

The research area concerning future sea levels is in an intensive phase within the scientific community, and there are large uncertainties about the size of future sea-level variations.

Regarding isostasy and ice sheet load, three different ice sheet reconstructions, including the one used in SKB's reference glaciation, have been analysed and compared with the aid of glacial isostatic adjustment (GIA) simulations (Schmidt et al. 2014). The reconstructions are compared directly in terms of ice volume, extent and ice thickness and in terms of the resulting isostatic response that the three ice load histories results in over Fennoscandia. The modelled isostatic response is then compared with measured values and distribution of today's isostatic uplift from GPS observations. Despite relatively large differences between some aspects in the reconstructions, the results show that all three models fit today's uplift rates about as well, but with different solid earth parameters in the GIA model. All three models, however, overestimate the uplift rate in south-western Fennoscandia. There are also larger differences between simulated and measured uplift rates in Finland and northmost Sweden and Norway. These results can be used to discuss the detailed realism in the studied ice sheet reconstructions, as well as to improve future reconstructions.

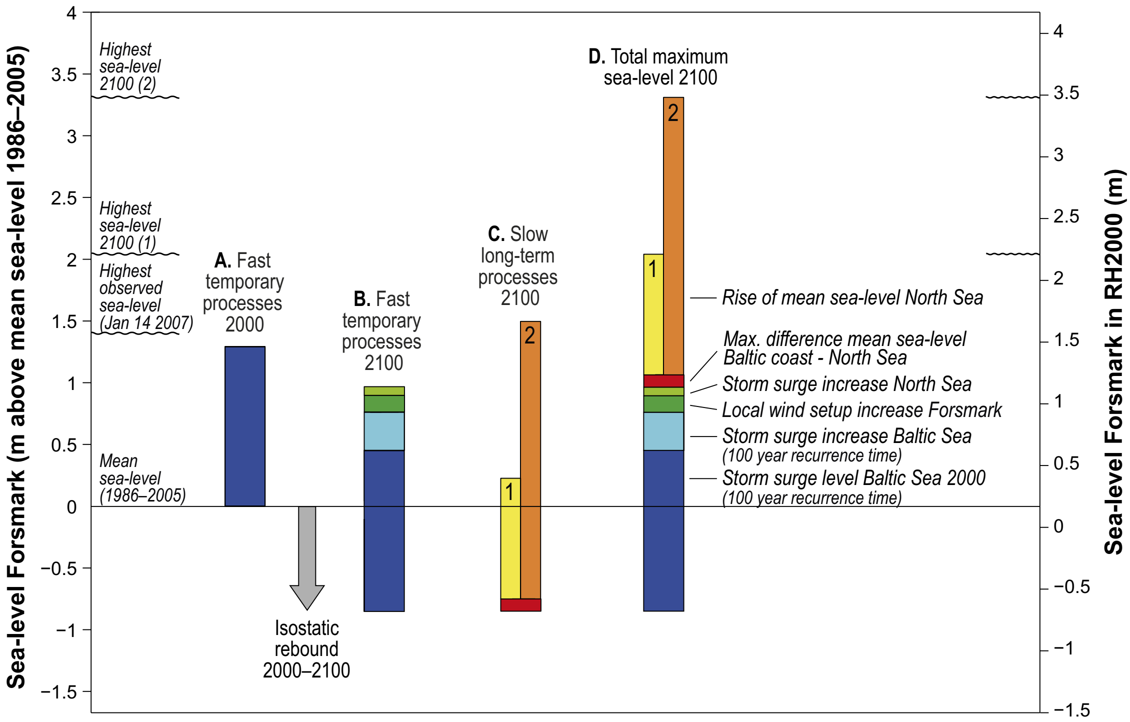


Figure 13-3. *Estimated maximum relative sea level in Forsmark in 2100. A: Today's maximum sea level caused by fast temporary processes during storms. B: Maximum sea level in 2100 caused by fast temporary processes during storms, and its sub-components, partly compensated for by isostatic uplift. C: Sea-level rise in 2100 caused by slow processes (increase of average sea level), 1: from process-based projections (IPCC 2013) and 2: other methods (Sriver et al. 2012). D: Total maximum sea level rise in 2100 with regard to fast temporary processes and slow non-temporary processes. Value C2, based on e.g. Sriver et al. (2012), constitutes a "worst case" scenario estimated from non-process-based models and estimates of ice sheet melting. Note that the isostatic rebound compensates for 0.84 metres of sea level rise from 2000 to 2100. Sea level is presented in metres above today's average level (for the period 1986–2005) and in the RH2000 height system.*

Programme

An update of the state of knowledge concerning sea level changes in the near future (up to year 2100, see Figure 13-3) will be carried out. It is needed for e.g. the excavation and operational period for the Spent Fuel Repository and the operating period for SFR. Potentially, efforts will be needed for updating simulations of regional/local oceanographic conditions in the Forsmark region, in cooperation with the discipline Surface ecosystems.

Furthermore, an update is planned of information and knowledge concerning sea level changes over the perspective of several thousands of years, including the highest level/longest period with submerged conditions that could be caused by pronounced global warming. This affects, among other things, landscape evolution and the timing when it is possible to drill wells in the future in the Forsmark area.

13.4 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 groundwater's flow pattern. Groundwater composition may also be affected by salt exclusion.

In the RD&D Programme 2013, plans were presented to conduct a validation of the permafrost model that has been used in the safety assessments for the Spent Fuel Repository (Hartikainen et al. 2010, SKB 2010a) and SFR (Brandefelt et al. 2013), and which are now used in the safety evaluation for SFL. This validation began during the preceding RD&D period.

Programme

The validation of the permafrost model that has been initiated will continue and is planned to be concluded during the RD&D period. The work includes the following parts: i) verification of current and former versions of the permafrost model by applying them on the study area for the GAP on western Greenland (see Section 13.5) where the permafrost depth and rock temperature are partly known; and ii) analysis of feedback between temperature in soil/rock and basal ice temperatures in the ice sheet, to see how this feedback affects the simulated depths of permafrost and frozen ground.

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. Within the framework of the planned safety evaluation for the SFL repository, research initiatives may be undertaken regarding the link between potential freezing of repository structures and the choice of repository concept and repository site.

13.5 Newly developed description of ice sheet hydrology from GAP

Current situation

The Greenland Analogue Project (GAP) study was conducted 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. The GAP has currently 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 (see Section 1.3 in Claesson Liljedahl et al. 2016), and a large number of contributions to symposiums. All these references are not presented here, but the reader is referred to the reports above.

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 when it comes to a number of different processes that are of great importance in assessments of long term repository safety, above all with respect to the boundary conditions for groundwater modelling, bentonite stability and landscape development. To this end, the GAP has contributed to an improved process understanding within four areas. One area concerns transient melting processes and where and how groundwater is formed underneath an ice sheet. A new conceptual model of ice sheet hydrology, with four distinct areas with different hydrological properties, has been developed (Figure 13-4). Field observations and modelling results suggest, among other things, that more than 75% of the studied part of the Greenland ice sheet has warm-based conditions, with the occurrence of free water, whereas 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, important as boundary conditions in future hydrogeological simulations of glaciation, see Figure 13-4 and, above all, Section 5.1 in Claesson Liljedahl et al. (2016).

Another area of improved process understanding concerns the temporal and spatial variations in water pressure at the base of the ice sheet. With the aid of pressure- and thermal data from measurements in 23 boreholes drilled through the ice sheet, the conceptual understanding of the basal groundwater-drainage system has been revised, see 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 larger part of the ice sheet and for most of the year.

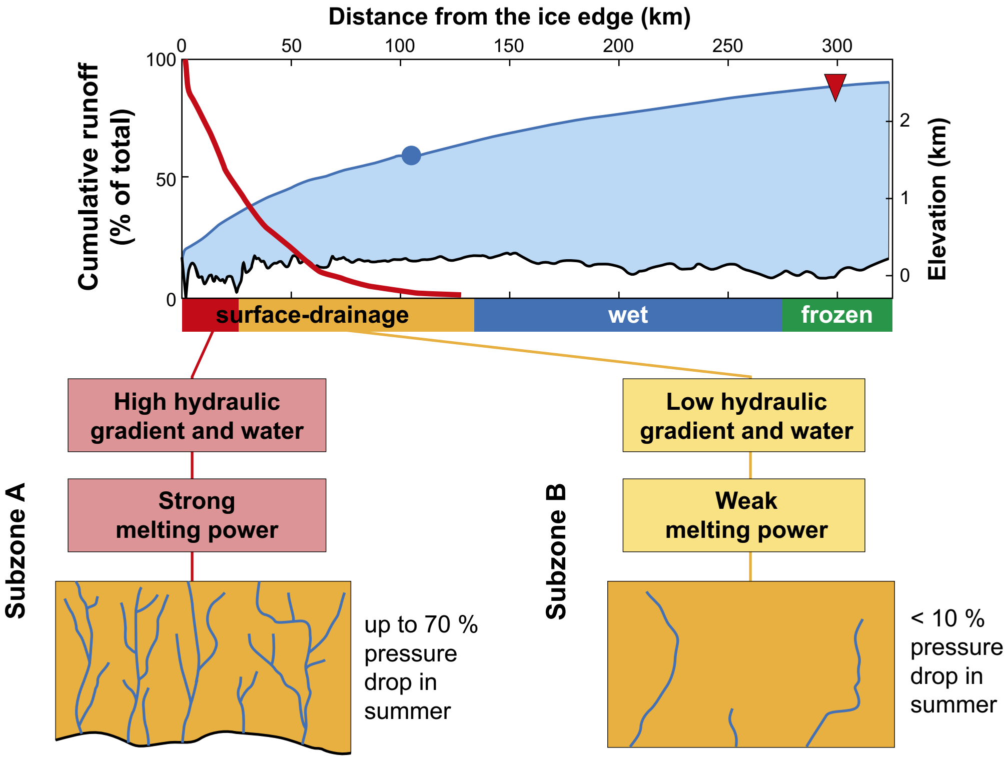


Figure 13-4. *Newly developed conceptual model of ice sheet hydrology, with four distinct zones with fundamentally different hydrological properties (from Claesson Liljedahl et al. 2016). Note that the zone dominated by surface drainage is divided into two parts: subzone A and subzone B. In subzone A (Transient conduit reach), the basal hydrology is dominated by a transient network of drainage channels developed during summer, up to 15–20 km from the ice sheet margin. The pressure in the channels has a very strong daily variation during the melting season (30–105 percent of the ice overburden pressure). The hydrology in subzone B (Distributed reach) still has a powerful water flow due to the input of surface meltwater, but the lower gradient of the upper ice sheet surface implies a general lack of discrete larger drainage channels. This is a main result in the GAP study; that growth of drainage channels during summer, from the ice sheet margin towards the internal parts, is highly limited by a reduced melting ability of the basal hydrological system (see Section 5.1 in Claesson Liljedahl et al. 2016).*

The GAP has also contributed to improved process understanding on the potential depths that glacial meltwater can penetrate to in the bedrock, and on the chemical composition of this water. Stable water isotopic signatures ($\delta^{2}H$ and $\delta^{18}O$) show that groundwater in 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, as well as 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 depend on 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 bedrock borehole at the ice sheet margin, with a limited groundwater flow beneath the permafrost. Below the permafrost, reducing conditions prevail.

Permafrost and taliks 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 through talik, located underneath the lake near one of the GAP bedrock boreholes. 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).

Programme

In its comments on the RD&D Programme 2013, SSM writes: "SSM takes a positive view of the Greenland Analogue Project (GAP) and the efforts to gain a better conceptual understanding of hydrology and hydrogeology under glacial conditions. SSM's judge that the study planned in order to transfer knowledge to Scandinavian conditions is important for the coming work with safety assessments for the planned final repositories." Furthermore, SSM writes: "SSM takes a positive view of the work being pursued within the GAP and the study of transferring the glacial-hydrological knowledge obtained from the GAP to Scandinavian conditions so that the reported information will be relevant to the safety assessments for the existing and planned final repositories."

The large field and modelling works within the GAP are completed, but some level of monitoring will continue during the period 2016–2018. The monitoring of the deep borehole that extends down to repository depth at the ice sheet margin will be maintained. In addition, the network of weather stations on the ice sheet will be maintained for at least the period 2016–2017.

SKB will also supplement the studies in GAP through a minor, in progress, glacial-hydrological study (ICE) within the same investigation area. ICE was initiated in 2014 and will be finished in 2016. The following is being studied within ICE:

1. **Short-duration very high subglacial pressures**: Research on valley glaciers has shown that very high but short-duration pressure pulses can 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 studied since the instrumentation in the ice boreholes has not had the capacity to record such short-lived pressure pulses.
2. **Pressure gradients on a scale corresponding to the ice thickness**: Results from the GAP show pressure gradients at the bottom of the ice sheet of the order 16 kPa/m. However, it is not known how far these high gradients can be maintained, and it is not known whether the gradients can exist over length scales of tens to hundreds of metres.
3. **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 is affected by this water pressure is unknown. This is important information as the degree of bedrock covered with water determines the size of the recharge area for groundwater.

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 under 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.

The extensive knowledge obtained from the GAP and GRASP projects concerning permafrost/taliks/surface hydrology in periglacial areas will also be applied in the assessment of post-closure safety in Forsmark, especially in the areas of hydrogeology and surface ecosystems.

Part III

Decommissioning of nuclear facilities

14 Premises for decommissioning of nuclear facilities

15 Planning for decommissioning at Uniper

16 Planning for decommissioning at Vattenfall

17 Planning for decommissioning of SKB’s nuclear facilities

18 Dependency and flexibility

19 Continued activities within decommissioning

14 Premises for decommissioning of nuclear facilities

The government decision regarding the RD&D Programme 2013 stated that the licensees of the nuclear power reactors, in consultation with SSM, should further develop the decommissioning planning of the facilities. This applies to both planned measures for dismantling and demolition and strategies for research and development. The decision also stated that the future RD&D programme needs to contain a clearer description of how information is distributed between the nuclear power companies and SKB, and how the coordination of these tasks takes place.

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 an 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 as well as management and disposal of the resulting radioactive waste is presented. The subsequent chapters provide a more detailed description of how the work will be pursued within each group and within each decommissioning project, with a focus on strategies and planned measures.

14.1 Concepts and requirements

Decommissioning of a nuclear power reactor 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 are under way 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. When SSM has approved an application for site release, 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 nuclear, which means that the remaining demolition and site remediation can take place without restrictions from the Nuclear Activities Act and the Radiation Protection Act.

Figure 14-1 shows schematically how the decommissioning of a nuclear power reactor is carried out in relation to the requirements on the facility during its life cycle. The upper part of the figure presents the activities that are planned to take place on 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 14-1. In conjunction with final shutdown, it is expected to be possible to revise the existing EIS as the transition period will largely be similar to previous operation. Prior to dismantling and demolition, on the other hand, a new EIS is required. The environmental impact statement, together with public consultations, constitutes the basis for a licence under the Environmental Code.

In accordance with the Nuclear Activities Act and the Radiation Protection Act and the current ordinances and regulations, among other things, 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 (SAR).

- Material according to the Euratom treaty, article 37.
- Step/sub-project notification.
- Decommissioning report.
- Inspection programmes for clearance and site release.

The decommissioning plan shall be reported to SSM, while the waste management plan and a safety analysis report (SAR) must be notified for formal approval according to SSMFS 2008:1. Furthermore, material will be prepared for information to the European Commission in accordance with the Euratom treaty, article 37. The formal report is submitted to the European Commission by SSM, while each decommissioning project compiles 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 protection measures, choice of technology and risk assessments and shall be formally safety reviewed.

When dismantling and demolition are completed, a decommissioning report shall be submitted to SSM, where experience from decommissioning and the final state of the site are described.

In addition to the above-mentioned requirements, there are also requirements related to decommissioning stipulated by other regulators. These requirements, which are not described in detail in this document, include for example laws and ordinances monitored by the Ministry of Environment and Energy, the Swedish Work Environment Authority, the Swedish Civil Contingencies Agency, the Swedish Transport Agency and the Building Committee in the municipality concerned.

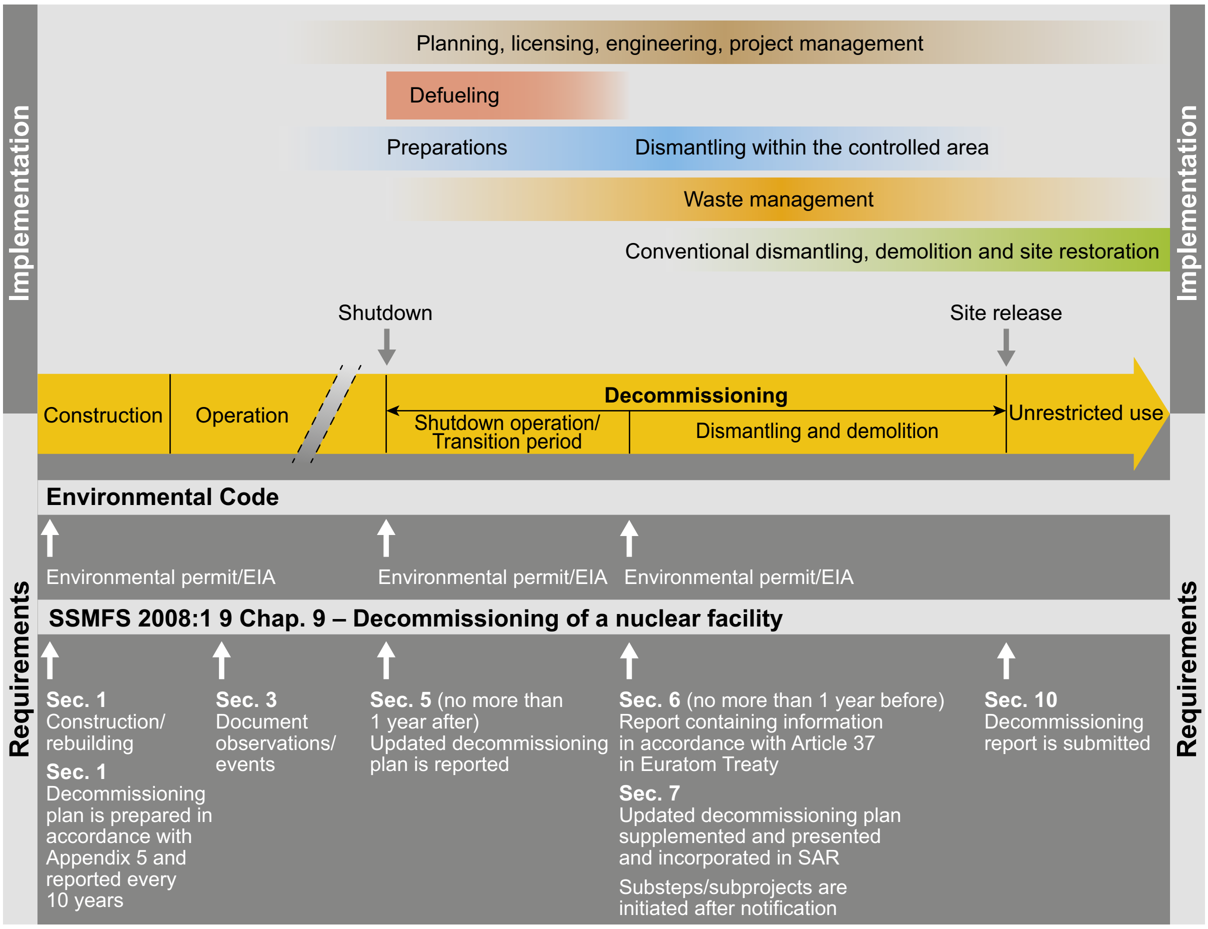


Figure 14-1. Overview of the different phases for execution of a power reactor's decommissioning as well as SSM's and the Environmental Code's requirements for decommissioning during the life of a nuclear facility.

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 detailed below.

14.2 Responsibility and division of roles

The licensee for a nuclear facility is responsible for decommissioning under the Nuclear Activities Act, the Radiation Protection Act, the Financing Act and SSM's Regulations. The licensee is responsible for the facilities according to this regulatory framework until clearance has occurred. The licensee is responsible for the radioactive waste until it has been cleared or until SSM has made a decision on closure of the final repository in question and the Government has granted exemption from responsibility under Section 10 of the Nuclear Activities Act (SFS 1984:3).

Barsebäck Kraft AB, Forsmarks Kraftgrupp AB, OKG Aktiebolag and Ringhals AB are licensees for the nuclear power reactors in Barsebäck, Forsmark, Oskarshamn and Ringhals. 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 Nuclear 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 is normally coordinated by SKB, whereas the practice for handling decommissioning issues varies slightly within the two industrial groups, Vattenfall and Uniper. This section describes working methods and distribution of work tasks.

14.2.1 Division of roles between the licensees and SKB

The licensee is responsible for decommissioning its nuclear power reactors. SKB has been contracted by the nuclear power companies to participate in the planning and execution of the future decommissioning. SKB's participation involves mainly compilation of the development needs identified by the licensees, coordination of general methods and procedures for transport and final disposal of radioactive waste, and compilation of the decommissioning-related costs reported by the licensees.

Under the Nuclear Activities Act, the nuclear power companies, working in consultation, must draw up a programme for the research and development activities and other measures needed to manage and dispose of the nuclear waste and the spent nuclear fuel 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 dismantling work

SKB is contracted to coordinate the general methods and procedures for transport and final disposal of radioactive waste that are needed for decommissioning. One task that SKB is responsible for, among other things, is developing waste type descriptions so that the waste meets the acceptance criteria that apply for each repository. The nuclear power companies produce waste type description specifications that form a basis for the waste type description. Furthermore, SKB has been tasked with developing new waste containers for the decommissioning waste which are common for all the nuclear power companies, see Chapter 6.

In order to achieve optimal national coordination, the nuclear power companies have jointly agreed on the tasks 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 2014b).

In the future, each nuclear power company will be responsible for the future decommissioning nuclear waste inventory, while SKB will be responsible for compiling the inventory and imposing 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 optimally conducted across the entire chain from dismantling and demolition to deposition and closure of the final repository.

Costs for decommissioning

Under the Nuclear Activities Act, the nuclear power companies are obliged to finance the measures needed to manage and dispose of the nuclear waste and the spent nuclear fuel and to decommission the facilities. On behalf of the nuclear power companies, SKB prepares a cost calculation every three years, in accordance with the Financing Act, see Section 1.5. Paid-in fees are managed by the state Nuclear Waste Fund.

Regarding the estimation of costs for decommissioning the nuclear power reactors, SKB previously had a greater responsibility when estimates were made together with the nuclear power companies. The tasks are now distributed so that the nuclear power companies estimate the costs for each nuclear power reactor and send these estimates to SKB. SKB compiles the estimates and produces an overall cost estimate and risk analysis, which then serve as a basis for the fees determined by the Government.

Transportation system

SKB is responsible for the transportation of spent nuclear fuel and radioactive 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. The ship and vehicles are used both for shipments of low- and intermediate-level waste and spent nuclear fuel. The different transport containers are specifically developed for each waste type. If new transport containers need to be developed, SKB is responsible for the execution of this.

Other radioactive waste can, if it is considered justified, be transported outside of SKB's system. This can for example involve large components such as intact reactor pressure vessels.

14.2.2 Distribution of work within the groups

For more efficient decommissioning activities within each group, coordination and cooperation also takes place on the group level. The licensee is ultimately responsible, but can, for example, have contracted parts of the planning to a dedicated group company or an organisational part within the group. Chapters 15–17 describe in greater detail the decommissioning work that is under way and planned within each group and within SKB.

14.3 National and international 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 certain international forums are also important.

On a national level, there is SKB's RD&D and Plan group where the nuclear power companies have an advisory role for SKB in the work with the RD&D programme and the cost calculation reports which is presented in the Plan report. The group deals with, among other things, issues of decommissioning and its planning.

To further support managers within SKB and the nuclear power companies concerning decommissioning and waste management, a forum for responsible managers has been established where SKB is the primary convener. This makes it possible to make decisions, within the mandate of the participants, in order to progress and prioritise the work conducted by SKB and the nuclear power companies.

SKB and the nuclear power companies are also participating in international forums in the field of decommissioning. Participation in the OECD-NEA's cooperative programme and the IAEA's programme are two examples. The work is being mainly pursued at the OECD-NEA in the Working Party on Decommissioning and Dismantling (WPDD) and at the IAEA in the International Decommissioning Network (IDN).

Barsebäck Kraft AB, AB SVAFO and SKB 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). TAG meets twice a year to exchange experience on different international decommissioning projects around the world. In the future, this forum will also be of interest for the decommissioning projects for Ringhals 1 and Ringhals 2 and Oskarshamn 1 and Oskarshamn 2.

15 Planning for decommissioning at Uniper

Uniper is a newly formed group where Barsebäck Kraft AB and OKG Aktiebolag are subsidiaries. From an owner perspective Uniper handles the operation and decommissioning of the power reactors in Barsebäck and Oskarshamn. The forms of collaboration within Uniper are currently being revised. The group's ambition is to make use of synergies between decommissioning projects at the two nuclear power plants. At present, and during this RD&D period, the majority of decommissioning-related work will be carried out within Barsebäck Kraft AB and OKG Aktiebolag. The ongoing and planned work is described in detail in the section for each facility below.

15.1 Barsebäck Kraft AB's planning for decommissioning

Barsebäck Kraft AB (BKAB) has since 2006 been conducting service operation of the reactors Barsebäck 1 (B1) and Barsebäck 2 (B2). Decommissioning of B1 and B2 is carried out as a joint project. Decommissioning is based on the strategy that the extension of SFR will be finished before dismantling and demolition begins. As a result of the postponement of SFR's extension, BKAB has changed strategy to interim-storage of the future decommissioning waste at the nuclear power plant or externally, awaiting transport to SFR. The final goal of decommissioning of the Barsebäck NPP is to meet the radiological criteria for clearance of buildings and release of the site so that the nuclear license can be terminated. How this will be achieved is described in general terms in the decommissioning plan for the Barsebäck NPP (Berglund and Lorentz 2016). The decommissioning plan is summarised below and Figure 15-2 gives an overview of the timetable for decommissioning of the reactors.

Overall planning

In 2015, a decision was taken to segment the reactor pressure vessel internals and store them in steel tanks in a new interim storage building at the plant, awaiting transport. The storage building has been constructed so that segmentation can begin in the autumn of 2016. Segmentation is expected to be completed in 2019 and will generate about 80 steel tanks.

The current strategy entails that the licensing process for dismantling and demolition will start in 2016 with the goal of starting dismantling and demolition in 2021.

BKAB has begun development of a radiological inspection programme for clearance of material and buildings and site release. The inspection programme will be linked to the 3D model of the facility that contains databases from the completed radiological survey.

The material developed by BKAB so far for decommissioning will be combined into a task for preliminary design. A prerequisite is that established technologies and methods shall be applied. During the preliminary design, an in-depth analysis, revision and evaluation are made of how decommissioning will be carried out. Examples of the content in the final report from preliminary design are steps/subprojects, schematic solutions, procurement procedures, economic calculations, risk and safety analyses, environmental aspects, waste management plans, training of contractors, dose budgets and timetables.

The results from preliminary design in turn lead to the Barsebäck plant owners being able to make a decision to start the design of dismantling and demolition of the plant. With the new strategy, preliminary design will begin in 2017 and design in 2018. This means that preparations such as establishment of production lines for handling of waste paths, auxiliary systems (ventilation, electricity etc.), interim storage facilities and other infrastructure can be completed so that dismantling and demolition can begin in 2021.

Dismantling and demolition will be carried out in specific steps such as extraction of the whole reactor pressure vessel, dismantling of the reactor building, the turbine building and the biological shield etc. The steps can be carried out in parallel or sequentially and the optimal execution will be established during design. After dismantling follow decontamination and clearance measurement of buildings. The preliminary timetable for dismantling, decontamination and clearance measurement of the buildings is from 2021 until 2025.

Figure 15-1. View of the Barsebäck NPP with the two BWR reactors B1 (closest in the picture) and B2.

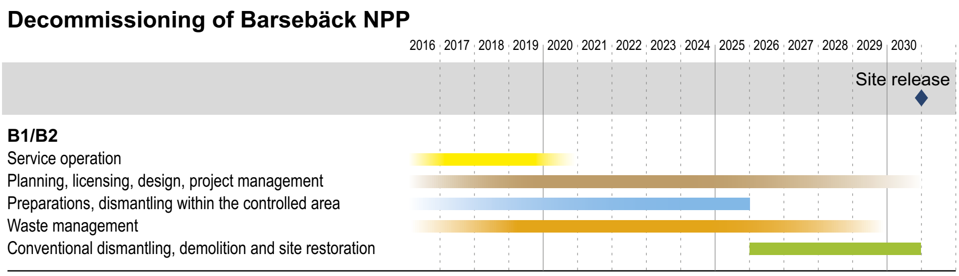


Figure 15-2. Schematic overview of Barsebäck Kraft AB's timetable for decommissioning.

After clearance of the buildings, conventional building demolition will commence to one metre below ground level. Demolition materials are used to fill culverts and other cavities in the facility construction. When the ground is prepared, clearance measurements will be performed to verify that the radiological criterion is fulfilled. The preliminary timetable for this is 2027 to 2028.

If SFR is not ready for operation during this period, the interim storage facility will remain a nuclear facility to be demolished and cleared when the waste packages have been transported away.

Waste management

The total waste quantity, both radioactive and non-radioactive, has been estimated by means of an inventory aimed at quantifying volumes and masses. Dismantling, demolition, waste management, transport and interim storage will be carried out as industrial processes with production lines. In order to avoid disruptions when handling the waste, activities such as clearance measurement and packaging in containers are designed based on the highest calculated flow. Buffer storage is introduced as needed to facilitate the logistics.

In order to avoid contamination of "clearable" waste, this should be dismantled first if the path ways cannot be separated from the radioactive waste.

Low radiation doses are prioritised, which means that radioactive waste is dismantled and transported away in large units, for example whole reactor pressure vessels. Clearance is pursued when benefits can be demonstrated without spending an unreasonable amount of resources. The concept may lead to slightly larger volumes of radioactive waste, but this is compensated by lower radiation doses to personnel, a shorter timetable and the generation of less secondary waste.

A near-surface repository for very low-level waste will not be established at the Barsebäck plant, taking into account that the site will be released from regulatory control in its entirety.

The radioactive decommissioning waste is interim-stored and thereafter transported to SFR, when the extended part of the final repository is in operation. The shipments from the interim storage facility to SFR are expected to be carried out over about two years and consist of whole reactor pressure vessels, ISO-containers, moulds and steel tanks.

Radioactive waste that can be cleared consists primarily of metal from the turbine auxiliary systems that can be sent for melting, and concrete structures that are reused on site as back filling when the ground is restored.

15.2 OKG Aktiebolag's planning for decommissioning

OKG Aktiebolag (OKG) plans for final shutdown of Oskarshamn 1 (O1) in conjunction with the maintenance outage in the summer of 2017. Oskarshamn 2 (O2), which has undergone an extensive modernisation programme during the last years, will not be restarted. A final shutdown is planned when the necessary environmental judgment has been obtained. Oskarshamn 3 (O3) will be operated until 2045. Thereafter, dismantling and demolition are planned to start during shutdown operation. For an overview of the timetable for decommissioning of the reactors, see Figure 15-4.

Decommissioning of OKG's facilities include the measures taken after final shutdown in order to remove radioactive material in land and buildings to such levels that the facilities and the site can be released for unrestricted use. The final objective is that the obligations from the Nuclear Activities Act and the Radiation Protection Act shall cease.

Decommissioning of OKG's facilities is to be carried out safely, efficiently and in an environmentally sustainable way. To achieve this, OKG has developed a strategic plan that includes all of OKG's facilities at the site. The plan presents the overall strategic goals and guidelines which serve as a basis for the future decommissioning. Physical connections and dependences to other licensees adjacent to the site are also described in general terms.

Overall planning

Based on the overall strategy, OKG has established decommissioning plans, one for O1 and O2 (Johannesson and Rannemalm 2016a), one for O3 (Johannesson and Rannemalm 2016b) and one for the shared facilities, Unit 0 (Johannesson and Rannemalm 2016c). The joint decommissioning plan for O1 and O2 is based on the physical connection between the facilities and that they will be decommissioned simultaneously or close to each other in time. In conjunction with the decommissioning of O1 and O2, certain shared facilities will be affected, for example the joint waste management building is at present planned to be decommissioned in conjunction with dismantling and demolition of O1 and O2. Parts of the radioactive waste arising thereafter, for example certain operational waste from O3, thus need to be managed in another manner. The decommissioning plans describe how OKG will reach the goals set up and how decommissioning will be conducted for each facility.

In 2015, the project Decommissioning Preparation Project (DPP) was initiated. The purpose was to plan and implement a new department responsible for decommissioning activities at OKG. The project would also prepare the material and carry out the preparations required for defueling of O1 and O2, followed by service operation and then dismantling and demolition. The new department with responsibility for decommissioning activities was established in August 2016 and handover from the DPP project continues during the autumn of 2016.

Defueling is estimated to take about 1.5 years for O1 and about 6 months for O2. O2's short defueling is due to the fact that the facility has not been in operation since 2013 and that transport of spent nuclear fuel to Clab began during the first half of 2016. For both O1 and O2, service operation is deemed necessary for a realistic timetable for the licensing process for dismantling and demolition. The length of service operation is judged to be 6 months for O1 and 1.5 years for O2. OKG will, however, strive to minimise or eliminate the time for service operation through good planning. Based on this a simultaneous dismantling and demolition of O1 and O2 is planned with an earliest start in the middle of 2019.

During the transition period, measures will be adopted to reduce the amount of radioactivity, and thereby reduce the radiological risk in each facility. Examples of such measures are removal of spent nuclear fuel, segmentation and removal of reactor internals and decontamination of primary reactor circuits. This means that some dismantling may be started immediately after final shutdown.

Dismantling and demolition of OKG's facilities will be divided into a number of steps that together with their constituent parts are described in the decommissioning plan for each facility. OKG will benefit from national and international experiences in its continued planning work and in the detailed design. OKG has also during the years in operation gained valuable knowledge of importance for decommissioning. In conjunction with the implementation of power uprate, OKG has developed and verified its methodology for managing large components. For example, reactor internals have been segmented, packaged and transported to OKG's rock cavern for waste (BFA) for interim storage. Moreover, large components such as the turbine from O3 have been dismantled, packaged, transported, melted and cleared from regulatory control. Logistics and planning of the whole waste stream have been verified.

Figure 15-3. *OKG nuclear power plant with the three BWR reactors O1, O2 and O3 from left to right in the picture.*

Waste management

The waste that is produced in connection with decommissioning is of the same type as the waste produced during operation, with the appreciable difference that the waste volumes will be considerably larger during decommissioning. This means that the capacity for waste management needs to be extended but that available techniques and methods can be used. The larger waste quantities during decommissioning, with regard to radioactive and conventional waste, impose stricter requirements on functioning waste logistics. OKG will with respect to waste management strive for redundancy in management and removal alternatives in order to reduce the risk of disturbances.

Since decommissioning of O1 and O2 is planned to be carried out before the extension of SFR is in operation, OKG in the continued planning work must investigate the need for possible interim storage of the arising radioactive waste. This applies, for example, to management of RPVs where OKG at present plans to dispose of them whole without internals.

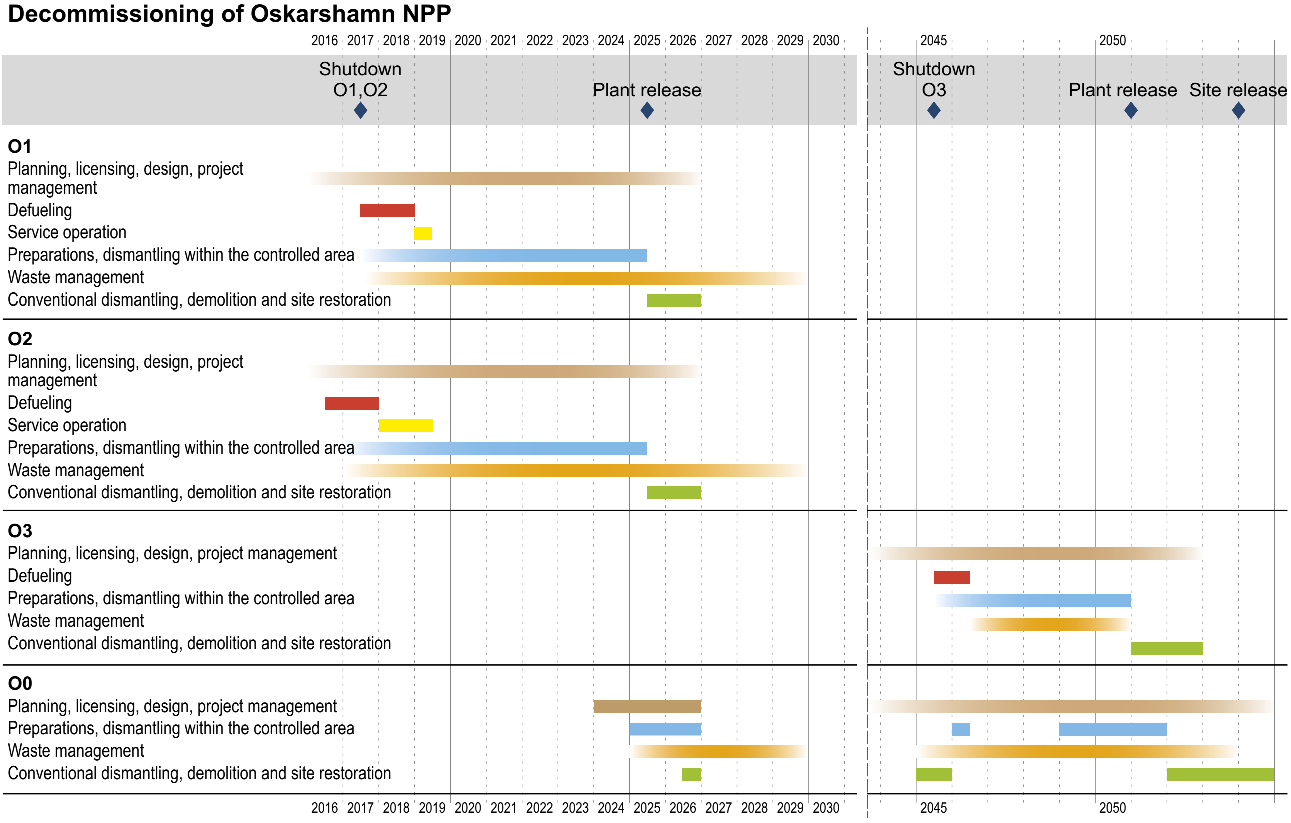


***Figure 15-4.** Schematic overview of OKG's timetable for decommissioning (O0 are shared facilities that are described separately).*

16 Planning for decommissioning at Vattenfall

At Vattenfall, decommissioning is planned to be organisationally separated from operation. This is done to ensure a business focus since the early decommissioning projects take place on facilities with continued production of electricity, and to concentrate the decommissioning competence in the group to a single organisational unit. The unit will ensure that safety is in focus and that planning is carried out efficiently with clear priorities within and between decommissioning projects.

The aim is that a new licensee, focused only on decommissioning, will take over the responsibility for the reactors after final shutdown as soon as the spent nuclear fuel has left the facilities.

In 2015, a unit was created within Vattenfall AB, the Business Unit Nuclear Decommissioning (BU-ND), to coordinate and pursue decommissioning within the group. After completed defueling, the operating licence is planned to be transferred to a company within BU-ND.

Analysis and planning for decommissioning are carried out by BU-ND on behalf of, and in close collaboration with, the licensee. The licensee ensures that there competence regarding knowledge of the facility and its history is available in the decommissioning project and is responsible for providing clear and realistic economic and technical prerequisites for the projects. The licensee also ensures that final shutdown takes place in an optimal way and that the fuel is transported away and physical separation of any facilities or systems that should remain at the site is performed. BU-ND works, in parallel with analysis and planning, for optimisation of decommissioning projects and provides strategic decommissioning-specific competence and strategic development support to the decommissioning projects.

16.1 Vattenfall's decommissioning planning

This section describes Vattenfall's decommissioning planning on a general level. The plans are valid for all nuclear decommissioning that is pursued within the group and can thereby be applied to both the Ågesta reactor and the power reactors in Forsmark and Ringhals. SKB's decommissioning planning is presented in a separate chapter, Chapter 17, since the decommissioning of SKB's facilities is planned to take place at a later time.

In order to clarify the strategic decisions, the section has been divided into two parts: project execution and waste management.

Project execution

In order to minimise the requirements and risks for execution of the decommissioning projects, prerequisites and extent must be well defined in advance. The timeline for dismantling and demolition shall be kept as short as possible with a coherent execution phase which will begin without unnecessary delay after final shutdown. This means that service operation shall be minimised and that dismantling and demolition lasts until the facility's final state is reached.

During defueling, in addition to fuel shipments, shutdown of systems and management and disposal of operational waste is planned, as well as preparatory activities for dismantling and demolition. These preparatory activities, for example decontamination of primary reactor circuits and segmentation of reactor internals from BWR reactors with long-lived activity content, are intended to make subsequent work as safe and efficient as possible. This is carried out for example to reduce the dose rate for workers and to improve the possibilities for flexibility on critical activities during dismantling. Plans are also made for the management of certain non-radioactive or low-level contaminated systems, such as turbine and generator systems, to optimise waste management prior to the start of dismantling and demolition on a large scale.

The decommissioning project organisation will have as a task planning and executing the project where contractors are used as the main work force during the implementation phase. With this, the project organisation will be kept small and effective.

The dismantling and demolition will be divided into a number of suitably delineated stages and sub-projects. 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 throughout decommissioning be optimised based on a joint ALARA and BAT perspective in order to minimise dose and maximise efficiency.

According to the current plan, RPVs from Vattenfall's facilities will be removed and deposited in the final repositories as whole. Removal of whole RPVs has never before been carried out in Sweden, but there is experience internationally, which will be studied in detail within the framework of the first decommissioning projects. Nationally, however, there is experience of handling of other large components, for example the steam generators. Management and final disposal of whole reactor pressure vessels is expected to minimise the dose to personnel, the time for handling and the impact on the environment.

To dismantle radioactive systems, mechanical methods that minimise the dispersion of contamination are primarily used, such as sawing and cutting. For non-radioactive systems, however, depending on the work operations, thermal methods may be used such as the use of cutting and plasma torches.

The final end-state of decommissioning is a released site that can be used for other industrial purposes. The final states for the different projects can, however, be an industrial plot that is possible to release from regulatory control, where some buildings and infrastructure are left, while other installations are dismantled. Activities for facilities and sites can take place in separate projects depending on what is optimal for each facility. The conventional building demolition occurs to about one metre below ground and remaining cavities are backfilled with demolition material. The uppermost ground layer is restored to the status that continued industrial activities on the site require.

Waste management

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 to not hinder progress in the project. Increased capacity can be achieved in several ways, for example by adaptation of 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 effective. This is achieved by classifying the waste before it is generated and sorting it directly at origin. Processing of radioactive waste shall be minimised and wherever possible carried out in connection with the origin of the waste. Creation of secondary waste should be justified, in other cases avoided. Large components should be disposed of as whole where this is justified from a cost and dose perspective. Time-consuming waste processing is only performed if benefits can be ensured.

The conventional waste that is produced during decommissioning also constitutes a challenge since, depending on its origin, it may need to be partially verified with respect to radioactivity content before it can be transported away for recycling or be used on the site as back filling material. 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 conventional demolition material that arises due to system dismantling, a clearance procedure in line with the existing clearance guideline is established.

Waste management, including storage and final disposal, should be optimised from a cohesive operational and decommissioning perspective within the Vattenfall group. This means that an increase of existing near-surface repositories for very low-level waste in Ringhals and Forsmark is planned in order to deposit both operational and decommissioning waste, when it is more advantageous than disposing of the waste in SFR or using other options. Increases of the capacity for managing the waste should be carried out where best suited from a Vattenfall group perspective.

The progress of the projects should be independent to a reasonable level of the capacity for waste treatment, waste transport and final disposal. For transport, the projects should primarily use SKB's transportation system, but they 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 decommissioning projects.

16.2 Ringhals AB's planning for decommissioning

Ringhals NPP is situated on Väröhalvön in the municipality of Varberg, in the county of Halland. The plant has four reactors, where 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, accesses roads etc., and occupies a total of 2.5 square kilometres, see Figure 16-1.

The reactors R1 and R2 were built in the 1970s and placed in a joint operations area. In the 1980s, R3 and R4 were built in an area originally separated from R1 and R2, but later linked together by a separate section. This simplifies the separation of the units and permits parallel operation and decommissioning at the site 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, buffer 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

Decommissioning of the reactors in Ringhals is described in two decommissioning plans, one for R1 and R2 (Bergman et al. 2016a) and one for R3 and R4 (Bergman et al. 2016b). These decommissioning plans are based on the overall strategy and goals presented in Section 16.1. Ringhals AB (RAB) is planning to operate the reactors R1 and R2 until the scheduled maintenance outage in 2020 and 2019. 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 premises for operation remain until 2041 and 2043. Defueling is currently judged to take about 16 months for R1 and about 22 months for R2, taking into consideration cooling requirements and the capacity for fuel transport.

Figure 16-1. *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 for example office buildings and a lunch restaurant. Just above R1 in the image is the plant's area for handling, conditioning and storage of radioactive waste.*

In December 2015, the decommissioning project for R1 and R2, project TYKO, was initiated, with the purpose of dismantling and demolishing the facilities. The focus in the near future is to assess the preconditions for decommissioning and evaluate in detail how the specific steps during decommissioning should be resolved in the best way. This includes assessment of for example licensing, demolition methodology, organisation and waste management. Section 19.2 describes the activities that will be analysed or conducted during the different project phases.

In conjunction with the decision for shutdown of R1 and R2, the project STURE was initiated at Ringhals. The purpose of the project is to prepare for decommissioning of R1 and R2 in a safe and efficient manner, at the same time as the operation of R3 and R4 can proceed according to plan. The project will plan and execute the physical separation of the reactor units that will continue in operation from the reactors that will be decommissioned, remove the spent fuel and drain systems as well as transport operational waste away from the plant. The project will also carry out decontamination of primary circuits after final shutdown and other preparatory activities needed for decommissioning, such as the preparation of infrastructure and increases of waste management capacity. Project STURE and project TYKO will cooperate prior to and during the execution of dismantling and demolition activities.

The general timetable for decommissioning of the Ringhals NPP is presented in Figure 16-2.

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 means that this alternative is the first choice for most waste streams.

The RPV from R1 can, due to its activity content, be deposited in SFR only after the most neutron-activated internals have been removed, segmented and packaged for interim storage prior to future deposition in SFL. Reactor internals such as steam dryer and steam separator will be disposed of together with the reactor pressure vessel in SFR.

The RPVs of R2, R3 and R4 need to be deposited of in SFL, due to their larger activation, which means that all internals are planned to be left in the RPV since the RPV and internals have the same destination for final disposal. Interim storage of the RPVs takes place at Ringhals NPP until the extension of SFR and SFL are in operation.

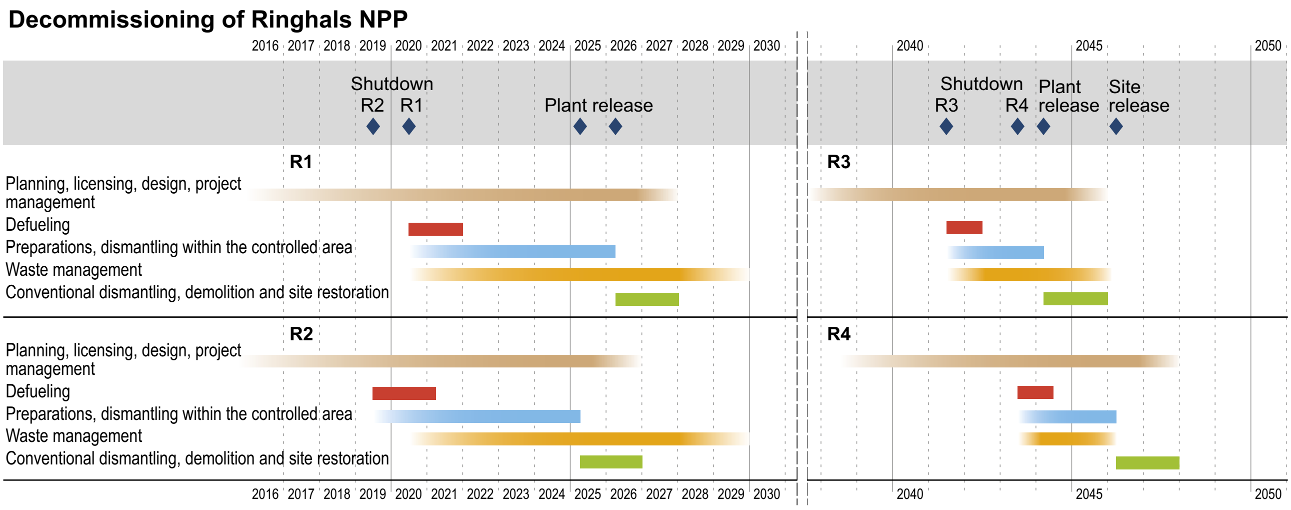


Figure 16-2. Schematic overview of Ringhals timetable for decommissioning.

16.3 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 the municipality of Östhammar in the county of Uppsala. There are three nuclear power reactors within the facility, Forsmark 1 (F1), Forsmark 2 (F2) and Forsmark 3 (F3), see Figure 16-3. The NPP also contains buildings for temporary accommodation, storage and workshop buildings and administrative buildings. A harbour has been constructed, which is used for example by ships transporting spent nuclear fuel and radioactive waste to SKB's facilities.

F1 and F2 are integrated facilities while F3 is situated separately to the 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.

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 begins at final shutdown; its length should be minimised as far as possible and is at the present predicted to be about 12 months.

Decommissioning of F1, F2 and F3 is described in the decommissioning plan (Runermark 2016) and is based on the overall strategy and the goals presented in Section 16.1. F1 and F2 are expected to be dismantled and demolished in a way that maximises synergy gains and minimises the need for facility separation or service operation. At the end of the decommissioning projects for F1 and F2, defueling is started for F3, which means that decommissioning is expected to take place in the area without interruption from the start of the first project until the last reactor is finally dismantled. Figure 16-4 presents the general timetable for decommissioning of the nuclear power plants in Forsmark.

Figure 16-3. *The Forsmark power station with the three BWR reactors F1, F2 and F3 from left to right in the picture.*

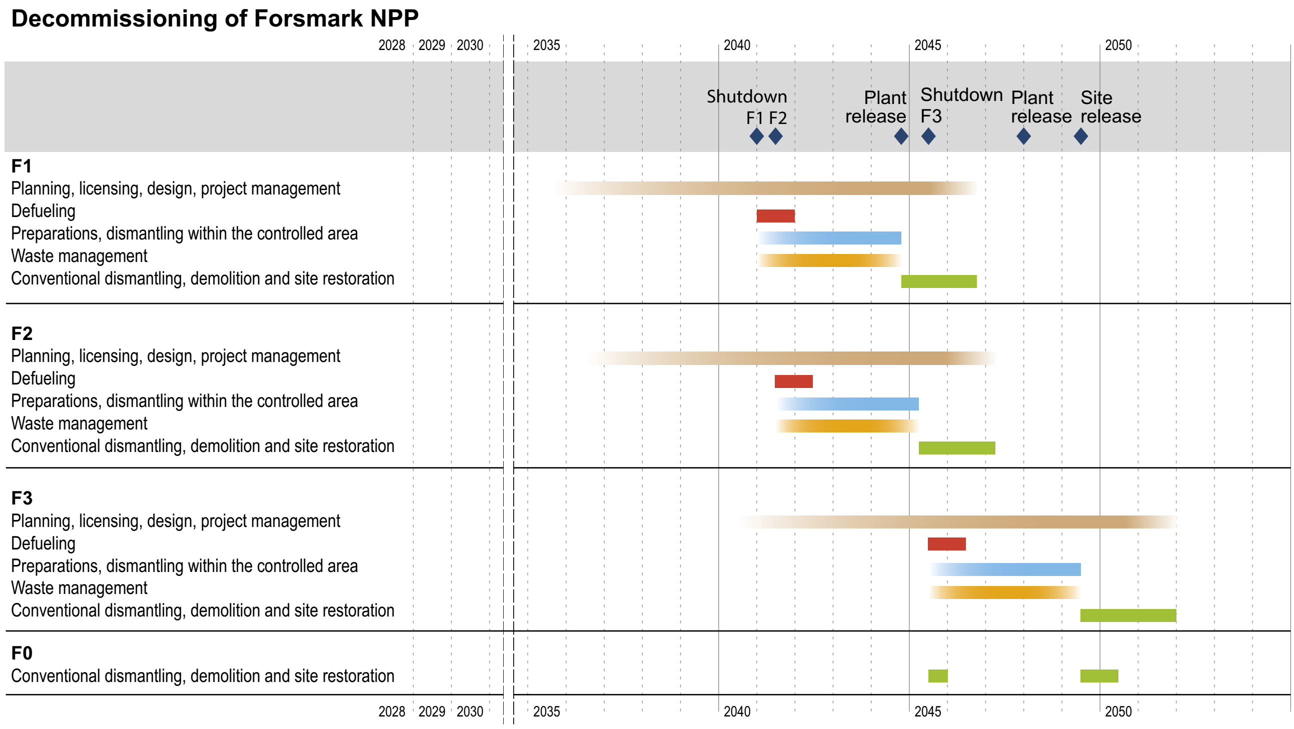


Figure 16-4. *Schematic overview of FKA's timetable for decommissioning (F0 are shared facilities that are described separately).*

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 to be deposited of in SFL. The short-lived low- and intermediate-level waste is sent directly to SFR after packaging.

16.4 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 facility parts, is situated in a rock cavern. 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, see Figure 16-5.

Ågesta NPP has been in service operation since 1974, which entails that the facility is shut down but certain operational and monitoring functions are maintained. AB SVAFO and the licensee Vattenfall each own 50 percent of the facility. AB SVAFO is conducting service operation on behalf of Vattenfall.

Because the Ågesta facility is situated in a rock cavern, space and opportunities are limited for handling and interim storage of waste on site. Ågesta's siting means that all transport must take place by road, with a possible exception for the RPV, which must initially be transported by road but thereafter possibly by ship. The siting near a densely populated area also means that the shipments that need to be carried out will affect nearby residents and facilities to some degree.

Overall planning

The environmental license for the ongoing service operation of the Ågesta reactor expires in 2020. The aim is to commence dismantling and demolition at the latest in conjunction with the expiration of the license. To comply with the general timetable, a decision was taken on project start for the Ågesta decommissioning project in November 2015. Decommissioning of the Ågesta reactor is further described in its decommissioning plan (Bohl Kullberg 2013).

In order to initiate dismantling and demolition at the latest when the current environmental license for service operation expires, the necessary licences and approval according to the Environmental Code, the Nuclear Activities Act and the Radiation Protection Act must be in place. The current assessment is that most critical element is the environmental licensing process. An objective is therefore to begin preparation of the application documents that are required for the application for an environmental licence, the environmental impact statement and consultation process as soon as the requisite material is available. Section 19.2 describes the activities that will be analysed or conducted during the different project phases. Figure 16-6 shows the timetable for planned decommissioning of the Ågesta reactor.

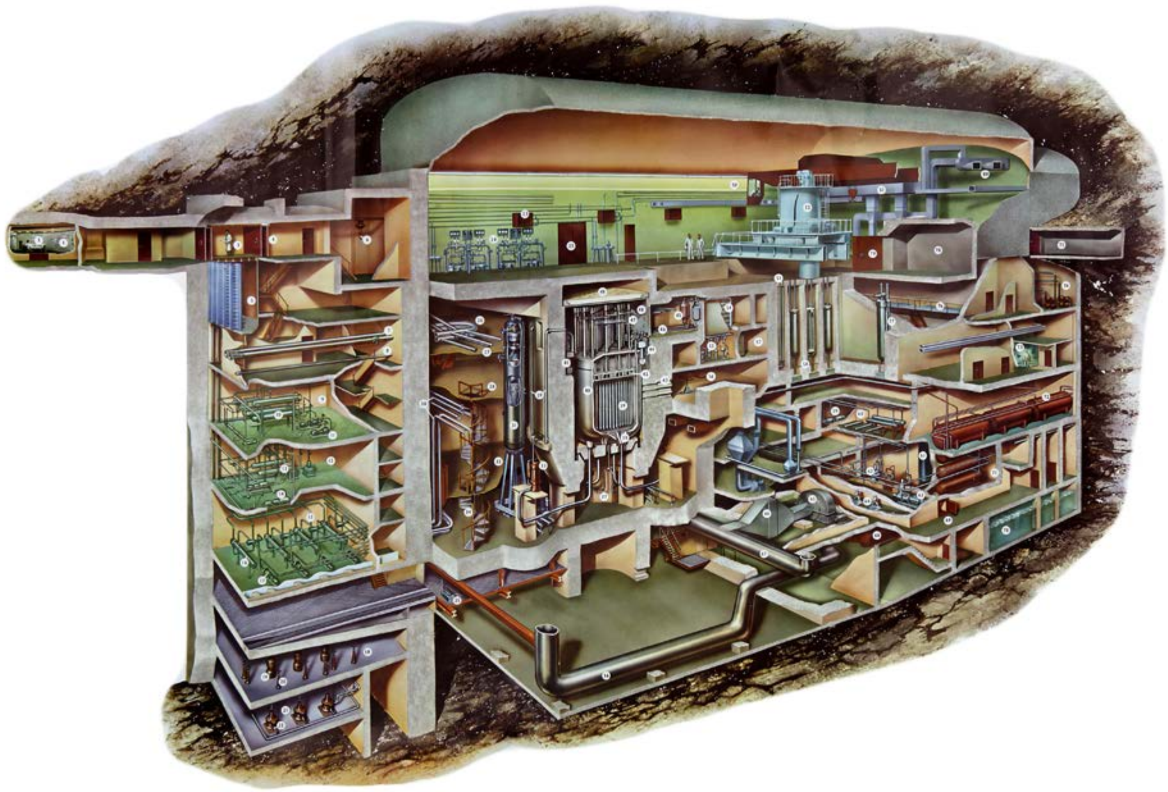


Figure 16-5. *Drawing of the Ågesta facility.*

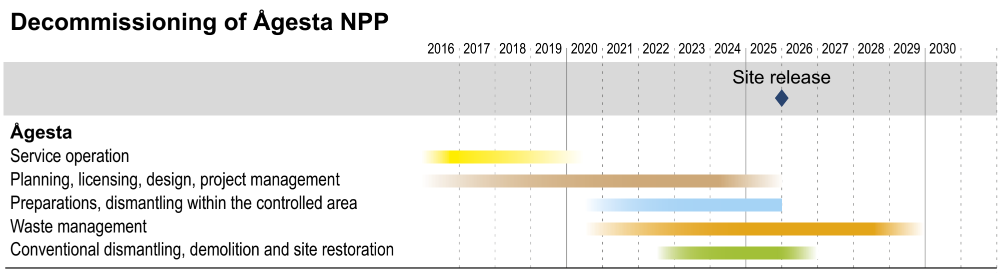


Figure 16-6. *Schematic overview of the timetable for planned decommissioning of the Ågesta reactor.*

Since the Ågesta reactor is an older facility that has been shut down for a long time and its owner and responsibility conditions have changed over time, conditions prevail in the sense that facility and operation documentation is spread geographically over several different archives within and outside of the Vattenfall Group. Furthermore, the documentation is old and inadequate. In order to provide the decommissioning project with as good prospects as possible, and thereby reduce the risk of future technical difficulties, the following measures have been taken:

- A project has been initiated to gather all relevant documentation from AB SVAFO and to digitalise selected parts of the documentation.
- A facility description will be produced, which provides a basis for e.g. definition of the extent and limitations of the project.
- Based on existing drawings, a 3D model is made of selected parts of the facility. Parts that are not modelled will be documented with photo technology in order to have a complete material for planning of dismantling and demolition.
- A radiological survey will be initiated, which will include both calculation of induced activity in and around the reactor pressure vessel and measurement of contamination in other parts of the facility.

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 steps have been identified up to final clearance or disposal through one of the available deposition alternatives. This means that all waste that is expected to arise during decommissioning of the Ågesta facility has a known final destination.

The existing decommissioning plan is based on in situ segmentation of the RPV at the facility. The current assessment is, however, that there are advantages to avoiding this segmentation, which means that alternative handling needs to be further studied, for example:

- Dismantling, transportation, interim storage and disposal of a whole reactor pressure vessel.
- Dismantling, transportation, interim storage and deposition of a reactor pressure vessel with the lid separated.
- Dismantling of a reactor pressure vessel on site and transport, interim storage and deposition in parts.

17 Planning for decommissioning of SKB's nuclear facilities

17.1 Central interim storage facility and encapsulation plant for spent nuclear fuel

SKB is the licensee for Clab and will continue to be so when the integration of the planned encapsulation facility is finished and the plant is renamed Clink. The decommissioning plan for Clink was updated in 2013, in conjunction with the compilation of supplementary material for the licence application for Clink (Calderon 2014a). Clink will be decommissioned when all spent nuclear fuel has been encapsulated and disposed of in the Spent Fuel Repository. The timetable depends on when the last nuclear power reactor is finally shut down. According to current planning, decommissioning of Clink could commence in around 2070 and be concluded within 5–7 years.

During the work of preparing the decommissioning plan for Clink, no reason has emerged why the decommissioning should be more complicated than for the other nuclear facilities whose decommissioning lies closer in time. It should be possible to carry out dismantling and demolition with a low dose to personnel, and the quantity of short- and long-lived radioactive waste that arises is expected to be limited. According to current plans, waste from dismantling and demolition will be sent to SFR for final disposal.

The goal of decommissioning is to remove all radioactive material and release the site. This means that all buildings, including equipment and land, will be cleared from regulatory control.

During 2013, SKB conducted a study for decommissioning of Clink in order to provide radioactive waste inventory data as a basis for the extension of SFR and a cost estimate for the Plan report (Edelborg et al. 2014).

In 2016, SKB plans to revise and update the decommissioning plan for Clab. Updating is done to harmonise with the regulations from SSM and to follow the established structure for a decommissioning plan (Calderon 2014b).

17.2 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 above-ground facility has been released and there are no radiological reasons to prevent the establishment of another industrial activity on the site. The parts of the facility that may be demolished in connection with decommissioning (the above-ground parts) are regarded as conventional, since they do not contain any radioactive material. A radiological survey of the facility will have to be done in order to rule out possible contamination of building parts that have been in contact with waste packages during operation, for example the terminal building. The goal of decommissioning is a released facility. How far demolition should be carried beyond that depends mainly on how the site will be used in the future.

The timetable for decommissioning of SFR is linked to when the last currently existing nuclear power plants and SKB's other nuclear facilities are dismantled and releases. Current plans call for 50–60 years of operation for the nuclear power plants and a few more years for Clink. Demolition of SFR could thereby commence in the mid-2070s.

17.3 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. Decommissioning will start in conjunction with repository closure, which takes place in the mid-2050s, see Section 3.3.4.

17.4 The Spent Fuel Repository

A decommissioning plan has been prepared for the Spent Fuel Repository and is included in the applications under the Nuclear Activities Act for final disposal of spent nuclear fuel and under the Environmental Code for the KBS-3 system (Hallberg and Tiberg 2010).

Decommissioning begins after operation is concluded, i.e. when all spent nuclear fuel has been deposed of and the deposition tunnels have been backfilled and sealed. Decommissioning entails closure of the remaining parts of the underground part and demolish of the surface part. Closure of the underground part is a part of the repository's barrier function and is of importance for long-term safety. SKB's work with closure is described in Section 10.8.1.

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.

18 Dependencies and flexibility

In order to be able to carry out decommissioning in accordance with existing plans, the establishment of prerequisites is required as well as some flexibility to external influences within the projects. During a decommissioning project, there is a risk that the project's continued activities are affected by external factors in the form of state and authority decisions, which are mainly outside of the defined project scope. The major external dependencies arise in conjunction with

- final shutdown,
- end of defueling,
- start-up of sub-projects/steps,
- release of the facility/site.

To complete these milestones, requirements are made on well-developed processes and close dialogues with authorities but also consequence analysis and preparation of a plan for the activities that can be carried out within the project to create flexibility in the planning.

Interfaces also arise in the nuclear power industry due to project dependencies with other industrial activities. These dependencies can to a large extent be handled by good planning, but if they are overlooked they can create great challenges for an ongoing project. The interfaces and dependencies arise both in the intersection between operation and decommissioning, and between decommissioning and SKB's waste management system for transport, interim storage and final disposal. They also arise due to the large extent of parallel decommissioning activities that are planned in the nuclear power industry. The major internal dependencies consists of

- separation of facilities,
- opportunity for preparations in-house at the facility,
- fuel transportation,
- interim storage capacity for fuel in Clab,
- management of reactor pressure vessels,
- management of long-lived waste,
- clearance and management of very low-level waste,
- critical resources nationally.

The present chapter describes dependencies, and how flexibility is created, in the interfaces between different decommissioning projects, between decommissioning projects and SKB, and between decommissioning projects and external actors. Section 18.1 first provides an overall picture of the decommissioning activities in the industry with a focus on describing interfaces, then presents the main dependencies together with a consequence description and the taken and planned measures to create flexibility.

18.1 Decommissioning activities during this RD&D period

The timetable for the nuclear power companies' and SKB's planned decommissioning of their facilities is presented on a general level in Figure 18-1. The RD&D period is dominated by activities at the reactors Barsebäck 1 and 2, Oskarshamn 1 and 2, Ringhals 1 and 2 and the Ågesta reactor.

The period includes measures associated with fuel handling at the reactors that are shut down prematurely. This imposes requirements on SKB's transportation system and Clab to provide capacity for transport and interim storage of the spent fuel from the final cores according to plan. Furthermore, measures will be adopted to achieve efficient and safe decommissioning. This includes preparations for, and in some cases execution of, segmentation of reactor internals, application for necessary licences prior to decommissioning and ensuring that all waste can be handled during the decommissioning projects.

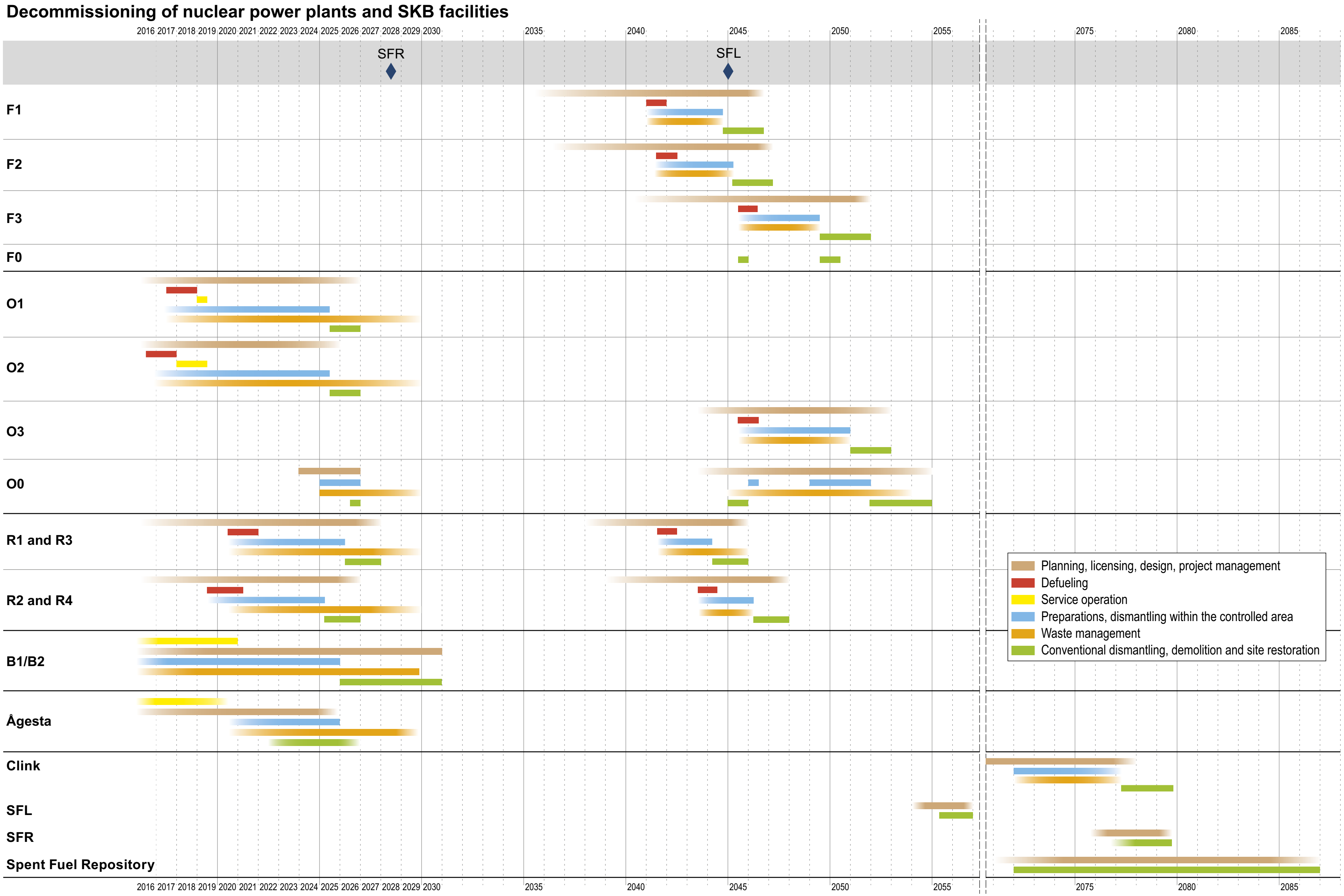


Figure 18-1. *Schematic overview of the nuclear power companies' and SKB's timetables for decommissioning (F0 and O0 are shared facilities on the sites reported separately).*

Radioactive waste management in turn requires that waste type descriptions for example be devised and approved for decommissioning waste, that handling and techniques for management of large components be developed, and that waste containers that are desirable from a decommissioning perspective be developed and licensed. Furthermore, path ways need to be available for the radioactive material that will not be disposed of by SKB, such as a licence for landfill disposal of very low-level decommissioning waste in near-surface repositories at the power plants.

Due to decommissioning of the first reactors being performed before there is reception capacity within the final repository system, i.e. before SFR is reopened after extension and before SFL has been built, the radioactive waste must be interim-stored prior to deposition (see the relation between milestones and decommissioning projects in Figure 18-1). The earlier start of the decommissioning projects, and the need for interim storage thus arising, mean that the decommissioning projects in the current planning become less sensitive to SKB's timetables than in the past. Since all nuclear activities are planned to cease at Barsebäck after the end of the decommissioning projects, there is still a dependency to remove the radioactive waste to permit release of the site. This means that a connection between decommissioning and commissioning of SFR's extension still remains and that the decommissioning projects are dependent on an external interim storage solution for long-lived waste.

Regarding the handling of long-lived waste, a large part of this waste will arise before the planned commissioning of SFL. Since the waste cannot be finally conditioned before the set of requirements for SFL is determined, which requires that the repository site is known, the long-lived waste needs to be placed in waste containers without final conditioning until final conditioning becomes possible. According to the current timetable final conditioning can commence at the earliest in conjunction with SKB obtaining a licence to build SFL, which is planned at the end of the 2030s, see Figure 3-5.

18.2 Flexibility concerning external dependencies

Principal external dependencies that arise during a decommissioning project are presented below together with a description of how they are handled in order to create flexibility and the measures that can be taken to minimise their consequences.

18.2.1 Final shutdown

The time for final shutdown is an important planning premise for a decommissioning project. All decommissioning projects will be dependent on a well-known time for final shutdown. The consequences of shutting down earlier than planned is higher for the reactors that will be decommissioned in the near future since there is no experience of large-scale decommissioning projects within the nuclear power industry and the authorities.

Decommissioning projects in the current plan include about five years of preparatory planning and design before final shutdown in order to ensure that, for example, licenses are in place, the project organisation has time to be established, the project execution has been analysed and an agreement with contractors has been established. This means that changes leading to an earlier final shutdown, scheduled to a point in time after the next five year period, do not affect decommissioning in terms of planning.

An earlier shutdown scheduled to a time within 5 years from the decision will, however, affect the execution of the decommissioning project, where the consequence depends on how large the restriction is for planning and design. In 2015, the time of final shutdown for a number of nuclear power reactors was moved forward to a point in time which to a varying degree is reached within 5 years.

In all cases, the earlier shutdown resulted in a shortened time for preparations. From the perspective of external dependencies, this has led to requirements on accelerated planning with a focus on producing material for the environmental licensing, as the environmental licensing process is critical since a reactor may not be finally shut down without a new environmental license issued. However, it should be mentioned that the implication of this is mainly that the activities that are planned for defueling cannot begin until the license is obtained. The reactor will be taken out of operation on the date planned even if this is done within the framework of the existing operating licence.

Since the environmental license prior to defueling is largely expected to be similar to the previous license during operation, an application for a changed license will be submitted with a revised environmental impact statement. The licensing process for this type of environmental license is judged to require about half a year for preparation of application documents and in an ideal case almost a year for scrutiny in the Land and Environment Court. There is always a risk for an appeal, which means that a decommissioning project must assume that the time from preparation of application documents to an approved environmental license may be as long as four years. During the period from final shutdown until a licence has been obtained, preparatory work for decommissioning will not be possible to carry out on the facility to the extent that is currently planned. The consequence of this is a prolongation of the project.

In a broader perspective, the next phase in the decommissioning project, defined by the end of defueling, see Section 18.2.2, will not automatically be affected in parity with a delay in the environmental licensing for final shutdown. This is because the next phase is mainly linked to the facility being out of spent nuclear fuel). The prospects for this are determined to a high degree by the cooling time for the final core and thereby depend mainly on the time for shutdown of the reactor, not when the environmental license for shut down operation is obtained.

Additional consequences of an earlier shutdown from the aspect of internal dependencies are discussed in Section 18.3.1.

18.2.2 End of defueling

The set of requirements that must be met after completed defueling is dependent on whether the planned activities consist of service operation or if dismantling and demolition begins.

If the end of defueling entails service operation, the safety analysis report only needs to describe service operation. If the end of defueling, however, means that dismantling and demolition begins, a more extensive revision of the safety analysis report is needed as well as a new licence under the Environmental Code.

If the time for the end of defueling is delayed, this implies that the requirement remains at an elevated level in relation to the actual status of the facility, provided that the fuel has been transported away. This means that work and staffing at the facility in practice still must be adapted to a nuclear power plant with fuel in place. The consequence is that work is impeded and the unjustified requirements give rise to increased costs.

When the extent of service operation is less than defueling, there is a possibility to limit the economic risk of a decommissioning project in the case where a licence for dismantling and demolition is delayed. This can be done by putting the facility in service operation awaiting the start of dismantling and demolition.

18.2.3 Start-up of sub-projects/steps

To begin dismantling activities in the facility, the decommissioning plan must be supplemented and reported to SSM and a report according to the Euratom treaty article 37 must be submitted. The latter shall be available to the Authority one year before the dismantling activities commence. Furthermore, it is required that the safety analysis report be revised and approved to cover the work that will be pursued and that a safety reviewed sub-project/step application has been sent to SSM. Finally, a reviewed and approved waste management plan for the radioactive waste generated during the sub-project is required. If any of these requirements are not fulfilled, the start-up of the sub-project/step is delayed.

A delay in the start-up of a sub-project/step entails a direct impact on the progress of the decommissioning project. Depending on how the content in the delayed sub-project/step affects other activities on the facility, the consequences can be larger or minor. A delay of an early sub-project probably entails a direct prolongation of the total time for decommissioning to a corresponding degree, which will increase project costs and lead to challenges in conjunction with contracting of suppliers, etc. For later sub-projects/steps, more time is expected to be available to develop input data and processes for review, furthermore more effective work methods and development of practice, both internally and externally, are expected to gradually simplify management as the projects progress.

The flexibility in conjunction with start-up of the first sub-project is limited. Sub-project extent can to some degree be adapted to time in order to increase the probability that preparatory work and review take place according to plan but otherwise, an efficient and predictable handling of the license applications is crucial for the success of the project. Flexibility in later stages can be created by carrying out a number of sub-projects in parallel and by letting sub-projects overlap in time in order to limit the risk of standstill in the project.

18.2.4 Release of facility

In order to finally obtain a decision for release, 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 verified regarding the presence of radioactive elements and that the measured concentrations are below stipulated clearance levels in accordance with SSMFS 2011:2. Furthermore, a decommissioning report over the execution of decommissioning, with descriptions of experience gained and the final state of the facility, must be compiled and submitted to SSM.

A delay in the time for releasing the facility means that conventional demolition activities at the facility, until further notice must be carried out under the same requirements as when it was a nuclear facility, which will lead to increased costs for the project. Greater flexibility regarding the time for release can be created by the facility being cleared progressively as contamination of materials, premises, buildings and land are found to be below clearance levels, whereby the measurement requirements prior to the final release decision can be limited. Furthermore, an early preparation of methods and expectations in consultation with SSM will reduce the risks of delays in conjunction with the final release of the site.

18.3 Flexibility concerning internal dependencies

Internal dependencies that can be uncertainties for a decommissioning project are presented below together with a description of how they are handled and the measures that can be adopted to minimise their consequences.

18.3.1 Separation of facilities prior to decommissioning

In order to simultaneously operate and decommission power reactors on a site, it is required that the activities can proceed in parallel with as few dependencies between them as possible. Since the oldest reactors at each site are to be decommissioned first and once also where commissioned first, there are often a number of dependencies between the younger reactors and the oldest. Some of these dependencies, for example, management of switchgears, operation of waste facilities and the support systems, must be separated in order for the younger reactors to continue their power operation.

Facility separation is performed from an operation perspective and is therefore not in a strict sense part of the decommissioning, but since safety and availability of reactors in operation has priority, a dependence is created between decommissioning and a completed facility separation. It is therefore important that dependencies between facilities are identified in an early phase and that adjustments are made in order to begin dismantling of the finally shutdown reactor units as soon as the licenses permit.

An earlier final shutdown entails shorter time to separate the facilities prior to decommissioning, which risks facility separation being critical for the project if efforts are not prioritised. The number of dependencies between a pair of reactor units is relatively low, making facility separation manageable within the time between the decision on final shutdown and the start of dismantling activities. However, there are more dependencies between reactors in a unit pair. In order to increase flexibility in decommissioning, it is advantageous that decommissioning of reactors in unit pairs takes place close to each other in time. The need for a more comprehensive separation is thereby minimised as well as the execution of dismantling activities.

18.3.2 Preparatory measures

During the transition period, preparatory measures are planned in the facility to increase safety and generally simplify dismantling and demolition. These measures consist of for example decontamination of primary circuits as well as segmentation and interim storage of reactor internals and core components that remains from the operation of the power reactor. Handling reactor internals has been previously conducted on all nuclear power plants during operation and is regarded as a proven technique.

The possibilities of conducting preparatory work during the early phases of decommissioning are limited by the environmental license. In the event that the activities are judged to be aimed towards dismantling and demolition, there may be additional requirements depending on what is being done in the facility, for example a notification according to the Euratom treaty, article 37.

By utilising the cooling time for the spent fuel and any additional time required to obtain licences for dismantling and demolition, for preparation of the facility for decommissioning, a greater flexibility is created for the first sub-projects/steps that are carried out during the dismantling and demolition phase. If the radioactive content in the different systems at the facility can be decreased, the general dose level will be lower, which both decreases the dose to personnel and reduces the background level for measurement of activity. The latter permits a more reliable characterisation work in the facility, which in turn improves precision in the planning of dismantling activities. In the event areas and buildings can be available by for example removing non-contaminated components, such as generators or turbine strings in PWR, the possibility for effective logistics increases during dismantling, which permits more rapid progress in the decommissioning project.

By managing internals in an early phase, there is an opportunity to empty reactor pools sooner, since segmentation takes place under water. Hence, the cleanup system for pools can be shut down, resulting in less radioactive waste in the form of ion exchange resins. This also permits the characterisation work related to reactor pools to start earlier, decreasing the continued risks for the decommissioning projects.

Another aspect to carrying out the preparatory measures in conjunction with shutdown is that there is good potential that needed expertise and experience from operation are available.

In the event preparatory measures cannot be carried out during the transition period, the decommissioning projects will be prolonged and cannot be carried out as effectively. In addition, there are risks for unnecessary dose to personnel awaiting the start of dismantling and demolition and that more radioactive waste will arise since the systems need to be in operation for a longer period.

18.3.3 Spent fuel transportation

The main prerequisite for being able to end defueling is that the spent nuclear fuel has been transported from the facility. Transport of fuel requires that the fuel has decayed to a decay heat that permits transport in fuel transport casks, and that transport casks and ship (m/s Sigrid) are available. The cooling time for the fuel depends on the burn-up, which in turn depends on, among other things, the number of fuel cycles. Some fuel assemblies can be transported earlier but the major parts of fuel shipments cannot begin until 9–12 months after final shutdown. With the existing fuel transport casks, it takes between three and five months to empty a core provided that m/s Sigrid is in shuttle service. This gives an expected shutdown time of between 12 and 24 months, where variations will occur depending on the reactor type, the size of the final core and the siting of the facility. At planned final shutdown, it is also possible to design the final core to minimise the necessary cooling time and the length of defueling.

The reactors in Barsebäck and Ågesta have already concluded defueling and are not dependent on fuel shipments. For O1 and O2, fuel transport is not expected to be a critical step in need of measures to create flexibility, since emptying of the final cores takes place in the near future with existing transport containers. For R1 and R2, however, the time limited certificates for the existing transport casks create a need for measures since fuel shipments from these facilities will occur after 2020 when the certificates have expired. There are two main possibilities; a renewed certificate in France for the old casks with validation in Sweden (carried out by SSM), or adapting the facilities for handling of the new transport casks, see Section 3.4.5.

The preferred solution is to apply for a licence and exemption to empty the final cores via the existing transport casks in order to avoid plant modifications and dependencies between decommissioning and delivery of new transport casks. SKB plans to carry out shipments according to this solution and has initiated a modification of existing transport casks to permit an extension of the certificate.

In the case where a re-certification or validation of existing casks is not possible, the possibility remains to modify the facilities. The modification is calculated to be possible to carry out during the cooling time for the spent fuel. Such a modification can thereby be decided upon up to the time of final shutdown without affecting the time for decommissioning. In the event fuel, regardless of reason, should leave the facility much later than planned, a prolonged period of shutdown operation can to some extent be used constructively provided that the prolongation is known in advance. Preparatory measures for decommissioning can in this case be carried out at the facility, which means that the consequences of the prolongation are less severe.

Other reactors, with 60 years of planned operating time, are not expected to be affected in the same way since fuel handling in these units already during operation will be adapted to the new system that is expected to be fully extended and well proven before decommissioning begins.

18.3.4 Interim storage capacity in Clab

Emptying of the final cores is also dependent on the capacity in Clab, see Section 3.4.3. The licence for Clab today covers interim storage of 8000 tonnes of spent nuclear fuel, which according to current forecasts will be reached in around 2023, i.e. close to the unloading of the final cores from R1 and R2. In the spring of 2015, SKB applied for a licence to increase the maximum inventory in Clab to 11000 tonnes. Decommissioning of particularly R1 and R2 may be dependent on this licence being obtained in time, if the remaining capacity in Clab, in case an increase is not possible, is prioritised for fuel from commissioned reactors, which would make emptying of the final cores impossible. An additional requirement that needs to be taken into account in the planning of capacity in Clab is the limitations on the amount fuel that is received per year according to the facility's safety analysis report.

The work load will be high at Clab in the early 2020s since many activities at the facility are planned to be performed at the same time. Loading in of fuel must therefore be carefully planned by SKB and the nuclear power companies so that shipments and reception are ensured. In the event of a longer interruption in fuel reception, the facilities that are finally shut down but not yet emptied of fuel will have an extended defueling. Similarly to the discussion above, this will lead to delays in the decommissioning project.

In the 2040s, when the younger reactors are decommissioned, Clab is not assumed to constitute a limitation. This is because fuel will start to be encapsulated and transported to the Spent Fuel Repository and thereby free up space in the pools. Thereby, space in the pools will be available with good margin prior to decommissioning.

18.3.5 Management of reactor pressure vessels

The only currently known new waste type that arises during decommissioning is reactor pressure vessels. The current plan is to dispose of whole reactor pressure vessels from both BWR and PWR reactors. This requires construction of a new deposition tunnel and a vault for reactor pressure vessels in SFR, see Section 3.3.3, and the corresponding adjustments in SFL. The decommissioning projects also require interim storage solutions, awaiting the extension of SFR and SFL being commisioned. Disposal as whole is expected to be both dose limiting and cost-effective for the decommissioning projects.

If it were to prove difficult to deposit the reactor pressure vessels as whole, or if it were undesirable for other reasons, the decommissioning projects and SKB need to plan for alternative management. The main alternative is segmentation at the power plant followed by packaging in waste containers. The consequence of the segmentation alternative is presently deemed to be that the decommissioning projects are extended and that the handling may lead to higher doses to personnel.

18.3.6 Management of long-lived waste

A large portion of the long-lived waste from the NPPs will arise before the planned commissioning of SFL. Long-lived waste may not be finally conditioned before the final set of requirements for SFL is established which is expected to be at the earliest when a licence to build SFL is obtained in the end of the 2030s. This means that the parts of decommissioning comprising execution of final disposal of the long-lived waste will not be carried out within the framework of early decommissioning projects.

In order to nevertheless be able to handle the long-lived waste that arises during decommissioning as completely as possible, sufficient planning premises need to be established. The ongoing planning of SFL, see Section 3.3.4, intends to give sufficient premises for the plants' decommissioning planning and will allow some conditioning of the long-lived waste during decommissioning projects. Interim storage will have to be carried out in waste containers without final conditioning.

Final conditioning can eventually begin during the design of SFL and be carried out in parallel with the construction of the facility. There are then possibilities for establishing a conditioning plant both locally at each facility and centrally. Up to the time of final conditioning, interim storage facilities will be required. In many cases, this can be solved locally on the site, but a need for a central interim storage solution is expressed by for example Barsebäck. The application for the extension of SFR includes the possibility of utilising the rock caverns that will be used for final disposal of decommissioning waste for interim storage of long-lived waste. Alternative solutions for storing waste with other licensees are also assessed.

18.3.7 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 streams are large within these waste categories, there are logistic challenges which create a need for well-defined routes for the waste in order to ensure adequate and efficient handling. In addition, international experience has shown that the quantities of very low-level waste and materials that need to be cleared are often larger than calculated, which imposes even higher demands on flexibility in the handling.

Examples of path ways that are desirable to create sufficient flexibility are: possibilities for disposal in near-surface repositories at the NPPs or controlled external landfill, possibility to incineration for energy recycling with controlled handling of ashes, possibility to clear and reuse metal scrap.

Both shallow land burial and more regular conditional clearance comprise handling methods for decommissioning waste which are not yet established. This requires a close dialogue with the concerned authorities and other stakeholders.

18.3.8 National planning for critical resources and functions

The planned execution of a number of decommissioning projects during this RD&D period 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 revision of competency requirements and competence availability is under way, in order to overview the situation and initiate adequate measures. In addition, a first dialogue with potential contractors has been initiated. The main decommissioning activities that are judged to require specific resources are segmenting the reactor internals, handling reactor pressure vessels as whole and dismantling 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.

In order to to obtain the necessary licences according to the plans, suitable processes need to be developed within the industry. This means that more coordination between the nuclear power companies is needed on these issues and that the licensees treat SSM, county administrative boards and municipalities in a uniform manner. It is also important that the decommissioning projects have a close dialogue with regulatory authorities. In the event revision of competency requirements and competence availability result in the conclusion that the projects will not be possible to carry out according to current planning, measures to reduce resource conflicts have to be adopted. This can, for example, entail that work within the projects is spread over longer time periods or that the entire project or sub-project is postponed in favour of progress in other activities with higher priority.

19 Continued activities within decommissioning

The decommissioning-specific activities that are described in the present RD&D programme are aimed at conducting development activities and taking the steps needed to be able to safely decommission the nuclear facilities. As the challenges in waste management during decommissioning are largely the same as those during operation, the decommissioning-specific activities mainly involve identifying differences from operation and safely handling the large volumes of waste that arise during dismantling.

This chapter presents conclusions from the development work during the preceding RD&D period on a general level, along with an account of the importance of the conclusions for the further planning. Furthermore, the remaining development work is presented that is identified in the decommissioning field, which is required to achieve the planning as given in Chapters 15–17. This includes both specific measures for the respective nuclear power companies and common needs.

19.1 Current situation

Below the decommissioning-related development activities that have been on-going during the past RD&D period are presented.

19.1.1 Industry-wide development work

The completed industry-wide development work which to a large extent has been pursued by SKB together with the nuclear power companies is presented below. Certain parts are described in greater detail in Part II, see references.

Management of reactor pressure vessels and large components

Several studies have been conducted to compare and shed light on the management of large components, such as whole reactor pressure vessels, so that the whole chain from dismantling to deposition takes place in an optimal manner. This is further described in Section 6.4.

Interim-storage facilities

A greater need for interim storage of short-lived radioactive waste has arisen as the dismantling and demolition is planned to start earlier than SFR can receive the waste. Another factor is that the capacity of the existing SFR is not sufficient for all operational waste arising. SKB has analysed whether interim on-site storage would be better for the NPPs or if centralised solutions are to be preferred. The conclusion is that it is more advantageous to the NPPs to interim-store their waste on site instead of a centralised solution operated by SKB. The need and opportunities for interim storage are further described in Section 3.3.3.

Near-surface repositories

In the area of near-surface repositories and management of very low-level waste, work has been carried out as described in Section 3.3.3. Among other things, SKB has studied if the forecasted decommissioning waste in the extended SFR could be handled in any other way, for example in near-surface repositories. The results show that SFR is the recommended alternative.

Clearance

Clearance has great importance for how decommissioning is planned and carried out, which has been a driving force behind SKB, together with the nuclear power companies and other licensees, developing a manual for clearance and site release during dismantling and demolition (Berglund et al. 2016). This work has also been reported to SSM and will serve as a basis for the nuclear power companies' further planning for clearance and site release in decommissioning.

Waste containers

Work has been carried out to develop waste containers and associated transport containers to facilitate an optimal handling of the waste during dismantling and demolition. This work is described in Section 6.5.

Structure for decommissioning plan

SKB has, together with the nuclear power companies, developed a common structure for the reporting of decommissioning plans (Calderon 2014b). The purpose is to support the licensees in the development of decommissioning plans as required according to the regulation in SSMFS 2008:1. This structure serves as a basis for the decommissioning plans that are updated during the period by each licensee and show a consensus and uniform presentation and interpretation of requirements. This also facilitates SSM to carry out its work as a regulatory authority.

International development work

During the period, SKB and the nuclear power companies have kept track of 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 cooperative programme but the IAEA's programme is also of importance. The latter focuses to a greater extent on the development of IAEA's Safety Standards which underlie 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. Some of the decommissioning aspects that OECD-NEA has worked with during the period are radiological survey and characterisation, knowledge and experience on site remediation of nuclear facilities and management of soil and groundwater that has been contaminated. Another focus area has been preparations for decommissioning that take place during operation and after shutdown to achieve optimum projects.

SKB and the nuclear power companies will continue to participate in the international networks within 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.

Pilot study at Energiforsk

A pilot study, Nuclear Decommissioning – A feasibility study of potential R&D initiatives for Energiforsk's nuclear programme, has been carried out with the purpose of studying desires regarding preparatory development of the nuclear power industry and its main stakeholders prior to a nuclear power plant being decommissioned and dismantled (Fors 2015). The focus was on identifying and defining the work that is suited for Energiforsk's nuclear power programme.

The three areas Energiforsk is recommended to develop are: Lessons learned and experience, logistics and conventional waste management. As a result of among other things the organisational rearrangement at Vattenfall with the creation of BU-ND and due to the decommissioning projects for the R1 and R2 being brought forward, work within the framework of Energiforsk has been paused until further notice. The results of the pilot study, however, provide a valuable basis for the project planning.

19.1.2 Development work at Uniper

Barsebäck Kraft AB

BKAB has through experience exchange developed an overall plan for decommissioning. The phases in the plan are based on the radioactive waste being removed from the facility in the order high-level to low-level, i.e. fuel, probes, control rods, internals and finally low-level waste. In combination with a decontamination of the primary circuits, it has resulted in that mostly low-level waste will occur during dismantling and demolition. This gives lower dose and less radiological risk for the personnel.

The following studies have been carried out: Demolition and dismantling of RPV as whole, segmentation of RPV and internals, radiological inventory during dismantling and demolition of Barsebäck 1 and 2, survey and categorisation of plant and environs with regard to radioactive contamination, management of large components, decontamination of building structures, demolition of all buildings, demolition and waste management logistics and 3D model of the facility.

OKG Aktiebolag

During 2014 project ProAct was carried out to prepare the licences required according to the Environmental Code and the Nuclear Activities Act to put Oskarshamn 1 into service operation in a planned and cost-effective manner.

The project comprised the following subareas:

- Environmental impact statement (EIS).
- Safety analysis report (SAR).
- Decommissioning plan.
- Separation O1 and O2 (technical measures).
- Competence, staffing, organisation and communication (measures within organisation and activities).

In view of the decisions to close Oskarshamn 1 and Oskarshamn 2 prematurely, in 2015 the Decommissioning Preparation Project, DPP, was started. The purpose of the project was to create conditions for setting up of a new department for decommissioning activities at OKG, and adapt the management system for this. The project's task was also to identify and work with critical activities related to decommissioning of Oskarshamn 1 and Oskarshamn 2, where work must take place immediately to not delay upcoming activities. The new department with responsibility for decommissioning was set up during August 2016 and the handover of project DPP's tasks is proceeding in the autumn of 2016.

19.1.3 Development work at Vattenfall

Principles for organisation and management of the decommissioning projects

As a step in developing Vattenfall's decommissioning activities, a project was initiated to devise an overall strategy for how the Vattenfall Group's Swedish decommissioning projects would be controlled and organised. The project, which went by the name Principles for Organisation and Management of Decommissioning Projects at Vattenfall, identified, based on national and international experiences, two fundamental principles for successful decommissioning of nuclear power:

1. Separate the organisation of decommissioning from operation at group level.
2. Gather the competence at group level within a new decommissioning organisation.

The recommendation from the project was thus that Vattenfall's organisational structure concerning decommissioning should be focused on what within the project was termed "Case 1": Decommissioning organisation as licensee. A policy decision was made in late autumn 2014. The new policy serves since then as a basis for both the build-up of the new organisation part BU-ND and for existing decommissioning planning.

Project AVANS

Project AVANS (Avvecklingsplanering inom Vattenfalls kärnkraftsverksamhet i Sverige – Decommissioning planning within Vattenfall's nuclear power operations in Sweden) was started during the spring of 2015, to produce an operations plan for an organisation to carry out decommissioning of the oldest reactors within Vattenfall and establish a platform for future decommissioning. The purpose was to create favourable conditions for the decommissioning organisation's continued activities and minimise the risk of increased decommissioning costs.

Project AVANS has focused on overall strategic aspects of decommissioning, and on identifying critical issues that must be solved in order to permit safe, optimised and effective decommissioning. The work has been divided into a number of sub-projects for handling areas such as licences, organisation and waste management. The conclusions of the sub-projects contributed to an overall assessment concerning critical aspects that need to be considered and managed in the continued decommissioning planning for Vattenfall's nuclear power plants. The outcome of the AVANS project includes the planned concept for decommissioning execution presented in Section 16.1, which today serves as a basis for all decommissioning planning within the group.

Decommissioning planning in 4D

A project concerning decommissioning planning in 4D has been conducted with the purpose of investigating the possibilities of using planning in programmes with 3D models where visualisation of dose and work operations is possible by simulation over time. The results show a large potential for this type of planning, but also that the market is immature and requires further technology development. Since the first decommissioning projects within Vattenfall take place in the near future there is not currently time for this development; it should be produced in the future. For projects in the near future, the present range and level of knowledge regarding 3D and 4D should be applied.

Ringhals 50 group

In conjunction with the completion of Ringhals' decommissioning study (Hansson et al. 2013), a working group called Ringhals 50 group was initiated, with representatives from management, HR, economy, technology, operation and communication at Ringhals NPP. As decommissioning was still more than ten years in the future, the existing operation plans did not cover decommissioning aspects which risked leading to suboptimal operations. The purpose of the working group was therefore to prepare the organisation for future decommissioning. The work was proactive and intended broadly to start considering decommissioning within all operations.

The result of work in the 50 group was, among other things, a staffing plan, an overview of connections and dependencies between reactor unit pairs and communicating the decommissioning plan and the decommissioning study. In conjunction with the early shut down of Ringhals 1 and Ringhals 2, the group ceased its activities as the tasks passed to project STURE.

Project STURE

In conjunction with the shutdown decision for Ringhals 1 and 2, project STURE (Säker och Trygg Utfasning av Reaktor 1 och 2 - Safe and Secure Outage of Reactors 1 and 2) was started. The purpose of the project is to identify, analyse and implement measures required so that Ringhals 3 and 4 will be able to be operated further when Ringhals 1 and 2 are decommissioned. Apart from physical separation measures STURE is also responsible for the licensing process that is required for shutdown of Ringhals 1 and Ringhals 2. In addition, the project includes organisational and economic aspects of the decommissioning decision. The project is under way and will be responsible for all decommissioning planning at Ringhals NPP during this RD&D period.

19.2 Programme

Future activities with a focus on the work during the coming six-year period are presented below. The section has been separated into industry-wide development in Sections 19.2.1 to 19.2.2, and group internal development in Sections 19.2.3 to 19.2.4.

19.2.1 Industry-wide development work on waste and final disposal

The impact of nuclear power decommissioning on the design of the final repository system is described below with a focus on the work that is required to permit decommissioning in line with existing plans.

Interim storage of long-lived waste

In order to be able to release Barsebäck NPP after completed decommissioning there is an interest in avoiding long term storage of radioactive waste at the facility. When SFR has been put in operation after the extension, the possibility of interim storage of long-lived waste there until SFL is opened is being planned. The need for interim storage of long-lived waste in SFR is thereby a direct result of facilities being decommissioned. The development work that is required to permit interim storage forms part of the extension of SFR, see Section 3.3.4.

Conditioning of long-lived waste

A large portion of the long-lived waste from the NPPs will arise before the planned commissioning of SFL. This means that sufficient planning premises need to be established in conjunction with the NPPs project planning for decommissioning 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 NPPs decommissioning planning, see Section 6.3.

Management of reactor pressure vessels and large components

A special waste package that will arise during decommissioning of facilities is reactor pressure vessels. In both the BWR and PWR case, disposal of the tanks as separate packages is currently being planned. Questions concerning BWR pressure vessels are handled within the framework of the extension of SFR, where existing application documents include pressure vessel disposal as whole, see Section 6.4. A comparative analysis between different handling alternatives for PWR reactor pressure vessels is being performed jointly by SKB and Vattenfall, see Section 6.4.

One of the questions regarding disposal of whole reactor pressure vessels is whether it will be possible and desirable to also deposit internals in place in the pressure vessel. This question will be studied by SKB and the licensees.

Being able to handle certain radioactive waste as large components in a decommissioning project may entail advantages from a dose and cost perspective. This requires, however, that the entire handling chain for the components is adapted for this. In analogy with the handling of whole reactor pressure vessels above there is an interest in investigating the possibilities of disposal of other large components. The programme for this is presented in Section 6.4.

Design of concrete structures and materials for SFL

The future SFL will dispose of two different types of long-lived waste: firstly activated metal components from the NPPs; and secondly contaminated legacy waste. The first waste type already occurs to some extent in the operation of facilities during exchange of BWR control rods and core internals in the reactors, but the majority of the waste will arise when the facilities are decommissioned. The part of SFL constructed to dispose of activated core components from the NPPs is thereby associated with decommissioning. The repository part, according to the current concept, will consist of a rock cavern where concrete is the main barrier. The concept and repository will be further developed within the framework of the work described in Section 9.3.

Method development for difficult-to-measure nuclides

Both the further development of the calculation model for the difficult-to-measure nuclides molybdenum-93, technetium-99, iodine-129 and cesium-135 and the question of relative accumulation of long-lived nuclides on system surfaces with activity build-up over long periods of time, will have implications for the decommissioning planning, see Section 6.1.2. Method development for difficult-to-measure nuclides will thereby be monitored from a decommissioning perspective.

Very low-level waste

Management of very low-level waste comprises a challenge during decommissioning. The character of the waste stream is varied and various efforts will be required to optimise handling and final destination. An important part consists of ensuring removal of the waste, where disposal in SFR constitutes a possibility. The development work that is required to permit final disposal is part of the extension of SFR (see Section 3.3.3).

Waste packaging and waste transport containers

Double-mould and/or *tetra-mould* can offer efficient handling and flexibility for conditioning of decommissioning waste as larger components can be packed in them. Efforts to realise these by developing manufacturing documents and further investigating its prerequisites are continuing within the work of SKB, see Section 6.5.2.

ATB 1T, which during the decommissioning projects will be used for transport of steel tanks, 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 6.5.3.

If new waste containers are developed, corresponding transport containers will also need to be developed. For *double-mould* and *tetra-mould* the work of producing a tailored transport container is included, see Section 6.5.3.

Damaged fuel

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 damaged fuel that exists in the facility. A project with the objective of disposing of all damaged fuel at the facilities has started and will be concluded during this RD&D period, see Section 7.1.

19.2.2 Other industry-wide development

The industry-wide work needed prior to decommissioning, but that is not directly associated with SKB's transportation and final repository systems, is presented below. The work may be coordinated by SKB, but is not entirely bound to do so in case the nuclear power companies prefer another cooperation form.

Harmonised licensing

A fundamental prerequisite for being able to execute the planned decommissioning projects is that the required 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. The work of developing more harmonised licensing will be pursued during this RD&D period.

IT support for waste documentation

During the entire life cycle of a nuclear facility, it is of importance, and a requirement, to have records of the waste produced at the facility. In the coming planning and design, record-keeping needs to be revised regarding radioactive waste that arises during dismantling and demolition. How waste data is propagated from the licensees to SKB and the final repositories where the waste is deposited and which system is supposed to provide support for this needs to be assured. The work is industry-wide but will be coordinated by SKB.

19.2.3 Development needs at Uniper

Barsebäck Kraft AB

Ongoing studies and tasks aim to establish the requirements for decommissioning.

Planned studies are dismantling preparations for supporting systems at decommissioning and decontamination of process equipment in the waste facility.

OKG Aktiebolag

In view of the decommissioning of Oskarshamn 1 and Oskarshamn 2 being brought forward, intensive planning is under way at OKG; this also applies to the planning of OKG's development work linked to decommissioning. The work is largely intended to identify the needs of research that exist for decommissioning and clarify the sequence for these studies. This will be carried out based on national and international experiences and OKG's own experience from operating the units.

OKG has in its general decommissioning strategy for example identified the following areas where the need for studies and analysis exists:

- Waste management strategy.
- Dismantling and demolition strategy.
- Organisational and competence strategy.
- Facility and production strategy.
- Radiological strategy.
- Business and purchases strategy.

In certain areas, studies have already started, for example strategic studies are under way regarding waste management and dismantling and demolition sequences. In the coming work for further studies, for example, studies in waste and demolition logistics and radiological survey and categorisation of the facilities are also planned.

19.2.4 Development needs at Vattenfall

A large part of the development work within the Vattenfall Group will be carried out as a part of the project for decommissioning of Ringhals 1, Ringhals 2 and Ågesta. Within the next three years, the work in the analysis phase, the planning phase and most of the establishment phase will be carried out for these projects. This entails formulating and analysing data and project prerequisites for developing the concept, sequence and detailed timetable for dismantling and demolition as well as transport and waste management with the necessary documentation and licences.

During the subsequent three-year period, the projects go into the dismantling and demolition phase. The areas that will be studied further relating to the actual execution of dismantling and demolition as well as waste management are described more specifically below.

In addition to the more routine project work, more general issues will be investigated over the coming years at Vattenfall; these are described under the heading Optimised Decommissioning at the end of this section.

Decommissioning concept and execution

In the formulation of a sequence for dismantling and demolition activities, there are several conceivable approaches to evaluate. Either the most radioactive parts can be dismantled first to reduce the radiation doses in the facility as quickly as possible or the least contaminated parts can be dismantled first. The advantage is then to reduce the risk of cross contamination and moreover the work with the more radioactive parts is also facilitated by areas being available. Another example is the selection of whether facility parts should be dismantled from an availability perspective in order to simplify logistics (outside and in) or if systems should be dismantled from below and above to minimise falling risks. The choice of methods, such as when mechanical or thermal methods are best suited, also needs to be analysed. The alternatives that are most advantageous will be studied from a dose and safety point of view as well as what is most optimal for minimisation of project risks and waste generation.

A fundamental input parameter for the planning and execution of decommissioning is via survey and characterisation to acquire necessary knowledge on the occurrence of radioactive elements in the facility. During the RD&D period, survey and characterisation work will be studied so that it is carried out in an optimal way based on the stipulated requirements and how decommissioning is planned.

Waste management

Alternative approaches in waste management which need to be studied are for example whether the waste should already be cut up to fit in the waste packages at source, or if the facility should be dismantled in as large segments as possible to maximise the progress made in dismantling, after which waste conditioning takes place at a centralised facility. An optimisation of decommissioning projects with respect to the conditioning opportunities, logistics optimisation and dose will be carried out.

In the design of waste management, the ratio between installed process capacity and established buffer and storage capacity also needs to be evaluated. This optimisation, which will be carried out during the RD&D period, will be a trade-off between a waste-oriented decommissioning process, which risks affecting dismantling activities, and a dismantling-oriented process, where waste management risks becoming a limitation.

Handling of waste which is in the area between very low-level and clearable waste constitutes a challenge as the due to the low contamination level. To optimise waste management, Vattenfall will investigate when different alternatives such as conditional clearance, recycling or deposition are advantageous from environmental, dose load and economic points of view.

Optimised decommissioning

Within BU-ND work is planned to proceed during this RD&D period to organise, control and optimise the Vattenfall Group's decommissioning activities. Essentially, the work will focus on the following areas:

- **Organisation and control**: As a continuation of the projects that have been carried out at Vattenfall during 2014 and 2015 regarding organisation and management, see Section 19.1.3, further work will be pursued to construct a model for the decommissioning project's organisational structure.
- **Internal competence and strengths**: As a step in ensuring that there are sufficient resources with the correct skills available at Vattenfall in order to be able to complete the planned activities for disposing of the waste, a strategic competence plan will be devised. The plan will identify any gaps and activities that need to be initiated to cover these. The work has been initiated and will continue during this RD&D period.
- **Supplier market**: To stimulate a mature supplier market with sufficient resources and capacity to be able to execute the practical dismantling and demolition work that will be subcontracted, a stimulus plan will be developed and implemented during this RD&D period.
- **Purchase concepts:** Due to the large dependence of decommissioning projects on effective procurement, a strategy for procurement will be devised. The strategy will be translated into a plan in which, among other things, the possibilities with, and potential of, a central function for purchases, evaluation and experience accumulation regarding contractors will be evaluated.
- **Cost calculations:** For the purpose of preparing a project cost calculation suitable as material prior to, and in the follow-up of decommissioning projects and their constituent steps, a development project in the area has been initiated and will be further developed during the coming RD&D period.

Part IV

Other issues

20 Preservation of information and knowledge through generations

21 Disposal in deep boreholes

20 Preservation of information and knowledge through generations

In its day-to-day work and until the closure of the final repositories, SKB manages and preserves documents, data and information in the long term in accordance with requirements from SSM and the National Archives. In these requirements long term entails longer than 25 years. When SKB's activities cease, the archives, arranged and listed, must be handed over to the National or regional archives according to SSM's regulations.

SKB has also worked with questions concerning archiving and information preservation in the long term, several generations into the future, for several years. In the consultations prior to submitting applications for the encapsulation plant and the final repository for spent nuclear fuel, the issues on preservation of information and knowledge of the Spent Fuel Repository and who is responsible for it post-closure, were frequently raised. In the referral procedure of the application for the KBS-3 system under the Environmental Code, wishes have also been 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 be considered most urgent for the Spent Fuel Repository, but also need to be considered for SFR and SFL. Practically, the solutions for information preservation need to be in place when the Spent Fuel 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 so far in the future.

The structured mode of working SKB is conducting today regarding management and preservation of documents, data and information is a valuable platform for the future assessment and the choice of which information needs to be preserved. SKB believes that the only meaningful plan of action is to have a way of working that aims to keep the issue updated, develop the work and disseminate knowledge of the need. A prerequisite for succeeding is an interest and involvement from several societal sectors.

20.1 Complete work

SKB has had a long-standing collaboration with Linnaeus University, studying the following areas:

- Human evolution and long-term future.
- Historical awareness and learning for the future.
- The idea about the future in the cultural heritage sector and management of radioactive waste.

This part of the collaboration was concluded in 2015 and the main result is that through participating in conferences and publishing articles, the question of how to preserve information and knowledge on a final repository for radioactive waste has been introduced to sectors in society that work with preservation issues (Holtorf 2013, Holtorf and Högberg 2013, 2014, 2015a, b). Experience from the work shows, among other things, that the institutions working to preserve objects and culture for "the future" usually do not have a picture of what the future may look like and what it might be interested by. It is important to create a living heritage, which will also continue to be considered worthwhile preserving.

Furthermore, a researcher at the Centre for Theology and Religious Studies at Lund University has at SKB's request published two articles on the difficulties with transferring information on hazardous nuclear waste in writing (Wikander 2015a, b). It is pointed out that, among other things, a crucial factor for whether languages can be understood is how many documents that are preserved and what they look like. Languages that are related to others are easier to interpret than isolated languages. History further shows that the risk of being misunderstood is significant. Wikander paints a scenario where an extremely advanced future underestimates us and believes that we were primitive cavemen who could never have mastered nuclear power technology. To consider the problem, he believes that we need to try to imagine what people in the future will think of us – and our language.

20.2 Current and future work

SKB is participating in the international projects Records, Knowledge and Memory across Generations and Assembling Alternative Futures for Heritage, which are presented in Sections 20.2.1 and 20.2.2. Furthermore, SKB is involved in a joint project with its French counterpart, Andra, concerning information preservation. Andra has an extensive programme for the next few years, for example concerning archaeology of the landscape, continuity of institutional organisations and consequences of social breakdowns (war, natural disasters etc.).

20.2.1 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 through generations (Records, Knowledge and Memory across Generations, RK&M). Phase I was finalised in 2014 with an international conference in Verdun where, among other things, it was concluded that there is a large interest in the question even outside the realm of society that works directly with the management of radioactive waste. Many proposals and thoughts around long-term preservation of information and knowledge were presented. The conference was divided into three parts based on which time perspective is considered: short term (while the final repository is in operation), medium-long term (the period after closure during which society can be assumed to have some kind of control), and the long term (for the Spent Fuel Repository several thousand years). This is the same time perspective that is used in OECD-NEA's working group.

The work in phase II is under way and is expected to be finalised in April 2018. The goal is to present a form of "tool box" and an international consensus on proposals for which "tools" can be used to best preserve documents, information and knowledge on final repositories for radioactive waste over the different time perspectives. Among other things, the prospects are analysed and discussed for storage in traditional archives and cross archiving, international mechanisms, time capsules, markers (on the surface and underground), creation of tradition and heritage and how and to whom responsibility for the final repository and the preservation of information and knowledge can be transferred after closure.

A key issue is which type of data and information should be saved with the different time perspectives. SKB is involved in a practical attempt to produce a common structure for a short aggregated description, about 40 sides – Key Information File of a final repository. The main target group is future laymen. The description must for example contain the siting of the final repository, the content, its properties and a summary of the assessment of post-closure safety. It must also provide a description of radioactivity and nuclear power plants and the context in which they have been used. France, the USA and Sweden will present concrete documents describing the French above-ground final repository for short-lived low- and intermediate-level waste (Centre de stockage de la Manche) that was closed in 1994, the Swedish Spent Fuel Repository respectively the American WIPP. The basis for the description of the Spent Fuel Repository consists of the applications under the Environmental Code and the Nuclear Activities Act.

SKB is also participating in another OECD-NEA project, Radioactive Waste Repository Metadata Management (RepMet), aimed at producing a common view of which meta data should be saved.

20.2.2 Assembling Alternative Futures for Heritage

Assembling Alternative Futures for Heritage (AAFH) is an interdisciplinary research programme aimed at developing a wide, international and multisectoral framework for understanding "heritage" in its most expansive sense. AAFH is a collaboration between the University College London, University of Exeter, University of York and Linnaeus University in Kalmar. The programme started in the spring of 2015 and is planned to continue for four years. SKB is invited to participate as a result of the collaboration we have previously had with Linnaeus University.

Although heritage might be considered strictly defined, it is possible to claim that all community functions that in practice work with questions concerning the future or issues involving the past, can be considered to engage in some kind of cultural heritage. The aim of the research programme is to explore different guidelines for heritage, both with a critical and a comparative perspective, to improve the academic understanding of the social, cultural, material, ecological, political and ideological contexts in which “heritage” is transmitted. In this way, the programme aims to facilitate a more democratic and informed dialogue concerning cultural heritage, between different community functions and companies and their target groups as well as contribute to the development of new politics favourable for more innovative, common, resilient and sustainable strategies to preserve the cultural heritage. The work will involve academics in different disciplines within arts, humanities, social sciences and natural sciences, including anthropology, archaeology, environmental science, cultural geography, history, creative arts and ecological humanities.

The research programme is divided into four themes:

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

SKB will as an industry partner in particular be involved in the theme Preparations for an uncertain future together with One Earth – New Horizons Message project. New Horizons is an unmanned space probe that NASA sent towards Pluto to photograph and gather information. The journey started in January 2006 and the probe reached Pluto in July 2015. Once the assignment is complete and all collected information has been sent to NASA, the idea is that a message from Earth will be placed on the computers. The space probe will continue its journey through space forever, or until someone/something finds it.

The point of contact with the message on New Horizons is that there is the same uncertainty regarding format and content in a message to an unknown receiver. How can the message be made sustainable and understandable?

The first workshop for the research programme was held in Sweden in March 2016 and, among other things, the following issues were discussed:

- How does the naming of places (archives, museums, zoos, botanical gardens, “sites”, landscapes, final repositories for radioactive waste, etc.) contribute to the way in which they are understood and evaluated?
- How can the different architectonic and technical forms within the different areas the research programme includes, affect each other?
- What different forms are available for preservation ex-situ and how do they relate to the various methods and processes that are performed within them?
- What different consequences are there with preservation ex-situ and in-situ?

SKB’s participation was a first opportunity to obtain direct contact with individuals who, in one way or another, work professionally with preservation of objects and knowledge and benefit from their experience and knowledge in relation to the question of how to preserve information and knowledge on a final repository for radioactive waste.

21 Disposal in deep boreholes

In the case of disposal in deep boreholes, the most important safety function is the isolation and retardation of radioactive elements that the rock offers, which to a large extent assumes that the groundwater is stagnant at great depths. An important reason for the stagnant conditions is that the groundwater has high salinity, and thereby high density, resulting in a stable stratification that counteracts remixing with the lighter fresh water above.

Deposition in deep boreholes has some potential advantages. Deposition takes place at great depth, which means a long distance and probably also low groundwater flow up to the ground surface for radionuclides that spread from the near field to the water in the rock. This can in turn imply long travel times to the biosphere unless rapid transport pathways are created via the deposition hole or via zones in the rock combined with motive forces due to for example gas evolution.

The probability of inadvertent intrusion is also lower for disposal in deep boreholes due to the greater depth, which leads to lower risk that people in the future for example will inadvertently drill into the deposition area. Furthermore, the waste/fuel will be more difficult to retrieve for malicious purposes due to the great depth.

The great depth, however, also brings disadvantages such as difficulties in characterising the surrounding rock, increased risk for mishaps during deposition and retrieval for intended purposes.

21.1 Current situation

21.1.1 Carry out deposition

The concept of disposal of spent fuel in deep boreholes entails typically that steel canisters with spent nuclear fuel (one PWR or two BWR assemblies per canister) are emplaced in boreholes at 3–5 kilometres' depth. This requires a borehole diameter of at least 44.5 cm at repository depth. Approximately 80 deposition holes of this type would be required to dispose the spent fuel from the Swedish nuclear power plants, provided that 400 canisters are placed in each hole. In view of the heat generation from spent nuclear fuel, boreholes should be situated with about 100 metres' spacing (Marsic and Grundfelt 2013). The top part of the borehole is sealed. The concept is described in more detail in SKB (2014i) and is based on, among other things, work carried out by Sandia National Laboratories, one of United States Department of Energy's laboratories, during 2009 and 2011 (Brady et al. 2009, Arnold et al. 2011).

A possible principle design of a facility for disposal of spent fuel in deep boreholes with 80 boreholes is shown in Figure 21-1.

In the case of disposal in deep boreholes mishaps may occur with consequences that are difficult to rectify. For example, a canister may get stuck in the hole and break before it has reached disposal depth. A broken canister could get stuck in a position with flowing groundwater, without being surrounded by a protective buffer, with risk for contamination of borehole liquid and tools that are used to restore the situation (Grundfelt 2013). The probability of an accident with release of activity in connection with the disposal of 400 canisters in a borehole has been calculated to between 1×10^{-4} and 7×10^{-3} depending on disposal technique (Hardin et al. 2016).

21.1.2 The evolution of the repository after closure

Figure 21-2 shows schematically dispersion of radioactivity from a KBS-3 repository and for disposal in deep boreholes. The time point illustrated could be at the end of the present interglacial temperate period or the end of the next glacial period.

The results from SR-Site show that the canisters in the KBS-3 repository can be expected to be leaktight for a very long time (SKB 2011b). No activity dispersion has therefore been added around the KBS-3 repository in the figure.

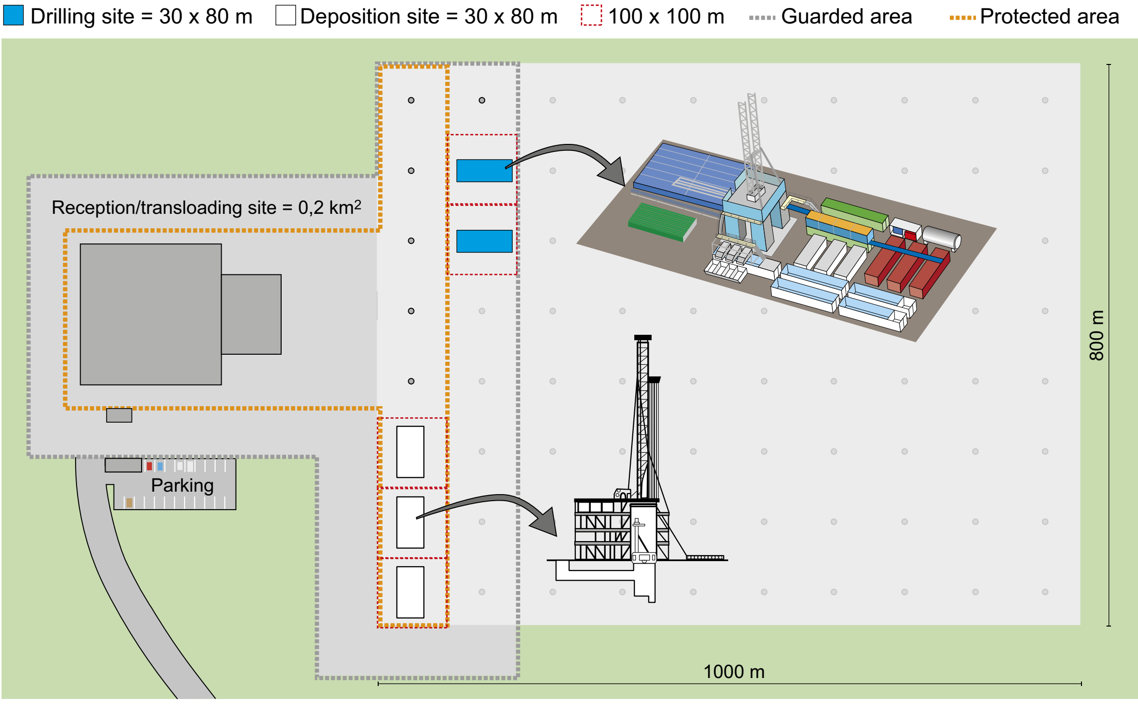


***Figure 21-1.** Possible principle design of a facility for disposal of spent fuel in deep boreholes. Note that the facility layout and other design only comprise simplified illustrations.*

In the case of disposal in deep boreholes, a steel canister has been assumed. Other canister material has been considered, but the aggressive environment in the deposition zone with expected temperatures of 80–100 °C and salinities of 50–100 g/L means that even the life of a more corrosion-resistant canister material is greatly limited (SKB 2014i).

Directly after deposition in a deep borehole, steel canisters are expected to start corroding. This has two consequences; thinned material in the canisters and hydrogen gas obtained as a reaction product. Material thinning leads eventually to the bottom-most canisters in a canister string being expected to lose so much of their strength that they give way. In this case the fuel assemblies inside the broken canister will probably also be deformed, whereby there is a risk that uranium pellets are exposed, leading to the borehole and its environment being contaminated with radionuclides. Temperature increase and gas formation both produce driving forces for upward flow in the area around the hole, which can lead to an upward transport of radionuclides. The extent of this has not been analysed for the conditions that are relevant to Sweden, but are illustrated schematically in Figure 21-2. In studies performed in the USA, with the conditions that apply there, very low doses have been calculated (Lee et al. 2012). They do not, however, taken into account the driving forces that can be produced from the gas formation.

Corrosion of the canisters may lead to the fuel geometry being changed by pellets leaving the fuel rods and falling downwards in the hole. This could lead to criticality. However, it is currently difficult to assess how plausible such an evolution would be and what consequences this could result in.

21.1.3 Work carried out since RD&D 2013

In the supplement to the application for licences for the Spent Fuel Repository under the Environmental Code that was submitted to the Land and Environment Court in September 2014, three documents were included dealing with the deposition of spent fuel in deep boreholes (SKB 2014i, j, Thegerström 2014). The report (SKB 2014i) is an update of the report where final disposal of spent nuclear fuel by the KBS-3 method and disposal in deep boreholes are compared with regard to a number of central issues such as deposition and closure, nuclear safety in the handling of encapsulated spent nuclear fuel and long-term safety of a closed repository. An overall conclusion from SKB (2014i) is that there is nothing today to suggest that a switch to planning for disposal in deep boreholes would result in a safer final disposal of the spent nuclear fuel than is offered by the KBS-3 method.

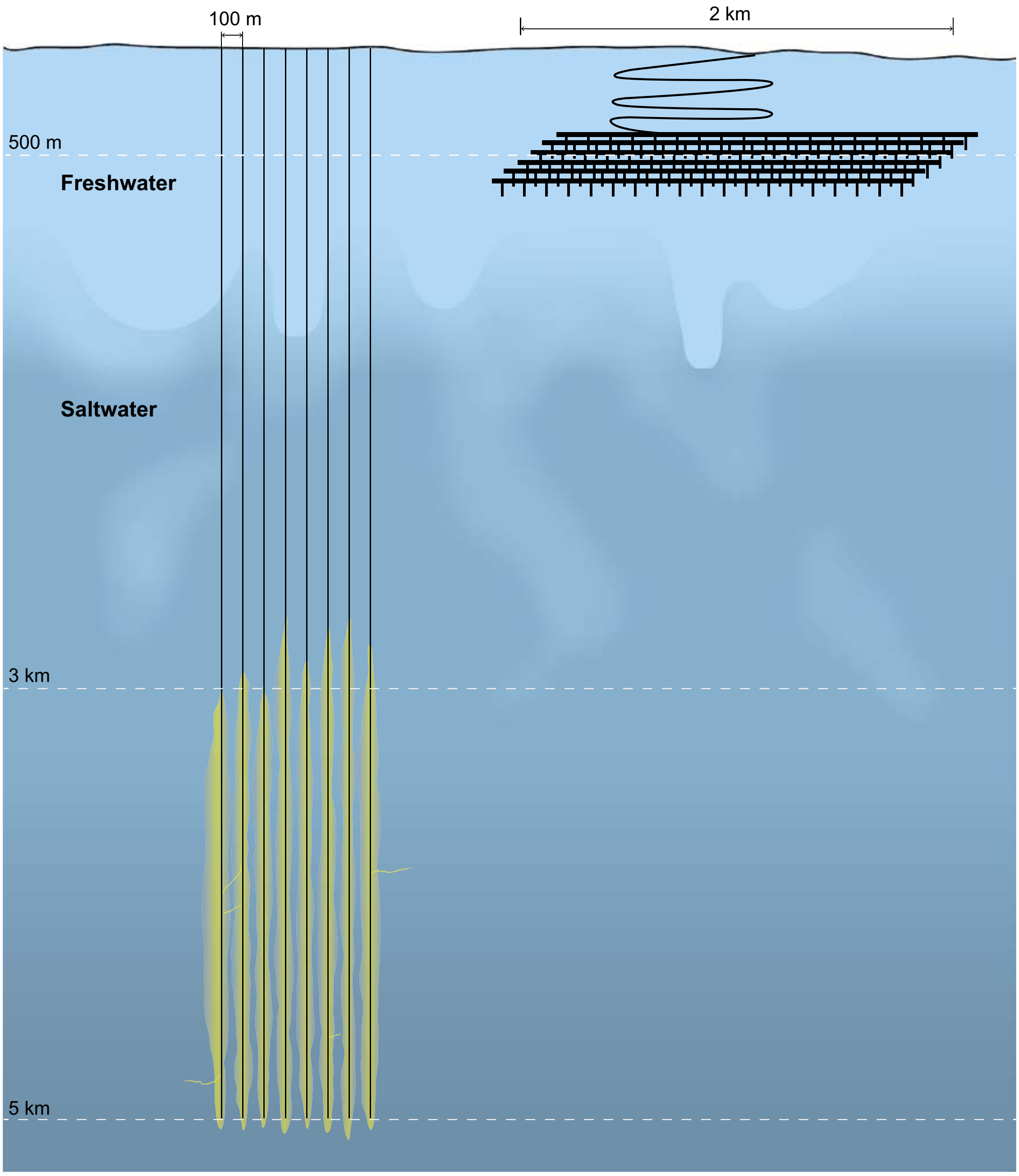


Figure 21-2. *Schematic future activity dispersion situation (highlighted in yellow) for a KBS-3 repository and for disposal in deep boreholes. The time point illustrated could be at the end of the present interglacial temperate period or the end of the next glacial period. At that time point no activity dispersion is expected from the KBS-3 repository.*

To keep up with the development regarding deep boreholes, SKB has participated in conferences and workshops, for example in the workshop that U.S. Nuclear Waste Technical Review Board (NWTRB) arranged in Washington in October 2015, with the theme "International Technical Workshop on Deep Borehole Disposal of Radioactive Waste" (NWTRB 2015, 2016).

The United States Department of Energy (U.S. DOE) has set aside funds to drill a demonstration borehole down to about 5 000 metre depth, as well as work with method development during the period 2015–2019. The purpose of NWTRB's workshop was to highlight the U.S. DOE's research and development programme for deep boreholes. It is not yet decided where the demonstration hole will be drilled.

At the workshop it emerged that the U.S. DOE is not considering disposal in deep boreholes for commercial spent nuclear fuel, but that there are a number of other waste types that potentially could be considered (NWTRB 2015, p 268). NWTRB concludes that even if it is possible to dispose of certain types of waste in deep boreholes, there is still a need for a geological repository. NWTRB also states that the U.S. DOE expresses that all waste types that are studied for disposal in deep boreholes can also be deposited in a geological repository (NWTRB 2016).

Disposal in deep boreholes comprised the theme of a session at the conference Waste Management 2016 in Phoenix, Arizona. The contributions to this session came mainly from Sandia National Laboratories and University of Sheffield. The presentations from Sandia concerned risk analyses of the deposition procedure (Freeze et al. 2016, Hardin et al. 2016) within the framework of preparations prior to the aforementioned test programme, while the contributions from Sheffield (Gibb 2016, Travis and Gibb 2016) described research on methods to improve the containment of the deposited waste and sealing of deep boreholes. One of the contributions to the session was based on SKB's work in the area (Grundfelt et al. 2016) and described mechanisms for potential dispersion of radionuclides from a borehole repository.

In June 2016, SKB participated in the conference "International Meeting on Deep Borehole Disposal of High-Level Radioactive Waste", arranged by the University of Sheffield. Nothing new emerged there regarding the possibilities of disposing of spent fuel in deep boreholes.

21.2 Evaluation and further work

21.2.1 SKB's judgement

SKB's stands by its assessment from previous RD&D programmes: that disposal in deep boreholes is not a realistic method for final disposal of spent nuclear fuel. A KBS-3 repository can, in contrast to deep boreholes, be constructed, operated and sealed in a manner that can be controlled and verified at every step. A repository according to the deep boreholes concept is associated with great uncertainties regarding the evolution of the repository after closure. Furthermore, it can be noted that no country in the world recommends disposal in deep boreholes as a preferred alternative for disposal of spent nuclear fuel.

21.2.2 Continued work

SKB intends to continue to follow developments within the areas drilling of and deposition in deep boreholes. There is, however, no justification for pursuing an independent research programme in the field. Available resources will instead be concentrated on realising a final repository according to the KBS-3 method.

SKB is following developments in the Swedish Deep Drilling Program (SDDP 2016) and thereby is updated of any results that are of relevance to the deep borehole concept.

SKB has observed that interest for disposal in deep boreholes has increased recently in the USA as the U.S. DOE has set aside funds for drilling of a demonstration borehole down to about 5 000 metre depth. SKB intends to follow the drilling, but at the same time concludes that the geological conditions are different in the USA and in Sweden.

SKB has further the intention to continue to monitor and in relevant contexts participate in international fora regarding deep boreholes.

References

SKB's (Svensk Kärnbränslehantering AB) publications can be found at www.skb.com/publications. SKBdoc documents will be submitted upon request to document@skb.se.

Aalto P, Aaltonen I, Ahokas H, J Andersson, Hakala M, Hellä P, Hudson J, Johansson E, Kemppainen K, Koskinen L, Laaksoharju M, Lahti M, Lindgren S, Mustonen A, Pedersen K, Pitkänen P, Poteri A, Snellman M, Ylä-Mella M, 2009. Programme for repository host rock characterisation in the ONKALO (ReRoC). Posiva Working Report 2009-31, Posiva Oy, Finland.

Abarca E, Idiart A, de Vries L M, Silva O, Molinero J, von Schenck H, 2013. Flow modelling on the repository scale for the safety assessment SR-PSU. SKB TR-13-08, Svensk Kärnbränslehantering AB.

Aggarwal S, Addepalli V, Smart N, 2015. Further metallographic analysis of MiniCan SCC test specimens. SKB R-15-11, Svensk Kärnbränslehantering AB.

Andersson E (ed), 2010. The limnic ecosystems at Forsmark and Laxemar-Simpevarp. SKB TR-10-02, Svensk Kärnbränslehantering AB.

Andersson K, 2013. Copper corrosion in nuclear waste disposal. A Swedish case study on stakeholder insight. Bulletin of Science, Technology & Society 33, 85–95.

Andersson L, Karnland O, Hedström M, 2014. Swelling pressure of ion-exchange resins at SFR – Experimental results. SKBdoc 1450664 ver 1.0. Svensk Kärnbränslehantering AB.

Andersson-Östling H C M, Sandström R, 2009. Survey of creep properties of copper intended for nuclear waste disposal. SKB TR-09-32, Svensk Kärnbränslehantering AB.

Anunti Å, Larsson H, Edelborg M, 2013. Decommissioning study of Forsmark NPP. SKB R-13-03, Svensk Kärnbränslehantering AB.

Aquilonius K (ed), 2010. The marine ecosystems at Forsmark and Laxemar-Simpevarp. SR-Site Biosphere. SKB TR-10-03, Svensk Kärnbränslehantering AB.

Arnold B W, Brady P V, Bauer S J, Herrick C, Pye S, Finger J, 2011. Reference design and operations for deep borehole disposal of high level radioactive waste. SAND2011-6749, Sandia National Laboratories, New Mexico.

Aronsson P-O, 2016a. C-14 accumulated in ion exchange resins in Swedish nuclear power plants Part 2: Historical accumulations up to and including 2014 and predicted future accumulations from 2015 onwards. SKBdoc 1544816 ver 1.0, Svensk Kärnbränslehantering AB.

Aronsson P-O, 2016b. C-14 i jonbytarmassor från svenska kärnkraftverk: sammanfattning av historiska och framtida upptag. SKBdoc 1544992 ver 1.0, Svensk Kärnbränslehantering AB. (In Swedish.)

Aronsson P-O, Lillfors-Pintér C, Henning Å, 2016. C-14 accumulated in ion exchange resins in Swedish nuclear power plants Part 1: Results from analyses and calculation of accumulation. SKBdoc 1543560 ver 1.0, Svensk Kärnbränslehantering AB.

Arvidsson A, Josefsson P, Eriksson P, Sandén T, Ojala M, 2015. System design of backfill. Project results. SKB TR-14-20, Svensk Kärnbränslehantering AB.

Babaahmadi A, 2015. Durability of cementitious materials in the long-term contact with water. PhD thesis. Chalmers University of Technology, Sweden.

Backers T, 2005. Fracture toughness determination and micromechanics of rock under mode I and mode II loading. Scientific Technical Report STR 05/05, GeoForschungsZentrum, Potsdam, Germany.

Backers T, Stephansson O, 2012. ISRM suggested method for the determination of mode II fracture toughness. Rock Mechanics and Rock Engineering 45, 1011–1022.

Barreiro Fidalgo A, Sundin S, Jonsson M, 2014. Effect of bentonite on radiation induced dissolution of UO_2 in an aqueous system. Journal of Nuclear Materials 447, 73–76.

Barreiro Fidalgo A, Dahlgren B, Brinck T, Jonsson M, 2016. Surface reactions of H_2O_2, H_2, and O_2 in aqueous systems containing ZrO_2. The Journal of Physical Chemisty C 120, 1609–1614.

Bastviken D, Svensson T, Sandén P, Kylin H, 2013. Chlorine cycling and fates of ^{36}Cl in terrestrial environments. SKB TR-13-26, Svensk Kärnbränslehantering AB.

Becker J K, Lindborg T, Thorne M C, 2014. Influence of climate on landscape characteristics in safety assessments of repositories for radioactive wastes. Journal of Environmental Radioactivity 138, 192–204.

Bengtsson A, Edlund J, Hallbeck B, Heed C, Pedersen K, 2015. Microbial sulphide-producing activity in MX-80 bentonite at 1750 and 2000 kg m^{-3} wet density. SKB R-15-05, Svensk Kärnbränslehantering AB.

Bengtsson S B, 2013. Interception and storage of wet deposited radionuclides in crops: field experiments and modelling. PhD thesis. Swedish University of Agricultural Sciences.

Berastegui P, Hahlin M, Ottosson M, Boman M, 2016. Memo: Analyses on N3 and N8 samples from Micans. SKBdoc 1538718 ver 1.0, Svensk Kärnbränslehantering AB.

Beresford N A, Wood M D, Vives i Batlle J, Yankovich T L, Bradshaw C, Willey N, 2016. Making the most of what we have: application of extrapolation approaches in radioecological wildlife transfer models. Journal of Environmental Radioactivity 151, 373–386.

Berglund J-Å, Lorentz H, 2016. Avvecklingsplan för Barsebäcksverket. 1884576 / 8, Barsebäck Kraft AB. (In Swedish.)

Berglund J-Å, De la Gardie F, Herschend B, Lorenz H, 2016. Friklassning vid nedmontering och rivning av kärntekniska anläggningar. SKB R-16-13, Svensk Kärnbränslehantering AB. (In Swedish.)

Bergman E, Fors P, Knutsson A, 2016a. Avvecklingsplan för Ringhals 1 och 2, 1848196 / 7.0, Ringhals AB. (In Swedish.)

Bergman E, Fors P, Knutsson A, 2016b. Avvecklingsplan för Ringhals 3 och 4, 2337675 / 2.0, Ringhals AB. (In Swedish.)

Birgersson M, Goudarzi R, 2016. Vapor transport and sealing capacity of buffer slots ("sauna" effects). SKB TR-15-09, Svensk Kärnbränslehantering AB.

Björck M, 2015. Estimation of oxide growth on joint surfaces during FSW. SKBdoc 1402837 ver 1.0, Svensk Kärnbränslehantering AB.

Björck M, Elger R, 2013. Oxidation kinetics of copper at reduced oxygen partial pressures. SKBdoc 1410172 ver 1.0, Svensk Kärnbränslehantering AB.

Björkbacka Å, 2015. Radiation induced corrosion of copper. PhD thesis. Royal Institute of Technology, Stockholm, Sweden.

Björkbacka Å, Hosseinpour S, Johnson M, Leygraf C, Jonsson M, 2013. Radiation induced corrosion of copper for spent nuclear fuel storage. Radiation Physics and Chemistry 92, 80–86.

Björkbacka Å, Yang M, Gasparrini C, Leygraf C, Jonsson M, 2015. Kinetics and mechanisms of reactions between H_2O_2 and copper and copper oxides. Dalton Transactions 44, 16045–16051.

Blom A, Pedersen K, 2016. Analyser av kopparhalter i vatten. Microbial Analytics Sweden AB. SKBdoc 1523054 ver 1.0, Svensk Kärnbränslehantering AB. (In Swedish.)

Bodén A, Pettersson S, 2011. Development of rock bolt grout and shotcrete for rock support and corrosion of steel in low-pH cementitious materials. SKB R-11-08, Svensk Kärnbränslehantering AB.

Bodén A, Sievänen U, 2005. Low-pH injection grout for deep repositories. Summary report from a co-operation project between NUMO (Japan), Posiva (Finland) and SKB (Sweden). SKB R-05-40, Svensk Kärnbränslehantering AB.

Bohl Kullberg E, 2013. Avvecklingsplan för Ågestaanläggningen. N-13-008, Vattenfall AB. (In Swedish.)

Boman M, Berger R, Andersson Y, Hahlin M, Björefors F, Gustafsson T, Ottosson M, 2014. Corrosion of copper in water free from molecular oxygen. Corrosion Engineering, Science and Technology 49, 431–434.

Bradshaw C, Kapustka L, Barnthouse L, Brown J, Ciffroy P, Forbes V, Geras'kin S, Kautsky U, Bréchignac F, 2014. Using an Ecosystem Approach to complement protection schemes based on organism-level endpoints. Journal of Environmental Radioactivity 136, 98–104.

Brady P V, Arnold B W, Freeze G A, Swift P N, Bauer S J, Kanney J L, Rechard R P, Stein J S, 2009. Deep borehole disposal of high-level radioactive waste. SAND2009-4401, Sandia National Laboratories, New Mexico.

Brandefelt J, Näslund J-O, Zhang Q, Hartikainen J, 2013. The potential for cold climate conditions and permafrost in Forsmark in the next 60,000 years. SKB TR-13-04, Svensk Kärnbränslehantering AB.

Brandt F, Curti E, Klinkenberg M, Rozov K, Bosbach D, 2015. Replacement of barite by a $(Ba,Ra)SO_4$ solid solution at close-to-equilibrium conditions: a combined experimental and theoretical study. Geochimica et Cosmochimica Acta 155, 1–15.

Bréchignac F, Oughton D, Mays C, Barnthouse L, Beasley J C, Bonisoli-Alquati A, Bradshaw C, Brown J, Dray S, Geras'kin S, Glenn T, Higley K, Ishida K, Kapustka L, Kautsky U, Kuhne W, Lynch M, Mappes T, Martin T E, Mihok S, Møller A, Mothersill C, Mousseau T A, Otaki J M, Pryakhin E, Rhodes O E, Salbu B, Smith J, Strand P, Tsukada H, 2016. Addressing ecological effects of radiation on populations and ecosystems in order to improve protection of the environment against radiation: agreed statements from a Consensus Symposium. Journal of Environmental Radioactivity 158–159, 21–29.

Bring A, Fedorova I, Dibike Y, Hinzman L, Mård J, Mernild S H, Prowse T, Semenova O, Stuefer S L, Woo M-K, 2016. Arctic terrestrial hydrology: a synthesis of processes, regional effects and research challenges. Journal of Geophysical Research: Biogeosciences 121. doi:10.1002/2015JG003131

Bruno J, Spahiu K, 2014. The long-term effect of hydrogen on the UO_2 spent fuel stability under anoxic conditions: findings from the Cigar Lake Natural Analogue study. Applied Geochemistry 49, 178–183.

Brydsten L, Strömgren M, 2010. A coupled regolith-lake development model applied to the Forsmark site. SKB TR-10-56, Svensk Kärnbränslehantering AB.

Brydsten L, Strömgren M, 2013. Landscape development in the Forsmark area from the past into the future (8500 BC – 40,000 AD). SKB R-13-27, Svensk Kärnbränslehantering AB.

Bödvarsson R, 2012. Swedish National Seismic Network (SNSN). A short report on recorded earthquakes during the fourth quarter of the year 2011. SKB P-12-09, Svensk Kärnbränslehantering AB.

Bödvarsson R, Lund B, Roberts R, Slunga R, 2006. Earthquake activity in Sweden. Study in connection with a proposed nuclear waste repository in Forsmark or Oskarshamn. SKB R-06-67, Svensk Kärnbränslehantering AB.

Börgesson L, Hernelind J, 2014a. Modelling of the mechanical interaction between the buffer and the backfill in a KBS-3V repository. Updated design of backfill and deposition hole. SKB R-14-21, Svensk Kärnbränslehantering AB.

Börgesson L, Hernelind J, 2014b. Modelling of bentonite block compaction. SKB P-14-10, Svensk Kärnbränslehantering AB.

Börgesson L, Sandén T, Dueck A, Andersson L, Jensen V, Nilsson U, Olsson S, Åkesson M, Kristensson O, Svensson U, 2015a. Consequences of water inflow and early water uptake in deposition holes. EVA project. SKB TR-14-22, Svensk Kärnbränslehantering AB.

Börgesson L, Åkesson M, Kristensson O, Malmberg D, Birgersson M, Hernelind J, 2015b. Modelling of critical H-M processes in the engineered barriers of SFR. SKB TR-14-27, Svensk Kärnbränslehantering AB.

Calderon M, 2013. Avvecklingsplan för ett utbyggt SFR. Slutförvaret för kortlivat radioaktivt avfall. SKB P-13-35, Svensk Kärnbränslehantering AB. (In Swedish.)

Calderon M, 2014a. Avvecklingsplan för Clink. SKBdoc 1404607 ver 1.0, Svensk Kärnbränslehantering AB. Available at: http://www.stralsakerhetsmyndigheten.se/Global/Slutf%c3%b6rvar/Komplettering%20150120/4Bilaga_C_Avvecklingsplan_f%c3%b6r_Clink.pdf (In Swedish.)

Calderon M, 2014b. Struktur på avvecklingsplan för kärntekniska anläggningar ”guideline”. SKBdoc 1416378 ver 1.0, Svensk Kärnbränslehantering AB. (In Swedish.)

Chan T, Christiansson R, Boulton G S, Ericsson L O, Hartikainen J, Jensen M R, Mas Ivars D, Stanchell F W, Vistrand P, Wallroth T, 2005. DECOVALEX III BMT3/BENCHPAR WP4: The thermo-hydro-mechanical responses to a glacial cycle and their potential implications for deep geological disposal of nuclear fuel waste in a fractured crystalline rock mass. International Journal of Rock Mechanics and Mining Sciences 42, 805–827.

Chen J, Qin Z, Shoesmith D W, 2012. Copper corrosion in aqueous sulphide solutions under nuclear waste repository conditions. In Scientific basis for nuclear waste management XXXV: symposium held in Buenos Aires, Argentina, 2–7 October 2011. Warrendale, PA: Materials Research Society. (Materials Research Society Symposium Proceedings 1475), 465–470.

Chen J, Qin Z, Shoesmith D W, 2014a. Key parameters determining structure and properties of sulphide films formed on copper corroding in anoxic sulphide solutions. Corrosion Engineering Science and Technology 49, 415–419.

Chen J, Qin Z, Wu L, Noel J J, Shoesmith D W, 2014b. The influence of sulphide transport on the growth and properties of copper sulphide films during copper corrosion in anaerobic sulphide solutions. Corrosion Science 87, 233–238.

Choi J-H, Edwards P, Ko K, Kim Y-S, 2016. Definition and classification of fault damage zones: a review and a new methodological approach. Earth-Science Reviews 152, 70–87.

Christiansson R, Janson T, 2003. A test of different stress measurement methods in two orthogonal bore holes in Äspö Hard Rock Laboratory (HRL), Sweden. International Journal of Rock Mechanics and Mining Sciences 40, 1161–1172.

Claesson Liljedahl L, Kontula A, Harper J, Näslund J-O, Selroos J-O, Pitkänen P, Puigdomenech I, Hobbs M, Follin S, Hirschorn S, Jansson P, Kennell L, Marcos N, Ruskeeniemi T, Tullborg E-L, Vidstrand P, 2016. The Greenland Analogue Project: Final report. SKB TR-14-13, Svensk Kärnbränslehantering AB.

Colleoni F, Wekerle C, Masina S, 2014. Long-term safety of a planned geological repository for spent nuclear fuel in Forsmark – estimate of maximum ice sheet thicknesses. SKB TR-14-21, Svensk Kärnbränslehantering AB.

Colleoni F, Wekerle C, Näslund J-O, Brandefelt J, Masina S, 2016. Constraint on the penultimate glacial maximum Northern Hemisphere ice topography (≈140 kyrs BP). Quaternary Science Reviews 137, 97–112.

Corkhill C L, Myllykylä E, Bailey D J, Thornber S M, Qi J, Maldonado P, Stennett M C, Hamilton A, Hyatt N C, 2014. Contribution of energetically reactive surface features to the dissolution of CeO_2 and ThO_2 analogues for spent nuclear fuel microstructures. ACS Applied Materials & Interfaces 6, 12279−12289.

Cronstrand P, 2007. Modelling the long-time stability of the engineered barriers of SFR with respect to climate changes. SKB R-07-51, Svensk Kärnbränslehantering AB.

Cronstrand P, 2016. Long-term performance of the bentonite barrier in the SFR silo. SKB TR-15-08, Svensk Kärnbränslehantering AB.

Curti E, Puranen A, Grolimund D, Jädernas D, Sheptyakov D, Mesbah A, 2015. Characterization of selenium in UO_2 spent nuclear fuel by micro X-ray absorption spectroscopy and its thermodynamic stability. Environmental Science: Processes & Impacts 17, 1760–1768.

Dahlberg C, Öberg M, Faleskog J, 2014. Continuum modeling of nodular cast iron using a porous plastic model with pressure-sensitive matrix – Experiments, model calibration & verification. SKBdoc 1414719 ver 1.0, Svensk Kärnbränslehantering AB.

Damjanac B, Fairhurst C, 2010. Evidence for a long-term strength threshold in crystalline rock. Rock Mechanics and Rock Engineering 43, 513–531.

Darcel C, Le Goc R, Davy P, 2015. Onkalo POSE experiment – Effective elastic properties of fractured rocks. Posiva Working Report 2015-17, Posiva Oy, Finland.

Davy P, Le Goc R, Darcel C, Bour O, de Dreuzy J-R, Munier R, 2010. A likely universal model of fracture scaling and its consequence for crustal hydromechanics. Journal of Geophysical Research 115. doi:10.1029/2009jb007043

Davy P, Le Goc R, Darcel C, 2013. A model of fracture nucleation, growth and arrest, and consequences for fracture density and scaling. Journal of Geophysical Research: Solid Earth 118, 1393–1407.

De la Gardie F, Calderon M, 2014. Utredning av extern segmentering av reaktortank vid nedmontering och rivning av kärnkraftverk. SKBdoc 1432771 ver 1.0, Svensk Kärnbränslehantering AB. (In Swedish.)

Dessirier B, 2016. Numerical modeling of groundwater and air flow between compacted bentonite and fractured crystalline rock. PhD thesis. Stockholm University.

Dillström P, 2014. Summary of important characteristic parameters for the BWR- and PWR-insert, based on performed strength and damage tolerance analyses. Report 50017500-1, rev 4, Inspecta Technology AB. SKBdoc 1288292 ver 1.0, Svensk Kärnbränslehantering AB.

Dillström P, Bolinder T, Bonnaud E, 2014. Analysis of PWR canister inserts using data acquired from PWR material. Report 50010510-1, rev 8, Inspecta Technology AB. SKBdoc 1288288 ver 2.0, Svensk Kärnbränslehantering AB.

Drake H, Hallbeck L, Pedersen K, Rosdahl A, Tullborg E-L, Wallin B, Sandberg B, Blomfeldt T, 2014. Investigation of sulphide production in core-drilled boreholes in Äspö Hard Rock Laboroatory. Boreholes KA3110A, KA3385A and KA3105A. SKB TR-13-12, Svensk Kärnbränslehantering AB.

Drake H, Tullborg E-L, Whitehouse M, Sandberg B, Blomfeldt T, Åström M E, 2015. Extreme fractionation and micro-scale variation of sulphur isotopes during bacterial sulphate reduction in deep groundwater systems. Geochimica et Cosmochimica Acta 161, 1–18.

Dueck A, Goudarzi R, Börgesson L, 2014. Buffer homogenisation, status report 2. SKB TR-14-25, Svensk Kärnbränslehantering AB.

Duro L, Grivé M, Gaona X, Bruno J, Andersson T, Borén H, Dario M, Allard B, Hagberg J, Källström K, 2012. Study of the effect of the fibre mass UP2 degradation products on radionuclide mobilisation. SKB R-12-15, Svensk Kärnbränslehantering AB.

Edelborg M, 2013. Jämförelse mellan alternativen hel respektive segmenterad reaktortank. SKBdoc 1335231 ver 1.0, Svensk Kärnbränslehantering AB. (In Swedish.)

Edelborg M, Anunti Å, Oliver L, Lundkvist N, Leveau N, 2014. Decommissioning study of Clink. SKB R-13-36, Svensk Kärnbränslehantering AB.

Elfwing M, Evins L Z, Gontier M, Grahm P, Mårtensson P, Tunbrant S, 2013. SFL concept study. Main report. SKB TR-13-14, Svensk Kärnbränslehantering AB.

Emborg M, Jonasson J-E, Knutsson M, 2007. Långtidsstabilitet till följd av frysning och tining av betong och bentonit vid förvaring av låg- och medelaktivt kärnavfall i SFR 1. SKB R-07-60, Svensk Kärnbränslehantering AB. (In Swedish.)

Engels S, Self A E, Luoto T P, Brooks S J, Helmens K F, 2014. A comparison of three Eurasian chironomid–climate calibration datasets on a W–E continentality gradient and the implications for quantitative temperature reconstructions. Journal of Paleolimnology 51, 529–547.

Ericsson L O, Thörn J, Christiansson R, Lehtimäki T, Ittner H, Hansson K, Butron C, Sigurdsson O, Kinnbom P, 2015. A demonstration project on controlling and verifying the excavation-damaged zone. Experience from the Äspö Hard Rock Laboratory SKB R-14-30, Svensk Kärnbränslehantering AB.

Eriksson D, Bultmark F, Andersson H, 2015. Gasevakuering genom betongbarriär i 2BMA. SKBdoc 1409731 ver 1.0, Svensk Kärnbränslehantering AB. (In Swedish.)

Eriksson P, 2014. Basic engineering of buffer production system. SKB P-14-11, Svensk Kärnbränslehantering AB.

Eriksson P, 2016. Investigation of alternatives to the buffer protection. SKB P-16-07, Svensk Kärnbränslehantering AB.

Esaki T, Du S, Mitani Y, Ikusada K, Jing L, 1999. Development of a shear-flow test apparatus and determination of coupled properties for a single rock joint. International Journal of Rock Mechanics and Mining Sciences 36, 641–650.

Evins L Z, Juhola P, Vähänen M, 2014. REDUPP. Final report. Posiva Working Report 2014-12, Posiva Oy, Finland.

Favalli A, Vo D, Grogan B, Jansson P, Liljenfeldt H, Mozin V, Schwalbach P, Sjöland A, Tobin S J, Trellue H, Vaccaro S, 2016. Determining initial enrichment, burnup, and cooling time of pressurized-water-reactor spent fuel assemblies by analyzing passive gamma spectra measured at the Clab interim-fuel storage facility in Sweden. Nuclear Instruments and Methods in Physics Research A 820, 102–111.

Fenton C H, Adams J, Halchuk S, 2006. Seismic hazards assessment for radioactive waste disposal sites in regions of low seismic activity. Geotechnical & Geological Engineering 24, 579–592.

Fors P, 2015. Nuclear decommissioning – A feasibility study of potential R&D initiatives for Energiforsk's nuclear programme. RAPPORT 2015:131, Energiforsk AB.

Freeze G, Brady P, Hardin E, MacKinnon R, Sevougian D, Stein E, Hadgu T, 2016. Safety considerations for deep borehole disposal of Cs/Sr capsules. Proceeding of Waste Management, Phoenix, Arizona, 6–10 March 2016. Paper 16294.

Funehag J, 2016. Efterinjekteringen av TASS-tunneln. Design, genomförande och resultat från efterinjektering. SKB R-11-09, Svensk Kärnbränslehantering AB. (In Swedish.)

Funehag J, Emmelin A, 2011. Injektering av TASS-tunneln. Design, genomförande och resultat från förinjekteringen. SKB R-10-39, Svensk Kärnbränslehantering AB. (In Swedish.)

Fälth B, 2014. Simulating earthquake rupture and off-fault fracture response. Lic. thesis. Uppsala University.

Fälth B, Hökmark H, 2006. Seismically induced slip on rock fractures. Results from dynamic discrete fracture modeling. SKB R-06-48, Svensk Kärnbränslehantering AB.

Fälth B, Hökmark H, 2011. Modelling end-glacial earthquakes at Olkiluoto. Posiva Working Report 2011-13, Posiva Oy, Finland.

Fälth B, Hökmark H, 2012. Modelling end-glacial earthquakes at Olkiluoto. Expansion of the 2010 study. Posiva Working Report 2012-08, Posiva Oy, Finland.

Fälth B, Hökmark H, 2015. Effects of hypothetical large earthquakes on repository host rock fractures. Posiva Working Report 2015-18, Posiva Oy, Finland.

Fälth B, Hökmark H, Munier R, 2007. Seismically induced shear displacements in repository host rock fractures. Proceedings of 9th Canadian conference on Earthquake Engineering, Ottawa, Canada, 26–29 June 2007.

Fälth B, Hökmark H, Munier R, 2008. Seismically induced slip on rock fractures – expanded study with particular account of large earthquakes. Proceedings of 42nd U.S. Rock Mechanics Symposium, San Fransisco, 29 June – 2 July 2008.

Fälth B, Hökmark H, Munier R, 2010. Effects of large earthquakes on a KBS-3 repository. Evaluation of modelling results and their implications for layout and design. SKB TR-08-11, Svensk Kärnbränslehantering AB.

Fälth B, Hökmark H, Lund B, Mai P M, Roberts R, Munier R, 2015. Simulating earthquake rupture and off-fault fracture response: application to the safety assessment of the Swedish nuclear waste repository. Bulletin of the Seismological Society of America 105, 134–151.

Ganchenkova M G, Yagodzinskyy Y N, Borodin V A, Hänninen H, 2014. Effects of hydrogen and impurities on void nucleation in copper: simulation point of view. Philosophical Magazine 94, 3522–3548.

Ganopolski A, Winkelmann R, Schellnhuber H J, 2015. Critical insolation–CO_2 relation for diagnosing past and future glacial inception. Nature 529, 200–203.

Gaucher E, Tournassat C, Nowak C, 2005. Modelling the geochemical evolution of the multi-barrier system of the Silo of the SFR repository. Final report. SKB R-05-80, Svensk Kärnbränslehantering AB.

Gibb F, 2016. Deep borehole disposal: options, issues and challenges. Proceeding of Waste Management, Phoenix, Arizona, 6–10 March 2016. Paper 16250.

Glamheden R, Fredriksson A, Röshoff K, Karlsson J, Hakami H, Christiansson R, 2007. Rock mechanics Forsmark. Site descriptive modelling, Forsmark stage 2.2. SKB R-07-31, Svensk Kärnbränslehantering AB.

Glaus M A, Van Loon L R, 2008. Degradation of cellulose under alkaline conditions: new insights from a 12 year degradation study. Environmental Science & Technology 42, 2906–2911.

Glaus M A, Van Loon L R, Achatz S, Chodura A, Fischer K, 1999. Degradation of cellulosic materials under the alkaline conditions of a cementitious repository for low and intermediate level radioactive waste. Part I: Identification of degradation products. Analytica Chimica Acta 398, 111–122.

Grahm P, Malm R, Eriksson D, 2015. System design and full scale testing of the Dome Plug for KBS-3V deposition tunnels. Main report. SKB TR-14-23, Svensk Kärnbränslehantering AB.

Grambow B, Duro L, Spahiu K, Kulik D, Brandt F, 2014. Slow processes in close-to-equilibrium conditions for radionuclides in water/solid systems of relevance to nuclear waste management. SKIN Final synthesis report. Deliverable 5.5. European Commission. Available at: http://web.emn.fr/x-subatech/skin/images/7/7b/FinalSynthesis.pdf

Grolander S, 2013. Biosphere parameters used in radionuclide transport modelling and dose calculations in SR-PSU. SKB R-13-18, Svensk Kärnbränslehantering AB.

Grundfelt B, 2013. Radiological consequences of accidents during disposal of spent nuclear fuel in a deep borehole. SKB P-13-13, Svensk Kärnbränslehantering AB.

Grundfelt B, Birgersson L, Setzman E, 2016. Views on long-term safety of deep borehole disposal. Proceeding of Waste Management, Phoenix, Arizona, 6–10 March 2016. Paper 16657.

Guo X, Szenknect S, Mesbah A, Labs S, Clavier N, Poinssot C, Ushakov S V, Curtius H, Bosbach D, Ewing R C, Burns P C, Dacheux N, Navrotsky A, 2015. Thermodynamics of formation of coffinite, $USiO_4$. Proceedings of the National Academy of Sciences of the United States of America, 112, 6551−6555.

Gärdenäs A, Eckersten H, Reinlert A, Gustafsson D, Jansson P-E, Ekström P-A, Avila R, Greger M, 2009. Tracey – a simulation model of trace element fluxes in soil–plant system for long-term assessment of a radioactive groundwater contamination. SKB TR-09-24, Svensk Kärnbränslehantering AB.

Haglind J, Egeltun P, 2014. Utredning av hantering av reaktortank – Tids- och kostnadsuppskattning. SKBdoc 1410596 ver 2.0, Svensk Kärnbränslehantering AB. (In Swedish.)

Hakala M, Siren T, Kemppainen K, Christiansson R, Martin D, 2013. In-situ stress measurements with the new LVDT-cell – method description and verification. Posiva 2012-43, Posiva Oy, Finland.

Hakami E, 2011. Rock stress orientation measurements using induced thermal spalling in slim boreholes. SKB R-11-12, Svensk Kärnbränslehantering AB.

Hakami E, Min K-B, 2009. Modelling of the state of stress. Preliminary site description Laxemar subarea – version 1.2. SKB R-06-17, Svensk Kärnbränslehantering AB.

Hakami H, 2006. Numerical studies on spatial variation of the in situ stress field at Forsmark – a further step. Site descriptive modelling Forsmark – stage 2.1. SKB R-06-124, Svensk Kärnbränslehantering AB.

Hallberg B, Tiberg L, 2010. Preliminär plan för avveckling – slutförvar för använt kärnbränsle. SKB P-10-30, Svensk Kärnbränslehantering AB. (In Swedish.)

Halldin Stenlid J, Johansson A J, Brinck T, 2014. Searching for the thermodynamic limit – a DFT study of the step-wise water oxidation of the bipyramidal Cu_7 cluster. Physical Chemistry Chemical Physics 16, 2452–2464.

Halldin Stenlid J, Johansson A J, Kloo L, Brinck T, 2016. Aqueous solvation and surface oxidation of the Cu_7 nanoparticle: insights from theoretical modeling. The Journal of Physical Chemistry C 120, 1977–1988.

Hansson T, Norberg T, Knutsson A, Fors P, Sandebert C, 2013. Ringhals Site Study 2013 – An assessment of the decommissioning cost for the Ringhals site. SKB R-13-05, Svensk Kärnbränslehantering AB.

Hardacre C J, Heal M R, 2013. Characterization of methyl bromide and methyl chloride fluxes at temperate freshwater wetlands. Journal of Geophysical Research: Atmospheres 118, 977–991.

Hardin E, Jenni K, Clark A, Cochran J, Finger J, Sevougian S D, 2016. Deep borehole disposal waste emplacement mode cost-risk study. Proceeding of Waste Management, Phoenix, Arizona, 6–10 March 2016. Paper 16346.

Harper J, Hubbard A, Ruskeeniemi T, Claesson Liljedahl L, Kontula A, Bougamont M, Brown J, Dirkson A, Dow C, Doyle S, Drake H, Engström J, Fitzpatrick A, Follin S, Frape S, Graly J, Hansson K, Harrington J, Henkemans E, Hirschorn S, Hobbs M, Humphrey N, Jansson P, Johnson J, Jones G, Kinnbom P, Kennell L, Klint K E S, Liimatainen J, Lindbäck K, Meierbachtol T, Pere T, Pettersson R, Tullborg E-L, van As D, 2016. The Greenland Analogue Project: Data and processes. SKB R-14-13, Svensk Kärnbränslehantering AB.

Hartikainen J, Kouhia R, Wallroth T, 2010. Permafrost simulations at Forsmark using a numerical 2D thermo-hydro-chemical model. SKB TR-09-17, Svensk Kärnbränslehantering AB.

Hedin A, Lilja C, Johansson J, Puigdomenech I, 2015. Samlad redovisning om kopparkorrosion i syrgasfritt vatten. SKBdoc 1473304 ver 1.0, Svensk Kärnbränslehantering AB. (In Swedish.)

Helmens K F, 2014. The Last Interglacial-Glacial cycle (MIS 5–2) re-examined based on long proxy records from central and northern Europe. Quaternary Science Reviews 86, 115–143.

Helmens K F, Salonen J S, Plikk A, Engels S, Väliranta M, Kylander M, Brendryen J, Renssen H, 2015. Major cooling intersecting peak Eemian Interglacial warmth in northern Europe. Quaternary Science Reviews 122, 293–299.

Hernelind J, 2014. Analysis of canister with unfavourable pressure load. SKBdoc 1419643 ver 1.0, Svensk Kärnbränslehantering AB.

Hernelind J, 2015. Analysis of creep in the KBS-3 copper canister due to internal and external loads. SKBdoc 1399768 ver 2.0, Svensk Kärnbränslehantering AB.

Herschend B, 2014. Long-lived intermediate level waste from Swedish nuclear power plants. Reference inventory. SKB R-13-17, Svensk Kärnbränslehantering AB.

Hoch A R, Lever D A, Shaw G, 2014. Uptake of carbon-14 in the biosphere: summary report. AMEC/004041/008 Issue 2, AMEC, UK.

Holtorf C, 2013. A comment on knowledge recovery: 'How we today can make things easier for historians/archaeologists of the future. In Gordon-Smith H, Schröder J, Pescatore C (eds). The preservation of records, knowledge and memory (RK&M) across generations: improving our understanding: RK&M workshop proceedings, 12–13 September 2012, Issy-les-Moulineaux, France, Paris: Nuclear Energy Agency, OECD, 48–50. Available at: http://www.oecd-nea.org/rwm/reports/2013/rwm-r2013-3.pdf

Holtorf C, Högberg A, 2013. Heritage futures and the future of heritage. In Bergerbrant S, Sabatini S (eds), Counterpoint: essays in archaeology and heritage studies in honour of professor Kristian Kristiansen. Oxford: Archaeopress, 739–746.

Holtorf C, Högberg A, 2014. Communicating with future generations: what are the benefits of preserving for future generations? Nuclear power and beyond. European Journal of Post-Classical Archaeologies 4, 315–330.

Holtorf C, Högberg A, 2015a. Contemporary heritage and the future. In Waterton E, Watson S (eds). The Palgrave handbook of contemporary heritage research. New York: Palgrave Macmillan, 509–523.

Holtorf C, Högberg A, 2015b. Archaeology and the future. Managing nuclear waste as living heritage. Radioactive waste management and constructing memory for future generations. Proceedings of the International Conference and Debate 1517 September 2014, Verdun, France. Paris: OECD Nuclear Energy Agency (NEA), 97–101.

Hora S, Jensen M, 2005. Expert panel elicitation of seismicity following glaciation in Sweden. SSI Rapport 2005:20, Statens strålskyddsinstitut (Swedish Radiation Protection Authority).

Hosseinzadeh Delandar A, 2015. Numerical modeling of plasticity in FCC crystalline materials using discrete dislocation dynamics. Lic. thesis. Royal Institute of Technology, Stockholm, Sweden.

Huybrechts P, 2010. Vulnerability of an underground radioactive waste repository in northern Belgium to glaciotectonic and glaciofluvial activity during the next 1 million year. Report 10/01, Departement Geografie VUB.

Hyman J D, Karra S, Makedonska N, Gable G W, Painter S L, Viswanathan H S, 2015. DFNWORKS: A discrete fracture network framework for modelling subsurface flow and transport. Computers & Geosciences 84, 10–19.

Höglund L O, 2001. Project SAFE. Modelling the long-term concrete degradation processes in the Swedish SFR repository, Svensk Kärnbränslehantering AB. SKB R-01-08, Svensk Kärnbränslehantering AB.

Höglund L O, 2014. The impact of concrete degradation on the BMA barrier functions. SKB R-13-40, Svensk Kärnbränslehantering AB.

Höglund L-O, Bengtsson A, 1991. Some chemical and physical processes related to the long-term performance of the SFR repository. SKB SFR 91-06, Svensk Kärnbränslehantering AB.

Hökmark H, Lönnqvist M, Fälth B, 2010. THM-issues in repository rock. Thermal, mechanical, thermo-mechanical and hydro-mechanical evolution of the rock at the Forsmark and Laxemar sites. SKB TR-10-23, Svensk Kärnbränslehantering AB.

Ibrahim B, 2015. The corrosion behaviour of Cu in irradiated and non-irradiated humid air. Master thesis. The University of Western Ontario, Canada.

Ibrahim B, Zagidulin D, Smith J M, Ramamurthy S, Wren J C, Shoesmith D W, 2015. Radiolytic corrosion of Cu nuclear waste containers. In Proceedings of the 17th Interntional Conference on Environmental Degradation in Nuclear Poser Systems – Water Reactors, Ottawa, 9–13 August 2015.

IPCC, 2013. Climate change 2013: The physical science basis: summary for policymakers. Contribution of Working Group I to the Fifth Assessment Report of the Intergovernmental Panel on Climate Change. Available at: http://www.ipcc.ch

Itasca, 2013. 3DEC – 3 Dimensional Distinct Element Code, User's manual, Version 5.0. Minneapolis, MN: Itasca Consulting Group, Inc.

Itasca, 2014a. UDEC – Universal Distinct Element Code, Version 6.0. Minneapolis, MN: Itasca Consulting Group, Inc.

Itasca, 2014b. PFCSuite – Particle Flow Code, Version 5.0. Minneapolis, MN: Itasca Consulting Group, Inc.

Jaeschke B, Smith K, Nordén S, Alfonso B, 2013. Assessment of risk to non-human biota from a repository for the disposal of spent nuclear fuel at Forsmark. Supplementary information. SKB TR-13-23, Svensk Kärnbränslehantering AB.

Jaremalm M, Köhler S, Lidman F, 2013. Precipitation of barite in the biosphere and its consequences for the mobility of Ra in Forsmark and Simpevarp. SKB TR-13-28, Svensk Kärnbränslehantering AB.

Johannesson E, Rannemalm T, 2016a. Oskarshamn 1 och 2 inklusive 0AVF – Avvecklingsplan. 2016-03988 utgåva 1, OKG AB. (In Swedish.)

Johannesson E, Rannemalm T, 2016b. Oskarshamn 3 – Avvecklingsplan. 2016-09146 utgåva 2, OKG AB. (In Swedish.)

Johannesson E, Rannemalm T, 2016c. Block 0 – Avvecklingsplan. 2016-09147 utgåva 1, OKG AB. (In Swedish.)

Johannesson L-E, 2014. KBS-3H. Manufacturing of buffer and filling components for the Multi Purpose Test. SKB P-14-07, Svensk Kärnbränslehantering AB.

Johannesson L-E, Kristensson O, Åkesson M, Eriksson P, Hedin M, 2014. Tests and simulations of THM processes relevant for the buffer installation. SKB P-14-22, Svensk Kärnbränslehantering AB

Johannesson L-E, Dueck A, Andersson L, Jensen V, 2015. Investigations of hydraulic and mechanical processes of the barriers embedding the silo in SFR. Laboratory tests. SKB TR-15-05, Svensk Kärnbränslehantering AB.

Johansson C, 2015. Samlat ställningstagande rörande återtag av avfallstyp S.14 samt kompensatoriska åtgärder. SKBdoc 1489807 ver 2.0, Svensk Kärnbränslehantering AB. (In Swedish.)

Johansson E, 2016. The influence of climate and permafrost on catchment hydrology. PhD thesis. Stockholm University.

Johansson E, Berglund S, Lindborg T, Petrone J, van As D, Gustafsson L-G, Näslund J-O, Laudon H, 2015a. Hydrological and meteorological investigations in a periglacial lake catchment near Kangerlussuaq, west Greenland – presentation of a new multi-parameter data set. Earth System Science Data 7, 93–108.

Johansson E, Gustafsson L-G, Berglund S, Lindborg T, Selroos J-O, Claesson Liljedahl L, Destouni G, 2015b. Data evaluation and numerical modeling of hydrological interactions between active layer, lake and talik in a permafrost catchment, Western Greenland. Journal of Hydrology 527, 688–703.

Johansson E, Stenberg L, Olofsson I, Karlzén R, 2015c. Utbyggnaden av Äspölaboratoriet 2011–2012. Karakterisering, projektering och tunneldrivning. SKB R-13-28, Svensk Kärnbränslehantering AB. (In Swedish.)

Jonsson M, Rydén H, 2014. Kravbild för oförstörande provning av kopparkapselns rör, lock och botten. SKBdoc 1414374 ver 1.0, Svensk Kärnbränslehantering AB. (In Swedish.)

Joyce S, Applegate D, Appleyard P, Gordon A, Heath T, Hunter F, Hoek J, Jackson P, Swan D, Woollard H, 2015. Groundwater flow and reactive transport modelling in ConnectFlow. SKB R-14-19, Svensk Kärnbränslehantering AB.

Juhlin C, Lund B, 2011. Reflection seismic studies over the end-glacial Burträsk fault, Skellefteå, Sweden. Solid Earth 2, 9–16.

Kalinowski M, 2015. Betongcylinder I-1B med ingjutna metallprover. CBI Uppdragsrapport, CBI Betonginstitutet. SKBdoc 1495415 ver 1.0, Svensk Kärnbränslehantering AB. (In Swedish.)

Kautsky U, Lindborg T, Valentin J (eds), 2013. Humans and ecosystems over the coming millennia: overview of a biosphere assessment of radioactive waste disposal in Sweden. Ambio 42, 381–526.

Kautsky U, Saetre P, Berglund S, Jaeschke B, Nordén S, Brandefelt J, Keesmann S, Näslund J-O, Andersson E, 2016. The impact of low and intermediate-level radioactive waste on humans and the environment over the next one hundred thousand years. Journal of Environmental Radioactivity 151, 395–403.

Keith-Roach M, Lindgren M, Källström K, 2014. Assessment of complexing agent concentrations in SFR. SKB R-14-03, Svensk Kärnbränslehantering AB.

Kienzler B, Lemmens K, 2015. First-nuclides: results on radionuclide release from used fuel. In Proceedings of the 15th International High-Level Radioactive Waste Management Conference (IHLRWM 2015), Charleston, South Carolina, 12–16 April 2015, 646–652.

Kim J, Dong H, Seabaugh J, Newell S W, Eberl D D, 2004. Role of microbes in the smectite to illite reaction. Science 303, 830–832.

King F, 2013. A review of the properties of pyrite and the implications for corrosion of the copper canister. SKB TR-13-19, Svensk Kärnbränslehantering AB.

King F, Lilja C, 2013. Localised corrosion of copper canisters in bentonite pore water. SKB TR-13-27, Svensk Kärnbränslehantering AB.

King F, Lilja C, 2014. Localised corrosion of copper canisters. Corrosion Engineering, Science and Technology 49, 420–424.

King F, Chen J, Martino T, Partovi-Nia R, Qin Z, Shoesmith D W, Lilja C, 2014. Sulphide mass-transport limited corrosion. SKBdoc 1422182 ver 2.0, Svensk Kärnbränslehantering AB.

Klinkenberg M, Brandt F, Breuer U, Bosbach D, 2014. Uptake of Ra during the recrystallization of barite: a microscopic and time of flight-secondary ion mass spectrometry study. Environmental Science & Technology 48, 6620−6627.

Konovalenko L, 2012. Element transport in marine coastal ecosystems – modelling general and element-specific mechanisms. Lic. thesis. Stockholm University.

Konovalenko L, 2014. Element transport in aquatic ecosystems: modelling general and element-specific mechanisms. PhD thesis. Stockholm University.

Konovalenko L, Bradshaw C, Kumblad L, Kautsky U, 2014. Radionuclide transfer in marine coastal ecosystems, a modelling study using metabolic processes and site data. Journal of Environmental Radioactivity 133, 48–59.

Konovalenko L, Bradshaw C, Andersson E, Lindquist D, Kautsky U, 2016. Evaluation of factors influencing accumulation of stable Sr and Cs in lake and coastal fish. Journal of Environmental Radioactivity 160, 64–79.

Kosec T, Kranjc A, Rosborg B, Legat A, 2015. Post examination of copper ER sensors exposed to bentonite. Journal of Nuclear Materials 459, 306–312.

Kristiansen P T, Massel F, Werme L, Lilja C, Duda L-C, 2015. Sulfidation of single-phase oxide on copper and as powder studied using soft X-ray spectroscopy. Journal of The Electrochemical Society 162, C785–C791.

Kumblad L, Kautsky U, 2004. Models for transport and fate of carbon, nutrients and point source released radionuclides to an aquatic ecosystem. SKB TR-04-13, Svensk Kärnbränslehantering AB.

Källbom R, Jonsson M, Rydén H, 2014. Kravbild för oförstörande provning av segjärnsinsats. SKBdoc 1414760 ver 1.0, Svensk Kärnbränslehantering AB. (In Swedish.)

Köhler S J, Lidman F, Laudon H, 2014. Landscape types and pH control organic matter mediated mobilization of Al, Fe, U and La in boreal catchments. Geochimica et Cosmochimica Acta 135, 190–202.

Lagerblad B, Trägårdh J, 1994. Conceptual model for concrete long time degradation in deep nuclear waste repository. SKB TR 95-21, Svensk Kärnbränslehantering AB.

Lagerblad B, Golubeva M, Cirera J R, 2015. Lämplighet för krossberg från Forsmark och SFR att användas som betongballast. SKB P-16-13, Svensk Kärnbränslehantering AB. (In Swedish.)

Lagerbäck R, Sundh M, 2008. Early Holocene faulting and paleoseismicity in northern Sweden. Research Paper C 836, Geological Survey of Sweden.

Lagerbäck R, Sundh M, Svedlund J-O, Johansson H, 2005. Forsmark site investigation. Searching for evidence of late- or postglacial faulting in the Forsmark region. Results from 2002–2004. SKB R-05-51, Svensk Kärnbränslehantering AB.

Lagerlund J, 2014. Utveckling av kringgjutningsbruk för 1BMA – Injekteringsförsök i betongspalt. Rapport U14:71, Vattenfall Research and Development AB. SKBdoc 1441135 ver 1.0, Svensk Kärnbränslehantering AB. (In Swedish.)

Lagerlund J, 2015a. Injektering av sprickor med kringgjutningsbruk för 1BMA. Rapport VRD-R03:2051, Vattenfall Research and Development AB. SKBdoc 1479371 ver 1.0, Svensk Kärnbränslehantering AB. (In Swedish.)

Lagerlund J, 2015b. Utveckling av kringgjutningsbruk för 2BMA. Rapport VRD-R01:2051, Vattenfall Research and Development AB. SKBdoc 1471727 ver 1.0, Svensk Kärnbränslehantering AB. (In Swedish.)

Lagerlund J, Thorsell P-E, Liljeby E, 2014. Utveckling av kringgjutningsbruk för 1BMA. Rapport U14:02, Vattenfall Research and Development AB. SKBdoc 1425185 ver 1.0, Svensk Kärnbränslehantering AB. (In Swedish.)

Lantmäteriet, 2015. Produktbeskrivning: GSD-Höjddata, grid 2+. Version 2.2. Gävle: Lantmäteriet. Available at: https://www.lantmateriet.se/globalassets/kartor-och-geografisk-information/hojddata/produktbeskrivningar/hojd2_plus.pdf (In Swedish.)

Larsson H, Anunti Å, Edelborg M, 2013. Decommissioning study of Oskarshamn NPP. SKB R-13-04, Svensk Kärnbränslehantering AB.

Lee J H, Arnold B W, Swift P N, Hadgu T, Freeze G, Wang Y, 2012. A prototype performance assessment model for generic deep borehole repository for high-level nuclear waste. In Proceedings of WM2012 Conference, Phoenix, Arizona, 26 February – 1 March 2012. Available at: http://www.wmsym.org/archives/2012/papers/12132.pdf

L'Hôpital E, Lothenbach B, Le Saout G, Kulik D, Scrivener K, 2015. Incorporation of aluminium in calcium-silicate-hydrates. Cement and Concrete Research 75, 91–103.

Li Y, 2015. *Ab initio* based modeling of defects and disorder in industrial materials. PhD thesis. Royal Institute of Technology, Stockholm, Sweden.

Li Y, Korzhavyi P A, 2015. Interactions of point defects with stacking faults in oxygen-free phosphorus-containing copper, Journal of Nuclear Materials 462, 160–164.

Li Y, Lousada C M, Soroka I L, Korzhavyi P A, 2015. Bond network topology and antiferroelectric order in cuprice CuOH. Inorganic Chemistry 54, 8969–8977.

Lidman F, Mörth C-M, Björkvald L, Laudon H, 2011. Selenium dynamics in boreal streams: the role of wetlands and changing groundwater tables. Environmental Science & Technology 45, 2677–2683.

Lidman F, Mörth C-M, Laudon H, 2012. Landscape control of uranium and thorium in boreal streams – spatiotemporal variability and the role of wetlands. Biogeosciences 9, 4773–4785.

Lidman F, Ramebäck H, Bengtsson Å, Laudon H, 2013. Distribution and transport of radionuclides in a boreal mire – assessing past, present and future accumulation of uranium, thorium and radium. Journal of Environmental Radioactivity 121, 87–97.

Lidman F, Köhler S J, Mörth C-M, Laudon H, 2014. Metal transport in the boreal landscape – the role of wetlands and the affinity for organic matter. Environmental Science & Technology 48, 3783–3790.

Lidman F, Peralta-Tapia A, Vesterlund A, Laudon H, 2016. $^{234}U/^{238}U$ in a boreal stream network – Relationship to hydrological events, groundwater and scale. Chemical Geology 420, 240–250.

Lidmar-Bergström K, Bonow J M, Japsen P, 2013. Stratigraphic Landscape Analysis and geomorphological paradigms: Scandinavia as an example of Phanerozoic uplift and subsidence. Global and Planetary Change 100, 153–171.

Lindblom E, Lund B, Tryggvason A, Uski M, Bödvarsson R, Juhlin C, Roberts R, 2015. Microearthquakes illuminate the deep structure of the endglacial Pärvie fault, northern Sweden. Geophysical Journal International 201, 1704–1716.

Lindborg T, Brydsten L, Sohlenius G, Strömgren M, Andersson E, Löfgren A, 2013. Landscape development during a glacial cycle: modeling ecosystems from the past into the future. Ambio 42, 402–413.

Lindborg T, Rydberg J, Tröjbom M, Berglund S, Johansson E, Löfgren A, Saetre P, Nordén S, Sohlenius G, Andersson E, Petrone J, Borgiel M, Kautsky U, Laudon H, 2016. Biogeochemical data from terrestrial and aquatic ecosystems in a periglacial catchment, West Greenland. Earth System Science Data Discussions. doi:10.1594/PANGAEA.860961

Lousada C M, Trummer M, Jonsson M, 2013a. Reactivity of H_2O_2 towards different UO_2-based materials: the relative impact of radiolysis products revisited. Journal of Nuclear Materials 434, 434–439.

Lousada C M, Yang M, Nilsson K, Jonsson M, 2013b. Catalytic decomposition of hydrogen peroxide on transition metaland lanthanide oxides. Journal of Molecular Catalysis A: Chemical 379, 178–184.

Lousada C M, Johansson A J, Korzhavyi P A, 2015. Thermodynamics of H_2O splitting and H_2 formation at the Cu(110)–water interface. The Journal of Physical Chemistry C 119, 14102–14113.

Lousada C M, Soroka I L, Yagodzinskyy Y, Tarakina N V, Todoshchenko O, Hänninen H, Korzhavyi P A, Jonsson M, 2016. Gamma radiation induces hydrogen absorption by copper in water. Nature Scientific Reports 6. doi:10.1038/srep2434

Lund B, Schmidt P, Hieronymus C, 2009. Stress evolution and fault stability during the Weichselian glacial cycle. SKB TR-09-15, Svensk Kärnbränslehantering AB.

Lund B, Buhcheva D, Tryggvason A, Berglund K, Juhlin C, Munier R, 2015. The Burträsk end-glacial fault: Sweden's most seismically active fault system. Proceedings of EGU General Assembly, Vienna, 12–17 April 2015.

Lundgren K, 2005. Jod-129: Uppskattning av aktivitet i driftavfall från svenska LWR. SKB R-05-79, Svensk Kärnbränslehantering AB. (In Swedish.)

Lundgren K, 2006. Mo-93, Tc-99 och Cs-135: Uppskattning av aktivitet i driftavfall från svenska LWR. 06-0031R, ALARA Engineering. SKBdoc 1393386 ver 1.0, Svensk Kärnbränslehantering AB. (In Swedish.)

Luterkort D, Gylling B, Johansson R, 2012. Closure of the Spent Fuel Repository in Forsmark Studies of alternative concepts for sealing of ramp, shafts and investigation boreholes. SKB TR-12-08, Svensk Kärnbränslehantering AB.

Luterkort D, Nyblad B, Wimeliius H, Pettersson A, Aghili B, 2014. SFR förslutningsplan. SKBdoc 1358612 ver 1.0, Svensk Kärnbränslehantering AB. (In Swedish.)

Löfgren A (ed), 2010. The terrestrial ecosystems at Forsmark and Laxemar-Simpevarp. SR-Site Biosphere. SKB TR-10-01, Svensk Kärnbränslehantering AB.

Lönnqvist M, Hökmark H, 2013. Approach to estimating the maximum depth for glacially induced hydraulic jacking in fractured crystalline rock at Forsmark, Sweden. Journal of Geophysical Research: Earth Surface 118, 1777–1791.

Lönnqvist M, Hökmark H, 2015. Assessment of method to model slip of isolated, non-planar fractures using 3DEC. In Proceedings of 13th International Symposium on Rock Mechanics (ISRM Congress 2015), Montreal, 10–13 May 2015. Canadian Institute of Mining, Metallurgy & Petroleum and ISRM.

Macdonald D D, Sharifi-Asl S, Engelhardt G R, 2014. Issues in the corrosion of copper in a Swedish high level nuclear waste repository. Phase III. Role of sulphide ion in anodic and cathodic processes – research report. Technical Note 201:57, Strålsäkerhetsmyndigheten (Swedish Radiation Safety Authority).

Mahmoudzadeh B, 2016. Solute transport through fractured rocks: the influence of geological heterogeneities and stagnant water zones. PhD thesis. Royal Institute of Technology, Stockholm, Sweden.

Maillot J, 2016. Comportement mécanique et hydraulique des réseaux de fractures complexes (multiéchelles) à partir de simulations numériques et de modèles théoriques. PhD thesis. Université de Rennes 1.

Maldonado P, Godinho J R A, Evins L Z, Oppeneer P M, 2013. Ab initio prediction of surface stability of fluorite materials and experimental verification. The Journal of Physical Chemisty C 117, 6639−6650.

Maldonado P, Evins L Z, Oppeneer P M, 2014. Ab initio atomistic thermodynamics of water reacting with uranium dioxide surfaces. The Journal of Physical Chemisty C 118, 8491−8500.

Mao F, Dong C, Sharifi-Asl S, Lu P, Macdonald D D, 2014. Passivity breakdown on copper: influence of chloride ion. Electrochimica Acta 144, 391–399.

Marsic N, Grundfelt B, 2013. Modelling of thermally driven groundwater flow in a facility for disposal of spent nuclear fuel in deep boreholes. SKB P-13-10, Svensk Kärnbränslehantering AB.

Martino T, Partovi-Nia R, Chen J, Qin Z, Shoesmith D W, 2014. Mechanisms of film growth on copper in aqueous solutions containing sulphide and chloride under voltammetric conditions. Electrochimica Acta 127, 439–447.

Mas Ivars D, Hakami H, 2005. Effect of a sub-horizontal fracture zone and rock mass heterogeneity on the stress field in Forsmark area – a numerical study using 3DEC. Preliminary site description Forsmark area – version 1.2. SKB R-05-59, Svensk Kärnbränslehantering AB.

Mas Ivars D, Pierce M E, Darcel C, Reyes-Montes J, Potyondy D O, Young R P, Cundall P A, 2011. The synthetic rock mass approach for jointed rock mass modelling. International Journal of Rock Mechanics and Mining Sciences 48, 219–244.

McEwen T (ed), Aro S, Kosunen P, Mattila J, Pere T, Käpyaho A, Hellä P, 2012. Rock suitability classification – RSC 2012. Posiva 2012-24, Posiva Oy, Finland.

Mikko H, Smith C A, Lund B, Ask M V S, Munier R, 2015. LiDAR-derived inventory of post-glacial fault scarps in Sweden. GFF 137, 334–338.

Min K-B, Stephansson O, Jing L, 2005. Effect of stress on mechanical and hydraulic rock mass properties – application of DFN-DEM approach on data from site investigation at Forsmark, Sweden. In Proceedings of ISRM International Symposium – EUROCK 2005, Brno, 18–20 May 2005.

Min K-B, Lee J, Stephansson O, 2013. Implications of thermally-induced fracture slip and permeability change on the long-term performance of a deep geological repository. International Journal of Rock Mechanics and Mining Sciences 61, 175–288.

Min K-B, Lee J, Stephansson O, 2015. Rock mechanics – Evolution of fracture transmissivity within different scenarios in SR-site. Main review phase. Technical Note 2013:37, Strålsäkerhetsmyndigheten (Swedish Radiation Safety Authority).

Molinero J, Trinchero P, Ebrahimi H, de Vries L, Luna M, Svensson U, Lichtner P, 2016. Development, testing and application of a high performance computing framework for reactive transport modelling in crystalline rocks (iDP). SKB R-15-17, Svensk Kärnbränslehantering AB.

Moreno L, Skagius K, Södergren S, Wiborgh M, 2001. Project SAFE. Gas related processes in SFR. SKB R-01-11, Svensk Kärnbränslehantering AB.

Munier R, 2006. Using observations in deposition tunnels to avoid intersections with critical fractures in deposition holes. SKB R-06-54, Svensk Kärnbränslehantering AB.

Munier R, Mattila J, 2015. MEMO on Terminology. POS-021685, ver 1.0 Posiva Oy, Finland.

Muschitiello F, Wohlfarth B, 2015. Time-transgressive environmental shifts across Northern Europe at the onset of the Younger Dryas. Quaternary Science Reviews 109, 49–56.

Muschitiello F, Andersson A, Wohlfarth B, Smittenberg R H, 2015a. The C_{20} highly branched isoprenoid biomarker – A new diatom-sourced proxy for summer trophic conditions? Organic Chemistry 81, 27–33.

Muschitiello F, Pausata F S R, Watson J E, Smittenberg R H, Salih A A M, Brooks S J, Whitehouse N J, Karlatou-Charalampopoulou A, Wohlfarth B, 2015b. Fennoscandian freshwater control on Greenland hydroclimate shifts at the onset of the Younger Dryas. Nature Communications 6. doi:10.1038/ncomms9939

Muschitiello F, Lea J M, Greenwood S L, Nick F M, Brunnberg L, Macleod A, Wohlfarth B, 2016. Timing of the first drainage of the Baltic Ice Lake synchronous with the onset of Greenland Stadial 1. Boreas 45, 322–334.

Myllykylä E, Lavonen T, Stennett M, Corkhill C, Ollila K, Hyatt N, 2015. Solution composition and particle size effects on the dissolution and solubility of a ThO_2 microstructural analogue for UO_2 matrix of nuclear fuel. Radiochimica Acta 103, 565–576.

Mårtensson P, 2015. Äspö Hard Rock Laboratory. Concrete and Clay. Installation report. SKB P-15-01, Svensk Kärnbränslehantering AB.

Mörner N-A, Tröften P E, Sjöberg R, Grant D, Dawson S, Bronge C, Kvamsdal O, Sidén A, 2000. Deglacial paleoseismicity in Sweden: the 9663 BP Iggesund event. Quaternary Science Reviews 19, 1461–1468.

Nardi A, Idiart A, Trinchero P, de Vries L M, Molinero J, 2014. Interface COMSOL-PHREEQC (iCP), an efficient numerical framework for the solution of coupled multiphysics and geochemistry. Computers & Geosciences 69, 10–21.

Natchimuthu S, Sundgren I, Gålfalk M, Klemedtsson L, Crill P, Danielsson Å, Bastviken D, 2015. Spatio-temporal variability of lake CH_4 fluxes and its influence on annual whole lake emission estimates. Limnology and Oceanography. doi:10.1002/lno.10222

Nilsson K, Byegård J, Selnert E, Widestrand H, Höglund S, Gustafsson E, 2010. Äspö Hard Rock Laboratory, Long Term Sorption Diffusion Experiment (LTDE-SD). Results from rock sample analyses and modelling. SKB R-10-68, Svensk Kärnbränslehantering AB.

Norrfors Knapp K, 2015. Stability and sorption capacity of montmorillonite colloids: investigation of size fractional differences and effects of γ-irradiation. PhD thesis. Royal Institute of Technology, Stockholm, Sweden.

NWTRB, 2015. United States Nuclear Waste Technical Review Board. Transcript of the International Technical Workshop on Deep Borehole Disposal of Radioactive Waste —Wednesday October 21, 2015. Available at: http://www.nwtrb.gov/meetings/2015/oct/15oct21.pdf

NWTRB, 2016. Technical evaluation of U.S. Department of Energy Deep Borehole Disposal Research and Development Program, January 2016. Available at: http://www.nwtrb.gov/reports/DBD.pdf

Nyberg J, Elhammer A, Sohlenius G, Kjellin B, Nordgren P, 2011. Results from marine geological investigations outside Forsmark. SKB P-11-39, Svensk Kärnbränslehantering AB.

Näslund J O, Brandefelt J, 2014. Timing of future glacial inception. In Haeberli W, Whiteman C (eds). Snow and ice-related hazards, risks, and disasters. Amsterdam: Elsevier, 347–353.

Ochs M, Colàs E, Grivé M, Olmeda J, Campos I, Jordi B, 2014. Reduction of radionuclide uptake in hydrated cement systems by organic complexing agents: Selection of reduction factors and speciation calculations. SKB R-14-22, Svensk Kärnbränslehantering AB.

Odeh R E, Owen D B, 1980. Tables for normal tolerance limits, sampling plans, and screening. New York: Marcel Dekker.

Ollila K, Myllykylä E, Tanhua-Tyrkkö M, Lavonen T, 2013. Dissolution rate of alpha-doped UO_2 in natural groundwater. Journal of Nuclear Materials 442, 320–325.

Olofsgård M, Baczynska M, 2016. Miljöpåverkan från hantering av BWR-reaktortankar – en LCA analys. Projekt-ID 60654442, ÅF. SKBdoc 1524819 ver 1.0, Svensk Kärnbränslehantering AB. (In Swedish.)

Olofsson I, Fredriksson A, 2005. Strategy for a numerical Rock Mechanics Site Descriptive Model. Further development of the theoretical/numerical approach. SKB R-05-43, Svensk Kärnbränslehantering AB.

Olofsson I, Christiansson R, Holmberg M, Carlsson A, Martin D, 2014. Application of the Observational Method in the Äspö Expansion project. SKB R-13-44, Svensk Kärnbränslehantering AB.

Olsson R, 1998. Mechanical and hydromechanical behaviour of hard rock joints: a laboratory study. PhD thesis. Chalmers University of Technology, Sweden.

Olvmo M, 2010. Review of denudation processes and quantification of weathering and erosion rates at a 0.1 to 1 Ma time scale. SKB TR-09-18, Svensk Kärnbränslehantering AB.

Pehkonen H, 2014. Design av gasskydd för friktionsomrörningssvetsningsmaskin. Master thesis. Linköping University. (In Swedish.)

Pettersson K, 2010. A study of grain boundary sliding in copper with and without an addition of phosphorus. Journal of Nuclear Materials 405, 131–137.

Pettersson K, 2012. A review of the creep ductility of copper for nuclear waste canister application. SSM report 2012:13, Strålsäkerhetsmyndigheten (Swedish Radiation Safety Authority).

Pettersson K, 2016. An updated review of the creep ductility of copper including the effect of phosphorus. SSM report 2016:02, Strålsäkerhetsmyndigheten (Swedish Radiation Safety Authority).

Pettersson S, 2013. Feasibility study of waste containers and handling equipment for SFL. SKB R-13-07, Svensk Kärnbränslehantering AB.

Petrone J, Sohlenius G, Johansson E, Lindborg T, Näslund J-O, Strömgren M, 2016. Using ground-penetrating radar, topography and classification of vegetation to model the sediment and active layer thickness in a periglacial lake catchment, Western Greenland. Earth System Science Data Discussions. doi:10.5194/essd-2016-19

Posiva, 2012. Olkiluoto site description 2011. Posiva Report 2011-2, Posiva Oy, Finland.

Potyondy D O, 2007. Simulating stress corrosion with a bonded-particle model for rock. International Journal of Rock Mechanics and Mining Sciences 44, 677–691.

Potyondy D O, Cundall P A, 2004. A bonded-particle model for rock. International Journal of Rock Mechanics and Mining Sciences 41, 1329–1364.

Puranen A, Granfors M, Ekeroth E, Spahiu K, 2016. Lessons learned from leaching of dry milled high burnup UO_2 fuel under H2 atmosphere. Scientific Basis for Nuclear Waste Management XXXIX. Warrendale, PA: Materials Research Society. (To be published.)

Pålbrink L, Rydman O, 2013. Frysnng av betong under inverkan av tvång: en experimentell studie av frostens inverkan på betongkonstruktionerna i slutförvaret för kortlivat radioaktivt avfall efter en permafrost. TVBM-5091, Lunds tekniska högskola. (In Swedish.)

Quiquet A, Colleoni F, Masina S, 2016. Long-term safety of a planned repository for spent nuclear fuel in Forsmark, Sweden and Olkiluoto, Finland – Phase 2: impact of ice sheet dynamics, climate forcing and multi-variate sensitivity analyses on the maximum ice thickness. SKB TR-16-02, Svensk Kärnbränslehantering AB.

Raiko H, Sandström R, Rydén H, Johansson M, 2010. Design analysis report for the canister. SKB TR-10-28, Svensk Kärnbränslehantering AB.

Railo A (ed) Laitinen I (ed), Mustonen S, Kosunen P, Joutsen A, Ikonen A, Nuijten G, 2015. Design and construction of the equipment and experimental deposition holes, in ONKALO Demonstration tunnel 1. Posiva Working Report 2015-25, Posiva Oy, Finland.

Railo A (ed) Laitinen I (ed), Mustonen S, Kosunen P, Joutsen A, Mellanen S, Ikonen A, Hollmen K, Nuijten G, 2016. Design and construction of the equipment and experimental deposition holes, in ONKALO Demonstration tunnel 2. Posiva Working Report 2016-27, Posiva Oy, Finland.

Razdan M, Trummer M, Zagidulin D, Jonsson M, Shoesmith D W, 2014. Electrochemical and surface characterization of uranium dioxide containing rare-earth oxide (Y_2O_3) and metal (Pd) particles. Electrochimica Acta 130, 29–39.

Ronneteg U, Grybäck T, 2014. Kvalificering av oförstörande provning. SKBdoc 1414464 ver 1.0, Svensk Kärnbränslehantering AB. (In Swedish.)

Ronneteg U, Grybäck T, 2015. Non-destructive testing of canister components and welds. SKBdoc 1434744 ver 2.0, Svensk Kärnbränslehantering AB.

Rosborg B, 2013a. Recorded corrosion rates on copper electrodes in the Prototype Repository at the Äspö HRL. SKB R-13-13, Svensk Kärnbränslehantering AB.

Rosborg B, 2013b. Post-test examination of a copper electrode from deposition hole 5 in the Prototype Repository. SKB R-13-14, Svensk Kärnbränslehantering AB.

Rosdahl A, Pedersen K, Hallbeck L, Wallin B, 2011. Investigation of sulphide in core drilled boreholes KLX06, KAS03 and KAS09 at Laxemar and Äspö. Chemical-, microbiological- and dissolved gas data from groundwater in four borehole sections. SKB P-10-18, Svensk Kärnbränslehantering AB.

Roth O, 2015. A study of instant release fractions and matrix dissolution rate of two fuel samples irradiated in Olkiluoto 1. Posiva Working Report 2015-19, Posiva Oy, Finland.

Roth O, Low J, Granfors M, Spahiu K, 2013. Effects of matrix composition on instant release fractions from high burn-up nuclear fuel. In Scientific basis for nuclear waste management XXXVI. Warrendale, PA: Materials Research Society. (Materials Research Society Symposium Proceedings 1518)

Roth O, Low J, Spahiu K, 2014. Effects of matrix composition and sample preparation on instant release fractions from high burnup nuclear fuel. In Scientific basis for nuclear waste management XXXVII. Warrendale, PA: Materials Research Society. (Materials Research Society Symposium Proceedings 1665). doi:10.1557/opl.2014.653

Roth O, Granfors M, Puranen A, Spahiu K, 2015. Release of ^{108}Ag from irradiated PWR control rod absorbers under deep repository. In Gin S, Jubin R, Matyáš J, Vance E (eds). Scientific basis for nuclear waste management XXXVIII. Warrendale, PA: Materials Research Society. (Materials Research Society Symposium Proceedings 1744). doi:10.1557/opl.2015.372

Rozalen M, Huertas F J, Brady P V, 2009. Experimental study of the effect of pH and temperature on the kinetics of montmorillonite dissolution. Geochimica et Cosmochimica Acta 73, 3752–3766.

Runermark C-G, 2016. Forsmarks aavvecklingsplan. F-0011130 Rev 0, Forsmarks Kraftgrupp. (In Swedish.)

Rutqvist J, Tsang C-F, 2008. Review of SKB:s work on coupled THM processes within SR-Can. External review contribution in support of SKI:s and SSI:s review of SR-Can. SKI Report 2008:08, Statens kärnkraftinspektion (Swedish Nuclear Power Inspectorate).

Rydberg J, Lindborg T, Sohlenius G, Reuss N, Olsen J, Laudon H, 2016. The importance of eolian input on lake-sediment geochemical composition in the dry pyroglacial landscape of western Greenland. Arctic, Antarctic, and Alpine Research 48, 93–109.

Saetre P, Nordén S, Keesmann S, Ekström P-A, 2013. The biosphere model for radionuclide transport and dose assessment in SR-PSU. SKB R-13-46, Svensk Kärnbränslehantering AB.

Salonen S J, Helmens K F, Seppä H, Birks H J B, 2013. Pollen-based palaeoclimate reconstructions over long glacial-interglacial timescales: methodological tests based on the Holocene and MIS 5d-c deposits of Sokli, northern Finland. Journal of Quaternary Science 28, 271–282.

Sandén T, Börgesson L, 2014. System design of backfill. Methods of water handling. SKB R-14-09, Svensk Kärnbränslehantering AB.

Sandén T, Olsson S, Andersson L, Dueck A, Jensen V, Hansen E, Johnsson A, 2014. Investigation of backfill candidate materials. SKB R-13-08, Svensk Kärnbränslehantering AB.

Sandström R, 2014a. The role of phosphorus for mechanical properties in copper. SKBdoc 1417069 ver 1.0, Svensk Kärnbränslehantering AB.

Sandström R, 2014b. The role of hydrogen in copper. SKBdoc 1420051 ver 1.0, Svensk Kärnbränslehantering AB.

Sandström R, 2016. Influence of phosphorus on the tensile stress strain curves in copper. Journal of Nuclear Materials 470, 290–296.

Sandström R, Wu R, Hagström J, 2016. Grain boundary sliding in copper and its relation to cavity formation during creep. Materials Science and Engineering: A 651, 259–268.

Schmidt P, Lund B, Näslund J-O, Fastook J, 2014. Comparing a thermo-mechanical Weichselian Ice Sheet reconstruction to reconstructions based on the sea level equation: aspects of ice configurations and glacial isostatic adjustment. Solid Earth 5, 371–388.

SDDP, 2016. SDDP - The Swedish Scientific Drilling Program. Available at: http://www.ssdp.se/

Selroos J-O, Appleyard P, Bym T, Follin S, Hartley L, Joyce S, Munier R, 2015. The stochastic-deterministic transition in discrete fracture network models and its implementation in a safety assessment application by means of conditional simulation. H53I-02, AGU Fall meeting, 14–18 December 2015, San Fransisco.

Selvadurai A P S, Suvorov A P, Selvadurai P A, 2014. Thermo-hydro-mechanical processes in fractured rock formations during glacial advance. Geoscientific Model Development Discussions 7, 7351–7394.

Shala S, 2014. Palaeoenvironmental changes in the northern Boreal zone of Finland based on lake sediment analyses: local versus regional drivers. PhD thesis. Stockholm University.

Shala S, Helmens K F, Luoto T P, Väliranta M, Weckström J, Salonen J S, Kuhry P, 2014a. Evaluating environmental drivers of Holocene changes in water chemistry and aquatic biota composition at Lake Loitsana, NE Finland. Journal of Paleolimnology 52, 311–329.

Shala S, Helmens K F, Jansson K N, Kylander M E, Risberg J, Löwemark L, 2014b. Palaeoenvironmental record of glacial lake evolution during the early Holocene at Sokli, NE Finland. Boreas 43, 362–376.

Sheppard S, Sohlenius G, Omberg L-G, Borgiel M, Grolander S, Nordén S, 2011. Solid/liquid partition coefficients (K_d) and plant/soil concentration ratios (CR) for selected soils, tills and sediments at Forsmark. SKB R-11-24, Svensk Kärnbränslehantering AB.

Shipsha A, 2013. Statistical data analysis of cast iron properties for PWR-inserts from tension, compression and fracture toughness testing. Report 50017490-1, rev 2, Inspecta Technology AB. SKBdoc 1414800 ver 1.0, Svensk Kärnbränslehantering AB.

Silva O, Abarca E, Molinero J, Kautsky U, 2015. An equivalent K_d-based radionuclide transport model implemented in COMSOL Multiphysics® Software. In Proceedings of Comsol Conference 2015, Grenoble, 14–16 October 2015.

SIS, 1976. SIS 28115: Vattenundersökningar – Bestämning av sulfidkoncentration hos renvatten och icke förorenat naturvatten. Stockholm: Swedish Standards Institute. (In Swedish.)

SKB, 2008. Site description of Forsmark at completion of the site investigation phase. SDM-Site Forsmark. SKB TR-08-05, Svensk Kärnbränslehantering AB.

SKB, 2009a. Design premises for a KBS-3V repository based on results from the safety assessment SR-Can and some subsequent analyses. SKB TR-09-22, Svensk Kärnbränslehantering AB.

SKB, 2009b. Site description of Laxemar at completion of the site investigation phase. SDM-Site Laxemar. SKB TR-09-01, Svensk Kärnbränslehantering AB.

SKB, 2010a. Climate and climate-related issues for the safety assessment SR-Site. SKB TR-10-49, Svensk Kärnbränslehantering AB.

SKB, 2010b. Design, construction and initial state of the underground openings. SKB TR-10-18, Svensk Kärnbränslehantering AB.

SKB, 2010c. Design, production and initial state of the canister. SKB TR-10-14, Svensk Kärnbränslehantering AB.

SKB, 2010d. Model summary report for the safety assessment SR-Site. SKB TR-10-51, Svensk Kärnbränslehantering AB.

SKB, 2010e. Ramprogram för detaljundersökningar vid uppförande och drift av slutförvar för använt kärnbränsle. SKB R-10-08, Svensk Kärnbränslehantering AB. (In Swedish.)

SKB, 2010f. Triumf NG Next Generation, ver 1.0.3.0. Svensk Kärnbränslehantering AB.

SKB, 2011a. Kärntekniska industrins praxis för friklassning av material, lokaler och byggnader samt mark. SKB R-11-15, Svensk Kärnbränslehantering AB. (In Swedish.)

SKB, 2011b. Long-term safety for the final repository for spent nuclear fuel at Forsmark. Main report of the SR-Site project. SKB TR-11-01, Svensk Kärnbränslehantering AB.

SKB, 2013a. Låg- och medelaktivt avfall i SFR. Referensinventarium för avfall 2013. SKB R-13-37, Svensk Kärnbränslehantering AB. (In Swedish.)

SKB, 2013b. Site description of the SFR area at Forsmark at completion of the site investigation phase. SDM-PSU Forsmark. SKB TR-11-04, Svensk Kärnbränslehantering AB.

SKB, 2013c. Svar till SSM på begäran om komplettering rörande lång återmättnadsfas. SKBdoc 1385067 ver 3.0. Svensk Kärnbränslehantering AB. (In Swedish.)

SKB, 2014a. Biosphere synthesis report for the safety assessment SR-PSU. SKB TR-14-06, Svensk Kärnbränslehantering AB.

SKB, 2014b. Climate and climate-related issues for the safety assessment SR-PSU. SKB TR-13-05, Svensk Kärnbränslehantering AB.

SKB, 2014c. F-PSAR SFR – Allmän del 1 kapitel 3 – Konstruktionsregler. SKBdoc 1220377 ver 2.0, Svensk Kärnbränslehantering AB. (In Swedish.)

SKB, 2014d. Model summary report for the safety assessment SR-PSU. SKB TR-14-11, Svensk Kärnbränslehantering AB.

SKB, 2014e. Preliminär typbeskrivning för hela BWR reaktortankar exklusive interndelar. SKBdoc 1262709 ver 3.0, Svensk Kärnbränslehantering AB.

SKB, 2014f. Svar till SSM på begäran om komplettering rörande fosfors inverkan på koppars krypegenskaper. SKBdoc 1457514 ver 1.0, Svensk Kärnbränslehantering AB. (In Swedish.)

SKB, 2014g. Svar till SSM på begäran om komplettering rörande kapselns mekaniska integritet. SKBdoc 1371849 ver 3.0, Svensk Kärnbränslehantering AB. (In Swedish.)

SKB, 2014h. Svar till SSM på begäran om komplettering rörande tillverkningsaspekter för ingående delar i kapseln. SKBdoc 1371851 ver 4.0, Svensk Kärnbränslehantering AB. (In Swedish.)

SKB, 2014i. Uppdatering av rapporten Jämförelse mellan KBS-3-metoden och deponering i djupa borrhål för slutligt omhändertagande av använt kärnbränsle. SKB P-14-21, Svensk Kärnbränslehantering AB. (In Swedish.)

SKB, 2014j. Uppdatering av rapporten Principer, strategier och system för slutligt omhändertagande av använt kärnbränsle. SKB P-14-20, Svensk Kärnbränslehantering AB. (In Swedish.)

SKB, 2015a. Handling of biosphere FEPs and recommendations for model development in SR-PSU. SKB R-14-02, Svensk Kärnbränslehantering AB.

SKB, 2015b. Safety analysis for SFR. Long term safety. Main report for the safety assessment SR-PSU. Revised edition. SKB TR-14-01, Svensk Kärnbränslehantering AB.

SKB, 2015c. Svar till SSM på begäran om komplettering avseende degraderingsprocesser för kapseln. SKBdoc 1398013 ver 4.0, Svensk Kärnbränslehantering AB. (In Swedish.)

SKB, 2016a. Lägesrapport om SKB:s pågående och planerade arbeten kring utveckling av kapseln. SKBdoc 1540089 ver 1.0, Svensk Kärnbränslehantering AB. (In Swedish.)

SKB, 2016b. Ytterligare information om kopparkorrosion i syrgasfritt vatten. SKBdoc 1540185 ver 1.0, Svensk Kärnbränslehantering AB. (In Swedish.)

SKB, 2016c. Äspö Hard Rock Laboratory. Annual Report 2015. SKB TR-16-10, Svensk Kärnbränslehantering AB.

Smart N R, Rance A P, Reddy B, Hallbeck L, Pedersen K, Johansson A J, 2014. In situ evaluation of model copper-cast iron canisters for spent nuclear fuel: a case of microbiologically influenced corrosion (MIC). Corrosion Engineering, Science and Technology 49, 548–553.

Smith C A, Sundh M, Mikko H, 2014. Surficial geology indicates early Holocene faulting and seismicity, central Sweden. International Journal of Earth Sciences 103, 1711–1724.

Smith J M, Wren J C, Odziemkowski M, Shoesmith D W, 2007. The electrochemical response of preoxidized copper in aqueous sulfide solutions. Journal of The Electrochemical Society 154, C431–C438.

SNSN, 2015. Svenska Nationella Seismiska Nätet. Uppsala Universitet, Institutionen för Geovetenskaper. Available at: http://www.snsn.se/ (In Swedish.)

Sohlenius G, Schoning K, Baumgartner A, 2013a. Development, carbon balance and agricultural use of peatlands – overview and examples from Uppland Sweden. SKB TR-13-20, Svensk Kärnbränslehantering AB.

Sohlenius G, Strömgren M, Hartz F, 2013b. Depth and stratigraphy of regolith at Forsmark. SR-PSU Biosphere. SKB R-13-22, Svensk Kärnbränslehantering AB.

Soroka I L, Tarakina N V, Jonsson M, 2016. Cuprous hydroxide: synthesis, structure and physical properties. SKB TR-16-03, Svensk Kärnbränslehantering AB.

Srivastava R M, 2002. Probabilistic discrete fracture network models for the Whiteshell Research Area. Ontario Power Generation Report 06819-REP-01200-10071-R00, Ontario Power Generation, Nuclear Waste Management Division, Canada.

Sriver R L, Urban N M, Olson R, Keller K, 2012. Towards a physically plausible upper bound of sea-level rise projections. Climatic Change 115, 893–902.

SSM, 2014a. Granskning och utvärdering av SKB:s redovisning av Fud-program 2013. Rapport 2014:12, Strålsäkerhetsmyndigheten (Swedish Radiation Safety Authority). (In Swedish.)

SSM, 2014b. Minnesantekningar från avstämningsmöte mellan SSM och SKB angående kontroll och provning av kapseln. SSM2011-1137-63, Strålsäkerhetsmyndigheten (Swedish Radiation Safety Authority). (In Swedish.)

SSM, 2015. Långsiktig säkerhet. Preliminär rapport. Dokument 13-3523, Strålsäkerhetsmyndigheten (Swedish Radiation Safety Authority). (In Swedish.)

SSM, 2016. Fud-utredning. Promemoria SSM2015-1136-2, Strålsäkerhetsmyndigheten (Swedish Radiation Safety Authority). (In Swedish.)

Stark K, 2015. Parallel assessment of effects on non-human biota using RESRAD-BIOTA. Technical Note 2015:45, Strålsäkerhetsmyndigheten (Swedish Radiation Safety Authority).

Stephens M B, Simeonov A, 2015. Description of deformation zone model version 2.3, Forsmark. SKB R-14-28, Svensk Kärnbränslehantering AB.

Stigsson M, 2016. Orientation uncertainty of structures measured in cored boreholes: methodology and case study of swedish crystalline rock. rock mechanics and rock engineering. Rock Mechanics and Rock Engineering. doi:10.1007/s00603-016-1038-5

Stigsson M, Munier R 2013. Orientation uncertainty goes bananas: an algorithm to visualise the uncertainty sample space on stereonets for oriented objects measured in boreholes, Computers & Geosciences 56, 56–61.

Sundell-Bergman S, Avilia R, Grusell E, Häggkvist K, deWith G, Maderich V, 2015. PREDO – PREdiction of DOses from normal releases of radionuclides to the environment. QP.50000-107745633, Vattenfall, Stockholm.

Sundin S, Dahlgren B, Roth O, Jonsson M, 2013. H_2O_2 and radiation induced dissolution of UO_2 and SIMFUEL in HCO_3^- deficient aqueous solution. Journal of Nuclear Materials 443, 291–297.

Svensson D, 2015. The bentonite barrier: swelling properties, redox chemistry and mineral evolution. PhD thesis. Lund University.

Svensson D, Hansen S, 2013. Redox chemistry in two iron-bentonite field experiments at Äspö Hard Rock Laboratory, Sweden: an XRD and Fe K-edge XANES study. Clays and Clay Minerals 61, 566–579.

Svemar C, Johannesson L-E, Grahm P, Svensson D, Kristensson O, Lönnqvist M, Nilsson U, 2016. Prototype Repository. Opening and retrieval of outer section of Prototype Repository at Äspö Hard Rock Laboratory. Summary report. SKB TR-13-22, Svensk Kärnbränslehantering AB.

Söderbäck B (ed), 2008. Geological evolution, palaeoclimate and historical development of the Forsmark and Laxemar-Simpevarp areas. Site descriptive modelling. SDM-Site. SKB R-08-19, Svensk Kärnbränslehantering AB.

Tang L, Bager D H, 2013. A study of consequences of freezing of concrete structures for storage of nuclear waste due to permafrost. SKB TR-12-13, Svensk Kärnbränslehantering AB.

Taniguchi N, Kawasaki M, 2008. Influence of sulfide concentration on the corrosion behavior of pure copper in synthetic seawater. Journal of Nuclear Materials 379, 154–161.

Tasdigh H, 2015. Assessment of the impact of fiber mass UP2 degradation products on nickel(II) and europium(III) sorption. Master thesis. Royal Institute of Technology, Stockholm, Sweden.

Taxén C, 2013. Ytprofiler på kopparkapslar från deponeringshål 5 och 6 i försöksserien Prototyp. SKB P-13-50, Svensk Kärnbränslehantering AB. (In Swedish.)

Taxén C, Sparr M, 2014. Corrosion properties of copper materials. SKB R-14-15, Svensk Kärnbränslehantering AB.

Taxén C, Lundholm M, Persson D, Jakobsson D, Sedlakova M, Randelius M, Karlsson O, Rydgren P, 2012. Analyser av koppar från prototypkapsel 5 och 6. SKB P-12-22, Svensk Kärnbränslehantering AB. (In Swedish.)

Thegerström C, 2014. SKB:s jämförande bedömningar av andra studerade metoder än den valda metoden, KBS-3. SKBdoc 1440497 ver 1.0, Svensk Kärnbränslehantering AB. (In Swedish.)

Thorsell P-E, 2013. Studier av frysningsegenskaper hos betong från 1 BMA. SKB P-13-07, Svensk Kärnbränslehantering AB. (In Swedish.)

Thuvander M, 2015. Investigation of the distribution of phosphorus in copper. Report 2015:11, Strålsäkerhetsmyndigheten (Swedish Radiation Safety Authority).

Thörn J, 2015. The impact of fracture geometry on the hydromechanical behaviour of crystalline rock. PhD thesis. Chalmers University of Technology, Sweden.

Toijer E, 2014. Assessment of primary damage and copper precipitation in cast iron in repositoiry conditions. Master thesis. Royal Institute of Technology, Stockholm, Sweden.

Travis K, Gibb F, 2016. Cleaning-up Hanford: a deep borehole concept for the Cs/Sr capsules. Proceeding of Waste Management, Phoenix, Arizona, 6–10 March 2016. Paper 16249.

Trinchero P, Painter S, Ebrahimi H, Koskinen L, Molinero J, Selroos J-O, 2016. Modelling radionuclide transport in fractured media with a dynamic update of K_d values. Computers & Geosciences 86, 55–63.

Tryggvason A, Linde N, 2006. Local earthquake (LE) tomography with joint inversion for P- and S-wave velocities using structural constraints. Geophysical Research Letters 33. doi:10.1029/2005GL025485

Tröjbom M, Grolander S, 2010. Chemical conditions in present and future ecosystems in Forsmark – implications for selected radionuclides in the safety assessment SR-Site. SKB R-10-27, Svensk Kärnbränslehantering AB.

Tröjbom M, Grolander S, Raguz V, Nordén S, 2013. K_d and CR used for transport calculations in the biosphere in SR-PSU. SKB R-13-01, Svensk Kärnbränslehantering AB.

Unosson M, 2014. A constitutive model for texture dependent deformation hardening and pressure dependent initiation of ductile failure in metallic materials. SKBdoc 1393179 ver 2.0, Svensk Kärnbränslehantering AB.

Unosson M, 2016. Numerical simulation of earthquake-induced rock shear through a deposition hole – crack initiation and propagation in an insert with pre-existing intertubular and edge crack planes. SKBdoc 1474363 ver 1.0, Svensk Kärnbränslehantering AB.

Vaughan D G, Comiso J C, Allison I, Carrasco J, Kaser G, Kwok R, Mote P, Murray T, Paul F, Ren J, Rignot E, Solomina O, Steffen K, Zhang T, 2013. Observations: cryosphere. I Stocker T F, Qin D, Platter G-K, Tignor M, Allen S K, Boschung J, Nauels A, Xia Y, Bex V, Midgley P M (eds). Climate change 2013: the physical science basis. Contribution of working group I to the Fifth Assessment Report of the Intergovernmental Panel on Climate Change. Cambridge: Cambridge University Press.

von Schenck H, Bultmark F, 2014. Effekt av bitumensvällning i silo och BMA. SKB R-13-12, Svensk Kärnbränslehantering AB. (In Swedish.)

von Schenck H, Kautsky U, Gylling B, Abarca E, Molinero J, 2015. Advancing the modelling environment for the safety assessment of the Swedish LILW repository at Forsmark. In Scientific basis for nuclear waste management XXXVII. Warrendale, PA: Materials Research Society. (Materials Research Society Symposium Proceedings 1744). doi:10.1557/opl.2015.372

Väliranta M, Salonen J S, Heikkilä M, Amon L, Birks H H, Helmens K F, Klimaschewski A, Kuhry P, Kultti S, Poska A, Shala S, Veski S, Birks H H, 2015. Plant macrofossil evidence for an early onset of the Holocene summer thermal maximum in northernmost Europe. Nature Communications 6. doi:10.1038/ncomms7809

Walke R C, Kirchner G, Xu S, Dverstorp B, 2015. Post-closure biosphere assessment modelling: comparison of complex and more stylised approaches. Journal of Environmental Radioactivity 148, 50–58.

Wallin A, Qvarfordt S, Borgiel M, 2016. Bottensubstratkartering i undersökningsområde 157_2 norr om SFR, Forsmark. SKB P-14-09, Svensk Kärnbränslehantering AB. (In Swedish.)

Wekerle C, Colleoni F, Näslund J O, Brandefelt J, Masina S, 2016. Numerical reconstructions of the penultimate glacial maximum Northern Hemisphere ice sheets: sensitivity to climate forcing and model parameters. Journal of Glaciology. doi:10.1017/jog.2016.45

Werner K, Sassner M, Johansson E, 2013. Hydrology and near-surface hydrogeology at Forsmark – synthesis for the SR-PSU project. SR-PSU Biosphere. SKB R-13-19, Svensk Kärnbränslehantering AB.

Wikander O, 2015a. Don't push this button: Phoenician sarcophagi, atomic priesthoods and nuclear waste. Vetenskapssocietetens årstidsskrift 2015, 109–124.

Wikander O, 2015b. Language, nuclear waste and society: the preservation of knowledge over vast periods of time and its relevance for linguistics. Lychnos 2015, 7–25.

Winkler S R, Eigl R, Forstner O, Martschini M, Steier P, Sterba J H, Golser R, 2015. Using the nuclear activation AMS method for determining chlorine in solids at ppb-levels and below. Nuclear Instruments and Methods in Physics Research Section B: Beam Interactions with Materials and Atoms 361, 649–653.

Wold S, 2014. Are colloids released from different materials in SFR to the saturating groundwater? School of chemical science and engineering, KTH. SKBdoc 1466123 ver 1.0, Svensk Kärnbränslehantering AB.

Wu R, Sandström R, 2015. Influence of cold work and notches on creep failure of Cu-OFP. SKB R-15-04, Svensk Kärnbränslehantering AB.

Wu R, Sandström R, Hagström J, 2015. Grain boundary sliding in phosphorus alloyed oxygen-free copper under creep. SKB R-15-14, Svensk Kärnbränslehantering AB.

Zanonato P L, Di Bernardo P, Grenthe I, 2014. A calorimetric study of the hydrolysis and peroxide complex formation of the uranyl(VI) ion. Dalton Transactions 43, 2378–2383.

Yang M, Jonsson M, 2014. Evaluation of the O_2 and pH effects on probes for surface bound hydroxyl radicals. The Journal of Physical Chemistry C 118, 7971−7979.

Yang M, Jonsson M, 2015. Surface reactivity of hydroxyl radicals formed upon catalytic decomposition of H_2O_2 on ZrO_2. Journal of Molecular Catalysis A: Chemical 400, 49–55.

Åkerblom E, 2015. Analys av Tc-99 och I-129 på vattenprov. N-15/337, Studsvik Nuclear AB. SKBdoc 1522753 ver 1.0, Svensk Kärnbränslehantering AB. (In Swedish.)

Åkesson M, Kristensson O, Börgesson L, Dueck A, Hernelind J, 2010. THM modelling of buffer, backfill and other system components. Critical processes and scenarios. SKB TR-10-11, Svensk Kärnbränslehantering AB.

Öhman J, Follin S, Odén M, 2014. SR-PSU Hydrogeological modelling. TD11 – Temperate climate conditions. SKB P-14-04, Svensk Kärnbränslehantering AB.

Appendix

Abbreviations

ABM	Alternative Buffer Materials. Experiments in the Äspö HRL where possible buffer materials are being investigated.
AKA study	Initial version of the KBS-3 repository – AKA Spent nuclear fuel and radioactive waste that delivered its final report in 1976.
ALARA	As Low As Reasonably Achievable. Keeping the radiation doses as low as reasonably achievable with regard to both economic and social factors.
APSE	Äspö Pillar Stability Experiment. Completed experiment in Äspö HRL for studies of how large a load the rock can withstand.
Asha	Indian bentonite from the Kutch region.
ASME	American Society of Mechanical Engineers. Provides standards.
ASTM	Standards within different technical fields issued by ASTM International, which is an American non-profit standards organisation with members from about 120 countries.
ATB	Waste transport container.
ATB 1T	A new container for transport of long-lived low- and intermediate-level waste in BFA tanks.
AVANS	Decommissioning Planning at Vattenfall's Nuclear Power Operations in Sweden.
B1	Nuclear power reactor Barsebäck 1.
B2	Nuclear power reactor Barsebäck 2.
BA	Combustible absorbers.
BELBAR	Bentonite erosion: effects on the long term performance of the engineered barrier and radionuclide transport. EU project.
BFA	Rock cavern on the Simpevarp peninsula for dry interim storage of operational waste.
BHA	Waste vault for the 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 a rock vault for low-level waste (1BLA) and a further four rock vaults (2–5BLA) are planned in the extended part of SFR.
BMA	Waste vault for intermediate-level waste in SFR. SFR contains a rock vault for intermediate-level waste (1BMA) and a further vault (2BMA) is planned in the extended part of SFR.
BTF	Rock vaults for concrete tanks in SFR, mainly intended for dewatered ion exchange resins.
BRT	Waste vault for whole reactor pressure vessels. Planned in the extended part of SFR.
BWR	Boiling Water Reactor. The reactors in Forsmark, Oskarshamn and reactor 1 in Ringhals.
CAPS	Count pressure Applied to Prevent Spalling. Completed experiment in Äspö HRL where the opportunities to reduce the risk of rock breakouts are investigated.
Cast	Carbon-14 Source Term. EU project that deals with questions concerning carbon-14 in radioactive waste.

CEC	Cation exchange capacity.
Cebama	Cement Based Materials: properties, evolution and barrier functions. EU project.
Clab	Central interim storage facility for spent nuclear fuel.
Clink	Central facility for interim storage and encapsulation of spent nuclear fuel.
CEMI	Centre of Excellence in Mining Innovations, Canada.
COMSOL	Calculation tools for modelling and simulation of complex physics-based systems. COMSOL Inc.
CoupModel	Model for linked heat and mass transfer in the system earth-plants-atmosphere.
CR	Concentration ratios.
CSH	Calcium silicate hydrate.
DFN	Discrete fracture network.
DFT	Density functional theory.
DOC	Dissolved organic carbon.
Domplu	Dome plug experiment. Full-scale test in the Äspö HRL in order to test and demonstrate the complete plug system. Part of the joint EU project DOPAS.
DPP	Decommissioning Preparation Project.
EDZ	Excavation Damaged Zone. The rock around a rock excavation where irreversible changes have taken place.
EmrasII	Environmental Modelling for Radiation Safety. IAEA project.
E.ON	E.ON Kärnkraft Sverige AB.
ERICA	Environmental risk from ionising contaminants. Tool for analysing biological effects of ionising radiation in habitats and ecosystems.
ESS	European Spallation Source, Lund.
F1	Nuclear power reactor Forsmark 1.
F2	Nuclear power reactor Forsmark 2.
F3	Nuclear power reactor Forsmark 3.
FE	Finite element.
FKA	Forsmarks Kraftgrupp AB.
FPI	Full perimeter intersection.
F-PSAR	Preparatory Preliminary Safety Analysis Report.
FSW	Friction Stir Welding.
GAP	Greenland Analogue Project. Project which SKB carried out between 2008 and 2014, together with Posiva and NWMO. The purpose was via observations from an existing inland ice sheet to enhance knowledge concerning how groundwater flow and groundwater chemistry in crystalline bedrock are affected by an ice sheet. The results are utilised for example in the assessment of post-closure safety of the Spent Fuel Repository.
GIA	Glacial Isostatic Adjustment.
GIS	Geographic Information Systems.
GRASP	Greenland Analogue Surface Project. SKB activity in order to identify differences in long-term change processes in surface systems between a cold and a temperate climate, partly to investigate how the hydrological properties and ecosystems properties vary depending on climate conditions.
HIDRA	Human Intrusion in Disposal of Radioactive waste. Network within IAEA that SKB is participating in.
Hint	Projects for handling of reactor internals (Barsebäck).

HM	Hydromechanical.
HPC	High Performance Computing.
IAEA	International Atomic Energy Agency.
IGD-TP	Implementing Geological Disposal – Technology Platform. European cooperation on nuclear waste disposal.
ISA	Isosaccharinic acid.
ISO container	Containers of sizes standardized by the International Organization for Standardization (ISO) which can be loaded on railways, trucks and cargo vessels.
KBS-3 method	The KBS-3 method has been given its name as it is based on the third report in the Nuclear Fuel Safety Project.
KBS-3H	The KBS-3 method with horizontal deposition.
K_d	Sorption coefficient, partitioning coefficient.
KTH	Royal Institute of Technology.
KTL	The Nuclear Activities Act.
LOT	Long Term Test of Buffer Material. Experiments in the Äspö HRL with the aim of finding out how primarily bentonite clay behaves at conditions similar to those in a final repository for spent nuclear fuel.
LTDE-SD	Long Term Diffusion Experiment – Sorption-Diffusion. Completed experiment in Äspö HRL.
Marfa	Transport model.
Matlab	Computer programme and programming language that is mainly used for mathematical and technical calculations, MathWorks.
MikeShe	The model for surface systems that was used in the modelling of SFR.
MIND	Microbiology In Nuclear waste Disposal, EU project
MIRARCO	Mining Innovation Rehabilitation and Applied Research Corporation, Canada.
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.
Nagra	Nationale Genossenschaft für die Lagerung von Radioaktiver Abfälle, Switzerland
NEA	Nuclear Energy Agency. A collaborative organisation for nuclear energy issues within OECD.
NWMO	Nuclear Waste Management Organization, Canada.
O1	Nuclear power reactor Oskarshamn 1.
O2	Nuclear power reactor Oskarshamn 2.
O3	Nuclear power reactor Oskarshamn 3.
OECD	Organization for Economic Co-operation and Development.
OKG	OKG AB.
Onkalo	The hard rock facility that Posiva has been building at Olkiluoto since 2004. Onkalo is used for research and development, but is also planned to constitute access to the final repository itself.
PHREEQC	Computer programme for transport and geochemistry modelling.

Posiva	Posiva Oy, Finland.
PSAR	Preliminary Safety Analysis Report.
PSU	SFR extension project.
PWR	Pressurised Water Reactor. The reactors R2, R3 and R4 in Ringhals and the Ågesta reactor.
R1	Nuclear power reactor Ringhals 1.
R2	Nuclear power reactor Ringhals 2.
R3	Nuclear power reactor Ringhals 3.
R4	Nuclear power reactor Ringhals 4.
RAB	Ringhals AB.
Redox	Redox is a chemical reaction in which the oxidation states of atoms are changed.
RNR	Radionuclide Retention Experiments. Completed experiment in Äspö HRL in order to investigate how the rock delays and filters elements.
SAR	Safety Analysis Report.
Scale	Calculation tool.
SEM	Scanning Electron Microscopy.
SFL	Final Repository for Long-lived Waste.
SFR	Final Repository for Short-lived Radioactive Waste.
SGU	Geological Survey of Sweden.
Sicada	Site characterisation database system. Database system to store and handle data from the different types of geoscientific investigations that SKB conducts. Data from experiments carried out at the Äspö HRL is also stored in the database.
SKB	Swedish Nuclear Fuel and Waste Management Company.
SNAB	Studsvik Nuclear AB.
SNSN	Swedish National Seismic Network.
SRM	Synthetic rock mass.
SR-PSU	Report of post-closure safety prior to the SFR extension project. Published in August 2015.
SR-Site	Report of post-closure safety for the final repository for spent nuclear fuel, published by SKB in March 2011.
SSM	Swedish Radiation Safety Authority.
SSMFS	Swedish Radiation Safety Authority's regulations.
STF	Safety-related technical specifications.
STURE	Safe and secure phasing out of reactors 1 and 2.
Suus	Safety during construction of the final repository for spent nuclear fuel (Spent Fuel Repository).
SVAFO	AB SVAFO. Owned by Ringhals AB, Forsmarks Kraftgrupp AB, OKG Aktiebolag and Barsebäck Kraft AB.
TASS	Tunnel for experiments in the Äspö HRL.
Task Force EBS	The Task Force on Engineered Barrier Systems. International cooperation between experts and modelling groups on questions concerning the engineered barriers in the future final repository.
THM	Thermo-hydro-mechanical
Triumf NG	Triumf New Generation. Triumf NG is a report and forecasting tool for calculation and prognosis of the amount of packages, materials and radionuclides in SFR.

TRUE	Tracer Retention Understanding Experiments. Completed experiments in Äspö HRL. Tracer tests on different scales to see to what extent results achieved on one scale are also valid for another.
TURVA-2012	Account of post-closure safety of the final repository for spent nuclear fuel in Olkiluoto, Finland, published by Posiva in 2012.
UFM	Nearly universal fracture model.
U.S. NRC	United States Nuclear Regulatory Commission.
VLF	Measurement with very low frequency radio waves for example for locating fracture zones in the rock.
VNF	Vattenfall Nuclear Fuel AB.
XRD	X-ray diffraction.

Arkitektkopia AB, Bromma, 2016

SKB is tasked with managing Swedish nuclear and radioactive waste in a safe way.

skb.se