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Initial state for the repository for the safety evaluation SE-SFL

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Preface

This report is one of the main references for the evaluation of post-closure safety for a proposed repository concept for the repository for long-lived low- and intermediate-level waste (SFL) in Sweden. The report describes the initial state for the safety evaluation of SFL, which is defined as the expected state of the repository and its environs immediately after closure. Important aspects of the initial state are the waste types, waste packaging, design features of each waste vault, allocation of waste packages to the waste vaults, material quantities, radionuclide inventories, main dimensions of the waste vaults, and control and inspection processes. The initial state is the basis for the reference evolution analysed in the SE-SFL Main report.

Jenny Brandefelt has been the project leader for SE-SFL and is responsible for the safety evaluation. The main contributing authors for this report have been Katrin Ahlford, Björn Herschend, Klas Källström, Per Mårtensson, and Henrik von Schenck (SKB).

A formal review of this report was performed by Maria Lindgren (Kemakta), Michael Thorne (Mike Thorne and Associates Ltd, United Kingdom) and Claes Johansson (SKB).

Solna, September 2019

Jenny Brandefelt
Project leader SE-SFL

Summary

The repository for long-lived waste (SFL) is planned to be constructed for the final disposal of low- and intermediate-level waste from Swedish nuclear facilities. SKB is planning to take SFL into operation around year 2045. Possible solutions for management and disposal of the Swedish long-lived low- and intermediate-level waste were examined in the SFL concept study and an approach to further assessment of post-closure safety was proposed (Elfwing et al. 2013). The next step in the development of SFL is the present safety evaluation. The purpose of this evaluation is to evaluate conditions in the waste, the barriers, and the repository environs under which the repository concept has the potential to fulfil the regulatory requirements for post-closure safety. Moreover, an objective is to provide SKB with a basis for prioritising areas in which the level of knowledge and efficiency of methods must be improved in order to perform a full safety assessment for SFL. This report constitutes one of the main references supporting the **Main report**, which summarises the evaluation of post-closure safety for SFL.

The initial state for the safety evaluation of SFL is defined as the expected state of the repository and its environs immediately after closure. This report describes the initial state of the repository which is based on the current knowledge on the properties of the waste and the repository components, and, an assessment of changes in these properties up to the time of closure. The estimated year of closure is 2075.

This report describes the following, which define the initial state of the repository assumed in the safety evaluation of SFL:

- Waste types.
- Waste packaging.
- Design features of each waste vault.
- Allocation of waste packages to the waste vaults.
- Material quantities.
- Radionuclide inventories
- Main dimensions of the waste vaults.
- Control and inspection processes.

An overview of suggested measures for the closure of SFL is also given. Overall, the report provides input to the evaluation of post-closure safety of the SFL repository and its environs.

Finally, the expected properties and condition of each system component at repository closure are described following prescribed lists of variables (parameters).

Sammanfattning

Slutförvaret för långlivat avfall (SFL) planeras att uppföras för slutförvaring av långlivat låg- och medelaktivt avfall från svenska kärntekniska anläggningar. SKB planerar att ta SFL i drift runt år 2045. Elfwing et al. (2013) undersökte möjliga lösningar för slutligt omhändertagande av det svenska långlivade låg- och medelaktiva avfallet och föreslog ett system att analysera vidare med avseende på säkerheten efter förslutning. Föreliggande säkerhetsvärdering utgör nästa steg i utvecklingen av SFL. Syftet med utvärderingen är att undersöka under vilka betingelser, med avseende på säkerhet efter förslutning, förvarets omgivningar, avfall och barriärer, som föreslaget koncept har möjlighet att uppfylla myndighetskraven. Denna rapport utgör en av huvudreferenserna till **huvudrapporten** som summerar utvärderingen av säkerheten efter förslutning för SFL.

Initialtillståndet, för säkerhetsvärderingen av SFL, är definierat som det förväntade tillståndet för förvaret och dess omgivning direkt efter förslutning. Denna rapport beskriver initialtillståndet för förvaret vilket bygger på nuvarande kunskap kring egenskaper hos avfallet och förvarskomponenterna, samt en bedömning av förändrade egenskaper under tiden fram till och med förslutning. Tidpunkten för förslutning uppskattas till år 2075.

Rapporten beskriver följande, vilket definierar initialtillståndet för förvaret som antas i säkerhetsvärderingen för SFL:

- Avfallstyper.
- Avfallsemballage.
- Bergsalarnas design.
- Fördelning av avfall mellan bergsalarna.
- Materialmängder.
- Radionuklidinventarium.
- Bergsalarnas dimensioner.
- Kontroll- och inspektionsprocess.

En översikt över föreslagna åtgärder för förslutning av SFL ges också. Sammantaget utgör den här rapporten underlag för analysen av säkerheten efter förslutning av SFL:s förvarssystem.

Slutligen beskrivs de förväntade egenskaperna samt deras kondition vid förslutning för varje systemkomponent i förvaret med hjälp av föreskrivna listor med variabler (parametrar).

Contents

1 Introduction

This report constitutes one of the main references supporting the safety evaluation for the repository for long-lived waste (SFL) in Sweden. The purpose of the SFL safety evaluation (SE-SFL) is to provide input to the subsequent, consecutive steps in the development of SFL. These consecutive steps include further development of the design of the engineered barriers and the site-selection process for SFL. Further, the outcomes of SE-SFL can be used to prioritise areas in which the level of knowledge must be improved in order to perform a subsequent, full safety assessment for SFL. This chapter gives the background to the project and an overview of the safety evaluation. Moreover, the role of this report is described in the context of the evaluation.

1.1 Background

The Swedish power industry has been generating electricity by means of nuclear power for more than 40 years. The Swedish system for managing and disposal of the waste from operation of the reactors has been developed over that period. When finalised, this system will comprise three repositories: the repository for short-lived radioactive waste (SFR), the repository for long-lived waste (SFL), and the Spent Fuel Repository.

The system for managing radioactive waste is schematically depicted in Figure 1-1. SKB currently operates SFR at Forsmark in Östhammar municipality to dispose of low- and intermediate-level waste produced during operation of the various nuclear power plants, as well as to dispose waste generated during applications of radioisotopes in medicine, industry and research. Further, SFR is planned to be extended to permit the disposal of waste from decommissioning of nuclear facilities in Sweden. The spent nuclear fuel is presently stored in the interim storage facility for spent nuclear fuel (Clab) in Oskarshamn municipality. Clab will be complemented by the Encapsulation Plant, together forming Clink. SKB has also applied to construct, possess and operate the Spent Fuel Repository at Forsmark in Östhammar municipality. The current Swedish radioactive waste management system also includes a ship and different types of casks for transport of spent nuclear fuel and other radioactive waste.

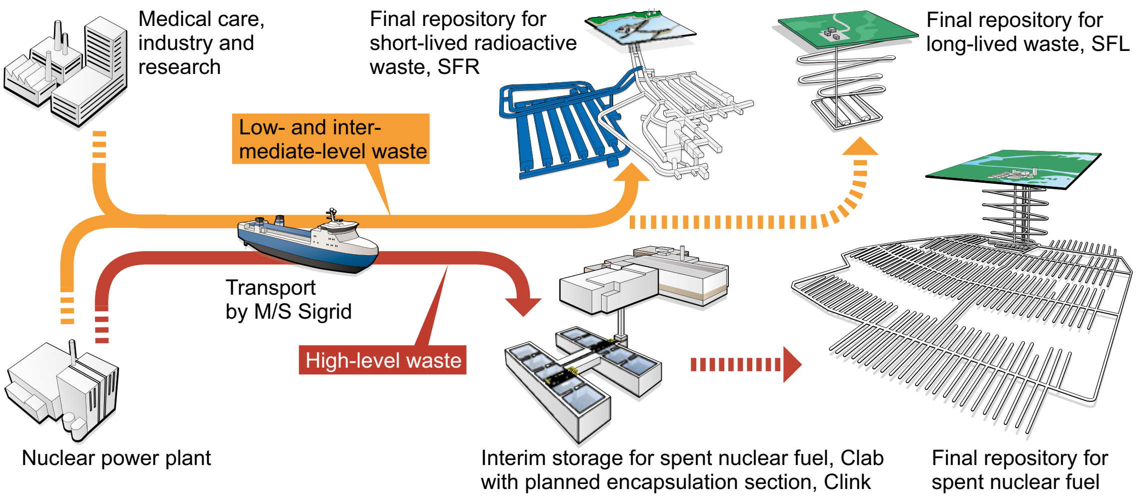


Figure 1-1. *The Swedish system for radioactive-waste management. Dashed arrows indicate future waste streams to facilities planned for construction.*

SFL will be used for disposal of the Swedish long-lived low- and intermediate-level waste. This comprises long-lived waste from the operation and decommissioning of the Swedish nuclear power plants, from early research in the Swedish nuclear programmes (legacy waste), and from medicine, industry and research, which includes the European Spallation Source (ESS) research facility. The long-lived low- and intermediate-level waste from the nuclear power plants consists of neutron-activated components and control rods and constitutes about one third of the waste planned for SFL. The rest originates mainly from the Studsvik site, where Studsvik Nuclear AB and Cyclife Sweden AB both produce and manage radioactive waste from medicine, industry and research. The legacy waste to be disposed in SFL is currently managed by the company AB SVAFO.

In 1999, a preliminary safety assessment for SFL was presented that focussed on a quantitative analysis of the environmental impact for a reference scenario (SKB 1999). Reflecting the comments from the authorities on the preliminary safety assessment, possible solutions for management and disposal of the Swedish long-lived low- and intermediate-level waste were examined in the SFL concept study (Elfwing et al. 2013). Among the considered alternatives a system was proposed as a basis for further assessment of post-closure safety. According to this concept, SFL is designed as a deep geological repository with two different sections:

- One waste vault for metallic waste from the nuclear power plants designed with a concrete barrier, BHK.
- One waste vault for the waste from Studsvik Nuclear AB, Cyclife Sweden AB and AB SVAFO designed with a bentonite barrier, BHA.

A schematic illustration of SFL is displayed in Figure 1-2. In SE-SFL, it is assumed that the waste vaults are located at 500 m depth. BHK is approximately 135 m long and BHA is approximately 170 m long. Both vaults have a cross sectional area of approximately 20 x 20 m^2.

1.2 The SE-SFL safety evaluation

There are two main objectives for SE-SFL. The first is to evaluate conditions in the waste, the barriers, and the repository environs under which the repository concept has the potential to fulfil the regulatory requirements for post-closure safety. The second is to provide SKB with a basis for prioritising areas in which the level of knowledge and adequacy of methods must be improved in order to perform a full safety assessment for SFL. This is in line with the iterative safety analysis process that the SFL repository program follows, in which the results from post-closure safety analyses and related activities (e.g. information from a site selection process and development of numerical methods) are used to successively inform and improve the analysis. In accordance with the Nuclear Activities Act (1984:3), important research needs for the SFL programme that emerge as a result of SE-SFL will be reported in the research development and demonstration (RD&D) programme. An important aspect of this is to ensure that the industry has well founded information to support long-term planning.

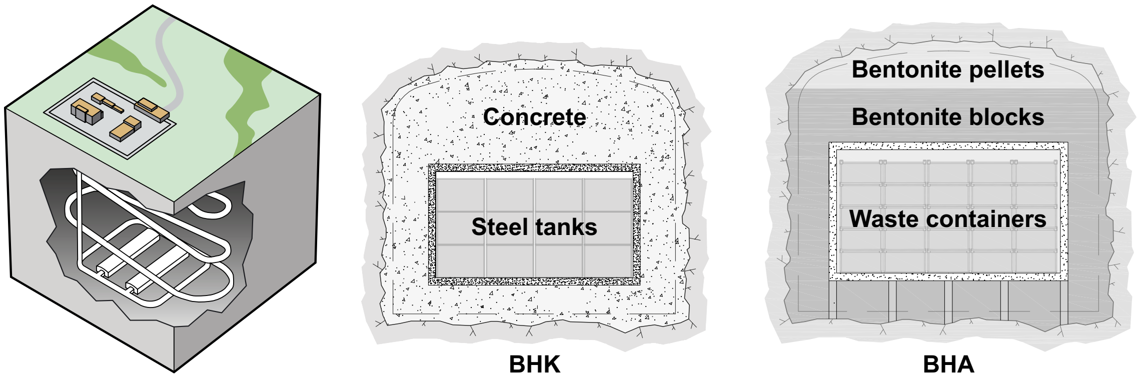


Figure 1-2. Preliminary facility layout and the proposed repository concept for SFL (left), one waste vault for metallic waste from the nuclear power plants (BHK, centre) and one waste vault for waste from Studsvik Nuclear AB, Cyclife Sweden AB and AB SVAFO (BHA, right).

The safety analysis methodology as applied in SE-SFL is a first evaluation of post-closure safety for the repository concept proposed by Elfwing et al. (2013) and is not part of a license application. As such, the methodology has been adapted to suit the needs of SE-SFL and thus differs from the methodology established by SKB for the most recent safety assessments for the extended SFR (SR-PSU; SKB 2015) and for the Spent Fuel repository (SR-Site; SKB 2011a). This also implies that the regulatory requirements on the methodology have not been applied rigorously, which would be needed for a safety analysis that is part of a license application. The evaluation is intentionally simplified as compared with SR-Site and SR-PSU, and more focus is given to aspects connected to the further development of the repository concept and related analyses. This is also reflected in using the term safety evaluation in contrast to safety assessment. The differences between SE-SFL and a full safety assessment are described in more detail in Section 2.1 of the **Main report**. The adaption of the methodology for the purposes of SE-SFL is described in Section 2.5 of the **Main report**.

To the extent applicable, SE-SFL builds on knowledge from SR-PSU and SR-Site. There are commonalities regarding the waste, engineered barriers, bedrock, surface ecosystems and external conditions relevant to post-closure safety. For instance, SE-SFL and SR-Site both address timescales of one million years (see Section 2.3 of the **Main report**). A further similarity is the proposed depth of 300–500 m. There are similarities between SFR and SFL regarding the waste and waste packaging and the proposed engineered barriers.

In SE-SFL, a first evaluation of a suitable repository design for disposal of the ESS waste is carried out. Since the information regarding the ESS inventory is not yet as well defined as for the other waste streams, the protective capability of the different waste vaults in relation to this waste is analysed separately.

No site has yet been selected for SFL and therefore data from SKB's site investigation programmes for the Spent Fuel Repository and for the extension of SFR have been utilized in SE-SFL. In order to have a realistic and consistent description of a site for geological disposal of radioactive waste, data from the Laxemar site in Oskarshamn municipality (see Figure 1-3), for which a detailed and coherent dataset exists, are used. Based on an initial hydrogeological analysis for SE-SFL, the example location for the SFL repository was selected to be a part of the rock volume that was earlier found most suitable for a potential Spent Fuel Repository within the Laxemar site (SKB 2011b).

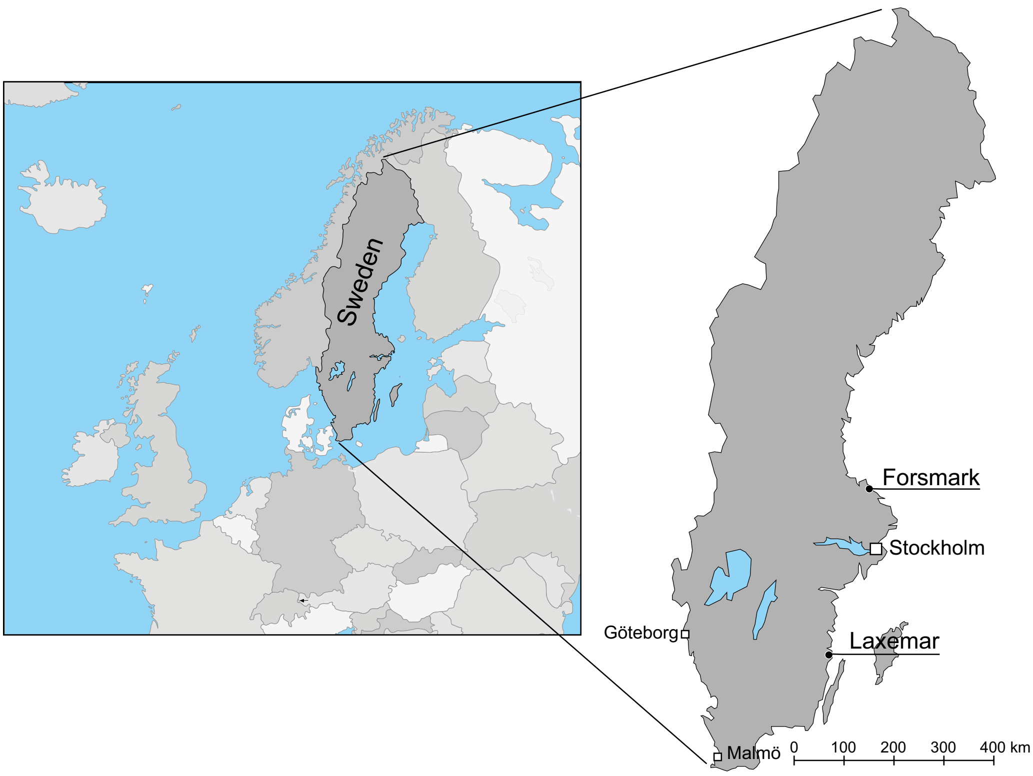


***Figure 1-3.** Map showing the location of Laxemar and Forsmark. Data from the site investigations in Laxemar, along with the data from the SR-Site and SR-PSU assessments from Forsmark, are used in SE-SFL in order to have a realistic and consistent description of a site for geological disposal of radioactive waste, for which a detailed and coherent dataset exists.*

SE-SFL is further developed in comparison to the previous assessments, which were mentioned in Section 1.1. Important improvements are an updated inventory and more elaborate account of internal and external processes. Moreover, the biosphere was, in the preliminary assessment, handled in a simplified manner, whereas it is handled in a more realistic way in SE-SFL. The availability of data from the Spent Fuel Repository site investigations also allows for more detailed representations of the geosphere. In general, SKB's experiences with safety analysis work have led to many developments since the late 1990s.

1.2.1 The SE-SFL report hierarchy

The **Main report** and main references in SE-SFL are listed in Table 1-1, also including the abbreviations by which they are identified in the text (abbreviated names in bold text). It can be noted that there are no dedicated process reports for SE-SFL. The SFR and SFL waste and repository concepts have many similarities, for instance the use of similar barrier materials and thus similar process interactions with the surrounding bedrock environment (Section 2.5.4 of the **Main report**). Therefore, the descriptions of internal processes for the waste (SKB 2014a) and the barriers (SKB 2014b) in SR-PSU are used in SE-SFL. For the bedrock system, the descriptions of internal processes for the geosphere in SR-Site (SKB 2010a) and SR-PSU (SKB 2014c) are used. There are also several additional references, which include documents compiled within SE-SFL. But there are also references to documents that have been compiled outside of the project, either by SKB or other similar organisations, or are available in the scientific literature. Additional publications and other documents are referenced in the usual manner. In Figure 1-4, the hierarchy of the **Main report**, main references and additional references within SE-SFL is shown.

Table 1-1. Main references in SE-SFL.

Abbreviation used when referenced in this report	Text in reference list
Main report	**Main report**, **2019.** Post-closure safety for a proposed repository concept for SFL. Main report for the safety evaluation SE-SFL. SKB TR-19-01, Svensk Kärnbränslehantering AB.
Biosphere synthesis	**Biosphere synthesis**, **2019.** Biosphere synthesis for the safety evaluation SE-SFL. SKB TR-19-05, Svensk Kärnbränslehantering AB.
Climate report	**Climate report**, **2019.** Climate and climate-related issues for the safety evaluation SE-SFL. SKB TR-19-04, Svensk Kärnbränslehantering AB.
FEP report	**FEP report**, **2019.** Features, events and processes for the safety evaluation SE-SFL. SKB TR-19-02, Svensk Kärnbränslehantering AB.
Initial state report	**Initial state report**, **2019.** Initial state for the repository for the safety evaluation SE-SFL. SKB TR-19-03, Svensk Kärnbränslehantering AB.
Radionuclide transport report	**Radionuclide transport report**, **2019.** Radionuclide transport and dose calculations for the safety evaluation SE-SFL. SKB TR-19-06, Svensk Kärnbränslehantering AB.

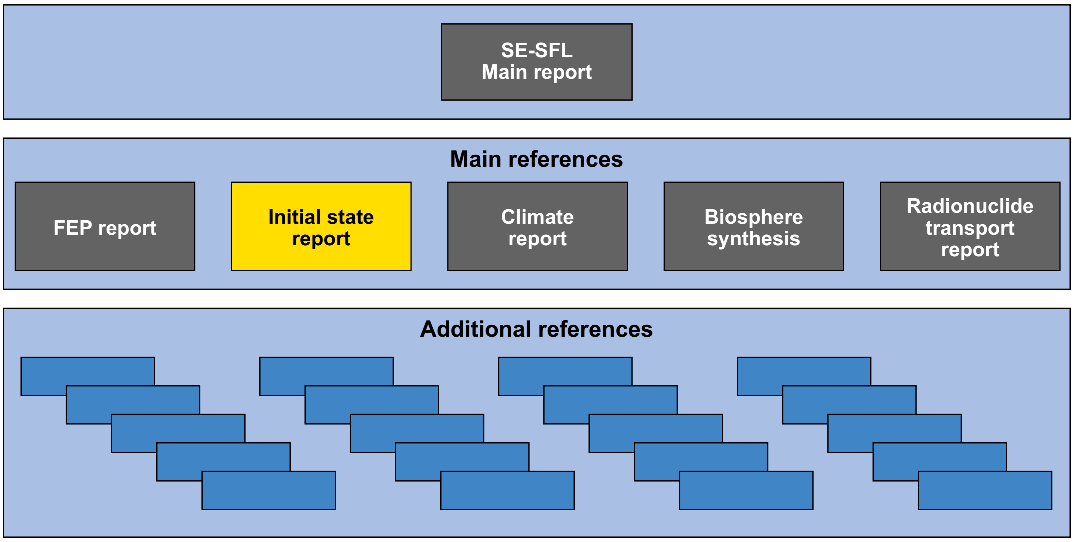


Figure 1-4. *The hierarchy of the Main report, main references and additional references in the safety evaluation of post-closure safety SE-SFL. The additional references either support the Main report or one or more of the main references.*

1.3 The role of this report in SE-SFL

This report describes the waste and design features for an appropriate initial state of the future SFL, relevant for the evaluation of post-closure safety for the proposed repository concept for SFL. The evaluation of post-closure safety for the proposed repository concept for SFL is performed according to a methodology comprising nine main steps, which are carried out partly concurrently and partly consecutively. The detailed evaluation methodology is described in Chapter 2 in the **Main report**.

1.4 Definition of the initial state

The initial state for the safety evaluation of SFL is defined as the expected state of the repository and its environs immediately after closure. This report describes the initial state of the repository which is based on the current knowledge on the properties of the waste and the repository components, and, an assessment of changes in these properties up to the time of closure. The assumed year of closure is 2075.

1.5 Structure of this report

This report comprises eight chapters. Following is a brief description of the contents:

Chapter 1 – Introduction. This chapter describes the background and the role of the report. Furthermore, definitions are given and explanation of abbreviations.

Chapter 2 – General considerations. In this chapter it is explained how different data in the **Initial state report** have been obtained and specified.

Chapter 3 – Waste. Descriptions of waste types, packaging, the allocation of waste packages to waste vaults, material quantities and radionuclide inventory in the waste packages are given in this chapter.

Chapters 4 and 5 – Waste vaults. In these chapters, designs and design considerations for the two waste vaults are described. In addition, the main dimensions are given in tables.

Chapter 6 – Closure components. An overview of suggested measures for the closure of SFL is given in this chapter.

Chapter 7 – Conditions at repository closure. The expected condition of the different system components at repository closure is defined by going through the lists of variables (parameters) that are required to describe the properties and condition of each system component.

Chapter 8 – Summary and conclusions.

1.6 Abbreviations used in this report

Terms and abbreviations used in this report are listed in Table 1-2.

Table 1-2. Explanations of abbreviations.

Term or abbreviation	Description
AB SVAFO	Company managing the Swedish legacy waste. Originally formed by Sydkraft, Vattenfall, Forsmark and OKG. Now owned by Forsmarks Kraftgrupp AB, Ringhals AB and OKG Aktiebolag.
BHA	Vault for legacy waste from the early research in the Swedish nuclear programmes, and smaller amounts of waste from medicine, industry and research.
B1 and B2	Barsebäck reactors 1 and 2.
BHK	Vault for reactor internals.
BWR	Boiling water reactor.
Clab	Central interim storage for spent fuel in Oskarshamn municipality.
Clink	Central interim storage and encapsulation plant for spent fuel (Clab is one part of Clink).
Conditioning	Those operations that produce a waste package suitable for handling, transport, storage and/or disposal (IAEA 2007).
Connecting tunnel	General term used for tunnels outside waste vaults.
ESS	European Spallation Source. Research facility in Lund municipality.
FEPs	Features, Events and Processes.
F1, F2 and F3	Forsmark reactor 1, 2 and 3.
Initial state	The state of the SFL repository and its environs immediately after closure.
KTL	Nuclear Activities Act (SFS 1984:3) (Kärntekniklagen).
Macadam	Macadam is crushed rock sieved in fractions 2–65 mm. Macadam has no or very little fine material (grain size < 2 mm). The fraction is given as intervals, for example “Macadam 16–32” is crushed rock comprising the fraction 16–32 mm.
NPP	Nuclear Power Plant.
O1, O2 and O3	Oskarshamn reactors 1, 2 and 3.
PWR	Pressurised water reactor.
QA	Quality assurance.
Repository and its environs	Broadly defined as the deposited radioactive waste and the surrounding packaging, the engineered barriers surrounding the waste packages, the host rock and the biosphere in the proximity of the repository.
R1, R2, R3 and R4	Ringhals reactors 1, 2, 3 and 4.
RPV	Reactor pressure vessel.
SFL	Repository for long-lived waste.
SFR	Repository for short-lived radioactive waste in Östhammar municipality.
SSM	Swedish Radiation Safety Authority (Strålsäkerhetsmyndigheten).
SSL	Radiation Protection Act (SFS 1988:220) (Strålskyddslagen).
System component	A physical component of the repository and its environs.
SNAB	Studsvik Nuclear AB.
SKB	Swedish Nuclear Fuel and Waste Management Company.
Waste form	The physical and chemical form after treatment and/or conditioning (IAEA 2007).
Waste package	Includes waste form and packaging .
Waste packaging	The outer barrier protecting the waste form. Includes the assembly of components (e.g. absorbent materials, spacing structures, radiation shielding, service equipment, etc) (IAEA 2007).
Waste type	In order to systematically classify the waste, different waste types have been defined and a code system developed.
Waste vault	Part of repository where waste is stored.

2 General considerations

2.1 General

In this section it is explained how different data in the initial state report were obtained and specified, and the QA measures that are foreseen to be taken to ensure the post-closure safety of the repository and its environs.

As SFL is in the planning stages, the description of the initial state is based on the proposed design (Elfwing et al. 2013). The initial state description is based on certain assumptions. The uncertainties in these assumptions are mainly associated with waste quantities in the repository, the radionuclide inventory and closure measures. For example, future waste quantities are based on forecasts and the technical solutions that will be used have not yet been finalised. Assumptions made in the safety evaluation and uncertainties relating to these assumptions are summarised in the **Main report**.

In 2016, the ownership of the facilities at the Studsvik site changed to also include Cyclife Sweden AB, as reflected in the general descriptions in Section 1.1. The main references to the present report were however already finished at that time. Hence, in the remainder of this report, the waste from Cyclife Sweden AB is included in the fraction from SNAB. The total amount of waste and the inventory of radionuclides from the Studsvik site is however intact and is therefore included in this initial state compilation.

2.2 Laws and regulations

The repository for long-lived low- and intermediate-level waste shall conform to the requirements of relevant laws and regulations.

The international treaties and national laws and regulations relevant for the design of a final repository for low- and intermediate-level radioactive waste are the following.

International treaty:

- Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste (SÖ 1999:60).
- EU Council Directive 2009/71/EURATOM (EU 2009).
- EU Council Directive 2011/70/EURATOM (EU 2011).

National laws:

- Nuclear Activities Act, KTL (SFS 1984:3).
- Radiation Protection Act, SSL (SFS 1988:220).

Regulations:

- Regulations concerning safety in nuclear facilities (SSMFS 2008:1).
- Regulations concerning physical protection of nuclear facilities (SSMFS 2008:12).
- Regulations concerning safety in connection with the disposal of nuclear material and nuclear waste (SSMFS 2008:21).
- Regulations on the protection of human health and the environment in connection with the final management of spent nuclear fuel and nuclear waste (SSMFS 2008:37).

In addition, the following laws must be followed during design, construction and operation of the repository:

- Environmental Code, MB (SFS 1998:808).
- Planning and Building Act, PBL (SFS 2010:900).
- Work Environment Act, AML (SFS 1977:1160).

2.3 System components

The repository and its environs are broadly defined as the deposited radioactive waste and the surrounding packaging, the engineered barriers surrounding the waste packages, the host rock and the biosphere in the proximity of the repository.

According to the **FEP** (Features, Events and Processes) **report** for the safety evaluation of SFL, the repository can be divided into a number of systems – i.e. the physical components that are described in the following section.

The waste vault for reactor internals, BHK, consists of several system components:

- Waste form, i.e. waste including conditioning material.
- Waste packaging, surrounding the waste form.
- Concrete structures.
- Grouting.
- Backfill of concrete.

The waste vault for legacy waste, BHA, also consists of several system components:

- Waste form, i.e. waste including conditioning material and when applicable the inner packaging.
- Waste packaging, surrounding the waste form.
- Concrete structures.
- Grouting.
- Backfill of bentonite clay.

Other system components in the underground facility are:

- Closure components.

In addition, the bedrock surrounding the repository and the surface environment in the repository area are defined as system components in the **FEP report**. The initial state for these components is described in the **Main report**.

2.4 System components and their functions

The primary safety function of the proposed concept is *retardation*. Retardation is achieved by limiting the groundwater flow through the waste by using large amounts of a material with low hydraulic conductivity, making diffusion the predominant transport process for radionuclides. Furthermore, diffusive transport between the waste and the surrounding rock is limited using concrete and bentonite in which the diffusive transport of radionuclides is very slow.

In addition, the use of concrete will create an alkaline environment, resulting in low corrosion rates for steel, which limits the release rate of activation products bound in metal. The concrete and bentonite also provide sorption sites for many of the radionuclides present in the waste. In addition, the bentonite can filter colloids to a substantial extent.

In the vault for reactor internals (BHK), the waste is surrounded by concrete system components. In the vault for legacy waste (BHA), the waste is surrounded by concrete and bentonite system components.

Both BHK and BHA will contain a large amount of different steels, which under anaerobic conditions will contribute to a low redox potential in the repository.

In addition to this, the natural barrier – the bedrock – will contribute to the primary safety function retardation by providing chemically favourable conditions, favourable hydrological and transport conditions, mechanically stable conditions and thermally favourable conditions. Concrete will be used as a construction material in the waste vaults, thus, sorption on the cement minerals in this concrete will also limit radionuclide releases.

The design principles of the barriers in the two waste vaults are summarised by Elfwing et al. (2013).

2.5 Inspection and control

Methods for testing and inspecting the materials and methods used during construction of SFL will be developed and defined during the detailed design phase of the repository. The control programme is to ensure that the repository will reach its desired initial state and retain the required post-closure properties. The aim is furthermore to confirm that the system components that have a function for the post-closure safety will comply with the requirements.

Inspection and control of the waste is described in Section 3.3.3.

3 Waste

3.1 General

Long-lived low- and intermediate-level waste from the NPPs typically consists of neutron-activated components from inside or close to (0.5–1.0 m) the reactor core. For most of these components, only a limited part is close enough to the core to accumulate a significant level of induced activity.

For the remaining parts of these components, i.e. the parts further away from the reactor core, the major source of activity is surface contamination. These parts can therefore be separated from the parts with significant levels of induced activity, cleaned and deposited in SFR. This procedure has been taken into account when the amount of waste for disposal in SFL has been estimated. However, since details on how to separate the long-lived from short-lived waste have not yet been specified, the total activity calculated for each component has been attributed to the parts that will be deposited in SFL.

The waste from SNAB and Cyclife to SFL comprises existing and future waste from research carried out at the Studsvik site as well as waste from decommissioning. Furthermore, wastes collected from other producers of radioactive materials in Sweden such as hospitals, industries and research facilities are included in the presented quantities.

The waste from AB SVAFO comprises legacy waste from the early operations at the Studsvik site as well as from Swedish military research. Another fraction of the waste from AB SVAFO originates from the operation and decommissioning of the Swedish research reactor R1, a research reactor at the Royal Institute of Technology in Stockholm, and the R2 and R2-0 research reactors at the Studsvik site. Also, the decommissioning waste from the remaining facilities is included in the presented waste quantities.

Decommissioning waste from the Ågesta reactor is also handled by AB SVAFO and will be deposited in SFL. The legacy waste from AB SVAFO is already conditioned and information about the radionuclide content as well as the content of other materials is limited.

Since the plans for the construction and operation of the European Spallation Source, ESS, are not yet finalised, only limited information is available concerning amount and composition of the waste from ESS.[1] The most active waste from ESS will be components that have been directly irradiated by protons and neutrons and other secondary particles. Some of the components will be exchanged at regular intervals during the operational period, thus giving rise to operational waste, whereas others will not be disposed of until the decommissioning phase. Due to the aforementioned uncertainties, the waste volume and amount of material from ESS are not taken into account in the safety evaluation of SFL. Instead, the list of radionuclides from ESS[1] has been screened for radionuclides that are specific to this waste (Crawford 2019). The resulting radionuclides have also been evaluated in the analysis, see Table 3-28 in Section 3.8.3.

3.2 Waste considerations

3.2.1 Waste acceptance criteria

There are no *waste acceptance criteria* (WAC) to date for the waste to be deposited in SFL. Therefore, the SFL waste that is being produced is not allowed to be finally conditioned i.e. it must be retrievable in order to make sure that the future WAC can be fulfilled.

3.2.2 Waste type descriptions

SKB has developed a system to facilitate the administrative handling of waste. The waste is divided into waste types and so-called *waste type descriptions* have been used since the late 1980s with the purpose of documenting the waste deposited in SFR. The waste type descriptions include an overview of

[1] **Persson P, 2018.** Characteristics of radioactive waste from ESS to be disposed in SKB facilities. Technical Report, Document number ESS-0036701, Revision 2(3), SKBdoc 1701742 ver 1.0, Svensk Kärnbränslehantering AB. Internal document.

the origin of the waste and the handling sequence, detailed descriptions of properties and characteristics of the waste, including the material codes, type of packaging and treatment methods etc. This includes a description of production data, results from investigations and calculations as well as control methods used. Descriptions of the controls on packaging, the waste form and the waste package are also given. Waste type descriptions will also be determined for each waste type to be disposed in SFL before disposal starts.

3.3 Handling and control of waste

The handling of the waste will include the following steps:

Production of the waste package

The production of the waste package will be performed according to the specified waste type description. The waste type description also includes other instructions such as the prescription for stabilisation with e.g. cement grout or concrete.

Storage

The waste type descriptions give details of the storage plans. They may also give instructions on, for example, transport to the storage site.

Transport

A system for transport of the waste will be set up before the repository is taken into operation.

Waste database register

Each waste package will be registered in a database and given a unique identification number which will be used in quality assurance procedures. After disposal the exact disposal position of the waste package will be registered in the waste register together with information about the waste content in the package.

3.3.1 Handling and control at the waste producer

The waste producers are responsible for:

- Waste conditioning and producing waste packages.
- Documenting production data.
- Measuring the activity of samples and/or measuring the surface dose rate of the waste package.
- Assigning a unique identity to the waste package marked on the waste packaging.
- Documentation in the shipping document.

The waste type descriptions are required to include details on the quality control procedures for the packaging, waste form and waste package. The waste producers are required to ensure that the producer of the packaging has a satisfactory programme for quality control. The quality of the waste form is primarily controlled by the surveillance of production, including both technical and administrative routines that influence the properties of the waste form. The producer must ensure that the activity content of and dose rates around the final waste package are measured and are within the specified guidelines.

3.3.2 Waste audits

SKB performs a quality audit of the waste handling at each nuclear facility every third year.

Every nuclear facility is required to have routines and instructions to ensure that each waste category is placed in the correct type of waste package and sent to the designated repository. Waste designated for SFL is yet not permitted to be finally conditioned and should be stored retrievably.

3.3.3 Handling and control at SKB

Before operations start in SFL, SKB will specify a programme for handling and control of the waste destined for SFL. This programme will comprise at least the following components:

- Determination of identity data and characteristics concerning e.g. contents of radionuclides and inspection of the shipping document.
- Measurement of surface dose rate.
- Measurement/control of surface contamination.
- Inspection for any signs of mechanical damage.

3.3.4 Waste register

Table 3-1 displays the information stored in the waste register for the SFR waste. Most likely the SFL waste register will contain the same type of information.

Table 3-1. Information stored in the waste register for SFR.

Information
Waste package ID
Waste type
Package type code
Waste material code
Package weight
Date of production
Nuclide content and total activity
Surface dose rate
Dose rate at 1 m
Date of measurement
Special information from the producer
Position in SFR after disposal

3.4 Waste packaging

A large portion of the waste scheduled for SFL, especially waste for BHA, has already been produced and is currently stored in different types of waste packaging which need to be handled by the SFL handling and transport systems. In this section, details on existing waste packaging and waste packaging planned for SFL are presented.

3.4.1 Waste packaging for BHK

Today, large steel tanks are used for storage of the core components from maintenance of the nuclear power plants. A handling system has been developed that allows for loading of the tanks under safe conditions in the reactor hall (SKB 2007, 2010b). It is planned to use these tanks also for final disposal of the waste for BHK.

The outer dimensions of the steel tanks are 3.3 x 1.3 x 2.3 m^3 (*length x width x height*) giving an outer volume of about 10 m^3. There are tanks available with wall thicknesses of 50, 100, 150 or 200 mm. The wall thickness is chosen to comply with the requirements determined by the activity and dose rate of the waste. Except for T50, i.e. the tank with 50 mm thick walls, the thickness of the bottom and lid is less than that of the walls. Additional shielding is in those cases provided by the bottom and top plate of the cassette in the tank. The internal dimensions and weights are shown in Table 3-2.

The waste is loaded in a cassette which is placed in the steel tank, see Figure 3-1. The cassette is designed for a loading of 12 tonnes of waste. However, experience from the nuclear power plants in Oskarshamn and Forsmark indicates that most of the tanks contain only about 3 tonnes with a maximum of 6–7 tonnes. In order to improve the packing density additional segmentation of the wastes would be required.

Both the cassettes and the steel tanks are made of carbon steel.

Table 3-2. Main data for steel tanks with various wall thicknesses (Pettersson 2013, Table 4-1).

Type of tank	T50	T100	T150	T200
Inner length (m)	3.2	3.1	3.0	2.9
Inner width (m)	1.2	1.1	1.0	0.9
Inner height (m)	2.2	2.2	2.2	2.15
Inner volume (m^3)	8.45	7.50	6.60	5.61
Outer length (m)	3.3	3.3	3.3	3.3
Outer width (m)	1.3	1.3	1.3	1.3
Outer height (m)	2.3	2.3	2.3	2.3
Outer volume (m^3)	9.87	9.87	9.87	9.87
Tank weight, empty (tonnes)	10.25	18.51	25.60	33.20
Cassette weight (tonnes)	2.82	3.57	5.46	5.81
Max waste weight (tonnes)	12.0	12.0	12.0	13.5
Max weight of tank, cassette and waste (tonnes)	25.1	34.1	43.1	52.5
Max total tank weight including grout (tonnes)	38.2	45.1	51.8	58.8

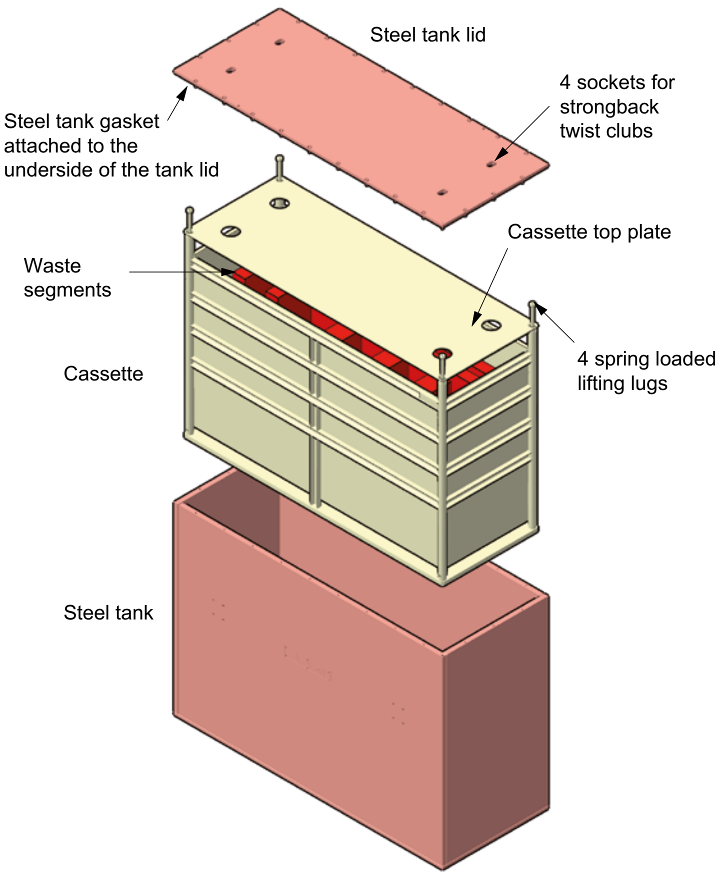


Figure 3-1. *Schematic illustration of the steel tank for neutron activated components. The steel tank has external dimensions 3.3 x 1.3 x 2.3 m^3 (length x width x height). The waste is placed in a cassette which is placed in the tank. A lid is bolted to the tank. The thickness of the steel walls is chosen to comply with the requirements determined by the radioactivity and dose rate of the waste.*

3.4.2 Existing waste packaging for BHA

Most of the legacy waste currently stored at the Studsvik site is stored in 200-litre drums containing a mixture of waste and concrete. To provide for safe storage and handling, these drums have recently been placed in new 280 litre protection drums. Solidified sludge is stored in steel drums containing mixers. Ashes, trash and scrap metal is stored in steel drums with inner 100 litre steel drums surrounded by concrete. Data on the different types of steel drums are presented in Table 3-3.

Some of the waste at the Studsvik site is also stored in steel and/or concrete moulds, see Tables 3-4 and 3-5 for details.

Table 3-3. Representative data for the steel drums used at SNAB and AB SVAFO.

	Value	Reference/Comment
200 L drum (SIS 846202)		
Height (m)	0.84	(Waste type description S.13)
Diameter (m)	0.57	(Waste type description S.13)
Thickness of walls (mm)	1	(Waste type description S.13)
Outer volume (m^3)	0.21	Calculated from dimensions above
Inner volume (m^3)	0.21	Chosen to be the same as outer volume
Disposal volume (m^3)	0.32	Calculated (4 drums stored on tray)
Outer surface area (m^2)	2.0	Calculated from dimensions above
Inner surface area (m^2)	2.0	Chosen to be the same as outer surface area
Steel weight (kg)	20	(Waste type description S.13)
100 L drum		
Height (m)	0.70	(Waste type description S.13)
Diameter (m)	0.46	(Waste type description S.13)
Thickness of walls (mm)	1	(Waste type description S.13)
Outer volume (m^3)	0.12	Calculated from dimensions above
Inner volume (m^3)	0.12	Chosen to be the same as outer volume
Outer surface area (m^2)	1.0	Calculated from dimensions above
Inner surface area (m^2)	1.0	Chosen to be as outer surface area
Steel weight (kg)	10	(Waste type description S.13)
280 L protection drum		
Height (m)	0.93	(Information from AB SVAFO)
Diameter (m)	0.68	(Information from AB SVAFO)
Thickness of walls (mm)	1.5	(Information from AB SVAFO)
Outer volume (m^3)	0.34	Calculated from dimensions above
Inner volume (m^3)	0.33	Calculated from dimensions above
Disposal volume (m^3)	0.60	(Pettersson 2013)
Outer surface area (m^2)	2.7	Calculated from dimensions above
Inner surface area (m^2)	2.7	Chosen to be the same as outer surface area
Steel weight (kg)	36	(Information from AB SVAFO)
Tray for four 200 L drums		
Thickness (mm)	4	(Pettersson 2013)
Length (m)	1.2	(Pettersson 2013)
Width (m)	1.2	(Pettersson 2013)
Weight (kg)	66	(Pettersson 2013)

Table 3-4. Representative data for concrete moulds (SKB 2014d).

	Value
Height (m)	1.2
Width (m)	1.2
Length (m)	1.2
Thickness of walls (m)	0.1
Outer volume (m^3)	1.7
Inner volume (m^3)	1.0
Disposal volume (m^3)	1.7
Outer surface area (m^2)	8.6
Inner surface area (m^2)	6.0
Concrete, volume (m^3)	0.7
Concrete, weight (kg)	1 840
Reinforcement, weight (kg)	274
Reinforcement, area (m^2)	11.8
Reinforcement, diameter (m)	0.012

Table 3-5. Representative data for steel moulds with concrete lids (SKB 2014d).

	Value
Height (m)	1.2
Width (m)	1.2
Length (m)	1.2
Thickness of walls (mm)	5
Outer volume (m^3)	1.7
Inner volume (m^3)	1.7
Disposal volume (m^3)	1.7
Outer surface area (m^2)	7.2
Inner surface area (m^2)	7.2
Steel weight (kg)	400

3.4.3 New waste packaging for BHA

The current plan is to dispose of the waste to SFL according to the packaging descriptions below, which comprises overpacks for standard moulds, standard 200-litre drums and 280-litre protection drums.

Waste packaging for standard moulds

The waste packaging for four standard moulds consists of a welded framework of square tubes, with sides made of corrugated steel panels that are designed to withstand the forces from grouting of the inner waste packages. The bottom plate consists of a flat plate with stiffeners arranged like a cross inside the waste packaging, see Figure 3-2. The stiffening plates also act as guides for the moulds when they are placed in the waste packaging. Guides for the moulds are also provided along the inner walls of the waste packaging.

In order to facilitate grouting, the waste packaging has been designed to fulfil the following conditions:

- Distance between moulds 50 mm
- Distance mould to waste-packaging wall 20 mm
- Distance from top of mould to top of waste packaging 80 mm

Figure 3-2. *3D-view of a waste packaging for standard moulds when empty. The stiffening plates for the bottom plate and guides for the moulds are visible.*

The waste packaging for moulds is assumed to be filled with grout to the top of the waste packaging. No steel lid will be used. The grout surface will be level with the top of the steel frame. Reinforcement bars will be placed on top of the moulds to prevent cracking of the top-most layer of the grout, see Figure 3-3.

Representative data for the waste packaging for standard moulds are shown in Table 3-6.

Figure 3-3. *3D-view of a waste packaging loaded with standard moulds. Reinforcement bars that will stabilise the future grout on top of the moulds are also shown.*

Table 3-6. Representative data for the waste packaging for standard moulds (Pettersson 2013, Table 5-1).

	Value	Comment
Outer height, without lifting brackets, (m)	1.296	
Outer width, without lifting brackets, (m)	2.690	
Outer length, without lifting brackets, (m)	2.690	
Inner height (m)	1.284	
Inner width (m)	2.49	
Inner length (m)	2.49	
Thickness of walls (mm)	4	
Thickness of bottom plate (mm)	12	
Outer volume (m^3)	9.38	
Inner volume (m^3)	8.60	
Total volume of moulds (m^3)	6.91	
Volume of grout (m^3)	1.69	Difference between the inner volume of waste packaging and outer volume of four moulds
Outer surface area (m^2)	21	Calculated from dimensions above
Inner surface area (m^2)	21	Chosen to be the same as outer surface area
Weight of empty waste packaging (kg)	1755	
Weight of waste, including moulds (kg)	20000	
Weight of grout (kg)	3377	Calculated for a grout density of 2000 kg/m^3
Total weight (kg)	25132	Calculated from weights

Waste packaging for standard 200-litre drums

The design of the waste packaging for standard 200-litre drums is like that of the waste packaging for standard moulds. The difference is the height of the waste packaging which is adapted to the height of the drums. To enable the handling of the drums in a rational manner, they are placed on trays – four drums on each tray – which are then placed in the waste packaging. For that reason, the guides are adapted for the trays, see Figure 3-4. The nominal distance between the trays is about 40 mm in order to facilitate grouting. The waste packaging for standard 200-litre drums will be filled with grout to the top of the waste packaging. No steel lid will be used. The grout surface will therefore be level with the top of the steel frame. Reinforcement bars will be placed on top of the drums to prevent cracking of the top-most layer of the grout, see Figure 3-5.

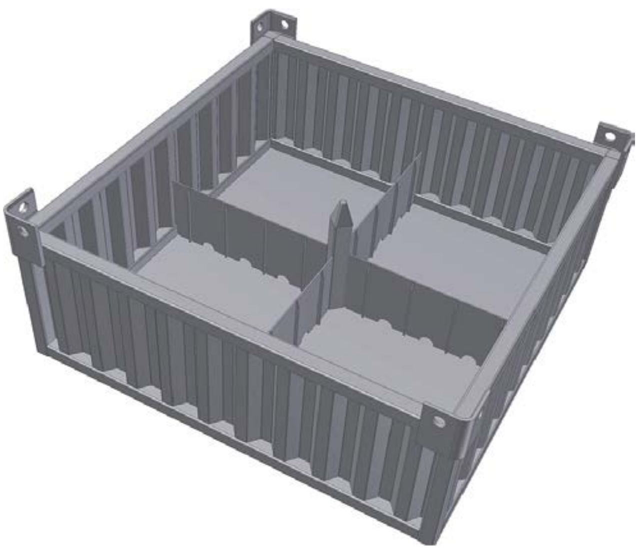


***Figure 3-4.** 3D-view of an empty waste packaging for standard 200-litre drums, showing the stiffening plates for the bottom plate and guides for the trays.*

Figure 3-5. *3D-view of a waste packaging loaded with four trays with four drums each. Reinforcement bars that will stabilise the future grout on top of the drums are also shown.*

Representative data for the waste packaging for standard 200-litre drums are shown in Table 3-7.

Table 3-7. Representative data for the waste packaging for standard 200-litre drums (Pettersson 2013, Table 5-2).

	Value	Reference/comment
Outer height, without lifting brackets, (m)	0.980	
Outer width, without lifting brackets, (m)	2.690	
Outer length, without lifting brackets, (m)	2.690	
Inner height (m)	0.968	
Inner width (m)	2.49	
Inner length (m)	2.49	
Thickness of walls (mm)	4	
Thickness of bottom plate (mm)	12	
Outer volume (m^3)	7.09	
Inner volume (m^3)	6.48	
Total volume of waste drums (m^3)	3.33	
Volume of grout	3.15	Calculated from inner volume and volume of drums
Outer surface area (m^2)	15.82	Calculated from dimensions above
Inner surface area (m^2)	15.82	Chosen to be the same as outer surface area
Weight of empty waste packaging (kg)	1602	
Weight of waste drums (kg)	8000	
Weight of grout (kg)	6312	Calculated from the volume and density of 2000 kg/m^3
Total weight (kg)	15914	Calculated from weights

Waste packaging for 280-litre protection drums

The design of the waste packaging for 280-litre protection drums is like that of the waste packaging for moulds and standard 200-litre drums. The differences are the height of the waste packaging and the inner dimensions, which are adapted to the height and diameter of the 280-litre protection drums, see Figures 3-6 and 3-7. It should be noted that before grouting of the waste, the lids of the 280-litre protection drums will be removed in order to fill the void between the 280-litre protection drums and the 200-litre drums containing the waste with grout. The waste packaging for 280-litre protection drums will be filled with grout to the top of the waste packaging. No steel lid will be used. The grout surface will therefore be level with the top of the steel frame. Reinforcement bars will be placed on top of the drums to prevent cracking of the top-most layer of the grout, see Figure 3-7.

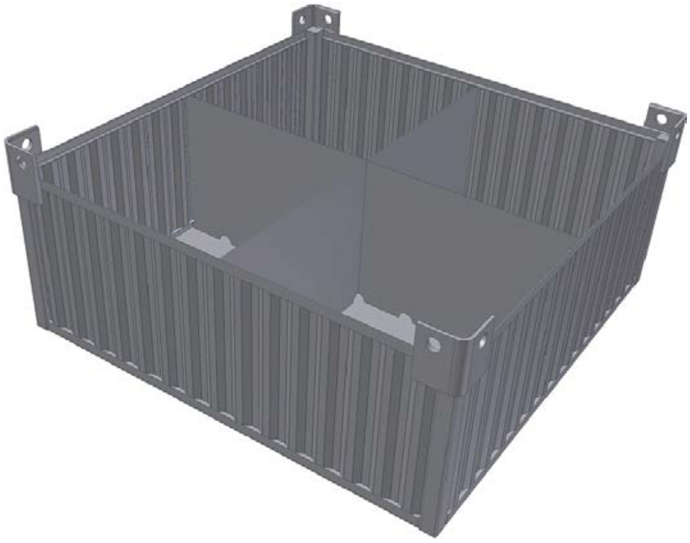


Figure 3-6. *3D-view of the waste packaging for 280-litre protection drums when empty. The stiffening plates for the bottom plate and guides for the trays are visible.*

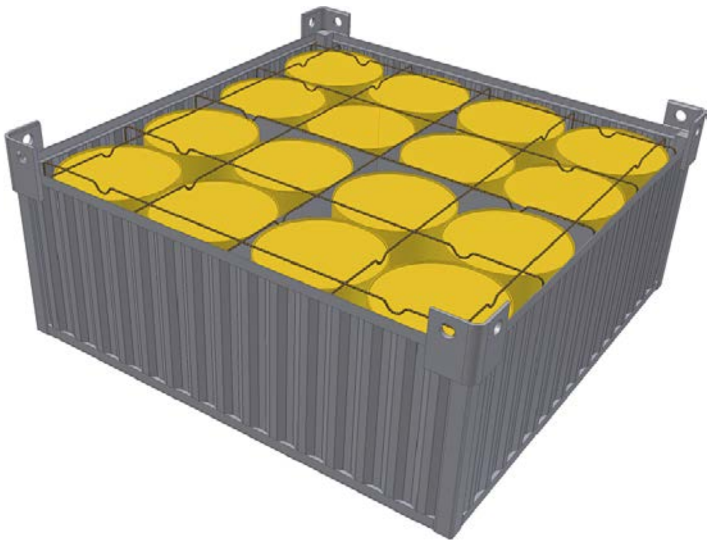


Figure 3-7. *3D-view of the waste packaging loaded with sixteen 280-litre protection drums. Reinforcement bars that will stabilise the future grout on top of the drums are also shown.*

Representative data for the waste packaging for 280-litre protection drums are shown in Table 3-8.

Table 3-8. Representative data for the waste packaging for 280-litre protection drums (Pettersson 2013, Table 5-3).

	Value	Reference/comment
Outer height, without lifting brackets, (m)	1.050	
Outer width, without lifting brackets, (m)	2.69	
Outer length, without lifting brackets, (m)	2.69	
Inner height (m)	1.038	
Inner width (m)	2.63	
Inner length (m)	2.63	
Thickness of walls (mm)	4	
Thickness of bottom plate (mm)	12	
Outer volume (m^3)	7.60	
Inner volume (m^3)	7.34	
Total volume of waste drums (m^3)	3.33	Waste drums = 200-litre drums in the 280-litre protection drums.
Void for grout (m^3)	4.02	Calculated from inner volume and volume of drums
Outer surface area (m^2)	16.95	Calculated from dimension above
Inner surface area (m^2)	16.95	Chosen to be the same as outer surface area
Weight of empty waste packaging (kg)	1480	
Weight of waste drums (kg)	8400	
Weight of grout (kg)	8031	Calculated from the volume and density of 2000 kg/m^3
Total weight (kg)	17911	Calculated from weights

3.5 Waste descriptions and distribution of waste in SFL

In SFL, the waste will be distributed between BHK and BHA. BHK will contain neutron-activated metallic waste from the Swedish nuclear power plants. BHA will contain waste generated or processed at the nuclear facilities at SNAB and the legacy waste from the Swedish nuclear programme stored at the facilities at AB SVAFO. For a description of the waste from ESS, see Section 3.1.

3.5.1 Waste for disposal in BHK

A large portion of the waste in the future SFL comes from the Swedish nuclear power plants. Radioactive waste is formed during nuclear fission in the reactor core with sequential neutron activation of the core material and material in close proximity of the core (Herschend 2014). The different components planned for disposal in BHK are described in the following sections.

Core shroud

The core shroud is an internal tank surrounding the core inside the BWR reactor pressure vessel. The core shroud is made of austenitic stainless steel and consists of a cylindrical part with an upper and lower flange. Weight and material specification for core shrouds from the different reactors are given in Table 3-9. For the F1/F2 and O1 reactors the core shrouds have been replaced during operation.

Table 3-9. Weight, material specification and number of core shrouds to be deposited in SFL. The reported weights correspond to the assumed activated part for each component. Data from Herschend (2014).

Reactor	Weight (kg)	Material	Number of components
B1/B2	24400	SIS2333	2
F1/F2	24000	SIS2353	4
F3	32000	SIS2352	1
O1	26600	SIS2353	2
O2	11000	SIS2333	1
O3	32000	SIS2352	1
R1	12300	SIS2333	1
Total	285300		12

Core grid

The core grid is mounted on top of the core shroud. Weight and material specification for core grids from the different reactors are given in Table 3-10.

Table 3-10. Weight, material specification and number of core grids to be deposited in SFL. The reported weights correspond to the assumed activated part for each component. Data from Herschend (2014).

Reactor	Weight (kg)	Material	Number of components
B1/B2	2500	SIS2353	4
F1/F2	8900	SIS2353	4
F3	6000	SIS2352	1
O1	4100	SIS2353	2
O2	2500	SIS2353	2
O3	6000	SIS2352	1
R1	7000	SIS2353	2
Total	84800		16

Core shroud head

The core shroud head lies directly on top of the core grid. It is made of austenitic stainless steel. Weight and material specifications for shroud heads from the different reactors are given in Table 3-11.

Table 3-11. Weight, material specification and number of core shroud heads to be deposited in SFL. The reported weights correspond to the assumed activated part for each component. Data from Herschend (2014).

Reactor	Weight (kg)	Material	Number of components
B1/B2	16500	SIS2352	4
F1/F2	19500	SIS2333	4
F3	31350	AISI 316L	2
O1	11000	AISI 316L	2
O2	14160	AISI 316L	3
O3	31350	AISI 316L	2
R1	25300	AISI 304L	2
Total	384480		19

Control rod guide tubes and control rods

The uppermost part of the BWR control rod tube together with the core plate at the top will be significantly neutron activated and will need to be disposed of as long-lived waste. The number of guide tubes in each reactor is the same as the reported number of control rods in Table 3-12.

BWR control rods are cruciform-shaped permanent absorbers used for power regulation. In the inventory, two types of control rods are accounted for. Control rods for power regulation during operation and control rods used during shutdown. The absorbing materials include boron carbide but also hafnium is used. The remaining weight of the control rod is treated as stainless steel. Table 3-12 shows the expected number of control rods and material quantities in the SFL waste.

Table 3-12. Material specification and estimated number of control rods to be deposited in SFL. The quantities only include the activated part of the control rods aimed for disposal in SFL. Data from Herschend (2014).

Reactor	Steel (kg per rod)	Hafnium (kg per rod)	Boron carbide (kg per rod)	Number of control rods
B1	73.2	3.2	10.2	218
B2	73.2	3.2	10.2	228
F1	73.2	3.2	10.2	555
F2	73.2	3.2	10.2	600
F3	74	3.2	10.2	814
O1	100.5	3.1	10.1	287
O2	81.7	3.1	10.1	453
O3	74	3.2	10.2	804
R1	74.1	-	10.5	467
Total	337 384	12 595	45 211	4 426

Detectors, guide tubes, boron plates and fuel assembly components

Several types of detectors are used to monitor the neutron flux in the BWRs. Most of the waste comes from local power range monitors (LPRM), which have fixed positions during reactor operation. There are additional detectors such as source range, intermediate range, short and intermediate range and wide range monitors (SRM, IRM, SIRM, and WRNM, respectively). There are also travelling in-core probes (TIP) that are movable inside the LPRM guide tubes. The detector type is not always specified for existing waste containing detector material.

Temporary absorbers in the form of boron steel plates have been used in the first cores of B1, O1, O2 and R1. Material and surface data for these has not been updated since the last reported reference inventory from 1998 (Lindgren et al. 1998).

In the existing long-lived waste, components from the fuel assemblies are present. These include fuel boxes, spacers, springs and handles. Both in terms of waste volume and activity, this waste category is dominated by fuel boxes. Table 3-13 shows the material composition of the fuel boxes as well as estimated weights of the boxes.

Table 3-13. Estimated material composition of fuel boxes to be deposited in SFL. The reported weights correspond to the contribution from each material to the waste mass of a single fuel box. Data from Herschend (2014).

Reactor	Steel (kg)	Zircaloy (kg)	Total (kg)
B1/B2	10.6	31.5	42.1
F1/F2	7.8	34.6	42.4
F3	10.6	31.5	42.1
O1	-	30.0	30.0

Miscellaneous scrap metal

In the present inventory there are small amounts (about 10.5 tonnes in total) of old scrap metal from reactor internals. The material composition varies, but in Herschend (2014) it is assumed to consist of stainless steel SIS2333. No activity has been attributed to this waste since all the activity originating from the core region is accounted for in other waste types.

In the waste there are also twelve start-up neutron sources from R1. No information about the material or radionuclide contents of these has been obtained. However, it is known that the waste is in the form of metal tubes (similar to PRM-tubes) and that the neutron source is Cf-252. It is assumed that there are 90 neutron sources present in the waste assigned to SFL.

PWR components and reactor pressure vessels

In terms of component weights and material, the three PWRs R2, R3 and R4 are considered identical. Table 3-14 summarises the weight and material composition of the PWR components per reactor in the long-lived waste including the RPVs themselves. In the initial state for BHK it is assumed that all PWR components and the RPVs are segmented before disposal.

Table 3-14. Weight and material specification for components from PWR to be deposited in SFL. The reported weights correspond to the expected waste mass from each component (Herschend 2014).

Component	Total weight per reactor (kg)	Weight of activated part per reactor (kg)	Weight considered as the initial state per reactor (kg)	Material
RPV	327 500	245 000	245 000	AISI 304 and carbon steel
Internals	135 500	94 000	94 000	AISI 304
Biological shield	588 000	370 318	588 000	Concrete and carbon steel (reinforcement)

The three Ringhals PWR reactor pressure vessels are cylinder shaped with an inner diameter of 4 m, and a total height of 13 m including the two half spheres at the top and bottom. The vessels are made from 200 mm thick carbon steel with an internal stainless steel cladding. The total weight of one RPV is 327 500 kg, out of which 245 000 kg is considered long-lived waste, see Table 3-14.

PWR internals consist of the core baffle, core barrel, thermal panels, upper internal assembly and the lower core plate. All components are made of AISI 304 stainless steel. The total weight of the PWR internals sums up to 135.5 tonnes out of which 94 tonnes is planned for disposal in SFL, see Table 3-14.

The biological shield is a reinforced concrete structure surrounding the RPV to protect the surroundings from gamma- and neutron radiation from the reactor core. The shield is made of reinforced concrete with a total weight of about 588 tonnes, 88 tonnes being steel reinforcement bars. All of this is planned for disposal in SFL, see Table 3-14.

Components from Ågesta

The Ågesta reactor is a pressurised heavy-water reactor (PHWR). The waste consists of the reactor pressure vessel including the head, thermal shields inside the reactor and control rods. Table 3-15 shows the different components, materials and weights.

Table 3-15. Weight and material specifications for components from the Ågesta PHWR to be deposited in SFL. The reported weights correspond to the expected waste. Data from Herschend (2014).

Component	Weight (kg)	Material
Reactor pressure vessel with lid and flange	140 200	2013-R3 with A24L cladding
Thermal shields and internals	125 000	A2MM
Coarse control rods	1 431	A24L and Ag/In/Cd
Fine control rods	380	A24L and Ag/In/Cd

Secondary waste

Secondary waste in the form of swarf has been reported for the segmentation of core components and reactor internals from Forsmark and Oskarshamn. A forecast of similar waste from the segmentation of reactor internals from Barsebäck has also been reported. This waste is accounted for in the volume of waste from core components. Forsmark and Oskarshamn have also reported amounts of actual secondary waste in the form of filters, abrasives and decommissioned segmentation equipment. This waste is stored in 1.2 x 1.2 x 1.2 m^3 moulds. The volume estimates for the secondary waste category have been based on the fact that the waste will be deposited in packaging with these dimensions.

Number of waste packages

The total number of steel containers to be deposited in BHK is 606, according to Herschend (2014, Table 5-9). In addition, 115 m^3 secondary wastes and 659 m^3 of the PWR biological shields are expected to be deposited in BHK, according to Herschend (2014, Table 5-3, 5-4 and 5-5).

3.5.2 Waste for disposal in BHA

Waste from SNAB and AB SVAFO

Low- and intermediate-level radioactive waste is currently stored at the Studsvik site. Some decommissioning waste that has been close to the core in the R2 reactor, such as control rods and other core components, is classified as long-lived and allocated to SFL. Equally, systems containing more than 10^{10} Bq C-14 are allocated to SFL (SKB 2014e).

The waste assigned to BHA consists of waste from research carried out at Studsvik and also of waste collected from other producers of radioactive materials in Sweden e.g. hospitals, industries and research facilities. The waste consists mainly of ion-exchange resins, precipitation sludge, exchanged components from the research reactor at Studsvik, tools, instruments, consumables, laboratory kit, ashes, glove boxes and radiation sources.[2] The amounts of materials in both the waste and the existing packaging in the different waste types are given in Table 3-16.

[2] **Herschend B, 2015**. Long-lived waste from AB SVAFO and Studsvik Nuclear AB. SKBdoc 1431282 ver 1.0, Svensk Kärnbränslehantering AB. Internal document.

Table 3-16. Quantities of different materials in the waste packages from SNAB and AB SVAFO to be deposited in BHA.[2]

Material	Waste (kg)	Existing packaging (kg)
Stainless steel	755 000	-
Zircaloy	2 400	-
Carbon steel	801 000	680 000
Zinc	200	-
Aluminum	164 000	-
Cadmium	1 000	-
Copper	700	-
Lead	2 900	27 000
Brass	500	-
Graphite	52 000	-
Textile	33 000	-
Paper	11 200	-
Wood	3 400	-
Plastic	87 600	-
Bakelite	100	-
Plexiglass	14 600	-
Rubber	4 800	-
Glass	1 700	-
Concrete	511 000	3 280 000
Misc. org.	78 100	-
Misc. inorg.	58 600	-
Ferrocyanide precipitate	7 300	-
Ashes	105 000	-
Ion-exchange resin	1 000	-
Thorium	2 700	-
Uranium	16 500	-
Plutonium	1	-
Mercury	100	-
Beryllium	300	-

Number of waste packages

The total number of waste packages to be deposited in BHA is 1 325.[3]

3.6 Total waste volumes for SFL

Waste for BHK

The total estimated deposition volume of long-lived waste from the Swedish NPPs has been calculated to about 6 800 m^3 (Herschend 2014, Table 3-17).

[3] **Herschend B, 2015**. Long-lived waste from AB SVAFO and Studsvik Nuclear AB. SKBdoc 1431282 ver 1.0, Svensk Kärnbränslehantering AB. (Table 5-23) Internal document.

Table 3-17. Total weight and volume for long-lived waste from the NPPs (Herschend 2014).

Waste type	Weight (tonnes)	Volume (m^3)
Core components BWR (existing)	386	1 048
Core components BWR (forecast)	83	203
Core components BWR (decom.)	588	1 392
Secondary waste	22	116
Control rods BWR (existing)	168	417
Control rods BWR (forecast)	323	791
Control rods (Ågesta)	2	5
RPV PWR	735	950
RPV PHWR (Ågesta)	140	188
Biological shield PWR	1 764	659
Reactor internals PWR	282	692
Reactor internals (Ågesta)	125	312
Total steel[1]	2778	
Total concrete	1764	
Sum (existing)	569	1 540
Sum (forecast)	414	1 036
Sum (decom.)	3 636	4 199
Total	4 619	6 774

[1] Excluding biological shield, secondary waste and the boron and hafnium in the control rods (according to Table 3-12).

Waste for BHA

The volume of the waste originating from SNAB and AB SVAFO sums up to approximately 4 140 m^3, occupying approximately 10 742 m^3 of repository storage volume, see Table 3-18.

Table 3-18. The total waste volume and the disposal volume for the waste in BHA.[2]

	Total (m^3)
Waste volume (waste package volumes)	4 140
Disposal volume (volume the waste packages will occupy in the vault)	10 742

3.6.1 Void and pore volumes

The void and pore volumes that are associated with the different waste vaults are presented in Table 3-19 and Table 3-20 respectively.

Table 3-19. Void and pore volumes (m^3) in different components of BHK.[4]

	Waste void	Waste grout pore volume	Caisson grout pore volume	Structure pore volume	Backfill pore volume	Backfill void	Total
BHK	1 200	800	800	300	11 900	0	15 000

Table 3-20. Void and pore volumes (m^3) in different components in BHA.[4]

	Waste void	Waste grout pore volume	Waste area grout (backfill)	Structure pore volumes	Backfill pore volume	Backfill void	Total
BHA	1 300	1 300	2 000	400	17 700	0	22 700

The void and pore volume in an individual waste package may vary due to the degree of filling, material composition and material quantities.

[4] **Herschend B, 2016**. Initial state SFL. SKBdoc 1441724 ver 1.0, Svensk Kärnbränslehantering AB. Internal document.

3.7 Total material quantities

3.7.1 Waste

The material inventory has been calculated from the reported and calculated waste amounts and the specification of construction materials for each component. Material specifications have been taken from the radionuclide inventory calculations for the decommissioning studies (Griffiths et al. 2008, Lindow 2012, Anunti et al. 2013, Larsson et al. 2013, Hansson et al. 2013) and are summarised in Herschend (2014). The material quantities that will be deposited in BHA are compiled[2] based on information and assumptions available from SNAB and AB SVAFO. Table 3-21 lists the material quantities that are planned for disposal in BHK and BHA, respectively.

Table 3-21. Estimated material quantities in the waste.

	Amount of material (tonnes)						
Material	Barsebäck	Forsmark	Oskarshamn	Ringhals	Ågesta	SNAB and AB SVAFO	Total
Stainless steel	200	570	541	439	266	755	2 770
Carbon steel	-	-	-	984	-	801	1 785
Boron steel	6	-	8	6	-	-	20
Boron carbide	5	20	16	5	-	-	46
Hafnium	1	6	5	-	-	-	13
Inconel	-	0.3	-	-	-	-	0.3
Zircaloy	-	-	17	-	-	2	19
Silver	-	-	-	-	1	-	1
Indium	-	-	-	-	0.2	-	0.2
Cadmium	-	-	-	-	0.1	1	1.1
Zinc	-	-	-	-	-	0.2	0.2
Aluminium	-	-	-	-	-	164	164
Copper	-	-	-	-	-	1	1
Lead	-	-	-	-	-	3	3
Brass	-	-	-	-	-	1	1
Uranium	-	-	-	-	-	17	17
Plutonium	-	-	-	-	-	0.001	0.001
Thorium	-	-	-	-	-	3	3
Mercury	-	-	-	-	-	0.1	0.1
Beryllium	-	-	-	-	-	0.3	0.3
Graphite	-	-	-	-	-	52	52
Textile	-	-	-	-	-	33	33
Paper	-	-	-	-	-	11	11
Wood	-	-	-	-	-	3	3
Plastic	-	-	-	-	-	88	88
Bakelite	-	-	-	-	-	0.1	0.1
Plexiglass	-	-	-	-	-	15	15
Rubber	-	-	-	-	-	5	5
Glass	-	-	-	-	-	2	2
Misc. org.	-	-	-	-	-	78	78
Misc. inorg.	-	-	-	-	-	59	59
Ferrocyanide precipitate	-	-	-	-	-	7	7
Ashes	-	-	-	-	-	105	105
Ion-exchange resins	-	-	-	-	-	1	1
Concrete	-	-	-	1 500	-	511	2 011
Misc. (filters etc)	-	1	21	-	-	-	22
Total	211	598	607	2 935	267	2 716	7 335

Surface area

The total surface area of the different metals in the waste has been calculated for BHK (Herschend 2014) and for BHA.[2] Table 3-22 shows the total surface area for the different metals.

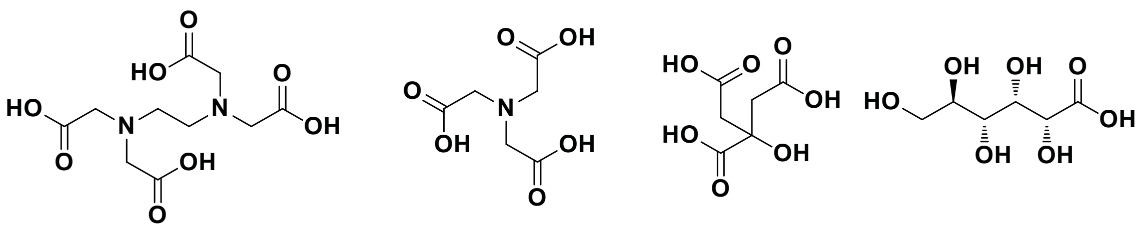
Table 3-22. Surface area (m^2) of different metals.

Material	BHK (Herschend 2014)	BHA[2]
Stainless steel	67 960	38 429
Carbon steel	1 313	40 763
Boron steel	7 472	-
Hafnium	946	-
Inconel	18	-
Zircaloy	1 797	146
Silver	7	-
Indium	1	-
Cadmium	0.4	44
Zinc	0	13
Aluminium	0	25 158
Copper	0	33
Lead	0	102
Brass	0	25

Organic complexing agents

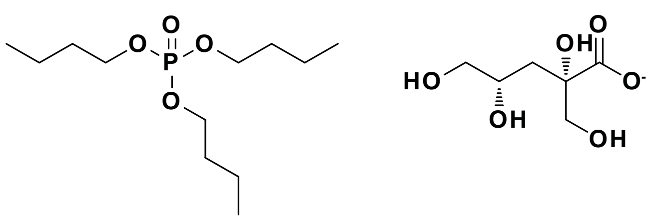
In BHK, the waste exclusively consists of metallic parts. Hence, no organic material that might form metal-organic complexes will be present in this waste vault. In BHA, the waste could possibly contain complexing agents such as ethylenediaminetetraacetic acid (EDTA), Nitrilotriacetic acid (NTA), citric acid and gluconate (Figure 3-8), which have been used as chelating agents in detergents through the years.

Another complexing agent that is potentially present in the legacy waste is tributyl phosphate (TBP) (Figure 3-9) that is used as a solution of tributyl phosphatein kerosene or dodecane for the liquid-liquid extraction (solvent extraction) of uranium-plutonium and thorium from spent uranium nuclear fuel rods dissolved in nitric acid.



EDTA **NTA** **Citric acid** **Gluconate**

Figure 3-8. *Chemical structure of ethylenediaminetetraacetic acid (EDTA), Nitrilotriacetic acid (NTA), citric acid and gluconate.*



TBP **α-isosaccarinate (ISA)**

Figure 3-9. *Chemical structure of tributyl phosphate (TBP) to the left and isosaccarinate (ISA) to the right.*

Cellulose, found for example in wood, paper and cotton tissues, is a precursor to isosaccarinate (ISA) that is formed by hydrolysis of cellulose in a Ca^{2+} rich alkaline environment. When cellulose degrades, two diastereomers are formed in almost equal quantities (α and β). In Figure 3-9 the α-isosaccharinate is shown. The difference between the two diastereomers is the absolute configuration of the hydroxyl group located α to the carbonyl carbon. In BHA it is estimated that about 47 tonnes of cellulose will be disposed of. The cellulose originates mainly from three different sources as shown in Table 3-23.

Table 3-23. The cellulose content that is estimated to be deposited in BHA.

Cellulose source	Waste mass (kg)	Cellulose content (%)
Textile	33 000	100
Paper	11 200	100
Wood	3 400	80

The amounts of complexing agents allowed for disposal will be regulated by the means of waste acceptance criteria (WAC). Microbes might utilise some of the material deposited in SFL as energy sources. Specifically cellulose is a favourable energy source for microbes but other sources could occur within the waste form. The microbial population initially present in the waste form depends on the origin of the waste form and whether microbes have been transported to the waste by the infiltrating groundwater.

Concentration of cellulose degradation products (CDP)

Of the total acids released in the degradation of cellulose, 3-deoxy-2-C-hydroxymethyl- D-*erythro*-pentonic acid (α-ISA) and 3-deoxy- 2-C-hydroxymethyl-D-*threo*-pentonic acid (*β*-ISA) are the most abundant. ISA shows strong complexing properties with divalent, trivalent and tetravalent cations resulting in soluble organometallic complexes that exhibits different retardation behaviour compared with the unbound cations. The degradation of cellulose can be modelled using the equations published by Glaus and Van Loon (2008). The concentration of dissolved ISA in BHA is shown in Table 3-24.[4]

Table 3-24. Dissolved (aq) ISA concentrations in BHA at different times.[4]

Time (y)	Concentration ISA in waste (M)
10	0
100	0.004
1 000	0.02 (saturated)
5 000	0.02 (saturated)
109 000	0.008
1 080 000	0.001

3.7.2 Waste packaging

In addition to the waste characteristics, the waste packages are of importance for the initial state of the repository. The waste packages contribute to the total amount of steel and concrete in the repository ensuring a low redox potential and providing both hydraulic resistance and pH control. The calculated void inside each waste package is also of importance when assessing concentrations of e.g. radionuclides and organic complexing agents. Table 3-25 shows the materials in the waste packaging for BHK and BHA respectively. For a more detailed discussion of the material quantities associated with the different packages, see Herschend (2014).[2]

Table 3-25. Material in waste packaging and the grout between the inner and outer waste packaging.

	BHK (Herschend 2014)	BHA[2]
Carbon steel (tonnes)	12 285	2 882
Carbon steel (m^2)	53 679	85 857[a] (200 991)[b]
Concrete and grout (tonnes)	5 415	8 649

[a] Surface area of the outer packaging.
[b] Surface area of the drums and moulds placed within the outer waste packaging for SFL, see Section 3.4.

3.8 Radionuclide inventory

Figure 3-10 shows the different information sources that have been used to gather information about the radionuclide inventory for the waste to be deposited in SFL and the calculation models used. Some of the information sources provide source terms for the calculation models whereas some of the data reported by SNAB and AB SVAFO were used directly in the radionuclide inventory.

The gathered information is preliminary and there are still remaining uncertainties in the estimated inventories, especially concerning the legacy waste.

The total radioactivity in SFL by the end of 2075 is estimated to be about 2×10^{17} Bq. Only 2 % of this total radioactivity is found in the waste planned for disposal in BHA.

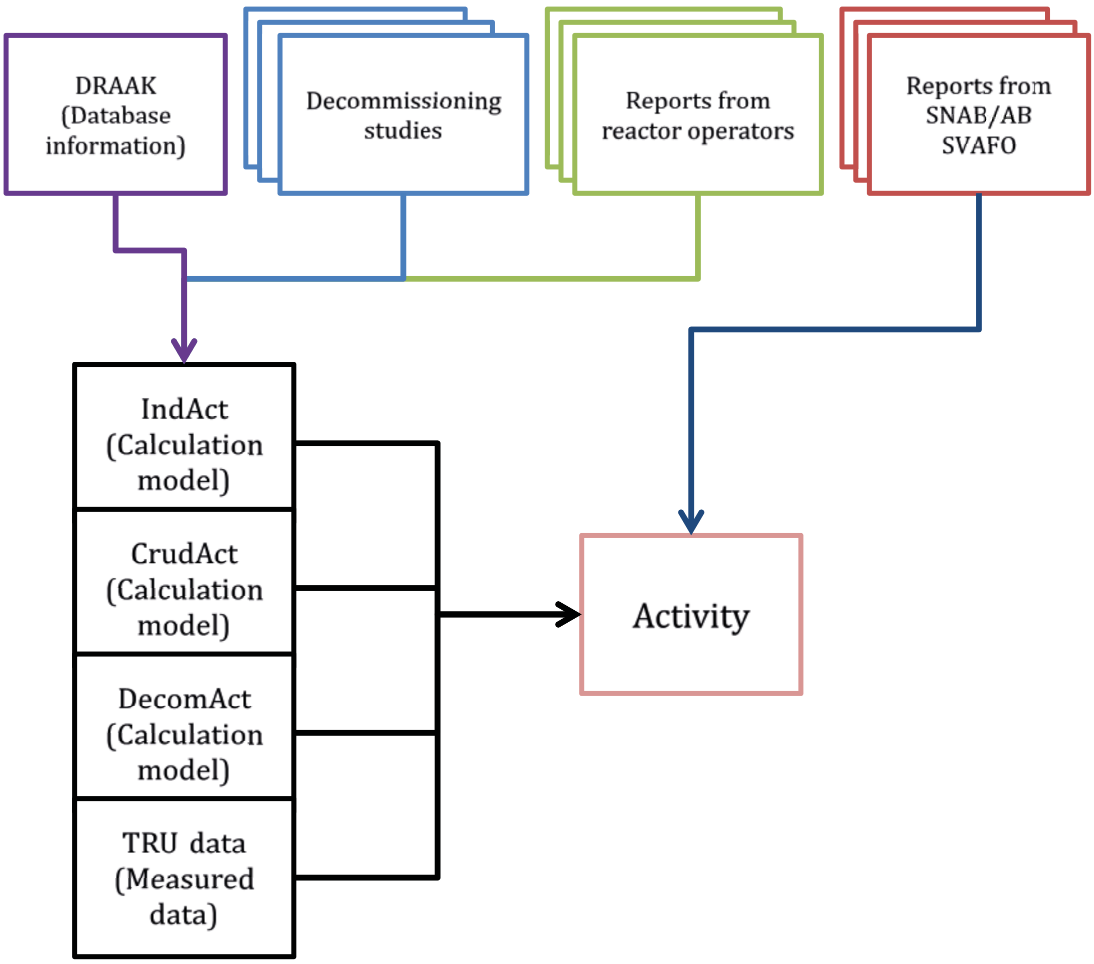


Figure 3-10. *Different information sources used to calculate the radionuclide inventory in SFL.*

3.8.1 Radionuclide inventory in BHK

The total radionuclide inventory in the reactor internals is summarised in Table 3-26 at the reference date 2075-12-31. The inventory is based on data from Herschend (2014) that has been corrected for some errors made in the previous calculations, resulting in a 0–13 % decrease in the total inventory of long lived activation products as compared with Herschend (2014). The radionuclide inventory has been further divided into four waste categories, based on the thickness of the component, see Appendix 1.

Table 3-26. Estimated radionuclide inventory in BHK 2075-12-31. Data is the sum of the activities given in Appendix 1.

Nuclide	Induced activity (Bq)	Contamination (Bq)
H-3	8.3E+14	-
Be-10	6.4E+04	-
C-14	3.6E+14	-
Cl-36	1.1E+11	-
Ca-41	3.2E+11	-
Fe-55	2.6E+14	4.7E+10
Co-60	5.5E+15	2.5E+12
Ni-59	1.7E+15	3.0E+12
Ni-63	1.8E+17	2.8E+14
Se-79	7.0E+10	1.0E+09
Sr-90	1.8E+11	4.7E+11
Zr-93	6.1E+11	7.5E+10
Nb-93m	8.6E+14	2.0E+13
Nb-94	1.4E+13	1.9E+11
Mo-93	2.2E+13	7.2E+09
Tc-99	3.4E+12	5.0E+09
Ru-106	1.4E+01	-
Ag-108m	1.9E+12	5.4E+11
Pd-107	1.3E+05	4.0E+07
Cd-113m	1.5E+09	-
Sn-126	1.6E+06	1.9E+07
Sb-125	1.7E+10	4.6E+08
I-129	2.1E+05	1.7E+08
Cs-134	2.3E+07	-
Cs-135	2.5E+06	-
Cs-137	2.0E+11	-
Ba-133	4.9E+09	-
Pm-147	1.9E+07	1.9E+07
Sm-151	4.4E+11	8.1E+09
Eu-152	3.4E+12	9.0E+06
Eu-154	6.0E+10	1.4E+09
Eu-155	3.4E+09	3.3E+07
Ho-166m	4.2E+09	3.3E+05
U-232	9.9E+06	1.6E+05
U-235	6.7E+02	4.8E+02
U-236	2.9E+06	9.9E+06
Np-237	4.1E+06	1.2E+07
Pu-238	1.9E+10	6.5E+10
Pu-239	8.7E+09	1.1E+10
Pu-240	7.1E+07	1.5E+10
Pu-241	1.2E+09	1.5E+11
Pu-242	1.1E+09	1.1E+10
Am-241	9.2E+08	5.2E+10
Am-242m	2.0E+06	3.8E+08
Am-243	7.5E+06	1.1E+09
Cm-243	1.1E+07	1.4E+08
Cm-244	1.1E+08	1.3E+10
Cm-245	7.2E+07	6.3E+08
Cm-246	6.6E+04	6.3E+06
Total	1.9E+17	3.1E+14

3.8.2 Radionuclide inventory in BHA

Legacy waste

The radionuclide inventory of the legacy waste is highly uncertain. The inventory is based on relatively sparse data from gamma spectroscopy and sampled surface contamination in combination with correlation procedures.[2]

The total radionuclide inventory in the legacy waste is summarised in Table 3-27 at the reference date 2075-12-31. The radionuclide inventory data also include activity calculated from reported actinide content (i.e. Th-232, U-235, U-238 and Pu-239). Decay products of Th-232 and U-238 have not been taken into account.

Existing operational waste

The operational waste is generally better characterised than the legacy waste and is also given in Table 3-27 at the reference date 2075-12-31.

Operational waste forecast

A forecast of operational waste from SNAB has been reported for each facility at the Studsvik site. The inventory also includes 400 drums of the S.09 type from AB SVAFO. These drums were previously intended for disposal in SFR, but have been excluded from the SFR inventory, and are therefore included here. The waste is assumed to be produced at a constant rate until 2045.

No additional estimates of activity in the forecasted operational waste have been reported by the producers. Hence, the radionuclide inventory is based on the average radionuclide inventory of corresponding waste types in the existing operational or legacy waste. Radioactive decay is calculated from a fixed reference time (2030-01-01) and the inventory is presented in Table 3-27 at the reference date 2075-12-31.

Decommissioning waste from SNAB and AB SVAFO

For the decommissioning waste from R2 and R2-0, data from the recent decommissioning plan for the R2 facility has been used to estimate the radionuclide inventory. For all other decommissioning waste, the radionuclide inventory is based on source terms for corresponding waste types in the existing waste. The inventory from decommissioning waste is presented in Table 3-27 at the reference date 2075-12-31.

Table 3-27. Estimated radionuclide inventory in BHA 2075-12-31.[2]

	Activity (Bq)					
	Legacy waste	Existing operational waste	Operational waste forecast	Decommissioning SNAB	Decommissioning AB SVAFO	Total
H-3	4.1E+13	3.5E+05	1.4E+06	3.4E+04	7.3E+10	4.1E+13
Be-10	3.5E+01	1.8E+02	1.1E+02	9.8E−01	2.7E+10	2.7E+10
C-14	5.8E+13	1.3E+07	2.6E+08	2.1E+08	5.5E+11	5.9E+13
Na-22	1.5E−02	1.2E−01	7.2E+01	3.4E+03	5.0E−08	3.5E+03
Cl-36	2.9E+13	6.5E+06	1.3E+08	1.0E+08	2.7E+11	3.0E+13
K-40	5.3E+09	1.1E+07	1.3E+08	1.0E+08	4.2E+07	5.5E+09
Fe-55	2.0E+07	1.2E+03	2.5E+06	2.2E+07	1.0E+07	5.5E+07
Co-60	2.8E+11	1.1E+07	1.2E+09	3.9E+09	1.5E+11	4.3E+11
Ni-59	3.8E+12	3.2E+08	5.6E+08	3.1E+08	2.4E+11	4.0E+12
Ni-63	1.8E+15	1.4E+10	2.3E+10	1.1E+10	2.9E+13	1.8E+15
Se-79	1.5E+06	1.3E+07	7.6E+06	2.9E+04	9.0E+05	2.3E+07
Sr-90	2.5E+13	9.2E+10	1.7E+11	7.1E+10	1.6E+11	2.5E+13
Nb-93m	6.1E+08	9.9E+06	2.5E+07	4.9E+05	4.9E+08	1.1E+09
Nb-94	3.6E+09	3.3E+07	2.0E+07	3.2E+05	2.2E+08	3.9E+09
Mo-93	6.4E+12	6.0E+07	1.2E+09	9.7E+08	6.3E+10	6.5E+12
Tc-99	1.8E+15	2.4E+11	4.8E+12	3.8E+12	1.7E+13	1.9E+15
Ru-106	7.1E−07	3.0E−10	1.7E−01	2.9E+03	5.5E−19	2.9E+03
Pd-107	3.5E+05	3.2E+06	1.9E+06	7.3E+03	-	5.5E+06
Ag-108m	2.4E+12	4.8E+07	3.0E+07	9.3E+04	1.9E+11	2.6E+12
Cd-113	-	-	-	-	-	-
Sn-126	1.8E+05	1.6E+06	9.5E+05	3.7E+03	-	2.7E+06
Sb-125	2.3E+05	2.1E+02	2.9E+05	1.4E+05	2.6E+02	6.6E+05
I-129	1.1E+06	9.6E+06	5.7E+06	2.2E+04	-	1.6E+07
Cs-134	9.6E+02	1.5E+01	3.3E+05	1.5E+07	2.9E−04	1.5E+07
Cs-135	3.5E+06	3.2E+07	1.9E+07	7.3E+04	-	5.5E+07
Cs-137	2.0E+13	5.2E+11	8.0E+11	1.5E+11	1.4E+11	2.2E+13
Ba-133	7.8E+04	1.6E+04	9.0E+04	2.3E+03	-	1.9E+05
Pm-147	4.0E+01	7.0E+03	1.2E+07	2.4E+06	-	1.4E+07
Sm-151	5.4E+08	5.2E+09	4.0E+09	1.8E+07	-	9.8E+09
Eu-152	9.2E+11	1.5E+09	5.5E+09	2.9E+08	1.9E+09	9.3E+11
Eu-154	2.3E+11	2.9E+08	2.1E+10	5.1E+10	2.0E+08	3.1E+11
Eu-155	2.8E+09	1.0E+06	1.2E+09	7.9E+09	3.3E+05	1.2E+10
Ho-166	-	-	-	-	-	-
Pb-210	-	-	-	-	-	-
Ra-226	3.3E+12	4.0E+08	7.4E+09	5.9E+09	2.9E+10	3.3E+12
Ac-227	-	-	-	-	-	-
Th-228	1.1E+10	1.3E−05	2.0E+02	5.0E+04	5.7E−07	1.1E+10
Th-229	-	-	-	-	-	-
Th-230	-	-	-	-	-	-
Th-232	4.2E+09	1.6E+07	3.2E+08	2.5E+08	3.5E+07	4.8E+09
Pa-231	-	-	-	-	-	-
U-232	1.1E+04	6.6E+05	5.6E+05	1.6E+03	-	1.2E+06
U-233	4.1E+02	-	-	-	3.7E+00	4.2E+02
U-234	5.9E+10	2.6E+08	1.9E+10	7.3E+04	-	7.8E+10
U-235	9.3E+09	2.2E+08	2.6E+09	1.4E+09	8.4E+04	1.4E+10
U-236	1.2E+09	1.5E+07	8.7E+06	2.3E+04	2.5E+04	1.2E+09
U-238	2.2E+11	3.8E+08	2.2E+10	3.2E+09	2.9E+06	2.5E+11
Np-237	3.4E+10	2.7E+07	1.2E+09	9.6E+08	1.8E+07	3.7E+10
Pu-238	2.7E+13	5.8E+10	8.6E+10	3.8E+10	2.0E+11	2.7E+13
Pu-239	1.6E+12	2.7E+10	1.7E+10	1.0E+09	1.4E+10	1.6E+12
Pu-240	5.5E+12	3.8E+10	2.7E+10	3.9E+09	7.0E+07	5.6E+12
Pu-241	3.7E+13	1.1E+11	9.3E+11	9.9E+11	8.2E+10	4.0E+13

	Activity (Bq)					
	Legacy waste	**Existing operational waste**	**Operational waste forecast**	**Decommissioning SNAB**	**Decommissioning AB SVAFO**	**Total**
Pu-242	1.4E+10	1.5E+08	1.2E+08	2.9E+07	1.3E+08	1.4E+10
Pu-244	-	-	-	-	-	-
Am-241	6.9E+12	2.8E+11	4.4E+11	2.1E+11	1.0E+10	7.8E+12
Am-242m	5.7E+06	3.3E+08	2.3E+08	6.3E+05	-	5.7E+08
Am-243	2.6E+07	1.5E+09	8.7E+08	2.2E+06	-	2.3E+09
Cm-243	2.4E+06	1.6E+08	2.1E+08	7.3E+05	-	3.7E+08
Cm-244	1.9E+10	5.7E+09	1.3E+10	2.4E+08	5.5E+07	3.7E+10
Cm-245	2.6E+05	1.5E+07	8.7E+06	2.2E+04	-	2.4E+07
Cm-246	6.9E+04	3.9E+06	2.3E+06	5.8E+03	-	6.2E+06
Total	3.9E+15	1.4E+12	7.4E+12	5.4E+12	4.8E+13	3.9E+15

3.8.3 Radionuclides from ESS

As mentioned in section 3.1, the radionuclides from ESS that are not already present in the BHK or BHA waste are included in a dedicated evaluation case, which evaluates if there is a preferred placement for these radionuclides in SFL. The ESS specific radionuclides are listed in Table 3-28.

Table 3-28. Radionuclides included in the transport calculations for the ESS waste.

Radionuclide		
Si-32	La-137	Tb-157
Ar-39	Gd-148	Tb-158
Ti-44	Eu-150	Re-186m

4 Waste vault for reactor internals, BHK

4.1 Design

In BHK, a barrier system based on concrete will be used. The system will comprise a concrete barrier in the form of six separate caissons, grout between and inside the waste packages as well as a concrete backfill of the vault. The design of the waste vault along with dimensions is described in Elfwing et al. (2013).

The primary safety function of this repository section is retardation (Elfwing et al. 2013). This is achieved by limiting the flow of groundwater through the waste and further by limiting the diffusive transport of substances to and from the waste. Finally, the concrete and grout provide a high sorption capacity for many radionuclides and maintain high pH and a favourable chemical environment in the repository.

The waste vault will have a width of approximately 20.6 m and height of approximately 19.6 m. An illustration of the cross section of the vault is shown in Figure 4-1. The length of the vault is approximately 134 m and six concrete caissons are planned based on the expected waste volume. The caissons provide radiation shielding during the operational phase. Each caisson has the lateral dimensions of 15 m x 16 m and a height of 8.4 m. The thickness of the caisson walls, floor and lid is 0.5 m.

When the waste packages have been placed in the caisson, the space between the waste packages will be filled with grout in order to stabilise the stack of waste packages and to reduce the void. In addition, the grout will stabilise the waste packages and improve the strength by reducing the deformations caused by the external forces on the walls, floor and lid during backfill and re-saturation of the repository. The grout also contributes to the high alkalinity in the vault and thus also the passivation of steel components in the waste packages, which thereby reduces the corrosion rate.

Main components of the engineered barrier system

The main components of the engineered barrier system are summarised below and shown in Figure 4-1:

- The structure is made of reinforced concrete with a thickness of 0.5 m.
- The space above the concrete structure is filled with concrete, with a thickness of 8.8 m.
- The space between the concrete structure and the rock wall is filled with concrete, with a thickness of 2.8 m.
- The space between the waste packages inside the concrete structure is filled with concrete or grout.
- The space below the structure is filled with concrete, with a thickness of 2.4 m.

Closing the vault for the metallic waste starts with the removal of installations and equipment. The space between the concrete caissons and the rock walls is then backfilled with concrete.

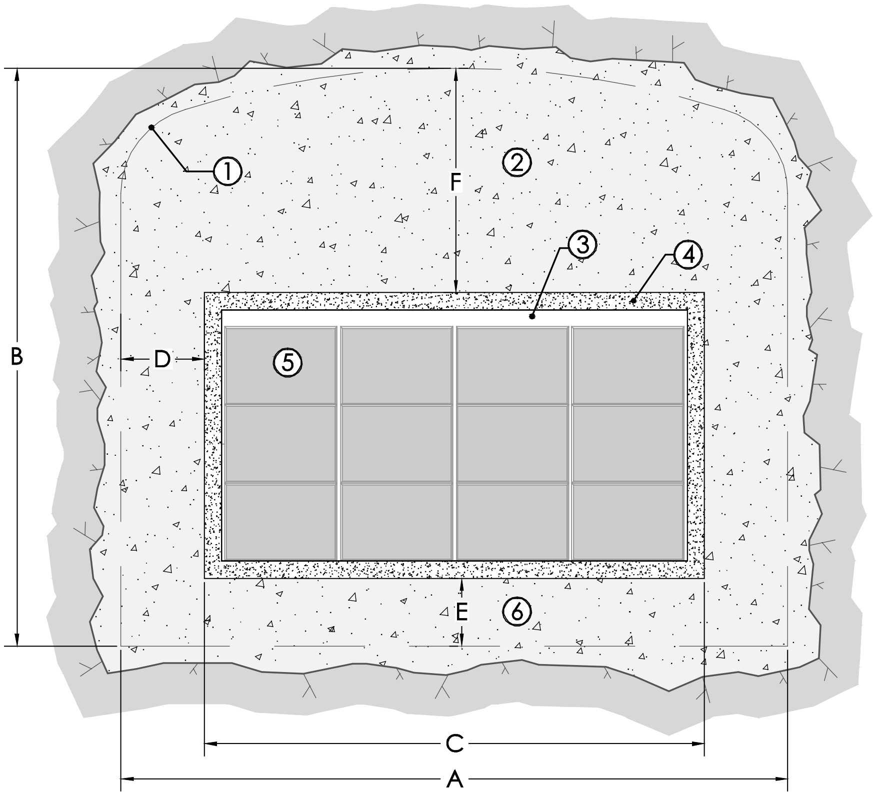


Figure 4-1. Schematic cross-sectional layout of BHK from Elfwing et al. (2013). Legend: 1) Theoretical tunnel contour. 2) Concrete backfill. 3) Grout. 4) Concrete structure (0.5 m). 5) Steel tanks. 6) Concrete. Approximate dimensions: A = 20.6 m, B = 19.6 m, C = 15 m, D = 2.8 m, E = 2.4 m, F = 8.8 m.

4.2 Design considerations

The presented design of this vault constitutes a solution that is technically feasible. However, it is foreseen that the design of sealing and closure of this vault can be further developed and optimised before closure of SFL.

Safety functions of the engineered barrier system

The concrete in the barrier – in both structures and backfill – will provide an alkaline environment in the repository section, which will create a passivating layer on the surface of the steel components and reduce the corrosion rate of steel (Elfwing et al. 2013). The low corrosion rate results in a low gas production rate, which is considered beneficial for the ability of the barriers to transport gas without negative effects on those barriers. The concrete has a high sorption coefficient for many radionuclides (except for the anions formed by e.g. chlorine and molybdenum). The diffusion rate through concrete is low for many important radionuclides. In fresh concrete, the hydraulic conductivity is also very low, which potentially makes diffusion the dominant transport mechanism for radionuclides through the concrete.

4.3 Dimensions of BHK

The main dimensions of BHK are given in Table 4-1.

Table 4-1. Dimensions for BHK extracted from CAD layout described in Kontio (2014).

Property	Value
Excavated rock cavity	
Total length (m)	134.1
Width (m)	20.6
Vertical cross-section area (m^2)	386
Height (m)	19.6
Excavated volume (m^3)	52 000
Distance from short-end rock wall to innermost concrete wall	
Length (m)	5.4
Waste disposal area	
Concrete structure (6 disposal caissons)	
Length outer (m)	104.7
Distance between caissons	1.5
Width outer (m)	15
Length outer (m)	16.2
Height outer (m)	8.4
Concrete lid (m)	0.5
Thickness outer walls (m)	0.5
Concrete floor (m)	0.5
Disposal caissons	
Width inner (m)	14
Length inner (m)	15.2
Height inner (m)	7.4
Bottom	
Concrete fill thickness (m)	2.4
Loading zone	
Length (m)	24

5 Waste vault for legacy waste, BHA

5.1 Design

BHA is designed for disposal of waste packages with a low activity but less well-known material composition and radionuclide inventory than the wastes scheduled for BHK. The waste vault comprises a concrete structure in which the waste is deposited, and which is surrounded by a thick layer of high-quality bentonite. This will function as a low permeability medium enclosing the waste. The rock walls and ceilings are to be lined with shotcrete if required. The design of the waste vault along with dimensions is described in Elfwing et al. (2013).

The primary safety function of this repository section is retardation (Elfwing et al. 2013). The barriers limit the flow of groundwater through the waste and thus, making diffusion the predominant transport process for radionuclides.

The waste vault will have a width of approximately 20.6 m and a height of approximately 18.4 m. An illustration of the cross section of the vault is shown in Figure 5-1. The length of the vault is approximately 170 m. A concrete structure of lateral dimensions of approximately 16 m x 140 m and a height of 8.4 m is planned based on the expected waste volume. The thickness of the walls, floor and lids is 0.5 m. The structure provides radiation shielding during the operational phase.

When the waste packages have been placed in the vault, the space between the waste packages will be filled with grout in order to stabilise the stack of waste packages and to reduce the void. In addition, the grout will stabilise the waste packages and improve the strength by reducing the deformations caused by the external forces on the walls, floor and lid during backfill and re-saturation of the repository. The grout also contributes to the high alkalinity in the vault and thus also to the passivation of steel components in the waste packages, which thereby reduces the corrosion rate. The concrete walls, lid, grout, the floor and the bentonite constitute chemical barriers that enhance sorption of radionuclides in the vault. The thick bentonite layer constitutes the main barrier that primarily reduces the flow of water through the repository but also increases the sorption capacity of the vault.

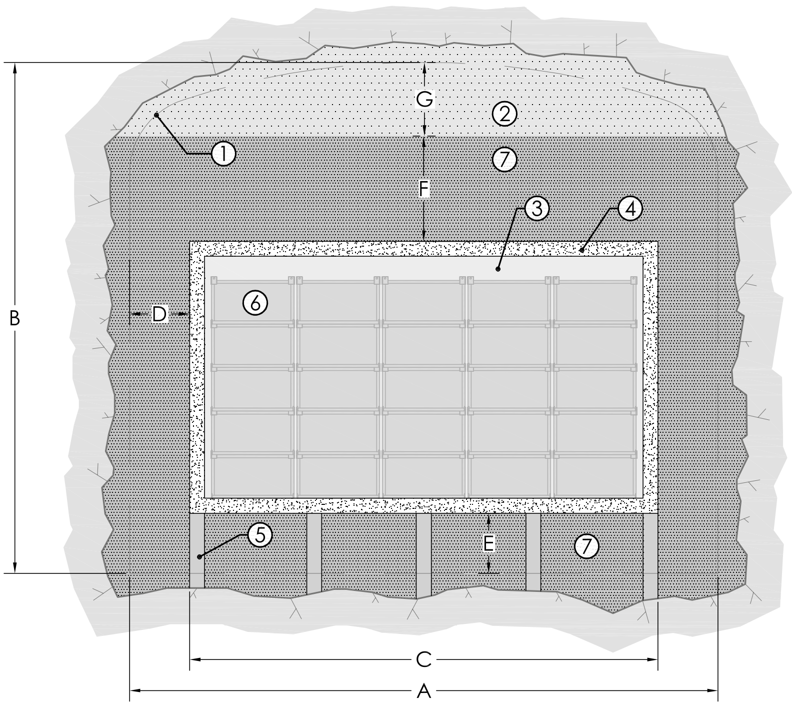


Figure 5-1. *Schematic cross-sectional layout of BHA from Elfwing et al. (2013). Legend: 1) Theoretical tunnel contour. 2) Bentonite pellets. 3) Grout. 4) Concrete structure (0.5 m). 5) Granite pillars. 6) Waste packages. 7) Bentonite blocks. Approximate dimensions: A = 20.6 m, B = 18.5 m C = 16 m, D = 2.3 m, E = 2.4 m, F = 4 m, G = 3.7 m.*

Main components of the engineered barrier system

A reinforced concrete base slab is placed on granite pillars, which stand on the floor of the vault. A concrete wall structure is cast onto the base slab. The waste is placed inside the structure. The purpose of the concrete structure is to provide radiation protection during the operational phase. After closure, it also contributes to high alkalinity in the vault and sorption sites for radionuclides.

At closure, empty voids between waste packages are grouted and a concrete lid is placed on the structure. Bentonite blocks are placed beneath the base slab as well as on the sides and on top of the concrete structure. The top part of the vault is filled with bentonite pellets. No bentonite is placed in the vault until the time of closure.

5.2 Design considerations

Safety functions of the engineered barrier system

The bentonite permits only a small advective flow, which makes diffusion the dominant transport mechanism for radionuclides through the bentonite. Late installation ensures good control of the condition of the bentonite at closure of the repository. The clay has beneficial mechanical properties and can be deformed without cracking as well as exhibiting self-healing properties. Bentonite is a natural material, and its long-term properties are well characterised from earlier research. Bentonite clay has the ability to effectively filter colloids. This is an advantage for this category of waste, since it contains more actinides compared with the metallic waste. Actinides, together with mixed organic material, could form different types of actinide-carrying colloids.

5.3 Dimensions of BHA

The main dimensions of BHA are presented in Table 5-1.

Table 5-1. Dimensions of BHA extracted from CAD layout described in Kontio (2014).

Repository dimensions	Value
Excavated rock cavity	
Total length (m)	169.3
Width (m)	20.6
Height (m)	18.5
Vertical cross-sectional area (m^2)	363
Excavated volume (m^3)	61 000
Distance from short-end rock wall to innermost concrete wall	
Length (m)	5.3
Waste disposal area	
Concrete structure	
Length outer (m)	140
Width outer (m)	16
Height outer wall (m) (above floor)	8.4
Concrete lid (m)	0.5
Thickness outer walls (reinforced) (m)	0.5
Concrete floor (reinforced) (m)	0.5
Bottom	
Bentonite fill thickness (m)	2.4
Loading zone	
Length (m)	24

6 Closure components

In a geological repository, the rock mass surrounding the vaults is part of the barrier system that will prevent the radioactive substances from harming human health or the environment. The tunnels that provide access to the waste vaults may impair the barrier function of the rock through the creation of open flow paths in the rock. These flow paths must be closed by installing closure components. The main purposes of the closure components are:

- Reducing the axial water transport in the tunnels.
- Reducing the water flow from the tunnels to the waste vaults.
- Supporting the rock and thereby preventing a collapse of the tunnel roof and walls.
- Preventing unintended access to the radioactive waste.

In this chapter, the closure components and methods used to close SFL are briefly outlined. At this stage, it is assumed that the design of the components and method for installation will not differ significantly between the different repository sections. The sealing sections of SFL are schematically illustrated in Figure 6-1. The suggested design in this report constitutes solutions that are technically feasible. However, it is foreseen that the design of closure components for each of the individual sections can be further developed and optimised before closure of SFL.

In BHK, the space between the caissons and the bedrock will be filled with concrete in order to further limit the water flow through the waste and contribute to high alkalinity and sorption sites for the radionuclides. In BHA, the space between the concrete structure holding the waste and the bedrock will be filled with bentonite in order to further limit the water flow through the waste.

To seal the waste vaults, tunnel sections adjacent to the vaults are to be filled with bentonite and confined by mechanical plugs. The bentonite acts as hydraulic seal to reduce the axial flow of groundwater through the waste vault. This solution is in accordance with previous investigations and concepts developed by SKB for the Spent Fuel Repository (Luterkort et al. 2012).

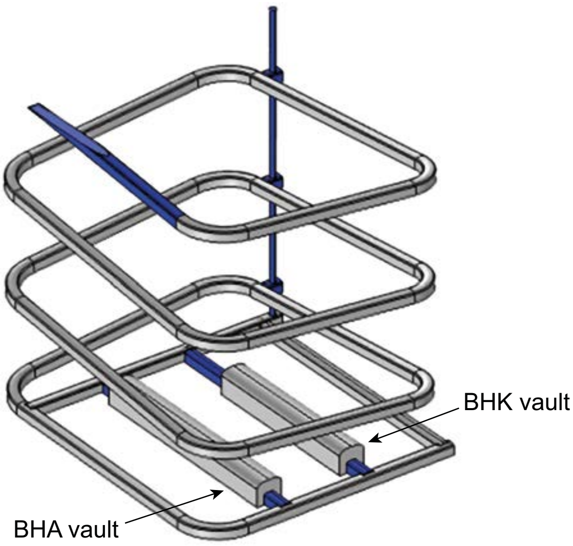


Figure 6-1. Sealing sections (blue) installed at closure in the tunnel sections connecting the vaults and access tunnels, the vertical shaft and in the access ramp.

The tunnels at repository level in connection with the sealed sections of the waste vaults are planned to be backfilled with crushed rock or a similar material. The access tunnel is also planned to be backfilled with crushed rock. In addition, a plug section will be installed, made up of a tight hydraulic section of bentonite to further reduce the groundwater flow through the tunnel, surrounded by concrete plugs as mechanical support. Figure 6-2 illustrates an example of such a technically feasible plug.

The first 50 m of the access tunnels is planned to be backfilled with boulders and a concrete plug will be cast to obstruct unintentional intrusion into the repository. Finally, the ground surface will be restored to match the surroundings.

The vertical shaft connecting different parts of the repository is planned to be sealed to restrict the flow of water. The suggested solution comprises a tight hydraulic section with bentonite surrounded by upper and lower concrete plugs for mechanical support.

In the present study, investigation boreholes are not considered in the modelling of the hydrogeology or radionuclide transport. After the site investigation program for SFL, it will be necessary to consider the presence of boreholes and to have a plan for sealing them.

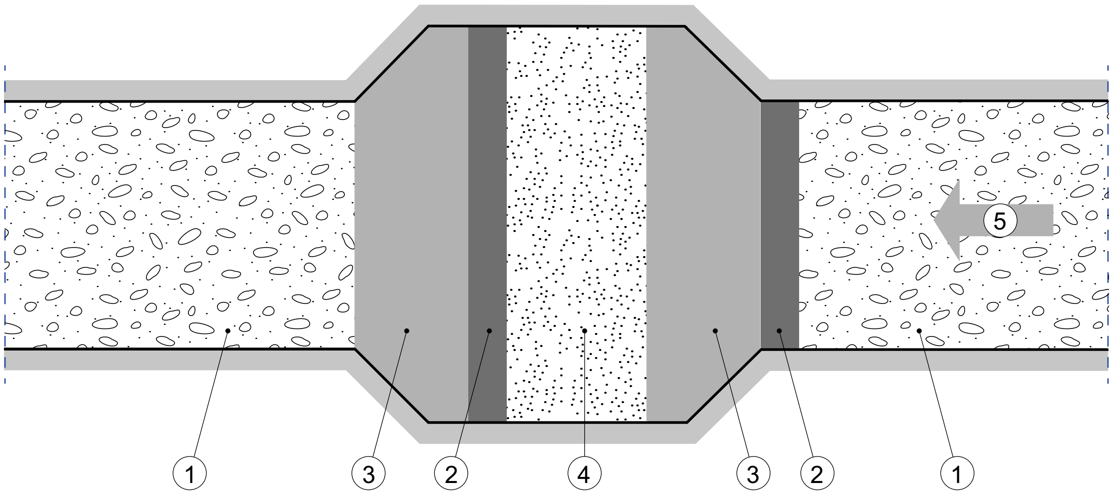


Figure 6-2. *Schematic design of a sealing plug (Luterkort et al. 2014). Legend: 1) Backfill of crushed rock or similar. 2) Retaining concrete walls. 3) Cast concrete. 4) Bentonite. 5) Backfill direction (from the waste vault and out). 6) Backfill material.*

7 Conditions at repository closure

7.1 Condition of the waste at repository closure

The initial state of the waste scheduled for disposal in BHK will be well known at the time of closure of the repository. All waste – PWR RPVs included – is assumed to be segmented, dried, and inspected, placed in steel tanks and grouted with cement with a well-defined water cement ratio (w/c) prior to disposal. For that reason, the metallic waste is expected to corrode to a negligible extent during the operational phase of the repository and no or negligible release of radionuclides will occur during this period.

The waste packaging (moulds and drums) in BHA was filled with a mixture of waste and concrete grout already in the 1960s and 1970s when the waste was first conditioned. Only limited information about the state of this waste is currently available.

The initial state of the waste to be deposited in SFL is described using a number of variables that are listed in Table 7-1. The list of variables has been adopted from the most recent safety assessment for SFR (SR-PSU; SKB 2014a).

Table 7-1. Variables used to describe the initial state of the waste form and their definitions from SKB (2014a).

Variable	Definition
Geometry	Volume and dimensions of the waste form and voids inside the waste packaging. Porosity and pore characteristics of the waste form. Amount and characteristics of fractures in the waste form.
Radiation intensity	Intensity of alpha, beta and gamma radiation.
Temperature	Temperature.
Hydrological variables	Magnitude, direction and distribution of water flow. Degree of water saturation. Amount of water. Water pressure. Aggregation state (water and/or ice).
Mechanical stresses	Stress and strain in waste form.
Radionuclide inventory	Inventory of radionuclides as a function of time within the waste package. Type, amount, chemical and physical form.
Material composition	Amount and surface characteristics of materials inside the waste package (excluding radionuclides). Type and amount of chemicals. Type, amount of organic materials and other substances that can be used by microbes as nutrients and energy sources. Type and amount of microbes and bacteria and other types of biomass.
Water composition	Composition of water including radionuclides. Redox potential, pH, ionic strength, concentration of dissolved species, type and amount of colloids and/or particles, amount and composition of dissolved gas. Type and amount of microbes and bacteria and other types of biomass. Density and viscosity.
Gas variables	Amount and composition including radionuclides. Volume, pressure and degree of saturation. Magnitude, direction and distribution of gas flow.

7.1.1 Geometry

The geometry of the waste scheduled for disposal in SFL is described in Chapter 3. The porosity of the waste form is dependent on the type of waste, whether the waste is conditioned and the type of conditioning, i.e. cement or concrete, as well as the dimensions of the waste packaging.

The inner volume of the steel tanks used for the waste for BHK varies in the range 5.6–8.5 m^3 (Table 3-2) and, from the information provided in Table 3-19, a porosity between 22–34 % depending on the thickness of the steel tank can be estimated. In the SE-SFL safety evaluation, an average porosity of 28 % will be used for all waste in BHK.

Applying the same methodology for BHA and combining the information provided in Tables 3-3 to 3-8 with the pore volumes presented in Table 3-20, a porosity between 27–34 % depending on the packaging used can be estimated. An average porosity of 31 % will be used for the waste in BHA within the SE-SFL safety evaluation.

The abundance and character of the fractures initially present in the waste that will be deposited in SFL are not known to date.

7.1.2 Radiation intensity

The energy resulting from decaying radionuclides in the SFL waste will be absorbed by surrounding materials and increases with time. If the total absorbed energy is sufficiently high it can affect the surrounding materials, such as concrete structures or organic material present in the waste. In order to assess the radiological impact on surrounding materials, the integrated absorbed energy from a selection of dominant γ-emitters has been calculated. The integrated dose has been calculated for a period of 400 years after closure. Beyond this point in time, the contribution from the initially dominant γ-emitters is negligible. Furthermore, the heat that the energy generates is not expected to cause any significant temperature increase in the repository (von Schenck 2018).

The waste in BHK mainly consists of reactor internals in steel tanks, surrounded by concrete grout. The absorbed dose from the BHK-waste has been calculated to be 3.11×10^5 Gy up to 400 years after closure (Forrest 2007).[4] In the literature, the approximate threshold level necessary to make any measurable damage on concrete has been reported to be 10^8 Gy from γ-irradiation (IAEA 1998). However, it has been reported that the compressive strength of concrete is decreased by 10 % upon γ-irradiation up to 6×10^5 Gy (Vodák et al. 2005). When calculating the absorbed energy, it is assumed that the decay energy is completely absorbed in the waste material and grout. Hence, the absorbed energy in the caisson walls can be neglected, especially if the radiation shielding of the steel tanks is taken into account. The grout has not been assigned any mechanical post-closure function. Hence, a 10 % degradation of its mechanical properties over a 400-year time period is insignificant.

Radiolysis products that may form are not considered to have any significant effect on the caisson walls due to their inherent reactivity and the fact that they most likely will react with the grout before leaving the waste form.

The absorbed energy from decaying radionuclides in the BHA-waste could possibly contribute to the degradation of the organic material present in the waste. The absorbed dose from the BHA-waste has been calculated to be 5.52×10^4 Gy 400 years after closure (Forrest 2007).[4] Especially the cellulose will have a potentially large impact on the safety functions of the repository since it degrades into isosaccharinate (ISA) under alkaline conditions. ISA is a strong complexing agent and can accelerate the transport of radionuclides out of the repository. Using the model published by Glaus and Van Loon (2008), it can be calculated that the cellulose will degrade completely to ISA by hydroxide-catalysed degradation in approximately 5 000 years. Beside the hydrolytic degradation, cellulose degradation can be induced by radiation. The energy required for radiolytic degradation to influence the cellulose can be estimated based on data found in the literature. The radiosterilisation of cellulose derivatives has been studied in order to determine how much energy can be absorbed before structural changes occur (Stawny et al. 2013). It was found that an absorbed dose exceeding 25 kGy may lead to changes in the physiochemical properties of cellulose and to the formation of radiolysis products. However, most of the physical properties did not change even after an absorbed dose of 200 kGy. The rate of ISA formation is therefore not expected to change due to energy absorption from decaying radionuclides.

Since the material inventory in BHA holds significant uncertainties, it cannot be ruled out that the irradiation from decay could have an impact on the degradation of organic materials other than cellulose.

7.1.3 Temperature

The temperature of the waste is set by the temperature of the surrounding bedrock and groundwater. The bedrock temperature in the Laxemar area at about 500 m depth is around 15 °C (Sundberg et al. 2008).

7.1.4 Hydrological variables

Immediately after closure and the cessation of drainage pumping, water will start to saturate the repository. The re-saturation of BHK is assumed to take tens of years, in accordance with saturation studies performed for SFR (Börgesson et al. 2015). The re-saturation of BHA is assumed to take thousands of years, which has been assessed for bentonite backfill saturation in SR-Site (Åkesson et al. 2010).

Initially all water within the waste form is in liquid form due to the local temperature, see Section 7.1.3.

7.1.5 Mechanical stresses

Mechanical stresses in the waste form can originate from the pressure from the grout, the backfill/bentonite and the groundwater when the repository has been water saturated. However, upon closure of the repository, the grout has hardened and thus does not exert any pressure on the waste. The grout also protects the waste from the forces caused by the backfill/bentonite and groundwater and thus, the mechanical stresses at the initial state are assumed to be negligible.

7.1.6 Radionuclide inventory

The radionuclide inventory in each waste vault is given in Section 3.8. The chemical form of each radionuclide is set by the surrounding environment, for example pH, Eh and presence of complexing agents. Initially small amounts of radioactive gases are present within the repository. As an example, some Rn-222 may be present due to the decay of U-238.

7.1.7 Material composition

The different waste-form materials are presented in Herschend (2014)[2] and a summary is given in Section 3.7.

Corrosion is the only considered process that influences the material composition during the operational phase.

For the concrete-embedded solids, i.e. scrap metal, the waste packages are assumed to be filled with a standard concrete.

7.1.8 Water composition

The concrete pore water composition is initially assumed to have the composition of fresh and/or leached cement, see Table 7-2.

Table 7-2. Analysis of pore water from fresh and leached cement (ion concentrations in mmol/L).

Parameter	Fresh cement (Lagerblad and Trägårdh 1994)	Leached cement (Engkvist et al. 1996)
pH	> 13	12.6
SO_4^{2-}	0.04	0.02
Cl^-	< 0.06	2
Na^+	28	3
K^+	83	0.1
Ca^{2+}	0.9	20
Si_{tot}	0.8	0.003
Al_{tot}	0.04	0.002
OH^-	114	36

The amount of radionuclides dissolved in the pore waters depends on the amount in each individual waste package. For the metallic waste, the amount of radionuclides initially dissolved in the pore water depends mainly on the degree of surface contamination. The contribution from induced radioactivity will be determined by the corrosion rate. Since negligible corrosion is assumed during the operational phase, the concentration of radionuclides from this source will be low.

No colloids will initially be present in the waste since colloids are not stable within waste forms that have a high ionic strength and are high in dissolved Ca^{2+} concentration, i.e. waste forms including cementitious materials.

The density and viscosity of the water is assumed to be the same as tabulated values found in the literature at the prevailing temperature.

7.1.9 Gas composition

As the repository fills with water, the most abundant gas in the interior of the waste packaging is the undissolved air that may be found in air pockets and pores. Small amounts of Rn-222 formed through the decay of U-238 might be present in the interior of the waste packaging.

Hydrogen gas may form through the anaerobic corrosion of metals. Initially, the corrosion of Al will contribute with the largest amounts of gas due to its high corrosion rate in an alkaline environment. However, in the long term the vast majority of the H_2 (g) formed in the repository will originate from the anaerobic corrosion of iron and steel.

During the operational phase no H_2 is expected to be formed in the repository, since the surface of the Al is protected by a stable surface oxide. The surface is thus passivated. Any corrosion of steel during this period will be aerobic and consequently no H_2 is formed. However, as soon as the vaults are re-saturated with water, the pH will increase and corrosion of Al will commence.

Low-molecular carbon containing molecules such as CH_4 and CO_2 may form through the degradation of organic matter in the waste. However, the gases formed during the operational phase of the repository will be evacuated from the repository and hence a very limited amount of these types of gaseous molecules will be present in the repository.

7.1.10 Assumed initial speciation of radionuclides

Induced radionuclides

For the induced activity found in metallic parts the speciation of the radionuclides will be governed by their chemical form found in the different steels. Steel consists mainly of iron and carbon, but usually contains one or more additional alloying elements. The radionuclides from metallic elements are initially present in their metallic form and therefore the corrosion process (anaerobic corrosion) will determine the speciation of the radionuclides released from the induced material.

Molybdenum-93 is an activation product with a half-life of about 4 000 years. The redox chemistry of molybdenum is relatively complex, but its main oxidation states are V, IV and III. Molybdates (Mo(VI)) tend to polymerise at concentrations above 10^{-4} M, and polynuclear complexes are also formed by the lower oxidation states. In addition, Mo(II) is known to form dimers with metal-metal bonds. An overview of the main (most stable) oxidation states is obtained from a Pourbaix diagram (Eh-pH predominance area diagram), see Figure 7-1.

The figure shows that negatively charged molybdates are the stable species in solution both for oxidising and for anoxic, mildly reducing conditions.

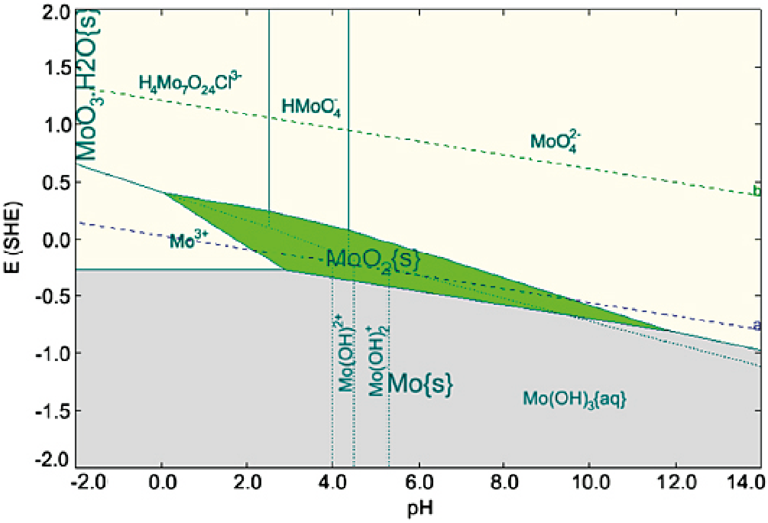


Figure 7-1. *Pourbaix diagram for Mo (total concentration 10^{-6} M.). From Wang et al. (2010).*

The C-14 in the metallic neutron-activated waste is considered to consist of various carbides formed by neutron activation of impurities in the steel. The carbides consist mainly of intermediate transition metal carbides. For such carbides, multiple stoichiometries are common, for example, iron forms a number of carbides, Fe_3C, Fe_7C_3 and Fe_2C. The best known is cementite, Fe_3C. Carbides of intermediate transition metals are assumed to be reactive. The carbides of Cr, Mn, Fe, Co and Ni, for example, are all hydrolysed by dilute acids and sometimes by water, to give a mixture of hydrogen and hydrocarbons (Greenwood and Earnshaw 1984). Therefore, the C-14 compounds released from the reactor pressure vessels are considered to be organic in nature.

C-14 may also be found as elementary carbon (0) originating from neutron activation of C-13 present as graphite impurities in various steels. An important contribution of this formation pathway is the C-14 formed from C-13 in the graphite moderators from the R1 research reactor. This C-14 activity is bound as graphite which is not soluble under present conditions. Further, thermodynamically stable graphite is considered as an immobilisation matrix and hence, no C-14 activity is assumed to be released from graphite.

The potential chloride content in steel originates from trace amounts of various chloride salts. The cross section for activation of stable Cl-35 to Cl-36 by thermal neutrons is large. Hence, if there is any Cl-35 present in the steel, it is likely to be transformed into Cl-36 upon neutron activation.

Loosely bound radionuclides

In BHA, the radionuclides are loosely bound to the surface of different materials. For radioactivity originating from loosely bound radionuclides the speciation in the initial state is set to be in the chemical form found under aerobic conditions. There might be local areas where anaerobic conditions will be prevailing during the operational phase but these will be hard to quantify. Hence, all loosely bound radionuclides are assumed to be in the oxidation state found under aerobic conditions.

7.1.11 Concentrations of radionuclides in the SFL repository

The amounts and concentrations of each radionuclide in each subsection of the waste vaults are given in Table 7-3 and 7-4, respectively. The concentrations are calculated based on the radionuclide inventory for each waste vault. In these calculations, it has been assumed that the repository has been saturated with water and that all radionuclides are immediately dissolved in the accessible water, hence the release rate for the induced activity is not controlled by the corrosion of the material.

Table 7-3. Concentrations of the different radionuclides in BHK. The concentrations are calculated for different volumes.

BHK	Amount (mol)	Concentration (mol/L)			
Nuclide		Waste	Waste volume	Caisson[a]	Vault
H-3	7.7E−01	3.9E−07	2.8E−07	2.5E−07	5.2E−08
Be-10	7.7E−06	3.9E−12	2.8E−12	2.5E−12	5.2E−13
C-14	1.7E+02	8.8E−05	6.2E−05	5.6E−05	1.2E−05
Cl-36	2.7E+00	1.4E−06	9.7E−07	8.7E−07	1.8E−07
Ca-41	2.6E+00	1.3E−06	9.2E−07	8.2E−07	1.7E−07
Fe-55	5.4E−02	2.7E−08	2.0E−08	1.7E−08	3.6E−09
Co-60	2.2E+00	1.1E−06	8.0E−07	7.1E−07	1.5E−07
Ni-59	1.1E+04	5.6E−03	4.0E−03	3.6E−03	7.4E−04
Ni-63	1.5E+03	7.7E−04	5.5E−04	4.9E−04	1.0E−04
Se-79	2.3E+00	1.2E−06	8.4E−07	7.5E−07	1.5E−07
Sr-90	1.4E−03	7.2E−10	5.1E−10	4.6E−10	9.4E−11
Zr-93	8.0E+01	4.0E−05	2.9E−05	2.6E−05	5.3E−06
Nb-93m	1.1E+00	5.4E−07	3.8E−07	3.4E−07	7.1E−08
Nb-94	2.4E+01	1.2E−05	8.5E−06	7.6E−06	1.6E−06
Mo-93	7.3E+00	3.7E−06	2.6E−06	2.4E−06	4.9E−07
Tc-99	5.8E+01	3.0E−05	2.1E−05	1.9E−05	3.9E−06
Ru-106	1.1E−15	5.6E−22	4.0E−22	3.6E−22	7.4E−23
Ag-108m	8.4E−02	4.3E−08	3.0E−08	2.7E−08	5.6E−09
Pd-107	2.0E−02	1.0E−08	7.2E−09	6.4E−09	1.3E−09
Cd-113m	1.6E−06	8.2E−13	5.9E−13	5.2E−13	1.1E−13
Sn-126	3.6E−04	1.8E−10	1.3E−10	1.2E−10	2.4E−11
Sb-125	3.6E−06	1.8E−12	1.3E−12	1.2E−12	2.4E−13
I-129	2.1E−01	1.0E−07	7.4E−08	6.6E−08	1.4E−08
Cs-134	3.6E−09	1.8E−15	1.3E−15	1.1E−15	2.4E−16
Cs-135	4.4E−04	2.2E−10	1.6E−10	1.4E−10	2.9E−11
Cs-137	4.5E−04	2.3E−10	1.6E−10	1.5E−10	3.0E−11
Ba-133	3.9E−06	2.0E−12	1.4E−12	1.3E−12	2.6E−13
Pm-147	7.5E−09	3.8E−15	2.7E−15	2.4E−15	5.0E−16
Sm-151	3.1E−03	1.6E−09	1.1E−09	9.9E−10	2.1E−10
Eu-152	3.4E−03	1.7E−09	1.2E−09	1.1E−09	2.3E−10
Eu-154	4.0E−05	2.0E−11	1.4E−11	1.3E−11	2.7E−12
Eu-155	1.2E−06	6.2E−13	4.4E−13	4.0E−13	8.2E−14
Ho-166m	3.8E−04	1.9E−10	1.4E−10	1.2E−10	2.5E−11
U-232	5.3E−08	2.7E−14	1.9E−14	1.7E−14	3.5E−15
U-235	6.1E−05	3.1E−11	2.2E−11	2.0E−11	4.1E−12
U-236	2.3E−02	1.2E−08	8.3E−09	7.4E−09	1.5E−09
Np-237	2.7E−03	1.4E−09	9.6E−10	8.6E−10	1.8E−10
Pu-238	5.6E−04	2.8E−10	2.0E−10	1.8E−10	3.7E−11
Pu-239	3.6E−02	1.8E−08	1.3E−08	1.2E−08	2.4E−09
Pu-240	7.3E−03	3.7E−09	2.6E−09	2.4E−09	4.9E−10
Pu-241	1.7E−04	8.5E−11	6.0E−11	5.4E−11	1.1E−11
Pu-242	3.4E−01	1.7E−07	1.2E−07	1.1E−07	2.3E−08
Am-241	1.7E−03	8.9E−10	6.3E−10	5.6E−10	1.2E−10
Am-242m	4.0E−06	2.1E−12	1.5E−12	1.3E−12	2.7E−13
Am-243	6.1E−04	3.1E−10	2.2E−10	2.0E−10	4.1E−11
Cm-243	3.3E−07	1.7E−13	1.2E−13	1.1E−13	2.2E−14
Cm-244	1.7E−05	8.7E−12	6.2E−12	5.6E−12	1.1E−12
Cm-245	4.5E−04	2.3E−10	1.6E−10	1.5E−10	3.0E−11
Cm-246	2.3E−06	1.2E−12	8.3E−13	7.4E−13	1.5E−13

[a] The volume inside the caissons, including the porosity of the caisson walls.

Table 7-4. Concentrations of the different radionuclides in BHA. The concentrations are calculated for different volumes.

BHA	Amount (mol)	Concentration (mol/L)			
Nuclide		Waste	Waste volume	Waste volume + concrete structure[a]	Vault
H-3	3.8E−02	1.5E−08	8.3E−09	7.7E−09	1.7E−09
Be-10	3.2E+00	1.2E−06	7.0E−07	6.4E−07	1.4E−07
C-14	2.5E+01	9.8E−06	5.5E−06	5.0E−06	1.1E−06
Na-22	6.9E−13	2.6E−19	1.5E−19	1.4E−19	3.0E−20
Cl-36	6.7E+02	2.6E−04	1.4E−04	1.3E−04	3.0E−05
K-40	5.3E+02	2.0E−04	1.1E−04	1.0E−04	2.3E−05
Fe-55	1.1E−08	4.3E−15	2.4E−15	2.2E−15	5.0E−16
Co-60	1.7E−04	6.6E−11	3.7E−11	3.4E−11	7.6E−12
Ni-59	2.3E+01	8.9E−06	5.0E−06	4.6E−06	1.0E−06
Ni-63	1.4E+01	5.3E−06	3.0E−06	2.7E−06	6.0E−07
Se-79	6.5E−04	2.5E−10	1.4E−10	1.3E−10	2.9E−11
Sr-90	5.5E−02	2.1E−08	1.2E−08	1.1E−08	2.4E−09
Nb-93m	1.4E−06	5.3E−13	3.0E−13	2.8E−13	6.1E−14
Nb-94	5.8E−03	2.2E−09	1.3E−09	1.2E−09	2.6E−10
Mo-93	2.0E+00	7.5E−07	4.2E−07	3.9E−07	8.6E−08
Tc-99	3.0E+04	1.2E−02	6.5E−03	6.0E−03	1.3E−03
Ru-106	2.3E−13	8.7E−20	4.9E−20	4.5E−20	1.0E−20
Pd-107	2.7E−03	1.0E−09	5.8E−10	5.3E−10	1.2E−10
Ag-108m	8.1E−02	3.1E−08	1.8E−08	1.6E−08	3.6E−09
Cd-113m	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Sn-126	4.7E−05	1.8E−11	1.0E−11	9.4E−12	2.1E−12
Sb-125	1.4E−10	5.3E−17	3.0E−17	2.7E−17	6.0E−18
I-129	2.0E−02	7.7E−09	4.3E−09	4.0E−09	8.8E−10
Cs-134	2.4E−09	9.2E−16	5.2E−16	4.8E−16	1.1E−16
Cs-135	9.5E−03	3.7E−09	2.0E−09	1.9E−09	4.2E−10
Cs-137	5.0E−02	1.9E−08	1.1E−08	9.9E−09	2.2E−09
Ba-133	1.5E−10	5.7E−17	3.2E−17	3.0E−17	6.6E−18
Pm-147	2.8E−09	1.1E−15	6.0E−16	5.5E−16	1.2E−16
Sm-151	6.7E−05	2.6E−11	1.4E−11	1.3E−11	2.9E−12
Eu-152	9.5E−04	3.6E−10	2.0E−10	1.9E−10	4.2E−11
Eu-154	2.0E−04	7.7E−11	4.3E−11	4.0E−11	8.8E−12
Eu-155	4.3E−06	1.6E−12	9.2E−13	8.5E−13	1.9E−13
Ho-166m	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Pb-210	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Ra-226	4.0E−01	1.5E−07	8.6E−08	7.9E−08	1.8E−08
Ac-227	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Th-228	1.6E−06	6.0E−13	3.4E−13	3.1E−13	6.9E−14
Th-229	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Th-230	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Th-232	5.1E+03	2.0E−03	1.1E−03	1.0E−03	2.3E−04
Pa-231	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
U-232	6.5E−09	2.5E−15	1.4E−15	1.3E−15	2.9E−16
U-233	5.0E−09	1.9E−15	1.1E−15	1.0E−15	2.2E−16
U-234	1.4E+00	5.6E−07	3.1E−07	2.9E−07	6.4E−08
U-235	7.2E+02	2.8E−04	1.6E−04	1.4E−04	3.2E−05
U-236	2.1E+00	8.2E−07	4.6E−07	4.2E−07	9.4E−08
U-238	8.3E+04	3.2E−02	1.8E−02	1.7E−02	3.7E−03
Np-237	5.9E+00	2.3E−06	1.3E−06	1.2E−06	2.6E−07
Pu-238	1.8E−01	7.0E−08	3.9E−08	3.6E−08	8.0E−09
Pu-239	3.0E+00	1.1E−06	6.4E−07	5.9E−07	1.3E−07
Pu-240	2.8E+00	1.1E−06	6.0E−07	5.5E−07	1.2E−07
Pu-241	4.3E−02	1.6E−08	9.2E−09	8.5E−09	1.9E−09
Pu-242	4.0E−01	1.5E−07	8.6E−08	7.9E−08	1.8E−08
Pu-244	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Am-241	2.6E−01	9.9E−08	5.5E−08	5.1E−08	1.1E−08
Am-242m	6.1E−06	2.3E−12	1.3E−12	1.2E−12	2.7E−13
Am-243	1.3E−03	5.0E−10	2.8E−10	2.6E−10	5.8E−11
Cm-243	8.3E−07	3.2E−13	1.8E−13	1.6E−13	3.6E−14
Cm-244	5.0E−05	1.9E−11	1.1E−11	1.0E−11	2.2E−12
Cm-245	1.5E−05	5.8E−12	3.3E−12	3.0E−12	6.6E−13
Cm-246	2.2E−06	8.6E−13	4.8E−13	4.4E−13	9.8E−14

[a] The volume inside the concrete structure, including the porosity of the concrete walls.

7.2 Conditions of the waste packaging at repository closure

The waste will be packed in the following two types of packaging:

- Steel tanks, see Section 3.4.1, containing mostly cement-stabilised metallic waste from the nuclear power plants for disposal in BHK.
- Waste packaging, see Section 3.4.3, containing the drums, moulds and odd containers, see Section 3.4.2, currently stored at the Studsvik site for disposal in BHA.

The condition of the waste packaging at repository closure is dependent on the time of storage prior to disposal and the climate in the storage facility as well as the climate in SFL and the time of storage in SFL prior to closure of the repository.

The most important mechanism that may affect the properties of the waste packaging is corrosion. In the repository, corrosion of the waste packaging can be rapid and extensive if significant levels of moisture and oxygen are available (SKB 2014f). Corrosion can be reduced by controlling the climate in the repository and reducing the intrusion of groundwater from the surrounding bedrock. The following can be anticipated.

Steel tanks for BHK: The steel tanks are provided with a surface coating which prevents corrosion. For that reason, it is anticipated that the surface of the steel tanks will not be affected by prolonged storage prior to closure of the repository.

Waste packaging for BHA: The waste packages currently stored at the Studsvik site will be placed in new waste packaging just prior to disposal. For that reason, the waste packaging is not anticipated to corrode prior to disposal.

Table 7-5 shows the variables that will be used to describe the initial state of the waste packaging that will be used in the future SFL.

Table 7-5. Variables used to describe the initial state of the waste packaging in SFL and their definition (from SKB 2014a).

Variable	Definition
Geometry	Volume and dimensions of the waste packaging. Porosity and pore characteristics of the waste packaging. Amount and characteristics of fractures in the waste packaging.
Temperature	Temperature.
Hydrological variables	Magnitude, direction and distribution of water flow. Degree of saturation. Amount of water. Water pressure. Aggregation state (water and/or ice).
Mechanical stresses	Stress and strain in waste packaging.
Material composition	Amount, composition and surface characteristics of materials in the waste packaging. Type and amount of chemicals. Extent of cement hydration in concrete. Type, amount of organic materials and other substances that can be used by microbes as nutrients and energy sources. Type and amount of microbes and bacteria and other types of biomass.
Water composition	Composition of water including radionuclides. Redox, pH, ionic strength, concentration of dissolved species, type and amount of colloids and/or particles, amount and composition of dissolved gas. Type and amount of microbes and bacteria and other types of biomass. Density and viscosity.
Gas variables	Amount and composition including radionuclides. Volume, pressure and degree of saturation. Magnitude, direction and distribution of gas flow.

7.2.1 Geometry

The volume and dimensions of the waste packaging are described in detail in Section 3.4. The steel in the packaging has no porosity and this variable is not relevant for the waste packaging used in SFL.

7.2.2 Temperature

The temperature in the waste packages is controlled by the temperature of the surroundings, see Section 7.1.3.

7.2.3 Hydrological variables

This variable is not relevant for the packaging used in SFL since steel has no porosity.

7.2.4 Mechanical stresses

The mechanical stresses on the waste packaging at repository closure are affected by pressures from the grout inside the waste packaging and from grout between the waste packages. However, as these materials will have hardened at the time of closure of the repository they will not exert any force on the waste packaging and the level of mechanical stresses will be low. Some mechanical stresses may be exerted upon the waste packaging from the load of grout and other waste packages. However, these forces are balanced by the grout inside the packaging and the resulting force is therefore very low.

7.2.5 Material composition

All waste packaging will be manufactured from carbon steel. The steel tanks in BHK will be painted with a corrosion resistant coating. The waste packaging in BHA might also be painted in the same manner.

7.2.6 Water composition

This variable is not relevant for the waste packaging in SFL since steel has no porosity.

7.2.7 Gas composition

This variable is not relevant for the waste packaging in SFL since steel has no porosity.

7.3 Conditions of the engineered barrier system components in BHK at repository closure

The concrete barriers include the grout surrounding the waste packages and the concrete structures including the bottom, lid and outer as well as inner walls in the repository. Shotcrete will be used to stabilise the rock during the operational phase.

Table 7-6 shows the variables used to describe the properties of the engineered barriers at the initial state.

Table 7-6. Variables used to describe the properties of the engineered barriers in BHK and BHA at repository closure, from SKB (2014b).

Variable	Definition
Geometry	Volume and dimensions of the barriers. Porosity and pore characteristics of the barriers.
Temperature	Temperature.
Hydrological variables	Magnitude, direction and distribution of water flow. Degree of saturation. Amount of water. Water pressure. Effective diffusivity. Aggregation state (water and/or ice).
Mechanical stresses	Stress and strain in the barriers.
Material composition	Amount, composition and surface characteristics of materials in the barriers. Type and amount of chemicals. Type, amount of organic materials and other substances that can be used by microbes as nutrients and energy sources. Type and amount of microbes and bacteria.
Water composition	Composition of water including radionuclides. Redox potential, pH, ionic strength, concentration of dissolved species, type and amount of colloids and/or particles, amount and composition of dissolved gas. Density and viscosity. Type and amount of microbes, bacteria and other types of biomass.
Gas variables	Amount, composition including radionuclides. Volume, pressure and degree of saturation. Magnitude, direction and distribution of gas flow.

7.3.1 Geometry

The design of the barrier system in BHK is described in Section 4.1. The dimensions are described in Section 4.3.

The porosity can vary between 9 and 15 % in the construction concrete, see for example Höglund (2014). For the evaluation of the post-closure safety a porosity of about 11 % is assumed for the concrete in the concrete structure and 30 % in the grout, respectively. No decision has yet been made on the quality of the concrete material used as vault backfill. Fractures caused during the hydration of the concrete in the concrete structure with a width of about 0.1 mm may be present.

7.3.2 Temperature

The temperature in the waste vault is set by the surrounding rock temperature, see also Section 7.1.3.

7.3.3 Hydrological variables

The amount of water present at the initial state in the cementitious materials in BHK will be determined by the composition of the materials as well as the climatic conditions in BHK during the operational phase of the repository.

The hydraulic conductivities of the different concrete types are given in SKB (2014d, Table 10-4).

The effective diffusivity in the concrete at the time of closure is assumed to be 3.5×10^{-12} m^2/s (Höglund 2014, Table 9-1).

Initially, all water within the concrete and grout are in liquid form due to the temperature, see Section 7.1.3.

7.3.4 Mechanical stresses

Mechanical stresses which may affect the development of the transport properties of the concrete structure and backfill in BHK will prevail over the entire operational phase of the repository. These stresses may be either load-independent such as the tensile stresses caused during the construction phase or load-dependent such as stresses caused by the loads from the grout during grouting of the waste or from the weight of the waste and the concrete structure.

The load-independent tensile stresses caused during the construction of the concrete structure will remain over the entire operational phase of the repository unless extensive fracture formation occurs. However, for fracturing to occur, the tensile stresses must exceed the tensile strength of the material. For that reason, tensile stresses in the range of the tensile strength of the concrete used will always prevail. This is true for all parts of the concrete structure as well as for the grout and the backfill concrete/grout.

The walls of the concrete structure are subjected to pressure from grout during grouting of the space between the waste packages as well as from the backfill material. The magnitude of this pressure can be carefully controlled by the setting time of the grout/concrete and the rate with which grouting/backfilling is done. At repository closure, these materials will have hardened and do not any longer exert any pressure on the concrete structures or on other parts of the engineered barrier materials.

The body load of the concrete itself together with the weight of the waste, grout and backfill will introduce stresses in the concrete. In the base slab in BHK, the stresses will be dominated by compressive stresses due to the weight of the above materials.

7.3.5 Material composition

The compositions of the concrete to be used in structural parts of BHK, the grout used to fill the space between the waste packages inside the concrete structure and the concrete used as backfill material in the waste vault have not been determined in detail. However, it is expected that a majority of the materials used will be based on standard formulations and that the amount of organic additives will be limited.

Standard Degerhamn Anläggningscement is assumed for the concrete in the concrete structures in BHK. The chemical composition of this cement is presented in Table 7-7. The composition of the grout to be used for filling the voids between the waste packages as well as for the vault backfill material has not yet been decided. As examples of composition of grout and concrete the formulations for the corresponding materials currently used in SFR are given in Table 7-8.

Table 7-7. Chemical composition of Degerhamn Anläggningscement, including both the oxide composition and the corresponding clinker mineral composition as given in Alemo (1992).

Component	Content % by weight
Ca	64
SiO_2	21
Al_2O_3	3.5
Fe_2O_3	4.6
MgO	0.7
K_2O	0.62
Na_2O	0.07
SO_3	2.2
Cl	< 0.1
Free $CaCO_3$	0.9
Corresponding clinker components	
Tricalcium silicate, C_3S	64.4
Dicalcium silicate, C_2S	10.9
Tricalcium aluminate, C_3A	2.0
Tetracalcium aluminate ferrite, C_4AF	13.9
Calcium sulphate (gypsum), $C\check{s}H_2$	3.7
Alkali hydroxides, N + K	0.7

Abbreviations used for the clinker components: C = CaO, S = SiO_2, A = Al_2O_3, F = Fe_2O_3, H = H_2O, $C\check{s}H_2$ = $CaSO_4{\cdot}2H_2O$, N = Na_2O, K = K_2O

Degerhamn Anläggningscement satisfies the requirements of EN 197-1 Cement-Part 1: Composition, specifications and conformity criteria for common cements and is in accordance with SS 13 42 02 – 03 for MH/LA (medium heat/low alkali). Anläggningscement has a low C_3A content and satisfies the requirements (i) for sulfate resistance of SR 3 type cement in EN 197-1, for low alkali cement in accordance with SS 13 42 03 and (ii) for cement with moderate heat development in accordance with SS 13 42 02, see Table 7-8 for the mixing proportions in SFR which are also assumed for SFL.

When one compartment has been entirely filled with waste the space between the waste packages will be filled with a concrete grout in order to reduce the void and also to stabilise the stack of waste packages as well as to increase the time for leaching of the cementitious components.

Table 7-8. Mixing proportions for concrete structures in the existing part of SFR, amounts given in kg/m^3.

Component	Construction concrete (SKB 2014d)	Grout (SKB 2014d)	Conditioning cement
Degerhamn anläggningscement	350	325	1180
Water	164.5	366	437
Ballast	1829 (total)	1302	-
	0–8 mm 920 kg/m^3 8–16 mm 374 kg/m^3 16–32 mm 535 kg/m^3		
Additives (anti-foaming, cellulose[1])	0.5 % Sika Plastiment BV-40 0.05–0.2 % Sika Retarder	6.5	-
Air	-	2.5 % by volume	-
w/c ratio	0.47 (0.46–0.49)	1.125	0.37

[1] Not allowed in future grout.

The ballast material will be selected to comply with Swedish standards on resistance to alkali-silica reactions.

The concrete structures within BHK are deemed not to contribute any significant amounts of nutrients for microbes compared with the nutrient amounts found in the infiltrating groundwater. The microbial population initially present in the concrete structures mainly depends on the microbial population established during the operational phase.

7.3.6 Water composition

The water composition in the pore systems of the different cementitious materials in BHK will at repository closure be entirely determined by the composition of the cement used for fabrication. At this stage, no interactions with the groundwater have occurred.

The composition of two different types of cement pore water is given in Table 7-2. Of these the composition of the pore water in BHK is expected to closely resemble that of the fresh cement described by Lagerblad and Trägårdh (1994).

Initially, no radionuclides, except for those naturally occurring in the groundwater, will be present in the water phase.

The density and viscosity of the free water is assumed to be the same as tabulated values found in the literature for the prevailing salinity and temperature in the repository.

Reducing conditions will prevail in SFL shortly after closure due to, for example, corrosion of the extensive amount of iron present in the repository (Duro et al. 2012). Colloids are judged not to be stable within the concrete structures for the same reasons as presented in Section 7.1.8.

7.3.7 Gas variables

Initially, dissolved gases could be present in the concrete system components originating from air that is present in the pores.

As the repository fills with water, the most abundant gas is the undissolved air that may be found in air pockets and pores in the materials surrounding the waste packaging, and concrete system components.

7.4 Conditions of the engineered barrier system components in BHA at repository closure

The engineered barrier system components are defined as the grout surrounding the waste packages, concrete structures and bentonite.

Table 7-6 shows the variables used to describe the properties of the engineered barriers at the initial state.

7.4.1 Geometry

The design of BHA is described in Section 5.1. The dimensions are given in Section 5.3.

The porosity in the concrete structure and grout is assumed to be the same as in BHK, see Section 7.3.1. It cannot be excluded that fractures (> 0.1 mm) exist initially in the grouting.

The diffusion-available porosity in the bentonite is assumed to be 43 % for cationic- and non-charged species and 17 % for anionic species (Ochs and Talerico 2004, p 90).

7.4.2 Temperature

The temperature in the waste vault is set by the surrounding rock temperature, see Section 7.1.3.

7.4.3 Hydrological variables

The amount of water present at repository closure in the cementitious materials in BHA will be determined by the composition of the materials as well as the climatic conditions in BHA during the operational period of the repository. The hydraulic conductivities of the different concrete types are given in SKB (2014d, Table 10-4). The water content of the bentonite is adjusted to about 15–20 % when the bentonite blocks are manufactured. At closure, the bentonite has not yet been saturated by the groundwater and can be considered to have the same level of humidity as when manufactured. For the analysis it is assumed that a MX-80 bentonite is used that has a target dry density of about 1 570 kg/m^3 and a saturated density of 2 000 kg/m^3, which implies a hydraulic conductivity of 1×10^{-13} m/s (SKB 2010c). The hydraulic conductivities of the different concrete types are given in SKB (2014d, Table 10-4).

The effective diffusivity of non-charged and hydrolysable elements (cations except from cesium) in the bentonite is assumed to be 1.2×10^{-10} m^2/s, whereas it is 1.0×10^{-11} m^2/s for anions and 3.0×10^{-10} m^2/s for cesium ions (Ochs and Talerico 2004, p 90).

Initially all water within the bentonite, concrete and grout is in liquid form due to the temperature, see Section 7.1.3.

7.4.4 Mechanical stresses

The mechanical stresses in the concrete structure in BHA will be affected by the large swelling pressure from the bentonite but this swelling pressure will not be fully developed until the bentonite has been saturated. Thus, the mechanical stresses from swelling of bentonite are assumed to be negligible at closure.

Bentonite can be plastically deformed and all forces will lead to a redistribution of the material rather than to a build-up of mechanical stresses. Hence, no mechanical stresses are expected in the bentonite.

7.4.5 Material composition

For details on the composition of the cementitious materials in this section, see Section 7.3.5. A MX-80 type of bentonite is used in the initial design work since it is a well-characterised material that was identified as one of two possible materials to be used in the repository for spent fuel in SR-Site (Karnland et al. 2006).

7.4.6 Water composition

The water composition in the pore systems of the different cementitious materials in BHA will at repository closure be entirely determined by the composition of the cement used for fabrication. At this stage, no interactions with the groundwater will have occurred.

The composition of two different types of cement pore water is given in Table 7-2. Of these, the composition of the pore water in BHA is expected to closely resemble that of the fresh cement described by Lagerblad and Trägårdh (1994).

The water initially in contact with the bentonite will have the composition of an undisturbed groundwater and the composition will be entirely determined by the properties of the bedrock and the conditions prevailing in the repository.

When the bentonite has been saturated with groundwater and the groundwater comes in contact with the concrete structure its composition will be altered, since the cement minerals, such as the portlandite and alkali hydroxides, will be dissolved.

Initially, no radionuclides, except for those naturally occurring in the groundwater, will be present in the water phase.

The density and viscosity of the free water is assumed to be the same as tabulated values found in the literature for the prevailing salinity and temperature in the repository.

Reducing conditions will prevail in SFL shortly after closure due to for example corrosion of the extensive amount of iron present in the repository (Duro et al. 2012). Colloids are judged not to be stable within the concrete structures for the same reasons as presented in Section 7.1.8.

7.4.7 Gas variables

Initially, dissolved gases could be present in the concrete system components originating from air that is present in the pores.

As the repository fills with water, the most abundant gas is the undissolved air that may be found in air pockets and pores in the materials surrounding the waste packages.

8 Summary and conclusions

This chapter gives a summary of the conditions of the repository at the initial state, defined as the expected state of the repository immediately after closure. The initial state of the repository is based on the current knowledge of the properties of the waste and the repository components, and, an assessment of changes in these properties up to the time of closure. The estimated year of closure is 2075.

8.1 BHK waste vault

In BHK, the radioactive waste is deposited in a concrete structure in the form of caissons. The steel tanks holding the waste are filled with grout to reduce the total void in the repository and to increase the stability of the steel tanks. Inside the caissons, the waste packages are embedded in grout prior to, or at the time of, closure. The space above the concrete structure and between the caissons and the rock wall is filled with concrete of the same quality as the walls and floor of the concrete structure.

Condition of the components

The conditioned waste is surrounded by the following components:

- Waste packaging.
- Grout in caissons.
- Concrete structure, caisson walls and bottom and concrete lid.
- Backfill of concrete.
- Closure components.

The waste packaging in this waste vault is made of steel. The integrity of the possible corrosion protective painting of the waste packaging cannot be guaranteed and corrosion during the operational phase cannot be excluded.

Other system components that are emplaced prior to closure are expected to be in the condition the items had at the time of emplacement, i.e. changes during the operational phase are assumed to be negligible.

In order to achieve the desired initial state of the concrete structure, a detailed control and inspection programme will be developed prior to the construction and operation phases of the repository.

8.2 BHA waste vault

In BHA, the radioactive waste is stored inside a concrete structure which is surrounded by a thick layer of bentonite clay. The waste packaging (drums and moulds) in which the waste is currently stored will be placed in new waste packaging and the voids between the original packages will be filled with grout. The waste packages will also be embedded in grout prior to closure. At closure of the repository, the space between the concrete structure and the bedrock will be filled with high quality bentonite blocks and pellets.

Condition of the components

The conditioned waste is surrounded by the following components:

- Waste packaging.
- Grout inside the concrete structure.
- Walls and bottom of concrete structure.
- Lid of concrete.
- Backfill of bentonite.
- Closure components.

The waste packaging in this waste vault is made of steel. The integrity of the possible corrosion protective painting of the waste packaging cannot be guaranteed and corrosion during the operational phase cannot be excluded.

Other system components that are emplaced prior to closure are expected to be in the condition the items had at the time of emplacement, i.e. changes during the operational phase are assumed to be negligible.

8.3 Closure components

Closure components are installed shortly before closure and are expected to be in good condition at closure.

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Appendix 1

Initial state activity in metal category including very thin metal parts in BHK.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Ac-227	0.00E+00	Eu-152	3.73E+04	Ra-228	0.00E+00
Ag-108m	5.40E+01	Gd-148	0.00E+00	Re-186m	0.00E+00
Am-241	9.20E+07	H-3	2.16E+07	Se-79	2.87E+04
Am-242m	8.34E+04	Ho-166m	8.45E+01	Si-32	0.00E+00
Am-243	1.06E+06	I-129	3.27E+04	Sm-151	1.08E+08
Ar-39	0.00E+00	K-40	0.00E+00	Sr-90	2.58E+10
Ba-133	7.21E+02	La-137	0.00E+00	Tb-157	0.00E+00
Be-10	0.00E+00	Mo-93	1.78E+11	Tb-158	0.00E+00
C-14-ind	1.30E+12	Nb-93m	1.67E+12	Tc-99	2.27E+10
C-14-inorg	0.00E+00	Nb-94	2.14E+10	Th-228	0.00E+00
C-14-org	0.00E+00	Ni-59	7.98E+12	Th-229	0.00E+00
Ca-41	0.00E+00	Ni-63	8.06E+14	Th-230	0.00E+00
Cd-113m	9.63E+03	Np-237	6.29E+05	Th-232	0.00E+00
Cl-36	4.12E+08	Pa-231	0.00E+00	Ti-44	0.00E+00
Cm-242	0.00E+00	Pb-210	0.00E+00	U-232	1.39E+06
Cm-243	1.50E+05	Pd-107	1.97E+04	U-233	0.00E+00
Cm-244	1.93E+07	Po-210	0.00E+00	U-234	0.00E+00
Cm-245	2.35E+04	Pu-238	3.16E+09	U-235	4.32E-01
Cm-246	9.13E+03	Pu-239	7.08E+06	U-236	4.70E+05
Co-60	9.06E+12	Pu-240	6.81E+06	U-238	0.00E+00
Cs-135	3.87E+05	Pu-241	1.66E+08	Zr-93	2.37E+06
Cs-137	2.79E+10	Pu-242	8.14E+04		
Eu-150	0.00E+00	Ra-226	0.00E+00		

Initial state activity in metal category including 1 mm thick metal parts in BHK.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Ac-227	0.00	Eu-152	1.86E+05	Ra-228	0.00
Ag-108m	2.94E+02	Gd-148	0.00	Re-186m	0.00
Am-241	8.02E+08	H-3	7.32E+14	Se-79	1.49E+05
Am-242m	1.86E+06	Ho-166m	4.38E+02	Si-32	0.00
Am-243	6.08E+06	I-129	1.70E+05	Sm-151	5.50E+08
Ar-39	0.00	K-40	0.00	Sr-90	1.44E+11
Ba-133	3.79E+03	La-137	0.00	Tb-157	0.00
Be-10	7.04E+02	Mo-93	9.27E+12	Tb-158	0.00
C-14-ind	5.06E+13	Nb-93m	5.84E+13	Tc-99	1.47E+12
C-14-inorg	0.00	Nb-94	7.88E+11	Th-228	0.00
C-14-org	0.00	Ni-59	3.68E+14	Th-229	0.00
Ca-41	8.80E-06	Ni-63	3.27E+16	Th-230	0.00
Cd-113m	1.03E+05	Np-237	3.22E+06	Th-232	0.00
Cl-36	2.32E+10	Pa-231	0.00	Ti-44	0.00
Cm-242	0.00	Pb-210	0.00	U-232	8.00E+06
Cm-243	1.03E+07	Pd-107	1.02E+05	U-233	0.00
Cm-244	8.11E+07	Po-210	0.00	U-234	0.00
Cm-245	7.19E+07	Pu-238	1.47E+10	U-235	6.59E+02
Cm-246	5.39E+04	Pu-239	8.40E+09	U-236	2.27E+06
Co-60	3.31E+14	Pu-240	6.21E+07	U-238	0.00
Cs-135	2.01E+06	Pu-241	9.38E+08	Zr-93	6.14E+11
Cs-137	1.57E+11	Pu-242	1.11E+09		
Eu-150	0.00	Ra-226	0.00		

Initial state activity in metal category including 1 cm thick metal parts in BHK.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Ac-227	0.00	Eu-152	2.87E+03	Ra-228	0.00
Ag-108m	8.34E+00	Gd-148	0.00	Re-186m	0.00
Am-241	9.73E+06	H-3	6.11E+09	Se-79	5.39E+03
Am-242m	1.24E+04	Ho-166m	1.32E+01	Si-32	0.00
Am-243	1.67E+05	I-129	5.15E+03	Sm-151	1.56E+07
Ar-39	0.00	K-40	0.00	Sr-90	2.98E+09
Ba-133	4.31E+01	La-137	0.00	Tb-157	0.00
Be-10	3.92E+03	Mo-93	6.65E+12	Tb-158	0.00
C-14-ind	1.35E+14	Nb-93m	4.99E+13	Tc-99	9.80E+11
C-14-inorg	0.00	Nb-94	8.70E+11	Th-228	0.00
C-14-org	0.00	Ni-59	5.53E+14	Th-229	0.00
Ca-41	0.00	Ni-63	5.41E+16	Th-230	0.00
Cd-113m	7.15E+03	Np-237	9.91E+04	Th-232	0.00
Cl-36	2.77E+10	Pa-231	0.00	Ti-44	0.00
Cm-242	0.00	Pb-210	0.00	U-232	1.93E+05
Cm-243	1.76E+04	Pd-107	3.10E+03	U-233	0.00
Cm-244	1.82E+06	Po-210	0.00	U-234	0.00
Cm-245	3.69E+03	Pu-238	4.52E+08	U-235	7.14E−02
Cm-246	1.44E+03	Pu-239	1.12E+06	U-236	7.40E+04
Co-60	5.08E+14	Pu-240	1.05E+06	U-238	0.00
Cs-135	6.10E+04	Pu-241	1.34E+07	Zr-93	7.02E+05
Cs-137	3.27E+09	Pu-242	1.28E+04		
Eu-150	0.00	Ra-226	0.00		

Initial state activity in metal category including 2 cm thick metal parts in BHK.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Ac-227	0.00	Eu-152	2.46E+04	Ra-228	0.00
Ag-108m	1.95E+12	Gd-148	0.00	Re-186m	0.00
Am-241	1.51E+07	H-3	5.77E+13	Se-79	7.00E+10
Am-242m	1.74E+04	Ho-166m	1.56E+01	Si-32	0.00
Am-243	1.94E+05	I-129	5.95E+03	Sm-151	2.43E+07
Ar-39	0.00	K-40	0.00	Sr-90	8.84E+09
Ba-133	6.57E+02	La-137	0.00	Tb-157	0.00
Be-10	0.00	Mo-93	6.36E+12	Tb-158	0.00
C-14-ind	1.71E+14	Nb-93m	7.46E+14	Tc-99	8.86E+11
C-14-inorg	0.00	Nb-94	1.19E+13	Th-228	0.00
C-14-org	0.00	Ni-59	8.15E+14	Th-229	0.00
Ca-41	3.79E+10	Ni-63	9.15E+16	Th-230	0.00
Cd-113m	1.43E+09	Np-237	1.14E+05	Th-232	0.00
Cl-36	5.32E+10	Pa-231	0.00	Ti-44	0.00
Cm-242	0.00	Pb-210	0.00	U-232	3.30E+05
Cm-243	5.02E+04	Pd-107	3.59E+03	U-233	0.00
Cm-244	9.45E+06	Po-210	0.00	U-234	0.00
Cm-245	4.28E+03	Pu-238	7.12E+08	U-235	4.42E-02
Cm-246	1.67E+03	Pu-239	1.29E+06	U-236	8.56E+04
Co-60	4.68E+15	Pu-240	1.23E+06	U-238	0.00
Cs-135	7.05E+04	Pu-241	1.03E+08	Zr-93	1.01E+08
Cs-137	9.33E+09	Pu-242	1.48E+04		
Eu-150	0.00	Ra-226	0.00		

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SKB is responsible for managing spent nuclear fuel and radioactive waste produced by the Swedish nuclear power plants such that man and the environment are protected in the near and distant future.

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