Technical Report

TR-13-14

SFL concept study

Main report

Mattias Elfwing, Lena Z Evins, Mikael Gontier, Pär Grahm, Per Mårtensson, Sofie Tunbrant

Svensk Kärnbränslehantering AB

December 2013

Svensk Kärnbränslehantering AB
Swedish Nuclear Fuel
and Waste Management Co

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

ISSN 1404-0344
SKB TR-13-14
ID 1335573

SFL concept study

Main report

Mattias Elfwing, Lena Z Evins, Mikael Gontier, Pär Grahm,
Per Mårtensson, Sofie Tunbrant

Svensk Kärnbränslehantering AB

December 2013

Keywords: SFL, Long-lived low and intermediate level waste, Concept study.

A pdf version of this document can be downloaded from www.skb.se.

Abstract

This report presents a study conducted to identify and evaluate possible solutions for management and disposal of the Swedish long-lived low and intermediate level waste. The aim is to examine all possible options, regardless of SKB's previous planning for the repository for long-lived waste (SFL). The study has identified laws and regulations from legislators and authorities and other requirements from owners and the public as a basis for the evaluation of the various alternative solutions.

The study includes the following steps:

- Identification of possible strategies.
- Identification of repository concepts for the implementation of a particular strategy.
- Identification of other system components which, together with the repository concept, form a complete system for management of SFL waste.
- Identification and selection of evaluation method and evaluation criteria.
- Evaluation and selection of repository concept for further assessment with respect to long-term radiological safety.

In the first step, different strategies for the disposal of long-lived low and intermediate level waste have been identified. In the next step, various concepts for realizing these strategies have been identified. Among the identified concepts, those that comply with laws and international conventions have been selected for further analysis. The repository concepts have been detailed with functions for storage, transportation etc required to create a system for management and disposal. A system consists of a complete set of facilities and components required to implement a particular strategy.

In parallel with this identification of strategies and repository concepts, requirements on the system and the system components have been identified from laws, regulations and international conventions. These requirements constitute the basis for the formulation of evaluation factors used to assess whether a concept meets the requirements and to compare the repository concepts that fulfil the requirements with each other. Furthermore, various studies have been conducted to highlight differences between concepts. These investigations support the final evaluation.

In the comparative evaluation, an overall assessment of the concepts has been undertaken using evaluation factors related to long-term safety, environment, technology, cost and time. The evaluation takes into account the whole handling chain from waste generation to the sealed repository and thus also includes waste packages, conditioning methods, etc. The need for research and technology development, as well as the construction and operating phases, have also been considered.

Following the evaluation, a system for the management and disposal of long-lived low and intermediate level waste is proposed to be further assessed with respect to long-term safety. The outcome of the assessment will determine whether the system meets the requirements on post-closure safety and may constitute SKB's main alternative for the future development and planning of SFL.

Proposed system for further assessment of long-term safety

The proposed system for management and disposal of the Swedish long-lived low and intermediate level waste consists of the following components:

Waste and containers

The waste is separated into two fractions:

- metallic waste from the nuclear power plants, and
- waste from AB SVAFO and Studsvik Nuclear AB.

The metallic waste from the nuclear power plants, including the BWR control rods, is segmented and placed in steel tanks and stabilized in the tanks with grout. The waste from AB SVAFO and Studsvik Nuclear AB is placed in containers designed for SFL and stabilized in the containers with grout.

System for external transportation

The system for external transportation is based on SKB's current system for external transportation. Future planning, including the future selection of the repository site, will decide which additional parts are needed for the transportation of SFL waste, e.g. waste transport containers.

Facilities for conditioning of waste

To implement the proposed solution up to three conditioning facilities will be built. The conditioning facilities will:

- segment BWR control rods and load the segments into steel tanks,
- stabilize metallic waste in steel tanks with grout, and
- load waste from AB SVAFO and Studsvik Nuclear AB into SFL containers.

The loading station for waste from AB SVAFO and Studsvik Nuclear AB should be located at the Studsvik site, since it already hosts this waste fraction. The location of the other facilities will be subject to further analysis, where co-location with the loading station is one of the options to be considered.

Storage facilities

Storage of SFL waste is managed by making use of existing storage capacity at the nuclear power plant sites and in Studsvik, and planned storage capacity in SFR. Additional storage facilities may be added in the future.

Repository and repository facilities

SFL is designed as a deep geological repository with two different sections:

- one section for the metallic waste from the nuclear power plants, and
- one section for the waste from AB SVAFO and Studsvik Nuclear AB.

The repository section for the metallic waste is designed with a concrete barrier. The waste is segmented, after which the parts are deposited in steel tanks and stabilized with grout. The steel tanks are emplaced in the repository. This section of the repository is backfilled with concrete, which acts as a barrier against hydraulic flow and contributes to a low diffusion rate and high sorption of many radionuclides. The concrete in the barrier will create an alkaline environment in the repository section, reducing the corrosion rate of the steel and thus limiting the release rate of radionuclides.

The repository section for the waste from AB SVAFO and Studsvik Nuclear AB is designed with a bentonite barrier. The waste is deposited in containers designed for SFL and stabilized with grout. These containers are emplaced in the repository. The section is backfilled with bentonite, but there will be no bentonite in the repository during the operating phase. The bentonite acts as a barrier by limiting the groundwater flow and making diffusion the dominant transport mechanism for radionuclides through the bentonite. Bentonite clay also has the ability to efficiently filter colloids.

Sammanfattning

I denna rapport redovisas en studie som genomförts för att identifiera och utvärdera möjliga lösningar för slutligt omhändertagande av det svenska långlivade låg- och medelaktiva avfallet. Syftet är att genomlysa alla tänkbara alternativ, oberoende av SKB:s tidigare planering för slutförvaret för långlivat avfall (SFL). Som grund för utvärdering av de olika lösningsalternativen har studien sammanställt de lagar och författningar som lagstiftare och myndigheter beslutat om samt övriga krav som ägare och allmänheten ställer.

Studien omfattar:

- Identifiering av möjliga strategier.
- Identifiering av slutförvarskoncept för att genomföra en viss strategi.
- Identifiering av övriga systemdelar som, tillsammans med slutförvarskonceptet, utgör ett komplett system för omhändertagande av SFL-avfallet.
- Identifiering och val av utvärderingsmetod och kriterier för utvärdering.
- Utvärdering och val av förvarskoncept för fortsatt analys med avseende på långsiktig radiologisk säkerhet.

I det första steget har olika strategier för slutligt omhändertagande av långlivat låg- och medelaktivt avfall identifierats. I nästa steg har olika koncept för att realisera dessa strategier identifierats och av dessa har de koncept som uppfyller lagar och internationella konventioner valts ut för vidare analys. Koncepten för slutförvaring har kompletterats med funktioner för mellanlagring, transport etc som krävs för att skapa ett system för hantering och slutförvaring. Ett system utgörs av en komplett uppsättning anläggningar och delar som krävs för att genomföra en viss strategi.

Parallellt med arbetet att identifiera strategier och förvarskoncept har krav på systemet och ingående delar identifierats utifrån lagar, föreskrifter och internationella konventioner. Dessa krav har utgjort basen för formuleringen av de utvärderingsfaktorer som används för att bedöma om ett koncept uppfyller kraven, och för att jämföra de slutförvarskoncept som uppfyller kraven med varandra. Vidare har olika utredningar genomförts för att belysa skillnader mellan koncepten. Dessa utredningar utgör underlag för den avslutande utvärderingen.

Vid den jämförande värderingen har en samlad bedömning av koncepten gjorts med stöd av utvärderingsfaktorer kopplade till långsiktig säkerhet, miljö, teknik, kostnad och tid. Utvärderingen har tagit hänsyn till hela hanteringskedjan från avfallets uppkomst till det förslutna förvaret, och innefattar således även avfallskollin, konditioneringsmetoder etc. Även behov av forskning och teknikutveckling liksom bygg- och driftskeden har beaktats.

Utvärderingen mynnar ut i ett förslag till ett system för hantering och slutförvaring av långlivat låg- och medelaktivt avfall, att analysera vidare med avseende på långsiktig säkerhet. Resultatet av analysen kommer att avgöra om systemet uppfyller kraven på säkerhet efter förslutning och kan utgöra SKB:s huvudalternativ för fortsatt utveckling och planering av SFL.

Förslag till system för vidare analys av långsiktig säkerhet

Studiens förslag till system för slutligt omhändertagande av det svenska långlivade låg- och medelaktiva avfallet består av följande delar:

Avfall och containrar

Avfallet separeras i två fraktioner:

- metalliskt avfall från kärnkraftverken, och
- avfall från AB SVAFO och Studsvik Nuclear AB.

Den metalliska avfallet från kärnkraftverken, inklusive BWR-styrstavar, segmenteras och deponeras i ståltankar och fixeras i tankarna med cementbruk. Avfallet från AB SVAFO och Studsvik Nuclear AB deponeras i slutförvaret i behållare utformade för SFL och fixeras i dessa behållare med cementbruk.

System för externa transporter

Systemet för externa transporter baseras på SKB:s nuvarande system för externa transporter. Framtida planering, inbegripet lokaliseringen av slutförvaret, kommer att avgöra vilka ytterligare delar som måste läggas till transportsystemet för att utföra transport av SFL-avfall, exempelvis avfallstransportbehållare.

Anläggningar för konditionering av avfall

För att genomföra den föreslagna lösningen uppförs upp till tre konditioneringsanläggningar. Konditioneringsanläggningarna ska:

- segmentera BWR-styrstavar och lasta delarna i ståltankar,
- fixera metalliskt avfall i ståltankar med cementbruk och
- lasta avfall från AB SVAFO och Studsvik Nuclear AB i behållare utformade för SFL.

Lastningsanläggningen för avfall från AB SVAFO och Studsvik Nuclear AB bör vara placerad i Studsvik, eftersom detta avfall redan mellanlagras där. Placeringen av övriga anläggningar är föremål för vidare analys, där samlokalisering med lastningsanläggningen är ett av de alternativ som bör övervägas.

Anläggningar för mellanlagring

Mellanlagring av SFL-avfall hanteras genom att använda befintlig kapacitet för mellanlagring vid kraftverken och i Studsvik och den planerade kapaciteten i det utbyggda SFR. Ytterligare anläggningar för mellanlagring kan tillkomma i framtiden.

Slutförvar och slutförvarsanläggningar

SFL utformas som ett geologiskt djupförvar med två olika förvarsdelar:

- en förvarsdel för metalliskt avfall från kärnkraftverken och
- en förvarsdel för avfallet från AB SVAFO och Studsvik Nuclear AB.

Förvarsdelen för metalliskt avfall utformas med en betongbarriär. Avfallet segmenteras, varefter delarna placeras i ståltankar och kringgjuts med cementbruk. Ståltankarna deponeras i förvarsutrymmet. Denna del av slutförvaret återfylls med betong, vilken fungerar som en barriär mot grundvattenflöde och bidrar till låg diffusionshastighet och hög sorption av många radionuklider. Betongen i barriären kommer att skapa en alkalisk miljö i förvarsdelen, vilket reducerar korrosionshastigheten hos stål och därmed begränsar frigörelsehastigheten av radionuklider.

Förvarsdelen för avfall från AB SVAFO och Studsvik Nuclear AB utformas med en bentonitbarriär. Avfallet placeras i behållare utformade för SFL och kringgjuts. Dessa behållare deponeras i förvarsutrymmet. Förvarsdelen återfylls med bentonit, men ingen bentonit placeras i förvarsutrymmet under driftfasen av förvaret. Bentoniten fungerar som en barriär genom att begränsa grundvattenflödet och göra diffusion till den dominerande transportmekanismen för radionuklider genom bentoniten. Bentonitlera har också förmågan att effektivt filtrera kolloider.

Contents

1 Introduction

The Swedish power industry has been generating electricity by means of nuclear power for about 40 years. The Swedish system for managing and disposal of the waste from operation of the reactors has been developed during that period of time. The system is based on the different types of nuclear waste with separate repositories for different types of waste. When finalized, the Swedish 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.

SKB currently operates SFR in Östhammar municipality, and an extension of SFR is planned, mainly to permit the disposal of decommissioning waste from the nuclear power plants and SKB's facilities. The system for disposal of the spent nuclear fuel consists today of the storage facility for spent nuclear fuel in Oskarshamn municipality (Clab) and will be completed by the construction and commissioning of the Encapsulation Plant and the Spent Fuel Repository. The current Swedish system also includes a ship (m/s Sigrid) and casks for transport.

SKB plans to dispose of the long-lived low and intermediate level waste in SFL. The waste comprises waste from the operation and decommissioning of the Swedish nuclear power plants, legacy waste from the early research in the Swedish nuclear programmes, and smaller amounts of waste from hospitals, industry and research.

The long-lived low and intermediate level waste from the nuclear power plants consists of neutron-irradiated components and control rods. The long-lived nuclides are formed from stable elements in, for example, steel when they are exposed to strong neutron radiation from the reactor core.

The total quantity of long-lived waste planned for SFL is estimated at about 16,000 m^3, of which about one third originates from the nuclear power plants. The rest comes from AB SVAFO and Studsvik Nuclear AB, who manage the legacy waste and the waste from hospitals, industry and research.

The planning for SFL is still in its early stages and the general plans are outlined in SKB (2013a). This study was undertaken to review the various alternatives available for the disposal of long-lived low and intermediate level waste and to identify feasible options for SKB. The strategy of geological disposal of nuclear waste has been sanctioned by the Swedish political system for many years. This has been disregarded for the present study and we are starting afresh by asking: *How* should we handle and dispose of the Swedish long-lived low and intermediate level waste? The aim of the study is to highlight and evaluate all possible alternatives and at the same time compile the premises, requirements and constraints that form the basis for the evaluation. The study follows the same kind of stepwise evaluation of alternatives as earlier system studies for the disposal of spent nuclear fuel undertaken by SKB (SKB 1992, 2000a).

Following the evaluation, a system for the management and disposal of long-lived low and intermediate level waste is proposed to be further assessed with respect to long-term safety. The outcome of the assessment will determine whether the system meets the requirements on post-closure safety and may constitute SKB's main alternative for the future development and planning of SFL. The evaluation will also further define the requirements on the waste, barriers, and the properties of the future site.

1.1 Legal framework

In Sweden, the legal framework that regulates the construction, ownership and operation of a nuclear facility for the disposal of long-lived low and intermediate level waste includes the following laws and conventions:

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

- Environmental Code (SFS 1998:808).
- Work Environment Act (SFS 1977:1160).
- Planning and Building Act (SFS 2010:900).
- Transport of Dangerous Goods Act (SFS 2006:263).

The regulations issued by the Swedish Radiation Safety Authority based on the Nuclear Activities Act and the Radiation Protection Act further define the legal requirements.

At the European level, the members of the European Union have adopted the following two directives that are relevant for nuclear waste management:

- The Nuclear Safety Directive or Council Directive 2009/71/EURATOM of 25 June 2009 establishing a Community framework for the nuclear safety of nuclear installations.
- The Council Directive 2011/70/EURATOM of 19 July 2011 establishing a Community framework for the responsible and safe management of spent fuel and radioactive waste.

The Nuclear Safety Directive has already been implemented in the Swedish legislation and is therefore integrated in the legislation listed above or the regulations issued by the Swedish Radiation Safety Authority. The directive on management of spent fuel and radioactive waste is not yet implemented at the national level.

At the international level, Sweden has ratified the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management (IAEA 1997), which is implemented in the Swedish legislation (SÖ 1999:60). Furthermore, the International Atomic Energy Agency (IAEA) has the authority to establish or adopt standards of safety for protection of health and minimization of danger to life and property. The IAEA Safety Standards provide a system of fundamental safety principles, safety requirements and safety guides for ensuring safety. They reflect an international consensus on what constitutes a high level of safety for protecting people and the environment from harmful effects of ionizing radiation.

1.2 Objective

The objective of the study is to identify conceivable alternatives for safe disposal of the Swedish long-lived low and intermediate level waste and to evaluate and compare those alternatives.

The evaluation will form the basis for SKB's decision on which concept or concepts will be further assessed. The goal is to suggest no more than two concepts for further assessment of long-term safety during 2014–2016.

1.3 Scope

The study covers the analysis, evaluation and selection of strategies and concepts for safe disposal of the Swedish long-lived low and intermediate level waste.

The study includes the following steps:

- Identify possible strategies for disposal.
- Identify repository concepts that implement a given strategy.
- Identify the other components which, together with the repository concept, form a complete system for management of the SFL waste.
- Identify criteria and method for comparison of repository concepts.
- Evaluate the repository concepts based on the chosen criteria and method.
- Report the findings of the study in a traceable way.

A repository system includes all management of the waste, from the origin of the waste to the disposal of the waste in the repository. This study includes storage and conditioning of waste, research and development necessary to implement a certain concept, construction of the repository, operation of the system, and disposal of the waste.

The study covers the management of all present and future long-lived low and intermediate level waste from the Swedish nuclear power plants, SKB's facilities, AB SVAFO and Studsvik Nuclear AB. The reference inventory from 1998 (SKBdoc 1416968) is used as a starting point for the work. Known changes in the waste flows are taken into account. Findings from the ongoing updating of the reference inventory are considered if possible. Updating of the reference inventory is, however, not a part of the study, but a separate activity.

The work also includes an assessment of the findings in the preliminary safety assessment presented in 1999 (SKB 1999a, b), see section 1.5.

1.4 International perspective

There is an international consensus regarding the main principles for the safe disposal of radioactive waste. This is expressed in documents such as the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management (IAEA 1997, SÖ 1999:60) and Disposal of Radioactive Waste – Specific Safety Requirements (IAEA 2011) from IAEA. The Joint Convention states the guiding principles of a national responsibility for the management of spent fuel and radioactive waste and that radioactive waste shall be disposed of in the country in which it was generated. Appropriate steps should also be taken to avoid imposing undue burdens on future generations.

The members of the European Union have adopted directives on nuclear safety (EU 2009) and on the responsible and safe management of spent fuel and radioactive waste (EU 2011). The latter directive confirms the above principles of the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management (IAEA 1997, SÖ 1999:60).

IAEA has formulated a classification system for radioactive waste directly related to the safety of disposal (IAEA 2009a). The basic principle is that the higher the activity of the waste, the longer distance from surface ecosystem and the more engineered barriers are required. It is not possible to give exact limits that differentiate classes; there is a gradual transition between them.

National regulations and conditions lead to various solutions and designs of repositories to fulfil the international agreements. The selection of a disposal alternative on the national level depends on many factors, both technical and administrative, such as radioactive waste management policy, national legislative and regulatory requirements, waste origin, characteristics and inventory, climatic conditions and site condition, public opinion, etc. It is up to the national authorities to set the limits based on the specific national situation.

The ways this has been or will be solved in different countries with long-lived waste are of specific interest in the development of technical solutions for SFL. This section provides a brief description of some selected national solutions for the disposal of long-lived waste. The selection gives some idea of the variety of solutions in operation and on the drawing board around the world.

Repositories and repository projects for long-lived waste in other countries

The Waste Isolation Pilot Plant (WIPP) in the USA is the only repository for long-lived low and intermediate level waste in operation. There are however several ongoing projects for developing and commissioning repositories for long-lived radioactive waste around the world. Among the more notable projects are:

- The Deep Geologic Repository at Bruce, Canada.
- The Industrial Centre for Geological Disposal in Bure, France.
- The Konrad repository in Salzgitter, Germany.

The Waste Isolation Pilot Plant, USA

The Waste Isolation Pilot Plant is located near Carlsbad in New Mexico, USA.

The repository was commissioned for disposal of long-lived radioactive waste from research and defence-related manufacturing. The waste is placed in various containers, e.g. 55 gallons drums or 100 gallon drums.

The repository is located in a salt rock formation that has been stable for 250 million years. The same natural barriers and self-sealing properties that have kept the salt intact for millions of years will also safely isolate the waste from the biosphere. Underground disposal rooms have been excavated 650 meters below surface. Canisters for intermediate level waste are placed in boreholes drilled into the walls of these underground disposal rooms. Barrels and boxes with low level waste are then stacked in rows on the floor in the same rooms. Bags of magnesium oxide are placed on top of and around the disposal containers as a chemical agent to maintain a high pH environment. The magnesium oxide will control the solubility of radionuclides and is an added measure of assurance to long-term repository performance.

The Waste Isolation Pilot Plan is commissioned for about 175,000 m^3 of which 96 percent is low level waste and the remaining consists of intermediate level waste. More information on the Waste Isolation Pilot Plant can be found in U.S. DOE (2012).

The Deep Geologic Repository at Bruce, Canada

The Deep Geologic Repository (DGR) is planned to be located at the Bruce nuclear power plant site near Kincardine in Ontario, Canada.

The DGR is planned to manage the low and intermediate level waste from the nuclear power plants at Bruce, Pickering and Darlington owned by Ontario Power Generation. The intermediate level waste consists primarily of reactor components, plus ion exchange resins and filters used to purify reactor water systems.

The proposed underground repository will consist of a series of tunnels and disposal rooms accessed through shafts. Low level waste and intermediate level waste will be placed in separate disposal rooms. The repository is proposed to be located about 680 metres below surface in low-permeability limestone, beneath a 200 metre thick layer of low-permeability shale. The geological conditions and the stable environment will safely isolate the waste from the biosphere. No engineered barriers, apart from the waste containers, are needed. Institutional control is to be performed during 300 years after closure.

The repository is designed for a capacity of 200,000 m^3 of low and intermediate level waste, of which 80 percent is low level waste. More information on the DGR can be found in OPG (2011).

The Industrial Centre for Geological Disposal in Bure, France

The French National Radioactive Waste Management Agency (Andra) is currently investigating the geology in the area of Bure, France. The aim is to verify that the site is a suitable geological host for the combined repository for high-level waste and long-lived intermediate level waste. The main feature of interest is the clay stone formation that exists at depth all over the site.

The long-lived intermediate level waste consists primarily of the metallic components of fuel assemblies, which are compacted and placed in stainless steel containers. These stainless steel containers with long-lived intermediate level waste will be put into concrete containers.

The repository will consist of several tunnels with disposal cells. The concrete containers will be stacked in disposal cells in the repository. The clay stone formation which is planned to host the repository is located at an average depth of 500 metres. There will be no engineered barriers in the repository, and long-term safety will depend on the tight and stable geological environment. Only the seals used to close the repository will call for the use of cement and bentonite.

The repository is planned to hold France's existing long-lived intermediate level waste as well as the forecasted long-lived intermediate level waste arising during the next 20 years, indicating a volume of 50,000−60,000 m^3 of long-lived intermediate level waste. More information on the Industrial Centre for Geological Disposal can be found in Andra (2005).

The Konrad repository in Salzgitter, Germany

The Konrad repository site is located near the city of Salzgitter in Lower Saxony in Germany. The Konrad repository will be used for the disposal of Germany's *radioactive waste with negligible heat generation*. The term *radioactive waste with negligible heat generation* includes low level waste and a large part of the intermediate level waste. The waste includes, for instance, used plant components and components such as pumps and pipes, ion-exchange resins, contaminated tools, protective clothing, laboratory waste, sealed radiation sources, sludge, and suspensions and oils.

The repository will be situated in a former iron ore mine, and new cavities will be excavated for the disposal of the waste. The waste will be placed in various containers of steel, cast iron or concrete. The planned disposal depth is approximately 800 metres below the surface or deeper. The repository is located beneath an almost 400 metre thick layer of argillaceous rock. This means that the disposal area of the Konrad repository has no hydraulically effective connections to near-surface groundwater, and hence is isolated from the biosphere. There are no engineered barriers in the repository. After operations have ceased, remaining cavities in the repository will be filled with a special concrete mixture and subsequently sealed.

The current licence allows for a maximum of 303,000 m^3 of waste to be deposited. More information on the Konrad repository can be found in Bundesamt für Strahlenschutz (2013).

1.5 Earlier work

1.5.1 Early planning

An initial design for a repository for long-lived low and intermediate level waste is described in the SKB report PLAN 82 (SKB 1982). This was followed by a more detailed design in PLAN 93 (SKB 1993). Based on the latter design, a prestudy was carried out in 1992−1995 for the purpose of testing the ability of the engineered barriers and the near-field to prevent and retard the release of radionuclides (Wiborgh 1995). The analysis also included some chemotoxic elements present in the waste, for example lead and beryllium. The prestudy included gathering waste data to compile the first SFL reference inventory.

Based on the results of this prestudy, a simplified design for the engineered barriers in the so-called SFL 3–5 repository was proposed. The reference inventory was updated in 1998 (SKBdoc 1416968) to serve as a basis for the preliminary safety assessment together with the simplified barrier design.

1.5.2 Preliminary safety assessment of SFL 3–5

The simplified design was based on experience from the construction and operation of the rock cavern for intermediate level waste (BMA) in SFR. The long-lived waste was planned to be placed in concrete structures in a rock cavern at about 300 metres depth. Two rock caverns were planned: SFL 3 (for legacy waste from the development of the Swedish nuclear programmes and operational waste from the central storage facility for spent nuclear fuel (Clab) and the Encapsulation Plant) and SFL 5 (for the neutron-irradiated components from the nuclear power plants). The short-lived decommissioning waste from Clab and the Encapsulation Plant was planned to be emplaced in the transport tunnels (SFL 4).

The waste was planned to be packaged in containers of steel and concrete. In the repository, the containers in SFL 3 and SFL 5 were to be surrounded by concrete walls and by backfill – concrete mortar was planned to be used inside the concrete walls of the enclosure, while crushed rock was planned to be used outside. The concrete and the host rock were the most important barriers to the escape of radionuclides. The role of the concrete was to contain the radioactive substances, to prevent – together with the outer draining backfill – water flow through the waste and to sorb dissolved radionuclides. The host rock would protect the repository while limiting the flow of water and the transport of radionuclides.

The preliminary safety assessment was presented in 1999 (SKB 1999a, b). The purpose of the assessment was to investigate the capacity of the facility to act as a barrier to the release of radionuclides and to shed light on the importance of the location of the repository site. The same hypothetical sites were used as in the safety assessment for the repository for spent nuclear fuel presented as SR 97 (SKB 1999c). These sites represent fairly different conditions in terms of hydrogeology, hydrochemistry and ecosystems. The safety assessment showed the importance of the conditions and properties of the site for the long-term radiological safety.

The primary conclusions from the study were:

- The radionuclides in the waste that are of the greatest importance for assessing safety are the ones that are highly mobile and long-lived. Their long life means that the barriers and the ecosystems must be regarded on a very long timescale.
- To reduce the uncertainty in calculated environmental impact, it is important to reduce the uncertainties in the estimates of the dose-dominant radionuclides Cl-36 and Mo-93. Studies that lead to a greater understanding of their accessibility in the waste, migration in the barriers and dose impact are also of importance.
- The properties of the site are of importance for safety. Two parameters emerge as being particularly important: the water flow at repository depth and the ecosystem in the areas on the ground surface where releases may occur in the future.
- An unfavourably high water flow in the rock around the repository can be compensated for by better barriers in the near field. However, their function must be sustained for a very long time. This requires materials that are durable in the chemical and mechanical environment of the repository.

1.5.3 The authorities' review of the preliminary safety assessment

The preliminary safety assessment was reviewed by the Swedish Nuclear Power Inspectorate (SKI) and the Swedish Radiation Protection Institute (SSI) with the aid of an international team of experts (SKI and SSI 2001, Chapman et al. 2000). Both the regulatory authorities and the team of experts draw the conclusion that a great deal of research and development work remains to be done before the level of knowledge in this field is comparable with that associated with the repository for spent nuclear fuel.

The authorities shared SKB's conclusion that there is a significant site-specific impact on the long-term safety, which is mainly related to the local groundwater flow velocity but also to current geochemical conditions. SKI and SSI called for a repository design that can be considered sufficiently robust with respect to the influence of site-specific factors and their long-term evolution. The authorities also called for a coherent account which justifies the proposed design from the perspective of long-term safety and recommended that SKB make a comparison with other possible repository designs for SFL 3–5. According to the authorities, the arguments presented for the proposed design appeared to be neither fully accounted for nor sufficiently well thought-out. The authorities stressed the urgency that data and parameters of the proposed design be better documented.

1.6 Outline of the report

Chapter 2 provides an overall description of the long-lived waste that is planned for disposal in SFL. The chapter outlines the waste categories and sources of the Swedish long-lived low and intermediate level waste. Important properties of the waste are summarized in the last section, to serve as premises for the repository design.

Chapter 3 presents the methodology used in the study. The workflow is described, including identification of possible alternatives, identification of requirements and evaluation factors, and subsequent investigation and evaluation.

Chapter 4 provides an overview of the SFL system. The system includes all the components necessary for the entire handling chain: waste and containers, system for external transportation, facilities for conditioning and storage of the waste, the repository, and the repository facilities. The last section elaborates on different safety functions that a repository for radioactive waste may have.

Chapter 5 presents handling alternatives for the long-lived low and intermediate level waste. The chapter contains an inventory of handling alternatives – including conditioning methods – that may be suitable for the different waste categories in question. The impact of the handling alternatives on for example exposures of workers, long-term safety and costs are assessed.

Chapter 6 presents the requirements on the SFL system and the evaluation factors identified by the study. The identified evaluation factors are used in Chapter 8 to evaluate the proposed repository concepts.

Chapter 7 presents the repository concepts that have been identified as possible solutions for disposal of long-lived low and intermediate level waste. The compilation of concepts is the result of the creative technique chosen for idea generation. Concept descriptions are presented for the concepts that have passed an initial screening.

Chapter 8 presents an evaluation of the repository concepts. The evaluation is done in two steps – a primary assessment of the proposed concepts followed by an expanded assessment of the concepts remaining from the first step. The primary assessment sorts out the concepts whose characteristics do not fulfil or are deemed unlikely of fulfilling certain requirements. In the expanded assessment, the remaining concepts are evaluated and compared using the evaluation factors.

Chapter 9 proposes a system for management and disposal of long-lived low and intermediate level waste, and in particular a repository concept for SFL, for further assessment. The chapter also summarizes the consequences that follow from the proposal.

Chapter 10 outlines the research programme needed to realize the proposed system, including both technology development and research and safety assessment.

2 Long-lived low and intermediate level waste

Long-lived low and intermediate level waste planned for disposal in SFL comprises four main categories:

- neutron-irradiated components such as reactor internals, core components and PWR pressure vessels from maintenance and dismantling of the Swedish nuclear power plants,
- BWR control rods from operation of the Swedish nuclear power plants,
- waste from early research in the Swedish nuclear programmes (currently managed by AB SVAFO), and
- waste from other sources such as industries, hospitals and research facilities including waste from operations in Studsvik.

This chapter is based on the updated SFL reference inventory of waste from the nuclear power plants (Herschend 2013), the RD&D programme (SKB 2013a), and on figures provided by AB SVAFO and Studsvik Nuclear AB to serve as basis for the updated reference inventory. Sections 2.2–2.4 present the different waste categories, along with volumes, activities, and material content. Chemotoxic waste is listed in section 2.5 and existing packages in section 2.6. The last section, section 2.7, summarizes the important properties of the waste, which also serve as design premises for the repository.

2.1 Background

A preliminary safety assessment for SFL 3–5 was presented in 1999 (SKB 1999a, b). One of the background documents was the reference inventory report (SKBdoc 1416968) that was compiled in 1998. In this report the waste was divided into three main categories: waste from nuclear power plants to be disposed of in SFL 5, legacy waste from the development of the Swedish nuclear programmes and operational waste from the central storage facility for spent nuclear fuel (Clab) and the Encapsulation Plant to be disposed of in SFL 3, and short-lived low and intermediate level waste from the dismantling of Clab and the Encapsulation Plant, which would take place after the closure of SFR, to be disposed of in SFL 4.

When plans were made to extend the operating time of the nuclear power plants beyond 2010, it was deemed necessary to compile a new reference inventory for SFL. Initially, the main focus in updating the reference inventory was on identifying the types of waste as well as total volumes and year of production. Three important observations were made (SKB 2010a):

- Most of the waste to be disposed of in SFL has already been produced. The existing waste mainly comprises the legacy waste.
- The quantity of core components and BWR control rods in SFL will increase compared with previous plans due to the extended operating time of the nuclear power plants.
- The total expected required repository volume is smaller compared with earlier estimates, since the waste from dismantling of Clab and the Encapsulation Plant, which was previously planned to be disposed of in SFL, is now planned for disposal in SFR.

The work has continued since 2010, and the focus has shifted towards identifying the important radionuclides in the waste and calculating the reference inventory. The new radionuclide inventory is based on the current plans for the operating time of the nuclear power plants (50 or 60 years of operation) and SFR (operation until 2075).

An estimated timetable for when the waste for SFL will be produced and the corresponding estimated repository volume required is presented in Figure 2-1.

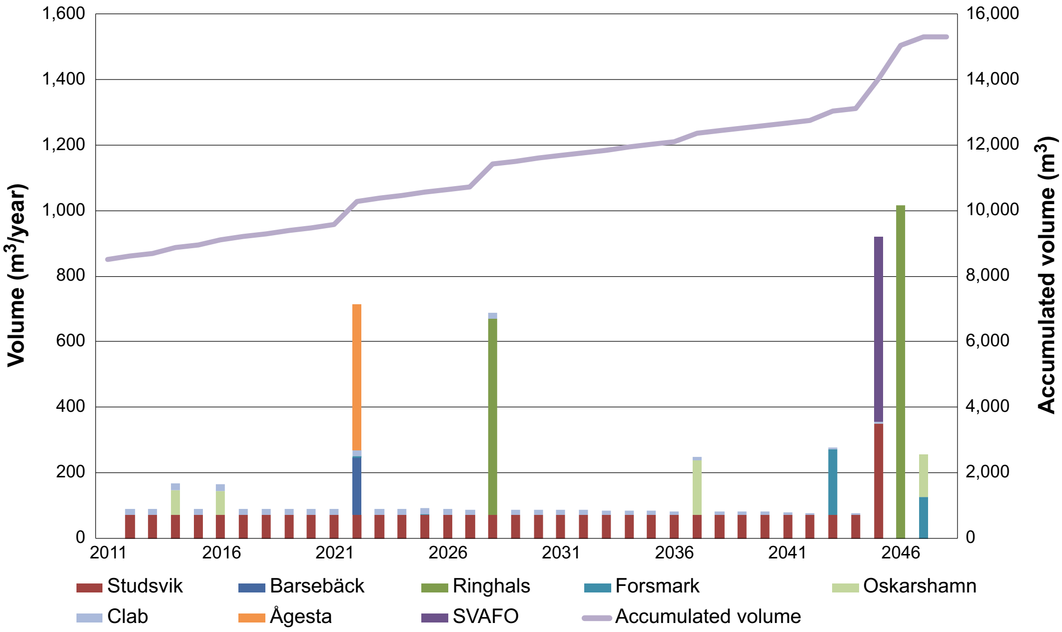


Figure 2-1. *An estimated timetable for when the waste for SFL will be produced and the corresponding estimated repository volume required (SKB 2013a).*

2.2 Waste from the Swedish nuclear power plants

The waste from the nuclear power plants comprises components with a significant content of long-lived radioactive isotopes. These components are typically located close to the core itself, where the neutron flux creates induced activity in the component material. The elevated levels of long-lived nuclides make the core components unsuitable for disposal in SFR.

The core components from the BWRs include the core support structure (moderator tank, moderator tank cover, core grid and the upper part of the control rod guide tubes) and the core spray. Also included are control rods, neutron detectors, guide tubes, boron plates and fuel boxes (including spacers etc.). No steam separators have been included in the summary, since they are planned for disposal in SFR.

The waste from the PWRs includes all reactor internals and the entire reactor pressure vessel.

The weights and disposal volumes of waste from the Swedish nuclear power plants are presented in Table 2-1. Sections of the core components containing little or no induced activity have been excluded from the weights and volumes estimate. These parts are better suited for disposal in SFR.

Table 2-2 shows the nuclide-specific activity of the waste from the Swedish nuclear power plants. The total activity of the waste from the Swedish nuclear power plants is estimated to be 2.1E+17 Bq in 2075.

2.3 Waste from SVAFO and Studsvik Nuclear

This waste category mainly consists of legacy waste from early research in the Swedish nuclear programmes (currently managed by AB SVAFO), and waste from other sources such as industry, hospitals and research facilities (currently managed by Studsvik Nuclear AB). Figures provided by AB SVAFO and Studsvik Nuclear AB to serve as basis for the updated reference inventory are presented in this section. It should be noted that there a large uncertainties regarding the content of the waste, mainly related to the scarce documentation of the legacy waste and to the forecasted operational waste from Studsvik Nuclear AB.

This section is divided into three parts: legacy waste, existing operational waste, and future operational and decommissioning waste.

Table 2-1. Waste from the Swedish nuclear power plants (Herschend 2013).

Waste category	Weight (tonnes)	Disposal volume (m^3)
Core components BWR (existing)	404	606
Core components BWR (forecast)	83	124
Core components BWR (dismantling)	595	893
Secondary waste	22	122
Control rods BWR (existing)	129	193
Control rods BWR (forecast)	344	516
Control rods PWR (Ågesta)	2	3
Reactor pressure vessel PWR	735	1,103
Reactor pressure vessel PWR (Ågesta)	140	210
Reactor internals PWR	282	423
Reactor internals PWR (Ågesta)	154	230
Sum (existing)	532	799
Sum (forecast)	449	640
Sum (dismantling)	1,614	2,418
Total	2,890	4,423

Table 2-2. Nuclide-specific activity in 2075 in core components and control rods (Herschend 2013).

Nuclide	Half-life (years)	Activity (Bq)	Nuclide	Half-life (years)	Activity (Bq)
H-3	12.3	7.92E+14	Cs-137	30	1.69E+11
Be-10	1,600,000	4.62E+03	Ba-133	10.5	5.10E+03
C-14	5,700	4.09E+14	Pm-147	2.62	2.89E+07
Cl-36	301,000	1.16E+11	Sm-151	90	8.24E+09
Ca-41	103,000	4.35E+10	Eu-152	13.5	8.84E+06
Fe-55	2.73	2.62E+14	Eu-154	8.59	1.73E+09
Co-60	5.27	5.54E+15	Eu-155	4.75	4.88E+07
Ni-59	76,000	1.94E+15	Ho-166m	1,200	2.05E+05
Ni-63	101	2.02E+17	U-232	69.8	9.50E+06
Se-79	1,100,000	8.03E+10	U-235	7.04E+08	6.64E+02
Sr-90	28.8	5.94E+11	U-236	23,700,000	1.16E+07
Zr-93	1,530,000	1.52E+11	Np-237	2,140,000	1.52E+07
Nb-93m	16.1	8.73E+14	Pu-238	87.7	7.64E+10
Nb-94	20,000	1.56E+13	Pu-239	24,100	1.54E+10
Mo-93	4,000	2.37E+13	Pu-240	6,563	1.36E+10
Tc-99	214,000	3.53E+12	Pu-241	14.3	1.47E+11
Ru-106	1.02	1.43E+01	Pu-242	374,000	1.66E+10
Pd-107	418	1.19E+05	Am-241	433	4.77E+10
Ag-108m	6,500,000	2.00E+12	Am-242m	141	3.82E+08
Cd-113m	14.1	1.24E+05	Am-243	7,365	1.02E+09
Sn-126	230,000	1.93E+07	Cm-243	30	1.41E+08
Sb-125	2.76	1.72E+10	Cm-244	18	1.21E+10
I-129	16,100,000	1.98E+05	Cm-245	8,500	9.26E+08
Cs-134	2.07	1.79E+06	Cm-246	4,730	6.06E+06
Cs-135	2,300,000	2.35E+06			

Legacy waste

Legacy waste constitutes the waste that falls under the Act on the financing of the management of certain radioactive waste etc. (SFS 1988:1597) – the so-called "Studsvik Act" – that defines the legacy waste (mainly produced pre-1991) and regulates the financial framework for the management and disposal of the legacy waste. Most of the long-lived low and intermediate level waste stored today at the Studsvik site was produced during the development of the Swedish nuclear programmes in the 1960s and early 1970s. Large parts of the waste are stored in drums filled with a mixture of waste and grout. By the time of conditioning, the waste was considered to have limited radiotoxicity and documentation of the waste is thus scarce.

Recently, AB SVAFO undertook an extensive examination programme during which 7,303 drums were examined using X-ray and measured for gamma radiation (Ekenborg 2012). During this work it was found that some drums also contained waste in the form of liquids as well as mercury. The examination programme also confirmed previous information regarding the presence of small amounts of plutonium and uranium. As part of this work the original drums were placed in new 280-litre steel drums as overpack. This was occasioned by the corrosion of old drums and the risk that the integrity of the original drums was compromised.

Figures provided by AB SVAFO and Studsvik Nuclear AB to serve as basis for the updated reference inventory are presented in Table 2-3, showing legacy waste types, numbers and packages. This adds up to a total of about 7,300 m^3 of legacy waste stored at the Studsvik site today.

Figures provided by AB SVAFO and Studsvik Nuclear AB estimate the total activity in the legacy waste to 1.0E+15 Bq is 2075. The radionuclide inventory contains for example Sr-90, Ra-226, Th-232, U-235, U-238 and long-lived transuranic nuclides such as Pu-239, Pu-241 and Am-241.

Existing operational waste

The waste from Swedish industry, hospitals, universities and research facilities is handled by Studsvik Nuclear AB, which prepares the waste for disposal and store it until a repository is available. AB SVAFO and Studsvik Nuclear AB has currently about 350 m^3 of operational waste planned for disposal in SFL in storage (see Table 2-4). The total activity content of the existing operational waste is estimated to 1.3E+12 Bq.

Table 2-3. Legacy waste stored at the Studsvik site.

Waste	Number	Package
Trash and scrap	679	5 × 80L drums in mould
Cement-solidified sludge and ion exchange resins	137	200L drum
Trash and scrap	7,418	100L drum in 200L drum in 280L drum
Ashes and dust	800	100L drum in 200L drum in 280L drum
Ashes and dust	167	100L drum in 200L drum
Trash and scrap	342	Berglöf boxes
Glove boxes	29	
Trash and scrap	48	200L drums in 400L drums
Processing waste and Thorium	46	200L drums
Uranium waste		2 half height containers plus 10 Berglöf boxes
Various containers	30	

Table 2-4. Existing operational waste from AB SVAFO and Studsvik Nuclear AB (31 December 2011).

Waste	Number	Package
Cement-solidified sludge	669	200L drum
Trash and scrap	244	100L drum in 200L drum
Ashes and dust	270	100L drum in 200L drum

Future operational waste and decommissioning waste

Future operational waste (until 2045) and waste from the dismantling of the facilities in Studsvik will render approximately 3,300 m^3 of waste planned for SFL. There is currently no information on the forecasted activity of this waste fraction.

Recently SKB has been involved in the planning for the European Spallation Source (ESS), which is planned to be located on the outskirts of Lund in southern Sweden. The ESS is a research facility where a powerful neutron source is used to probe specimens. The ESS operations will generate radioactive waste during operation and dismantling. The ESS is planned to open in 2019, and a preliminary waste management plan for the operation and dismantling of the facility has been prepared by ESS (Ene 2012). Parts of the ESS waste may be considered for disposal in SFL, but the ESS waste has not been considered in this study.

2.4 Material inventory

The material inventory of the neutron-irradiated components, including BWR control rods, from the Swedish nuclear power plants is presented in Table 2-5. The core components consist mainly of steel and stainless steel, with smaller amounts of other alloys, e.g. Zircalloy from fuel boxes. Besides steel, the control rods also contain boron carbide and hafnium (control rods from the Ågesta reactor contain silver, indium and cadmium). PRM detectors contain small amounts of fissile material (which is not included in Table 2-5).

Great uncertainties exist regarding the material composition of the waste from AB SVAFO and Studsvik Nuclear AB, especially the legacy waste. Table 2-6 shows the estimated material inventory of waste from AB SVAFO and Studsvik Nuclear AB. The reported material amounts correspond to the material compositions from the previous reference inventory (SKBdoc 1416968), extrapolated to the currently reported waste amounts. The list cannot be considered complete, given the lack of knowledge concerning the legacy waste and the fact that the data for future operational waste and decommissioning waste are lacking.

2.5 Chemotoxic waste inventory

The waste to be disposed of in SFL also contains substances which are chemotoxic rather than radioactive. The chemotoxic substances are mainly found in the legacy waste described in section 2.3, for example cadmium, lead, mercury, and beryllium (see Table 2-6). Some radionuclides, for example uranium, plutonium and thorium, could also be classified as chemotoxic.

Table 2-5. The material inventory of the neutron-irradiated components, including BWR control rods, from the Swedish nuclear power plants (Herschend 2013).

Material	Weight (kg)
Stainless steel	2,071,206
Steel	720,000
Boron carbide	55,646
Hafnium	15,539
Inconel	292
Zircalloy	3,946
Silver	458
Indium	86
Cadmium	228
Other (filters, segmentation equipment etc.)	22,197
Total	2,889,599

Table 2-6. The material inventory of waste from AB SVAFO and Studsvik Nuclear AB.

Material	Weight (kg)	Present matrix and packages (kg)
Stainless steel	146,302	
Zircalloy	8,379	
Steel/Iron	159,155	469,665
Zinc	301	
Aluminium	202,286	
Cadmium	1,181	
Copper	3,006	
Lead	10,406	27,160
Brass	2,027	
Graphite	301	
Textiles	18,064	
Paper	6,346	
Wood	2,803	
Plastics	47,471	
Bakelite	446	
Plexiglass	52,593	
Rubber	3,006	
Glass	6,352	
Concrete	4,412	4,192,850
Vermiculite	118	
Other organics	899	
Cement	322,400	
Ferrocyanid sediment	4,836	
Ashes	24,740	
Thorium	2,670	
Uranium	15,025	
Plutonium	1	
Mercury	68	
Beryllium	300	
Total	1,045,894	4,689,676

2.6 Existing packages

The existing waste packages are an important design premise for SFL, since much of the waste planned for SFL is already placed in different types of packages today. The packages are either filled with waste only, or a mixture of waste and grout. Waste stored in packages without grout can easily be retrieved if required, while retrieval of waste that has been mixed with grout presents a more challenging task.

A brief summary of containers currently used to store long-lived low and intermediate level waste is presented in this section.

Concrete mould

The concrete mould has external dimensions $1.2 \times 1.2 \times 1.2\ m^3$ and an internal volume of $1\ m^3$. This mould is a standard container for ion exchange resins in SFR but is also used for storage of used filters destined for SFL.

Steel mould

This mould has the same external dimensions as the concrete mould above.

Drum (200 litres)

Most of the legacy waste is stored in drums with approximate external dimensions 0.6 meter × 0.88 meter (*diameter* × *height*). Today there are about 7,500 drums in storage with waste that has been stabilized with grout.

Drum (80 litres)

The external dimensions of these drums are 0.38 meter × 0.88 meter (*diameter* × *height*). They are used for storage of highly radioactive material and often contain also a small lead container for radiation shielding. These drums are often stored in moulds for five drums for extra radiation protection. The waste can be retrieved from the drums as long as grout has not been used to stabilize the waste.

Steel tank

The steel tank has external dimensions 3.3 × 1.3 × 2.3 m^3 (*length* × *width* × *height*) and is mainly used for storage of core components and to some extent also for reactor internals destined for SFR. The waste is placed in a cassette which is placed in the tank. A lid is bolted to the tank. The waste in these tanks has not been stabilized with grout and can thus be retrieved and reconditioned.

Mould for five 80-litre drums

This mould has external dimensions 1.2 × 1.2 × 1.2 m^3 and contains five 80-litre drums, see above. The drums can be retrieved from the moulds if required.

Berglöf box

A number of open containers with different external dimensions, known as Berglöf boxes, are used for waste storage. The waste in these boxes has not been stabilized with grout and can thus be retrieved and reconditioned if required.

Unique containers

There are a number of unique containers with waste intended for SFL. These containers will need to be handled on an individual basis at the time of disposal.

2.7 Summary of waste properties

The long-lived low and intermediate level waste planned for disposal in SFL is diversified, and the purpose of this section is to summarize its key characteristics. The characteristics of the waste are important premises for the design of SFL.

Firstly, it is recognized that the total disposal volume of the long-lived low and intermediate level waste planned for disposal in SFL is only about 16,000 m^3. This should be compared to a planned total disposal volume in SFR (after extension) of 173,000 m^3 plus space for nine BWR tanks (SKB 2013a). However, the total activity in SFL is estimated to 2.1E+17 Bq in 2075, which is more than 200 times the activity in SFR. The content of long-lived nuclides is also considerably higher in the waste planned for SFL, compared to the waste in SFR.

One major fraction of the waste is the neutron-irradiated components – core components, reactor pressure vessels from PWRs, etc. – and the BWR control rods from the nuclear power plants. This part of the inventory adds up to approximately 3,000 tonnes of steel and stainless steel, corresponding to approximately 5,000 m^3 of disposal volume, and contains more than 99 percent of the estimated activity of the full SFL inventory.

One of the conclusions from the preliminary safety assessment in 1999 (SKB 1999a, b) is that the radionuclides in the waste that are of the greatest importance for assessing safety are the ones that are highly mobile and long-lived. Two such nuclides identified are Cl-36 and Mo-93, which have half-lives of 301,000 years and 4,000 years, respectively. These activation products are mainly induced activity bound in the metallic waste from the nuclear power plants, and will become available for release as the metallic components corrode.

The remaining part of the waste is the legacy waste from early research in the Swedish nuclear programmes and long-lived waste from hospitals, industry and research. This part of the inventory adds up to approximately 11,000 m^3 of disposal volume and contains less than 1 percent of the activity of the full SFL inventory. This part of the inventory also contains organic substances as well as chemotoxic waste comprising cadmium, lead, beryllium, and mercury.

The legacy waste from early research in the Swedish nuclear programmes is to a large extent already conditioned. The waste has been stabilized with grout in drums. Besides uranium, the waste also contains long-lived transuranic nuclides such as Pu-239, Pu-241 and Am-241. These nuclides are highly radiotoxic, but the preliminary safety assessment in 1999 (SKB 1999a, b) showed that the rock itself has a strong retention capacity for these nuclides. Disposal depth is an important factor for controlling the impact of these nuclides on man and the environment. It should be noted that there is limited information regarding the presence of several important radionuclides identified in SKB (1999a, b) – such as Cl-36 and Mo-93 – in the legacy waste.

Intermediate level waste requires, according to the IAEA classification of radioactive waste (IAEA 2009a), a greater degree of containment and isolation than that provided by near surface disposal. The content of long-lived nuclides is of particular importance in the assessment of suitable disposal options. Disposal in a facility at a depth of between a few tens and a few hundreds of metres is indicated for intermediate level waste (IAEA 2009a). Compared to near surface disposal, disposal at such depths has the potential to provide a long period of isolation from the environment, by offering a stable geology and also by limiting the likelihood of inadvertent human intrusion.

Due to the radioactive decay of the waste, the hazard of the radioactive inventory will decrease with time. The long life of key nuclides in the waste in question means that the barriers and the ecosystems must be regarded on a very long timescale. In the case of a repository for long-lived waste, the safety assessment should at least cover approximately one hundred thousand years or the period for a glacial cycle. For the time following one hundred thousand years, the safety assessment needs to show the safety of the repository in a more qualitative way, for a maximum time period of up to one million years (SSM 2008).

The timescale considered for a repository for long-lived low and intermediate level waste is thus of the same order of magnitude as the timescale for a repository for spent nuclear fuel. This directly influences certain design parameters for the repository, e.g. the ability to maintain sufficient barrier functions through glaciations.

3 Methodology

A system (not yet complete) exists in Sweden for management and disposal of radioactive waste. Still lacking are, for example, the Encapsulation Plant and the Spent Fuel Repository, as well as a system for management and disposal of the long-lived low and intermediate level waste – the SFL system. The SFL system comprises all the components necessary for the handling, transportation and disposal of the long-lived waste and will, when completed, be an integral part of the Swedish system for management and disposal of radioactive waste, see Chapter 4.

A main component of the SFL system is the repository itself, and a central part of this study is to identify and propose a design of a repository – a concept.

The methodology used in the study of repository concepts should make the identification and evaluation of concepts transparent and traceable. The principle for the methodology is that all initially identified concepts are evaluated against the identified relevant requirements. A concept that can be shown not to fulfil requirements or to have serious drawbacks compared with other concepts in terms of the requirements will be excluded from further evaluation.

3.1 General workflow

The general workflow for the identification and evaluation process is illustrated in Figure 3-1. It can be divided into three phases: initial identification, investigation and development, and evaluation.

The initial work is carried out in two parallel activities: identification of conceptual solutions and identification of requirements. In the investigation and development phase, the identified concepts are investigated, developed and further defined. The workflow ends with an evaluation of the concepts, with the aim of recommending no more than two systems for further assessment.

3.2 Identification of conceptual solutions for the components of the SFL system

The identification process starts by defining the different components necessary in the SFL system. Based mainly on the current Swedish system, components for management of SFL waste are identified. The components are presented in section 3.2.1.

Figure 3-1. *Illustration of the general workflow, from identification to evaluation.*

Solutions for the different components are identified in different ways, depending on the character of the component:

- The identification process for the **repository concepts** is guided by a stepwise process that has previously been used in for example SKB (2000a). The process offers a structured way to identify management strategies and concepts that implement those strategies. This concept identification process is described in section 3.2.2.
- **Handling alternatives**, including conditioning methods, that may be considered suitable for the long-lived low and intermediate level waste are indentified in the study as outlined in section 3.2.3.
- Identification of the **remaining components of the SFL system**, following from the identification of concepts and handling alternatives, is described in section 3.2.4.

3.2.1 Definition of the system and the system components

The SFL system comprises all the components and facilities necessary for the management and disposal of long-lived low and intermediate level waste:

- Waste and containers.
- System for external transportation.
- Facilities for conditioning of waste.
- Storage facilities.
- Repository (including its barrier system for long-term safety) and repository facilities (prior to closure).

The interface between the waste producers and SKB is "at the gates" of the waste producers. SKB is responsible for the system for external transportation and the producers are responsible for loading their waste onto the system for external transportation. Legal responsibility of the waste remains, however, with the waste producer until closure of the repository in accordance with Swedish law (SFS 1984:3).

The SFL system is described in Chapter 4, where the roles of the different components are described in detail.

3.2.2 Repository concepts

The initial identification of repository concepts can be described as a three-step process. In the first step the *strategy* for disposal is defined. This is followed by defining a *concept* used to realize a certain strategy, and finally the details of a specific *variant* of a certain concept can be specified. These steps are briefly described below, and the hierarchy of the workflow is illustrated in Figure 3-2.

The result of the identification process is reported in Chapter 7. The chapter presents the concept descriptions that serve as a basis for further evaluation. The concept descriptions contain vital information on the engineered barriers.

Step 1 – Strategy

The first step is to identify the *strategy* for management and disposal of the waste. Examples of different strategies are: launching the waste into space, sub-seabed disposal, or geological disposal, which is the strategy used for the repository for short-lived radioactive waste and the planned repository for spent nuclear fuel in Sweden.

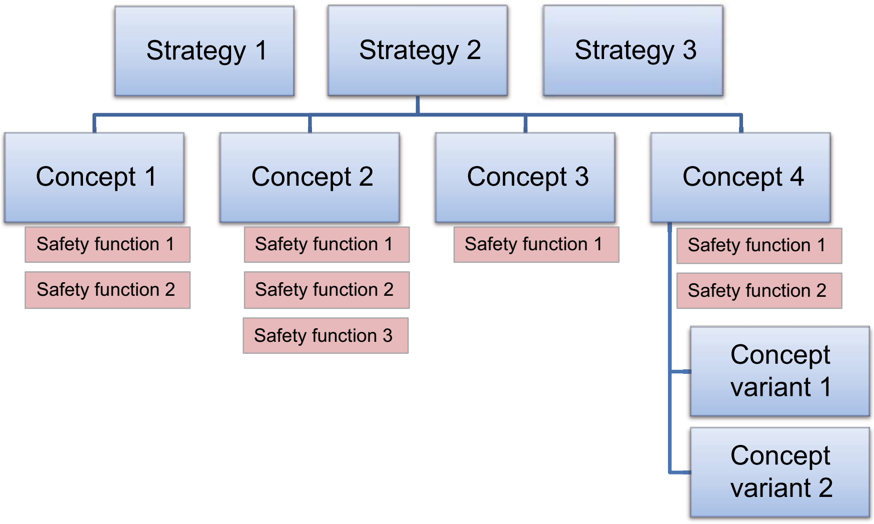


***Figure 3-2.** Schematic illustration of the different levels used in the work of identifying a repository concept. The study identifies strategies and different concepts within the strategies. The concepts may rely on one or several different safety functions. It is beyond the scope of the study to identify concept variants.*

Step 2 – Concept

The second step is to identify one or several *concepts* within each strategy. A concept is a technical solution to fulfil the chosen strategy. The concept relies on one or several *safety functions*. The safety functions provide information on what principles the concept is based on.

The repository for the short-lived radioactive waste (SFR) and the repository for the spent nuclear fuel are two examples of repositories based on different concepts within the strategy of geological disposal. SFR is based on the safety functions of limited inventory and retardation, while the repository for spent nuclear fuel is based on containment and retardation.

Most of the information about the choice of engineered barriers is collected in this step.

Step 3 – Concept variant

The last step is to describe the details for a concept, the repository *concept variant*. At this step, the specific details of the concept are specified.

As an example, KBS-3H (horizontal disposal of canisters) and KBS-3V (vertical disposal of canisters) are two variants within the concept of KBS-3. It is beyond the scope of this study to make distinctions between different variants of the identified concepts.

3.2.3 Handling alternatives

Beside the repository concepts, a number of handling alternatives for the waste, including conditioning methods, are identified in the study. The identification is based on experience from management of radioactive waste in general, rather than a guided process. The handling alternatives may include pre-disposal processing of the waste in order to achieve certain beneficial properties prior to emplacing the waste in the repository. The pre-disposal methods are not to be considered as stand-alone concepts but rather as possible add-ons to repository concepts. For example, melting as a processing method can be applied to metallic waste to reduce the disposal volume. The processed metallic waste will, however, still need a repository concept for disposal. The choice of pre-disposal processing method will have a direct effect on the system components *waste and containers* and *facilities for conditioning of waste*.

The identified handling alternatives are described in greater detail in Chapter 5. The handling alternatives range from a minimum of handling to advanced treatment of the waste. The description includes identified prerequisites and consequences of the method, for example the need for extended time for storage. The effect on the system of a specific handling alternative can subsequently be assessed in the final evaluation.

3.2.4 Remaining components of the SFL system

The identification of solutions for the remaining components follows from the choice of concept and handling alternatives as well as other planning premises, such as starting date of operations.

The remaining components are essential for the completeness of the system. They are, however, not the focus of the study, since the development of SFL still is in its early stages and the focus is on identifying feasible repository concepts. At this conceptual stage of the development process, the remaining components do not in many cases differ between repository concepts. The system for external transportation is, for instance, site-dependent rather than concept-dependent. If the solutions for the remaining components have an impact on the qualities of the SFL system, it is important to address this in the evaluation process.

3.3 Identification of requirements and evaluation factors

In order to exclude, evaluate and select a system as a future SFL candidate, it is necessary to know what requirements apply to the SFL repository system.

The objective of the work on requirements and evaluation factors is to identify what requirements the repository system has to fulfil, and from these requirements derive evaluation factors to be used in the evaluation of the concepts. The outcome of this work is reported in Chapter 6.

3.3.1 Workflow

The workflow from the identification of external requirements to the definition of evaluation factors is illustrated in Figure 3-3.

3.3.2 Identification of external requirements

External requirements applicable to the SFL system originate from different stakeholders, such as authorities and owners.

The initial step in the identification of applicable requirements is an inventory of regulatory requirements that apply to the SFL system. The identification of applicable laws and regulations is based on their relevance and possible influence on the repository concepts and the evaluation of the concepts. This means that the identification of applicable laws and regulations should not be seen as a complete set of requirements on the repository system. The list of applicable laws and regulations is presented in section 6.1.

Figure 3-3. *Illustration of the workflow from identification of external requirements to definition of evaluation factors.*

3.3.3 Derivation of requirements

Based on the external requirements identified from laws and regulations and requirements from the owners' point of view, a set of requirements applicable to the SFL repository system and its parts is derived. The requirements are presented in section 6.2 and in Appendix 1.

The derived requirements are SKB's interpretation of the functions and the properties a system for management and disposal of long-lived low and intermediate level waste must have in order to fulfil the requirements set up by the authorities as well as by the owners. The requirements are primarily formulated as *qualitative* requirements, but there are also *absolute* requirements. The qualitative requirements can be considered as ambitions and goals, e.g. to limit risks or using resources in an efficient way, whereas the *absolute* requirements can be answered by a simple "yes" or "no".

3.3.4 Definition of evaluation factors

The qualitative requirements can in some cases be quantified, e.g. by threshold values in different regulations. Given the limited scope of the concept study it is, however, not possible to verify the fulfilment of all the absolute or quantified requirements.

Instead of verifying all requirements, qualified judgements can be made of how easy or likely it is for a concept to fulfil the requirements. These judgements can be used in the evaluation of the concepts. As an example, one of the main requirements given in the regulations regarding the repository is the regulatory risk criterion, which implies that the maximum annual risk that any individual will be harmed shall be less than 10^{-6} after closure. It is beyond the scope of the concept study to verify the fulfilment of this threshold value; however the likelihood of any concept fulfilling this requirement can be qualitatively assessed. This is also reflected in the formulation of the requirements where threshold values are not included.

Based on the requirements, evaluation factors are formulated. To ensure a traceable evaluation process, each requirement is connected to at least one evaluation factor. As a result, requirements that are verifiable can be used in order to exclude concepts that do not fulfil the requirements. The remaining requirements are used for the evaluation of the remaining concepts, via the evaluation factors. The evaluation factors are described in section 6.3.

3.3.5 Best available technology – BAT

Originating from the Environmental Code and also included in the Nuclear Activities Act, the BAT principle plays a central role in the selection of a concept for the future repository. The BAT principle calls for the use of the best available technology in order to limit any negative impact on the environment and on human health, now and in the future.

The evaluation process described in this concept study includes the same aspects as the BAT principle and is a part of the work to meet the BAT requirement on a system level.

3.4 Assessment, investigation and development of conceptual solutions

The initial concepts are not described in detail. The concept descriptions focus on the principle of the safety functions of the barriers. In order to make a fair evaluation based on the criteria in the evaluation factors, further development and investigations are necessary.

Development of the concepts is an iterative process that is schematically described in Figure 3-4.

Based on the initially identified concepts, the requirements and evaluation factors, a preliminary assessment and identification of uncertainties is done for each concept. Already at this stage, some concepts are found not to fulfil the requirements and are discarded, see section 8.1. The work continues with investigations of the remaining concepts to find out more about each concept in terms of fulfilment of requirements and performance related to the evaluation factors.

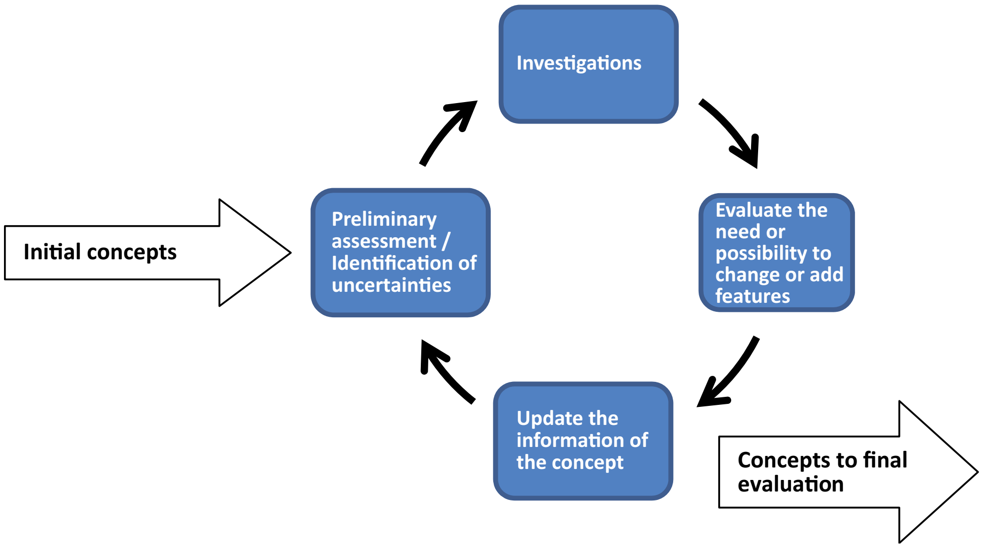


Figure 3-4. *Illustration of the work process: preliminary assessment – investigation – development.*

During the investigations, the initial concepts are in some cases modified by adding features or combining parts of concepts in order to fulfil the requirements or to strengthen the concepts. In particular, the option of including conditioning of the waste in order to obtain certain properties beneficial for the design of the repository is considered.

3.5 Evaluation method

The main purpose of this study is to propose a system for management and disposal of the long-lived low and intermediate level waste to be further assessed. The evaluation method used is intended to achieve transparency and traceability in the evaluation process.

A prerequisite in the evaluation of the concepts is fulfilment of all absolute requirements as described in section 3.3.3. If a concept fails to fulfil a requirement, it is excluded from further development and the expanded assessment of the evaluation. The exclusion is justified by references to the requirements or evaluation factors.

The second step of the evaluation – the expanded assessment of the remaining concepts – is done using the evaluation factors to compare the different concepts. For every evaluation factor, each concept is evaluated and compared to the other concepts. Due to the nature of the concept study, the focus is on the differences between concepts, rather than absolute numbers. The expanded assessment is described in section 8.3.

4 Overview of the SFL system

The SFL system is a system for disposal of long-lived low and intermediate level waste. The system consists of all the components necessary to facilitate the handling, transportation, storage and disposal of the long-lived waste and will, when completed, be an integral part of the Swedish system for management and disposal of radioactive waste.

The Swedish system for management and disposal of radioactive waste today comprises several facilities as well as transport and handling systems which together form a safe and efficient system for the management of radioactive waste, see Figure 4-1.

The main components today are the repository for short-lived radioactive waste, SFR, the central storage facility for spent nuclear fuel, Clab, and the transportation system in which the ship m/s Sigrid plays an important role. New components will be added to the system in the future, such as the Encapsulation Plant and the Spent Fuel Repository, and finally SFL.

Because SFL will be a part of an already existing Swedish system, some components of the future SFL will inherit properties from existing components, for example the transportation system. Other components will be designed for the SFL system alone, such as the repository itself. All components that may be part of the SFL system are described in this chapter:

- Waste and containers
- System for external transportation
- Facilities for conditioning of waste
- Storage facilities
- Repository (including its barrier system for long-term safety) and repository facilities (prior to closure)

The details of those components are elaborated on in greater detail in the following sections.

4.1 Waste and containers

4.1.1 Long-lived low and intermediate level waste

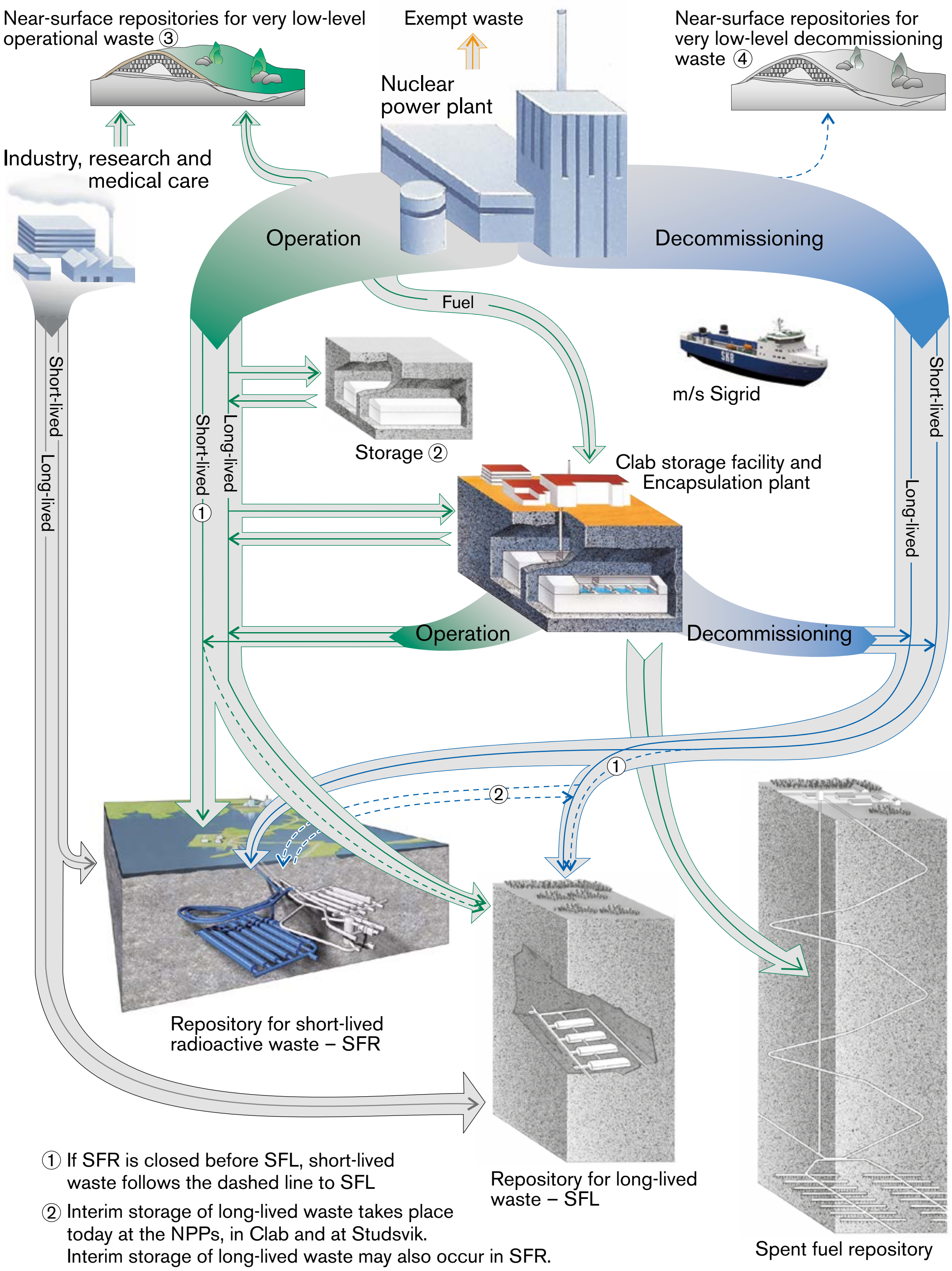
Long-lived low and intermediate level waste planned for disposal in SFL comprises three main categories: neutron-irradiated components such as BWR control rods, core components and PWR pressure vessels from maintenance and dismantling of the nuclear power plants, and waste from early research in the Swedish nuclear programmes and other sources such as industry, hospitals and research. Future changes in plans, if for instance the closing date of SFR is brought forward, may result in additional waste fractions (see Figure 4-1, note 1).

The details of the waste planned for disposal in SFL are outlined in Chapter 2.

4.1.2 Conditioning of waste

Treatment and conditioning processes are used to convert radioactive waste materials into a form that is suitable for its subsequent management, such as transportation, storage and disposal. For the course of this work the term *conditioning* is used to denote all processing, pre-treatment, treatment, and conditioning of waste.

The properties of the waste and the dimensions of the containers in which it is currently stored differ markedly between the waste categories. Some sort of conditioning will be required for almost all waste categories to prepare it for subsequent handling, transport, storage and disposal. The handling alternatives, including conditioning methods, for long-lived low and intermediate level waste are presented in Chapter 5.



① If SFR is closed before SFL, short-lived waste follows the dashed line to SFL

② Interim storage of long-lived waste takes place today at the NPPs, in Clab and at Studsvik. Interim storage of long-lived waste may also occur in SFR.

③ There are near-surface repositories at the NPPs in Forsmark, Oskarshamn and Ringhals. There is also a similar near-surface repository at Studsvik, to which some waste from industry, research and medical care is sent.

④ Possible alternative for very low-level decommissioning waste. A decision has not yet been made on this.

Figure 4-1. *The Swedish system for management and disposal of radioactive waste.*

Some conditioning of the waste may take place at the waste producer sites, for example to ensure safe transportation. Other methods may require a dedicated conditioning facility, which could be managed as a central facility. The role of such facilities in the SFL system is described in section 4.3.

4.1.3 Waste containers

Waste containers are the physical units that permit handling, transport, storage and disposal of the SFL waste. There are already a number of different containers in use for storage of the SFL waste, see Chapter 2 for details.

The waste containers can be classified into three different categories, depending on their contribution to radiological safety and long-term safety:

- Containers providing for safe management and disposal. They provide only limited radiological shielding and are not credited with a safety function in the assessment of the long-term safety of the repository.
- Containers providing adequate radiological shielding during handling and during the operating phase of the repository. They are also not credited with a safety function in the assessment of the long-term safety of the repository.
- Containers providing adequate radiological shielding during handling and during the operating phase of the repository, and which also have properties that enable them to be credited with a safety function in an assessment of the long-term safety of the repository.

Containers in the first category may be sufficient for low level waste, while the second category is necessary to ensure radiological safety during handling of core components classified as intermediate level waste.

4.2 System for external transportation

The system for external transportation provides transport of SFL waste from the waste producer to the facilities in the SFL system, as well as between the facilities in the SFL system. The system for external transportation is a vital part of the future SFL system to connect the nuclear power plant sites and Studsvik with the location(s) for storage, conditioning and disposal.

SKB's system for transport and handling currently comprises:

- A set of waste containers developed for the specific requirements of the individual waste types.
- A set of transport containers developed for the specific requirements of the different waste containers. The transport containers provide for radiation protection during transportation.
- Transport vehicles for surface transport.
- The ship m/s Sigrid for sea transport.
- Handling equipment at the different sites.

Specifically for SFL, steel tanks for storage, and possibly disposal, of core components and reactor internals are already in use. A transport container – ATB 1T – is under development for external transportation of the steel tanks.

SKB's existing system will clearly form the basis of SFL's system for external transportation. Certain components such as new waste containers and new transport containers may need to be added to transport waste destined for SFL. Another consideration is the fact that no site has yet been selected for SFL. In the event that SFL is located at an inland site, vehicles for long-distance overland transport, such as large trucks, or even vehicles for railway transport will be required.

4.3 Facilities for conditioning of waste

Today, long-lived low and intermediate level waste is stored in several different types of containers containing either only waste or a mixture of waste and grout, see sections 2.2 and 2.3.

Based on future decisions on acceptance criteria for different waste categories, conditioning methods for the waste and suitable containers for disposal of the waste will be selected. Following the selection of conditioning methods, requirements will be formulated for conditioning facilities within the SFL system. Some of the prerequisites associated with methods and facilities for conditioning are outlined in Chapter 5.

The sites of the waste conditioning facilities will be determined once suitable conditioning methods, disposal containers and transport containers have been identified. Conditioning may take place at the waste producers' sites or in centralized facilities.

4.4 Storage facilities

Since SFL is not planned to be put into operation until around 2045, facilities for storage of the waste prior to disposal will be required. The requirements on the storage facilities are determined by the activity levels, so that facilities for storage of both low level waste and intermediate level waste will be needed in the SFL system. As of today, the only facility for storage of long-lived low and intermediate level waste which is managed by SKB is Clab. However, several facilities for storage of long-lived low and intermediate level waste managed by other organisations are in use in Sweden today, see Table 4-1.

According to present plans, storage of steel tanks with long-lived low and intermediate level waste will be possible in one of the new rock caverns in the planned extension of the SFR repository. It is estimated that approximately 280 steel tanks will be able to be stored in SFR, representing a total storage volume of about 2,800 m^3.

Segmented core components and control rods currently stored in Clab may in the future be moved to other storage facilities, to make room for additional spent nuclear fuel in Clab. The need for additional storage facilities for core components and control rods will then be investigated.

The legacy waste managed by AB SVAFO as well as the waste from industry, hospitals and research managed by Studsvik Nuclear AB is in storage at the Studsvik site.

Table 4-1. Existing storage facilities that may be used for storage of long-lived low and intermediate level waste, and the total storage capacity of the facilities.

Storage	Capacity (m^3)
BFA rock cavern at Oskarshamn NPP	13,500
Storage at Forsmark NPP	550
Storage at Ringhals NPP	9,200
Storage on the Studsvik site	2,000
Rock cavern on the Studsvik site	5,000

4.5 Repository and repository facilities

The repository shall contain, prevent or retard the dispersion of radioactive substances so that the ionizing radiation does not cause harm to human health or the environment.

4.5.1 Repository

The repository is the area used for disposal of waste, including the engineered barriers. When all waste has been disposed of, technical equipment will be dismantled. A geological repository will then be backfilled and plugged, and the entrances will be sealed in a way that prevents intrusion and ensures that post-closure surveillance will not be needed.

4.5.2 Repository facilities

The repository facilities comprise the buildings, technical systems etc that are located in the vicinity of the repository to accommodate all the functions and equipment necessary for safe disposal of waste in the repository.

For geological repositories, the repository facilities usually consist of structures both above and below ground. The above-ground facilities include buildings housing technical support systems such as electricity and ventilation, security and system controls. They also include physical protection of the site, areas and systems for loading and unloading of waste transport containers, and the entrance to the underground area. The underground facilities include technical installations such as ventilation and climate control and systems for emplacing the waste in the repository.

4.5.3 Safety functions of the repository

The repository and its barrier system can be designed to incorporate different safety functions or combinations of safety functions. The identification of different safety functions follows from the identification of strategies and concepts as described in Chapter 3. Which safety function is most suitable depends on the waste category and the surrounding environment, such as the properties of the rock. The five major safety functions discussed in this work are:

- *Containment.* Separation of the radionuclides from their immediate environment, especially from water. The waste is placed in a tight container or canister that prevents dissolution and transport of radionuclides during the lifetime of the repository. The KBS-3 concept for spent nuclear fuel is an example of a concept based on this safety function.
- *Dilution.* The concentration of radionuclides is lowered by adding a solvent, such as water. Dilution can for instance be achieved by locating the repository at a place where the discharge area is a large lake, a river or an ocean.
- *Isolation.* Radionuclides are prevented from harming people and the environment by removing the waste permanently from the biosphere.
- *Limited inventory.* A limited radionuclide inventory intrinsically limits the potential hazards to people and the environment. This is achieved by disposal of only a limited amount of radionuclides in the repository.
- *Retardation*. Dissolution and transport of radionuclides are delayed by a system of engineered and natural barriers. The barriers can limit the flow of water but also provide sorption sites for the radionuclides, thus limiting the release rate.

5 Handling alternatives for waste

Treatment and conditioning processes are used to convert radioactive waste materials into a form that is suitable for its subsequent management, such as transportation, storage and disposal. Treatment and conditioning of waste are aimed at enhancing safety or reducing costs or both by changing the characteristics of the waste – for example by volume reduction, removal of radionuclides from the waste, or changing its composition – or by producing a waste package suitable for handling, transport, storage and/or disposal. Since all handling of the waste entails a risk for increased dose load, the handling chain should be planned as a whole. The benefits in terms of safety and handling must be weighed against the dose load to personnel during treatment and conditioning. In this study, the term *conditioning* is used to denote all processing, pre-treatment, treatment, and conditioning of waste.

The different waste types planned to be deposited in SFL have different properties. Neutron-irradiated components from the nuclear power plants consist primarily of steel and stainless steel. The control rods consist of stainless steel and neutron-absorbing material such as boron carbide. Both contain long-lived activation products created by neutron-irradiation. The legacy waste is largely already conditioned by embedding in cement and contains other long-lived radionuclides, such as uranium and fission products.

This chapter contains an inventory of handling alternatives – including conditioning methods – that may be suitable for the long-lived low and intermediate level waste in question. The impact of the handling alternatives on for example exposures of workers, long-term safety and costs are assessed.

The chapter is structured based on the waste categories:

- Metallic waste from the nuclear power plants, section 5.1.
- BWR control rods, section 5.2.
- Legacy waste, section 5.3.
- Waste from other sources such as industries, hospitals and research facilities, section 5.4.
- General methods, section 5.5.

The handling alternatives and conditioning methods are described, and the benefits and consequences of the methods are elaborated on.

5.1 Metallic waste from nuclear power plants

The metallic waste from the nuclear power plants – core components, reactor internals, PWR pressure vessels etc., but not BWR control rods – adds up to approximately 2,500 tonnes of steel and stainless steel. The activity of the neutron-irradiated components influences handling alternatives. The most demanding radionuclide from a radiation safety viewpoint during the first 70 years is Co-60. All handling requires radiation shielding, and any segmentation of these components is therefore preferably done in the pools at the nuclear power plants.

In the case of metallic waste from the nuclear power plants, five main alternatives can be formulated for handling and conditioning:

- Disposal of whole components.
- Segmentation and loading into steel tanks (BFA-tanks).
- Segmentation and loading into new containers.
- Segmentation and loading into long-term durable containers.
- Melting.

This section describes these five different handling alternatives assessed for the metallic waste from the nuclear power plants.

5.1.1 Disposal of whole components

In keeping with the ALARA principle, components may be handled in one piece rather than being segmented. The extensive handling associated with segmentation of, for example, a reactor pressure vessel may not be justified in terms of dose burden, and should be carefully evaluated against the option of handling the reactor pressure vessel in one piece.

In practice, this approach is considered a viable option only for sufficiently large components, which need extensive segmentation efforts. For smaller components, where segmentation is less extensive, segmentation and loading in containers is the preferred option due to the benefits in subsequent handling, transportation, storage and disposal.

As of now, Ringhals AB is planning to deposit intact PWR pressure vessels in SFL (including reactor internals). This option is discussed in this section.

Description

PWR pressure vessels are removed from the reactor containment, transported to the repository and deposited in the repository in one piece. In order to provide for safe transport and handling, radiation shielding, preferably a metallic shield, has to be applied to each of the vessels prior to any handling.

Prerequisites

Handling of radioactive components of the size of a reactor pressure vessel will be a challenging task. The prerequisites for handling of the vessels according to the described method are presented briefly in this section.

Installation of the radiation shield. As a first task in this process the reactor pressure vessels must be provided with a radiation shield in order to provide radiation protection during all subsequent handling steps. The shield is preferably a thick steel liner that is attached to the PWR pressure vessel by means of a suitable but not yet determined method.

Lifting of the reactor pressure vessels. The reactor pressure vessel can be removed from the reactor containment by means of a large crane. Even though this involves lifting of an object weighing about 540 tonnes (reactor internals and shielding included), it is not considered technically demanding as national and international experience exists from this kind of work.

Overland transport is preferably done by self-propelled modular trailers which can carry heavy loads and can be tailored to each individual transport. There is national and international experience of this kind of operation and the task is considered relatively simple.

Sea transport requires a ship with a big enough cargo capacity. Considering the relatively limited size of the reactor pressure vessels and international experience from sea transport of large objects, the task is considered relatively straightforward, although special arrangements will be needed.

Transport tunnels in the repository. In order to transport intact reactor pressure vessels through the transport tunnel down to disposal depth, the dimensions of the tunnel will have to be increased. The cross-sectional area of the tunnel has to be roughly doubled, compared to if the objects were segmented at the nuclear power plant site and transported in containers.

Rock cavern in the repository. A separate rock cavern needs to be built for PWR pressure vessels, similar to the rock cavern for BWR pressure vessels in the planned extension of SFR. The dimensions of the rock cavern will need to be commensurate with the dimensions of the reactor pressure vessel.

Filling of the reactor pressure vessels in the repository. In order to create a stable and alkaline environment inside the reactor pressure vessels, they will probably have to be filled with grout. This process is not technically challenging but requires careful planning and logistics. One of the main difficulties may be managing the gas that is expelled from the vessel when it is filled with grout. This gas may contain radioactive gases or particles that have to be prevented from contaminating the repository.

Radiation shielding in the repository. Radiation shielding will probably be required in the rock cavern. One suggestion is to build a concrete shield before disposal of the PWR pressure vessels.

Consequences

Environmental impact. The main environmental impact stems from having to increase the dimensions of the transport tunnel down to disposal depth. The total excavated rock volume is doubled compared to the case with segmented parts in containers. Since the tunnel will be 3–5 kilometres long, this will have a substantial impact on the total excavated rock volume.

Dose impact during handling. It is likely that personnel have to work in the vicinity of the reactor pressure vessels during many steps of this operation. Careful planning is needed to manage the handling chain from removal of the reactor pressure vessel at the nuclear power plant site to the backfilled rock cavern.

Impact on long-term safety. No significant impact on the long-term safety of the repository is expected.

The cost. Removing the pressure vessel from the reactor containment without any prior segmentation is likely to be cheaper than if it is segmented. This must be weighed against the expected increase in the construction cost of the repository. The main cost increase relates to the nearly doubled cost for rock excavation of the tunnel from the surface down to disposal depth, due to the roughly doubled tunnel area required for transport of the PWR pressure vessels in one piece. An extra rock cavern for the reactor pressure vessels also adds to the construction cost. However, the total disposal volume is likely to be smaller compared to segmentation and disposal in containers.

5.1.2 Segmentation and loading into steel tanks (BFA-tanks)

This conditioning method is aimed at creating an efficient and safe system for handling, transport, storage and disposal. The neutron-irradiated steel components are segmented and placed in steel tanks. The use of the existing handling system, of which the steel tank is a component, is warranted by the ALARA principle – the waste is handled remotely and finally emplaced in robust containers that can be handled remotely.

The handling alternative implies that steel tanks (Figure 5-1; details in Pettersson 2013) currently used for storage will be disposed of in SFL. The steel tanks form part of a system that has been used by Swedish nuclear power plant operators for lifting out segmented core components for dry storage. The steel tanks are available in different models with different wall thicknesses that can be chosen based on the activity of the waste. Due to the good radiation-shielding properties of the steel, the steel tank is proven to be a well-suited package for handling and storage of core components. One option is to stabilize the waste with grout before the tanks are placed in the repository.

Description

The neutron-irradiated components are segmented using different techniques, usually in the pools on the nuclear power plant sites. The segmented parts are placed in the waste cassette, which is placed at the bottom of the pool. A radiation protection hood is used to lift and transport the filled waste cassette from the segmentation pool to the steel tank which is located in a transport box on the pool side. A vacuum drying cover is then placed on the steel tank and connected to a pump unit to remove water from the steel tank. When the content of the steel tank is dry, the cover is removed and the tank lid is moved in place onto the tank. When the tank lid is bolted to the tank, the transport box is closed and transported to storage or disposal.

Today, there are segmented components from maintenance stored either dry in steel tanks on the nuclear power plant sites or wet in Clab or on the nuclear power plant sites, see section 2.2. The method implies that waste currently stored in pools is segmented and placed in steel tanks. During dismantling of the reactors, the neutron-irradiated components will be segmented in the same way as during maintenance.

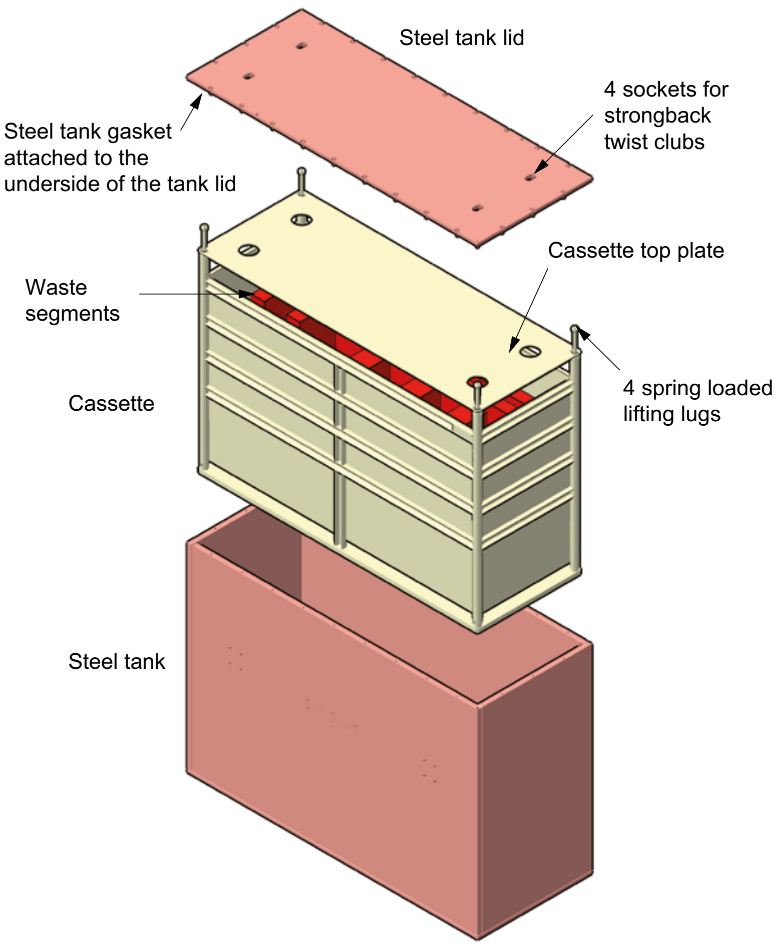


Figure 5-1. *Schematic figure of the steel tank for neutron-irradiated components. The steel tank has external dimensions 3.3 × 1.3 × 2.3 m*^{3} *(length × width × height). The waste is placed in a cassette which is placed in the tank. A lid is bolted to the tank. The thickness of the steel walls can be adjusted to comply with the requirements determined by the activity level of the waste.*

Prerequisites

The following are needed for this method:

Transport containers need to be suited for transport of the waste in its current state. A transport container – ATB 1T – is under development for external transportation of the steel tanks. ATB 1T is required for transportation of steel tanks with segmented parts that have not been stabilized with grout or similar.

Facility for conditioning. If the waste needs to be stabilized before disposal, one or more conditioning facilities need to be available. The facilities must provide for stabilizing of the waste in the existing steel tanks. The tanks need to be opened and filled with grout.

Consequences

Dose impact during handling. Since waste will not be treated outside the waste containers, the dose to the personnel is expected to be limited. The dose load to personnel is dependent on the decay time of the waste before handling.

Impact on long-term safety. No significant impact on the long-term safety of the repository is expected. The use of steel tanks will significantly increase the volume of steel in the repository, which increases both the rate of gas formation from anaerobic corrosion and the total volume of gas produced. The steel tank cannot be credited with the safety function of containment in the assessment of long-term safety of the repository.

The cost for each steel tank is between SEK 450,000 and SEK 950,000 depending on wall thickness. Approximately 450 steel tanks are needed, which gives an estimated total cost for the containers of SEK 300 million. The handling system is already in operation.

5.1.3 Segmentation and loading into new containers

This conditioning method is aimed at creating an efficient and safe system for handling, transport, storage and disposal. The neutron-irradiated steel components are segmented and placed in new containers, other than the steel tanks used in section 5.1.2. The use of shielded containers is warranted by the ALARA principle – the waste will be fixed in robust shielded containers that can be handled remotely.

A set of containers to facilitate efficient handling has been developed on a conceptual level (Pettersson 2013). By designing the set of containers with uniform lateral dimensions, it is possible to handle the containers using the same strongback and stack them in a safe manner during storage and in the repository. The set of containers includes one container with increased wall thickness in order to provide radioactive shielding during handling and storage (Figure 5-2). The benefit of using a new shielded containers compared to the steel tanks would be the possibility of having one single handling system for all SFL waste.

The shielded container for core components is fabricated using standard welding technique and the lid is bolted to the body of the container. However, due to the standard welding technique and the bolted lid, this container cannot be credited with the safety function of containment in the assessment of long-term safety.

Description

Segmentation of neutron-irradiated components is performed according to the methods described in section 5.1.2, and placed in the container. The waste is stabilized with grout in the container.

Neutron-irradiated components currently stored in pools are segmented to fit into the new containers. The waste is placed in the containers and stabilized with grout in the container.

The neutron-irradiated components, which are currently stored in steel tanks, are lifted out of the tanks and placed in the new shielded containers for SFL. The waste is stabilized with grout in the container.

Details concerning the containers such as weights and dimensions can be found in Pettersson (2013).

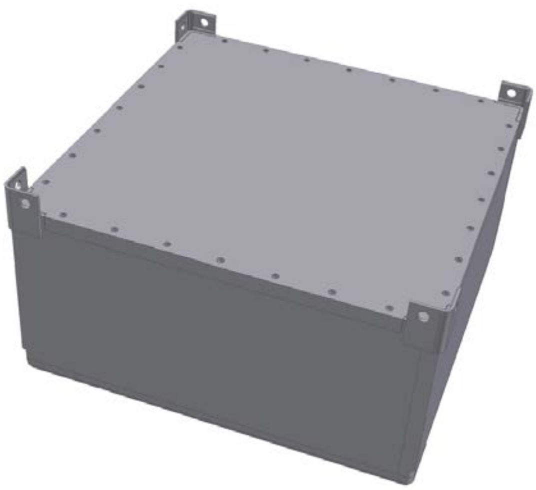


Figure 5-2. *Shielded container for neutron-irradiated components. The lateral dimensions are the same as for the containers shown in Figures 5-3, 5-5 and 5-6. The thickness of the steel walls can be adjusted to comply with the requirements determined by the activity level of the waste. The lid is bolted to the body of the container.*

Prerequisites

Repacking of intermediate level waste into new containers for SFL will be a rather straightforward process as long as segmentation of the waste is not required. In this section the most important prerequisites for the loading and repacking of the waste into new containers for SFL are discussed.

Handling equipment. This handling alternative requires that equipment for lifting out segmented core components from the nuclear power plant pools based on the new shielded container is available.

Repacking facilities. This method requires that one or more repacking facilities are available. The facilities must provide for repacking and stabilizing of waste already stored in steel tanks into new containers. Segmentation is expected to be needed, although for a limited amount of the waste in existing steel tanks. Since intermediate level waste, in particular the core components, will be treated outside the containers, this facility needs to include a hot cell or similar to provide for radiation shielding during operation.

Transport containers. A new set of transport containers need to be designed for the new waste containers for SFL. The external dimensions and weights of these containers must comply with SKB's current transportation and handling system. In Pettersson (2013), it is suggested that two new transport containers be designed carrying one shielded container or two containers for drums or moulds, respectively.

Consequences

Handling of secondary waste. Repacking of the waste will lead to the formation of secondary waste in the form of slightly surface-contaminated containers, such as the steel tanks for core components. These steel tanks need to be decontaminated prior to reuse or recycling. Since the expected level of contamination is low, it is presumed that this procedure can be undertaken for example at the present facility operated by Studsvik Nuclear AB at the Studsvik site. The amount of secondary waste stemming from this conditioning method that must ultimately be disposed of is judged to be very small.

Dose impact during handling. Since waste will be treated outside the waste containers during repacking of waste in existing steel tanks, it is not unlikely that the dose to personnel will be higher than for the handling alternative involving the steel tank (section 5.1.2). The dose load to personnel is above all dependent on the decay time of the waste before handling. However, as it is expected that the repacking will be done in a hot cell the dose is expected to be low.

Impact on long-term safety. No significant impact on the long-term safety of the repository is expected. The use of new containers will like the steel tank option (section 5.1.2) significantly increase the volume of steel in the repository, which increases both the rate of gas formation from anaerobic corrosion and the total volume of gas produced. The container cannot be credited with the safety function of containment in the assessment of long-term safety of the repository.

The cost for each container is approximately the same as for steel tanks, and the number of containers is roughly the same. The total cost for the containers is thus similar to the steel tank option – an estimated SEK 300 million. The cost for a new handling system adds to the total cost.

5.1.4 Segmentation and loading into long-term durable containers

The aim of the conditioning method is to add the safety function *containment* to the repository. The long-term durable container will have the same role as the copper canister in the KBS-3 method, keeping the waste contained for as long as necessary to maintain the long-term safety of the repository.

The metal components with induced activity contain the highly mobile and long-lived nuclides identified in the preliminary safety assessment (SKB 1999a, b) as being of particular concern for the long-term safety of the repository, e.g. Cl-36 and Mo-93. The containment offered by the long-term durable container will allow the long-lived nuclides to decay before the waste itself starts interacting with the groundwater.

The long-term durable container is obtained by manufacturing a container from thick metal plates. The metal plates, including the lid, are welded using special welding methods to create welded joints that are the same thickness as the metal plates. The metal in the container will corrode in the repository, but if placed in a suitable environment this process is very slow. The waste will decay with time before the waste itself starts interacting with the groundwater.

Besides the beneficial long-term properties, the container also provides radiation shielding during handling, storage and transportation.

Description

The containers are manufactured in a conventional workshop, except for the final welded joint that attaches the lid. Segmentation of neutron-irradiated components is performed according to the methods described in section 5.1.2, and placed in the container. Stabilizing of the waste in the container is an option. Finally, the lid is welded to the body of the container, Figure 5-3.

Prerequisites

Repacking of the intermediate level waste into containers intended for subsequent sealing will be a rather straightforward process as long as segmentation of the waste is not required.

The most important prerequisites for the method of repacking of the waste into long-term durable containers for SFL are discussed in this section.

Method for welding of thick steel plates. The welding of thick – approximately 100 mm – steel plates is considered to be the most critical aspect of this conditioning method. In order to be able to ascribe containment properties to this container in the assessment of long-term safety, the corrosion properties of the welds need to be equal to those of the parent metal.

There are a few special, yet commercial, welding methods available that are expected to meet the requirements. Compared to conventional welding, they are more time-consuming and involve additional steps such as pre-heating and post-welding heat-treatments to reduce internal stresses. These welding methods are discussed further in Pettersson (2013). Furthermore, quality control procedures to verify the results of the fabrication process also need to be developed.

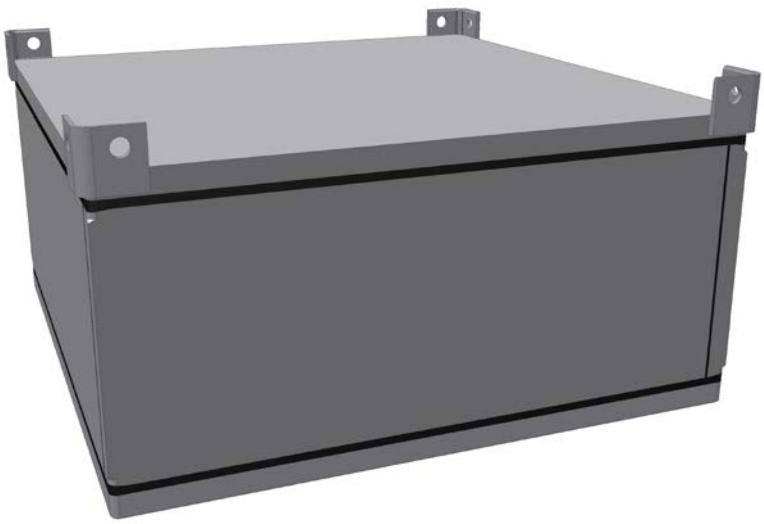


__Figure 5-3.__ The long-term durable container. The lid is welded to the body of the container. All steel plates are joined by a welding process that ensures uniform material properties across the weld. The thickness of the joint is the same as that of the steel plate.

Repository environment. This method has to be combined with a repository concept in which the container is stable over the required period of time. An alkaline anoxic environment is required to limit the corrosion rate of the long-term durable container. This can be achieved by placing the container in, for example, a concrete-based repository, such as described in section 7.2.2.

Facility for conditioning of waste. A conditioning facility that has capabilities for repacking of the waste, possible stabilization of the waste and subsequent welding of the lid to the body of the container must be available. Since intermediate level waste, in particular core components, will be treated outside the containers, the facility must include a hot cell or the like to provide radiation shielding during operation.

Gas formation in the container. Gas can be formed in a container due to chemical processes or radiolysis that require no interaction with the surrounding groundwater. In order to guarantee the long-term function of the sealed container, it must be ensured that the gas pressure in the container does not exceed the tensile strength of the parent metal or of the welds. Waste acceptance criteria must be developed to meet this requirement. Methods for drying the waste prior to packing may need to be developed.

Transport containers. A new set of transport containers need to be designed for the long-term durable container. The external dimensions and weights of these containers must comply with SKB's current transportation and handling system. In Pettersson (2013), it is suggested that two new transport containers be designed carrying one shielded container or two containers for drums or moulds, respectively.

Consequences

Impact on long-term safety. The most important consequence of the described method is that containment properties can be ascribed to the container in the assessment of the long-term safety. The long-term durable container will contain the waste for a very long time, allowing the long-lived nuclides to decay. This is expected to improve the long-term safety properties of the repository.

Handling of secondary waste. Repacking of the waste will lead to the formation of secondary waste in the form of slightly surface-contaminated containers, such as the steel tanks for core components. These steel tanks need to be decontaminated prior to reuse or recycling. Since the expected level of contamination is low, it is presumed that this procedure can be undertaken at, for example, the present facility operated by Studsvik Nuclear AB at the Studsvik site. The amount of secondary waste stemming from this conditioning method that must ultimately be disposed of is judged to be very small.

Dose impact during handling. Since waste will be treated outside the waste containers during repacking of waste in existing steel tanks, it is not unlikely that the dose to personnel will be higher than for the handling alternative involving the steel tank (section 5.1.2). The dose load to personnel is above all dependent on the decay time of the waste before handling. However, as it is expected that the repacking will be done in a hot cell the dose is expected to be low.

The cost for each sealed container is expected to be high, approximately SEK 1 million, for a container with 100 mm wall thickness. Roughly half the cost is attributable to the sophisticated and time-consuming fabrication method. However, the use of this method may reduce the demands on the other engineered barriers and possibly also reduce the cost of their construction.

The research and development efforts needed to realize this method are deemed extensive. The cost for research and development of the welding method, characterization and quality control of welds is thus expected to be very high.

5.1.5 Melting

The main mechanism for release of activity from activated metallic waste is corrosion of the waste. Corrosion enables nuclides previously bound in the matrix of the material to be released and transported out of the repository to the biosphere. Anaerobic corrosion also gives rise to large quantities of hydrogen gas.

One of the main benefits of melting of metallic waste is reduction of the accessible surface area by the formation of homogeneous metal ingots. The reduction in the surface-to-volume ratio reduces the corrosion rate calculated as kg/year, even though the corrosion rate in µm/year is not affected. This extends the decay time before the activity is released, leading to a reduced impact on the biosphere.

The feasibility of the method has been assessed by Huutoniemi et al. (2012). The report contains information about the method itself and the constraints imposed by that method. The conceptual design, construction and operation of a plant for melting of metallic intermediate level waste are presented.

Description

Melting of inactive material or material with a very low level of activity is a straightforward process which is done on a daily basis in foundries all over the world. Melting of long-lived intermediate level waste would not differ technically from melting of less active or inactive material, but the safety precautions have to be much more stringent.

The waste will be segmented at the waste producers' sites in order to allow transportation and will arrive at the melting plant in shielded steel containers. At the melting plant, further segmentation may be needed prior to melting in order for the waste to fit into the charging basket for the furnace.

The melting sequence is as follows:

- The waste is loaded into the furnace from the shielded charging basket.
- During the melting process, the melt must be stirred to ensure proper melting. Due to the high dose rate, this must be performed by a manipulator, either automated or manually remote-controlled.
- The generated slag is scooped up by the stirring mechanism and loaded into a container for cooling prior to further conditioning.
- The molten metal is poured into one or several moulds. The ingot is fitted with a lifting hook before it solidifies.

The lifting hook permits remote lifting of the ingot for transport to a packaging area, where the ingot is placed in a waste container. Samples from the melt are taken during processing for determination of the activity content.

Secondary waste is collected and packaged for disposal.

Prerequisites

Melting of intermediate-level waste must be considered rather hazardous and must be subject to stringent safety precautions. The most important prerequisites are summarized in this section.

Storage of waste prior to melting. It is considered necessary to set dose requirements on the waste to allow handling. In the case of metallic intermediate level waste, a significant decay time has to be imposed on the waste prior to treatment. In the case of melting of core components, reactor internals and PWR pressure vessels, approximately 75 percent of the waste can be treated after a decay time of 50 years (Huutoniemi et al. 2012). Approximately half of that mass is attributed the PWR pressure vessels.

Liquids and trapped gases. The metal charged to the melting furnace must not under any circumstances contain liquids, moisture or enclosed volumes. The presence of liquids and moisture in scrap metal can cause severe accidents when the material is melted.

Remotely operated handling equipment and hot cell. Remotely operated equipment must be used for segmentation. There is also a need to inspect and, if needed, physically manipulate the material in order to make it more suitable for melting. Two examples of this are the need to open closed compartments and to compact tubular objects in order to reduce the risk of floatation during melting. A hot cell will probably be needed for pre-treatment.

New melting plant. The conclusion drawn in Huutoniemi et al. (2012) is that a new melting plant needs to be built in order to handle intermediate level waste. Based on the authors' experience in operating a low level waste melting plant, their conclusion is that without technical improvements such a facility is not feasible today.

The main technical challenge is, not unexpectedly, the furnace itself. As stated in Huutoniemi et al. (2012), experience from operation of melting furnaces for mainly low level waste have shown that operational disruptions and accidents occur to such an extent that they should be expected.

When evaluating the technical feasibility of the melting plant, both normal and non-normal operating conditions need to be considered. Normal operating conditions, with no disturbances, are not considered to be critical for the technical feasibility of the melting plant. However, considering non-normal operation, several factors may affect the feasibility of the facility. The main factors are loss of electric supply, ladle breakthrough and vapour explosions.

Consequences

Dose impact to operators during handling. The dose burden during normal operation is expected to be low due to the fact that the whole normal operation of the facility is expected to be performed by remote-controlled equipment or by manipulators from a shielded area.

The main challenge is in non-normal operation. Due to the potential consequences of an accident, it is important to be able to show that probabilities are low and/or that normal operation can be restored with an acceptable radiological impact.

Dose impact to critical group outside the facility. Based on the safety assessment for the current melting plant at the Studsvik site, the average dose burden to a member of the critical group from releases to air from a new melting facility is expected to be below the regulatory limit (Huutoniemi et al. 2012).

Environmental impact. Apart from the radiological effects described above, the plant is expected to have effects similar to those of comparable industrial facilities. Due to the low capacity of the melting plant, its total environmental impact will be small. The main impact is energy usage. The energy consumption of the facility for intermediate level waste is expected to be similar to that of the current melting plant for low level waste at the Studsvik site.

Impact on long-term safety. The reduced surface-to-volume ratio of the melted material has a positive effect on long-term safety by reducing the amount of corroded material per unit time and thereby the release rate of radionuclides from the waste.

Reduction of the volume. Melting is currently applied to low level radioactive scrap metal for the purpose of reducing the disposal volume. The volume reduction is mainly achieved by releasing the metal from regulatory control after melting. The specific activity of the intermediate level waste planned for SFL is high, making it difficult for the material to be released from regulatory control. The main benefit is therefore the production of a homogeneous waste form with a smaller volume and a reduced surface-to-volume ratio. A smaller waste volume also reduces the required capacity of the repository.

Secondary waste. The melting process will result in secondary waste, primarily slag from the furnace. The slag contains a significant fraction of the heavy long-lived nuclides present in the primary waste and needs to be disposed of in SFL.

The cost for construction and decommissioning of the melting plant is expected to be high, in the order of SEK 250–350 million, with an estimated annual operating cost of about SEK 10 million. The waste will also have to be stored in a storage facility for about 50 years to allow it to decay prior to conditioning, unless it can be stored at the nuclear power plant sites.

5.2 BWR control rods

Used BWR control rods are stored today in the central storage facility for spent nuclear fuel (Clab) or in the pools at the nuclear power plants. The entire programme will result in approximately 4,000 BWR control rods. The control rods consist of stainless steel and neutron-absorbing material such as boron carbide. All handling requires radiation shielding, and the most demanding radionuclide from a radiation safety viewpoint during the first 70 years is Co-60. Two handling alternatives for the BWR control rods have been identified:

- Disposal of whole control rods.
- Segmentation and loading into containers.

The handling alternatives are described in this section, and the benefits and consequences of the methods are discussed.

5.2.1 Disposal of whole control rods

The most straightforward handling alternative for BWR control rods is to deposit the control rods in their current state and form. This method is justified by the ALARA principle, since by minimizing the handling steps needed, the dose burden on the personnel is expected to be lower.

The handling alternative entails BWR control rods to be placed in copper canisters and deposited in the planned repository for spent nuclear fuel.

One alternative is to design a new long waste container in which whole control rods can be placed, similar to the container used in SKB (1999a, b).

Method

BWR control rods are transferred from their current location in Clab to the Encapsulation Plant in a similar way as is planned for the spent nuclear fuel. The BWR control rods are emplaced in copper canisters and subsequently follow the same route as the spent nuclear fuel to disposal in the Spent Fuel Repository.

Prerequisite

Adaptation of the Encapsulation Plant. The Encapsulation Plant needs to be adapted to handle BWR control rods as well. The BWR control rods will be placed in copper canisters that are sealed in a similar fashion as is planned for the spent fuel.

Consequences

BWR control rods in the repository for spent nuclear fuel. The BWR control rods need to be emplaced in the repository for spent nuclear fuel, which will increase the disposal volume required in the planned repository for spent nuclear fuel. There are approximately 4,000 BWR control rods in the entire nuclear programme. Assuming that 12 BWR control rods can be fitted into a copper canister, there will be approximately 350 additional canisters to be disposed of in the planned repository for spent nuclear fuel. The possibility of increasing the packing density is discussed in section 5.2.2. Canisters with BWR control rods can likely be deposited with a smaller distance compared to canisters with spent fuel, since the heat generation is negligible.

Dose impact during handling is expected to be limited.

Environmental impact during handling. The environmental impact is deemed small. The process does not make use of chemicals. Release of substances to the environment is negligible. The impact is mainly related to the use of materials and energy.

Impact on long-term safety. It is likely that the long-term safety is improved by using the KBS-3 method for disposal of the BWR control rods.

Cost. Given that the system for management and disposal of spent nuclear fuel will be realized, the cost for this alternative is mainly related to the cost of additional labour, canisters and transportation.

5.2.2 Segmentation and loading into containers

The physical dimensions of the BWR control rods restrict dense packing, both in Clab and containers. The proposed conditioning method aims to reduce the dimensions of the BWR control rods, permitting dense packing in a selected container, both for storage under dry conditions and for disposal.

One possible method for segmentation of BWR control rods has been assessed by Pettersson (2013), and is presented in this section.

Description

Used BWR control rods are kept today in wet storage, primarily in Clab. The handle has been removed prior to transportation and storage, leaving a 4.2 metres long cruciform-shaped part, about 0.27 metre × 0.27 metre in size (Pettersson 2013).

The method involves three steps, as shown in Figure 5-4:

1. Removal of the stem of the control rod.
2. Segmentation of the control rod into four blades.
3. Optional further segmentation of the individual blades into shorter parts, which will fit into a container suitable for SFL.

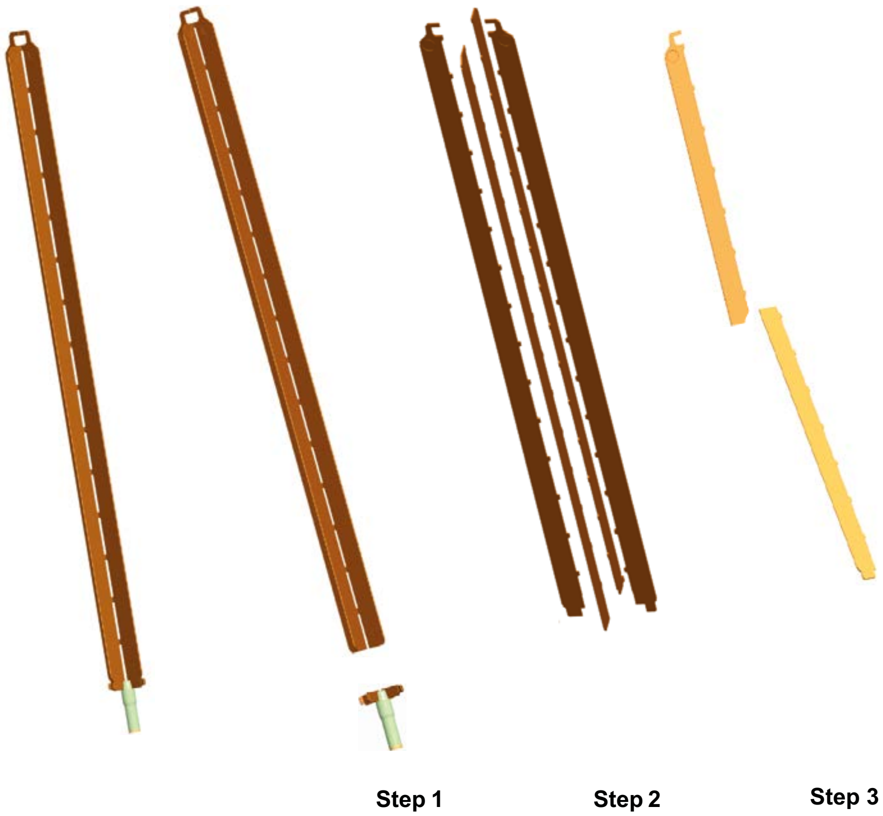


Figure 5-4. *Schematic illustration of the proposed method for segmentation of BWR control rods: 1) Removal of the stem of the control rod, 2) segmentation of the control rod into four blades and 3) (optional) further segmentation of the individual blades into shorter parts.*

The first cut involves removal of the stem in order to prepare the control rod for further segmentation. This procedure is preferably done by means of a saw.

The second cut is along the centre of the cruciform-shaped control rod to create four blades, each about 4 metres long and about 0.14 metre wide. This operation can be done without the release of any boron carbide.

The third step involves segmenting the steel blade into two equally long parts. Release of boron carbide is anticipated during this step, since one or several channels filled with boron carbide will be punctured.

After segmentation, the control rods are placed in suitable containers. Depending on whether the segmentation comprises two or three steps the requirements on the dimensions of the waste containers will vary. As an example, the cast iron insert of a copper canister used for disposal of spent nuclear fuel can be packed with a large number of control rods if the control rods are segmented into four blades. This will increase the packing density compared to the alternative presented in section 5.2.1. Disposal in containers described in sections 5.1.2 and 5.1.3 requires all three steps in the segmentation process.

Prerequisites

Segmentation of the control rods will require the development of new equipment, but also a facility in which this process can be carried out. Due to the high activity of the control rods, segmentation is likely to be done under water in a pool or in a hot cell but apart from that no major technical obstacles are expected. The following prerequisites have to be fulfilled:

Equipment for segmentation. Equipment for removal of the stem, segmentation of the control rod into four separate blades, and finally for the segmentation of each individual blade is needed. Design suggestions for this type of equipment are found in Pettersson (2013).

Facility for conditioning of waste. The most important consideration in designing the facility is probably whether segmentation should be performed in a dry hot cell or under water in a pool. If segmentation is done adjacent to Clab, where the control rods are currently stored in pools, under water is probably the preferred option.

Adaptation of the Encapsulation Plant. If the segmented blades are to be emplaced in copper canisters for disposal using the KBS-3 method, the Encapsulation Plant needs to be adapted to handle BWR control rod blades as well. The BWR control rod blades will be placed in copper canisters that are sealed in a similar fashion as is planned for the spent fuel.

Consequences

Dose impact during handling. Segmentation of control rods will probably entail the release of radioactive substances, at least during the third and final step of this process, during which at least one of the boron-carbide-filled channels is punctured. Tests show that there is no positive gas pressure in the control rods (Pettersson 2013). The method is likely to entail the release of both boron carbide and tritium gas. Methods for handling the release of powder and gas into the pool water are commercially available.

Environmental impact during handling. The environmental impact is deemed very small. The process does not make use of chemicals and only limited amounts of energy. Release of substances to the environment is negligible.

Impact on long-term safety. Segmentation of the control rods is expected to have only a limited impact on the long-term safety of the repository. If the segmented blades are to be emplaced in copper canisters for disposal using the KBS-3 method, it is likely that the long-term safety is improved.

Packing density. Segmentation of the control rods will increase the packing density in the waste containers significantly, and as a consequence the required repository volume for the control rods will also be reduced. Approximately 90 BWR control rods are expected to fit into a standard Clab cassette after a two-step segmentation (Pettersson 2013). If 90 BWR control rods can be fit into one copper canister, the approximately 4,000 BWR control rods in the entire nuclear programme will need an estimated 50 canisters.

BWR control rods in the repository for spent nuclear fuel. If the segmented blades are to be emplaced in copper canisters for disposal using the KBS-3 method, the required disposal volume in the planned repository for spent nuclear fuel will increase. Canisters with BWR control rods can likely be deposited with a smaller distance compared to canisters with spent fuel since the heat generation is negligible.

The cost is mainly dependent on whether the segmentation process can be undertaken in or adjacent to an existing facility, such as Clab, or if a new separate facility has to be built. From a cost perspective, a facility adjacent to Clab is beneficial to avoid handling and transport of the control rods prior to segmentation. If the segmented blades are to be emplaced in copper canisters for disposal using the KBS-3 method, costs for additional copper canisters adds to this.

5.3 Legacy waste

The legacy waste from early research in the Swedish nuclear programmes is today managed by AB SVAFO. This waste is to a large extent already conditioned. The waste is currently placed in different types of containers such as drums, moulds, large tanks and smaller boxes, as shown in Chapter 2.

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

- Handling without further conditioning.
- Repacking into new containers.
- Retrieval and sorting.
- Vitrification.

This section describes the handling alternatives assessed for the legacy waste.

5.3.1 Handling without further conditioning

The most straightforward conditioning method is to deposit the waste in SFL in its current state and form, such as it is described in Chapter 2. The dose to personnel is expected to be limited, by minimizing the conditioning needed.

In reality, however, parts of the waste are still expected to require emplacement in suitable containers in order to ensure safe transport and handling.

Method

The waste is accepted for disposal in SFL in its current state and form. Requirements on the waste packages are only imposed by the needs of safe transportation and handling.

Prerequisite

Transport containers. The transport of a majority of the existing waste packages – drums and moulds – can be handled within SKB's current transportation and handling system. For other waste packages, special arrangements within the current system will be needed, unless new transport containers are to be designed.

Consequences

Requirements on the repository. The strength of individual packages must be taken into consideration when packing and stacking different types of packages in a disposal room.

Dose impact during handling. Handling of individual drums and waste packages of various sizes is time-consuming and inappropriate from a handling viewpoint. The dose to the personnel is dependent on the high level of manual labour required to carry out this method. The dose impact is thus likely to be higher than if the waste is handled in a more efficient way.

Impact on long-term safety. This handling alternative is expected to have only a limited impact on the long-term safety of the repository.

Cost. Handling will be rather labour intense since several different waste containers will be handled during transportation and disposal. The cost is dependent on the high level of manual labour required to carry out this method.

5.3.2 Repacking into new containers

This conditioning method is aimed at creating an efficient and safe system for handling, transport, storage and disposal. The legacy waste is currently placed in different types of containers. The objective of this method is to use new standardized containers with uniform dimensions as overpacks for all types of legacy waste. The insides of the containers are tailored to accommodate existing waste packages, such as drums and moulds.

A set of containers to facilitate efficient handling has been developed on a conceptual level (Pettersson 2013). By designing the set of containers with uniform lateral dimensions, it is possible to handle the containers using the same strongback and stack them in a safe manner during storage and in the repository. The use of standardized containers is warranted by the ALARA principle – once the original packages are placed in the containers, the waste will be fixed in robust containers that can be handled remotely.

Description

The legacy waste currently stored in moulds and drums is packed into the new containers (see Figures 5-5 and 5-6). The waste packages are stabilized with grout in the container. The new containers will serve as overpacks for the moulds and drums and provide the mechanical strength needed for safe transport and handling of the waste.

The set includes containers for:

- 200-litre drums (Figure 5-5),
- 280-litre drums, and
- standard moulds (1.2 × 1.2 × 1.2 m^3) (Figure 5-6).

Details concerning the containers such as weights and dimensions can be found in Pettersson (2013).

Prerequisites

In this section the most important prerequisites for the method of repacking of the waste into new containers for SFL are discussed.

Repacking facility. This method requires that one repacking facilities is available. The facility needs to handle the legacy waste containers, which will be placed in standardized containers and stabilized for safe handling and transportation.

Transport containers. A new set of transport containers needs to be designed for the new waste containers for SFL. The external dimensions and weights of these containers must comply with SKB’s current transportation and handling system. In Pettersson (2013), it is suggested that two new transport containers be designed carrying one or two waste containers, respectively.

Figure 5-5. Waste container for waste currently stored in 200-litre drums. Its lateral dimensions are 2.69 metres × 2.69 metres, which is the same as for the containers shown in Figures 5-2, 5-3 and 5-6. The height of the container is 0.96 metre. The drums are stabilized in the container using grout. The container has no lid, since the grout is intended to fill the full volume of the container. Reinforcement bars to improve the mechanical properties of the grout are shown in the figure.

Figure 5-6. Waste container for waste currently stored in moulds. The lateral dimensions are the same as for the containers shown in Figures 5-2, 5-3 and 5-5. The height of the container is 1.30 metres. The moulds are stabilized in the container with grout. This container also has no lid, since the grout is intended to fill the full volume of the container. Reinforcement bars to improve the mechanical properties of the grout are shown in the figure.

Consequences

Dose impact during handling. Since handling of the waste is expected to be more efficient, once the original waste packages have been emplaced in the new containers, the dose to personnel is likely to be lower than if the waste is handled without further conditioning.

Impact on long-term safety. This handling alternative is expected to have only a limited impact on the long-term safety of the repository.

Cost. The main cost is related to the unit cost of the new containers. Since handling of the waste is expected to be more efficient, the handling cost is likely to be lower than if the waste is handled without further conditioning.

5.3.3 Retrieval and sorting

The nuclide and material inventory of the legacy waste is not fully characterized. Great uncertainties are thus associated with the reference inventory and the initial state at the time of closure of the repository. The aim of this handling alternative is to reduce the uncertainties and facilitate characterization of the legacy waste.

The method entails that the existing waste packages with conditioned waste are crushed and the waste sorted prior to further handling. Thanks to this method, a number of well-characterized waste fractions will be obtained. The new information will reduce the uncertainties associated with the legacy waste, which is beneficial for the assessment of the long-term safety of the repository. Sorting also makes it possible to treat certain waste fractions by means of other conditioning methods described in this chapter.

The sorting procedure is similar to the one undertaken by AB SVAFO during the period 1986–2002 to empty the storage facility AT in Studsvik.

Description

The containers are opened and the grout is crushed in an enclosed area. The waste is then sorted into a predetermined number of fractions by remote-controlled equipment. The waste may then be further processed by means of other conditioning methods described in this chapter.

Prerequisite

Sorting of the legacy waste is expected to be a time-consuming activity. However, provided sufficient capacity is made available, the technical challenge will be less daunting. The prerequisite for sorting of the legacy waste is discussed in the following section.

Facility for conditioning of waste. Sorting of the waste requires a conditioning facility with equipment for remote handling. A hot cell must be available for handling certain fractions of the legacy waste. Since the waste is currently stored in drums and is thus already segmented into small pieces, it is not expected that any equipment for additional segmentation of the waste components will be required.

The hot-cell facility managed by AB SVAFO in Studsvik is an example of a facility with suitable capability. However, due to the large volumes of legacy waste, a new plant needs to be built in order for this alternative to be feasible.

Consequences

Dose impact during handling. As with all other types of handling of radioactive material, it is expected that personnel operating the conditioning facility will be exposed to some radiation dose. However, the dose to personnel is expected to be kept low since crushing and sorting is expected to be done in hot-cell or similar.

Impact on long-term safety. The new information will reduce the uncertainties associated with the legacy waste, which is beneficial for the assessment of the long-term safety of the repository. By separating different waste fractions, it is possible to avoid unfortunate combinations of waste in the repository. Liquids in the legacy waste can be collected and solidified.

Acceptability. The probability of obtaining approval for disposal of the legacy waste is likely to increase if knowledge of the waste is improved.

The cost for a sorting facility is expected to be high (of the order of a few hundreds of millions of SEK), but it is not possible at present to give any exact figures. However, it is expected that sorting will lead to a significant reduction in the required disposal volume, since a large portion of the grout surrounding the waste today can be released from regulatory control. This also reduces the required repository volume. The effect on the total building cost of the repository is however limited since the volumes of the disposal areas are small compared to the total excavated volume for the repository and repository facilities.

5.3.4 Vitrification

The aim of this conditioning method is to transform the legacy waste into a more homogeneous and stable waste form, thus enhancing the long-term safety of the repository. The method involves the use of heat to vitrify the conditioned legacy waste and cast it into solid glass-like blocks.

As an alternative to direct vitrification of entire drums, the waste can be sorted as described in section 5.3.3 prior to vitrification. This will allow characterization of the legacy waste prior to vitrification, thus reducing the uncertainties. Vitrification can then be applied to certain waste fractions, improving the stability of the waste.

Description

The method involves melting the waste, drums and grout in a plasma-arc furnace or the like. The high-temperature process (up to 20,000 °C) results in a stable glass-like material. Sometimes a vitrification medium, like glass, is added in the process to form the desired final product. The vitrification process can be applied to the entire drum or selected waste categories. If the process is preceded by crushing and sorting of the material as discussed in section 5.3.3, more homogenous waste fractions will be obtained for vitrification.

Prerequisites

Vitrification of the legacy waste is considered a rather complicated task, surrounded by a large number of safety considerations. The most important prerequisites are summarized in this section.

Vitrification plant. A plant is required for vitrification of radioactive material. Today a few such plants exist in the world, e.g. Zwilag in Switzerland. No such plant exists in Sweden today.

The design of the plant is dependent on the exact process regime, especially the process temperature. The temperature is in turn dependent on the heterogeneity of the waste. If a single process is used for all the existing legacy waste, a higher temperature will probably be needed than if certain waste fractions are selected for processing. Higher process temperatures lead to increased outgassing and more secondary waste, such as filters etc.

Sorting of the waste. As an alternative to direct vitrification, the waste can be sorted prior to vitrification. In this way, the waste categories that are not suited for vitrification can be handled separately. This results in a more homogeneous waste product from the vitrification process, which is beneficial for subsequent handling. On the other hand, sorting of the waste is expected to lead to smaller volumes to be vitrified. A sufficiently thorough sorting process could make the vitrification step redundant.

Repository environment in which the waste form is stable. The glass-like waste resulting from the vitrification process still needs to be emplaced in a repository. Some glasses are known to be less stable in an alkaline environment. This may put constraints on the use of concrete or other cementitious materials in the construction of the engineered barriers in the repository.

Consequences

Dose impact during handling. As with all other types of handling of radioactive material, it is expected that personnel operating the vitrification facility will be exposed to some radiation dose. However, the dose to personnel is expected to be kept low by the use of sufficient shielding and remotely operated equipment.

The dose to a member of the critical group outside the facility is expected to be similar to that experienced outside a melting plant, as discussed in section 5.1.4, which would mean a limited dose to the public.

Environmental impact during handling. Apart from the radiological effects described above, the plant is expected to have effects similar to those of comparable industrial facilities. The amount of waste is limited, which means a small-scale plant. Since the capacity of the plant is also low, the total environmental impact will be small. The main impact is energy usage, which is expected to be quite high per unit weight, but small in absolute numbers due to the small volumes.

Secondary waste. This technique leads inevitably to secondary waste such as filters from flue gas cleaning and waste from dismantling and demolition of the plant, which also requires disposal.

Impact on long-term safety. It is expected that the increased stability of the waste form compared to the untreated material will have beneficial effects on the long-term safety of the repository. The initial state will, however, still be associated with great uncertainties unless the waste is characterized prior to processing. If sorting and characterization of the waste is done prior to processing, this uncertainty can be overcome.

The cost for construction and decommissioning of a vitrification plant is uncertain, but is expected to be high – at least as high as the cost for a melting plant (SEK 250–350 million). The R&D costs for developing suitable processes must also be considered.

5.4 Waste from other sources

This waste category includes operational and decommissioning waste from Studsvik Nuclear AB, operational and decommissioning waste from AB SVAFO, as well as waste from industries, hospitals, universities and research facilities, which is also managed by Studsvik Nuclear AB. Parts of the waste is stored in Studsvik today, parts are estimated future waste from operations and decommissioning.

Two main alternatives can be formulated for this waste category:

- Handling without further conditioning.
- Loading into new containers.

This section describes the handling alternatives assessed for this waste category.

5.4.1 Handling without further conditioning

The most straightforward conditioning method is to deposit the waste in SFL in its current state and form, such as it is described in Chapter 2. The dose to personnel is expected to be limited, by minimizing the conditioning needed.

Method

The waste is accepted for disposal in SFL in its current state and form. Requirements on the waste packages are only imposed by the needs of safe transportation and handling.

Prerequisite

Transport containers. The transport of the existing waste packages – drums and moulds – can be handled within SKB's current transportation and handling system.

Consequences

Requirements on the repository. The strength of individual packages must be taken into consideration when packing and stacking different types of packages in a disposal room.

Dose impact during handling. Handling of individual drums is time-consuming and inappropriate from a handling viewpoint. The dose to personnel is dependent on the high level of manual labour required to carry out this method. The dose impact is thus likely to be higher than if the waste is handled in a more efficient way.

Impact on long-term safety. This handling alternative is expected to have only a limited impact on the long-term safety of the repository.

Cost. Handling will be rather labour intense since individual drums will be handled during transportation and disposal. The cost is dependent on the high level of manual labour required to carry out this method.

5.4.2 Loading into new containers

This conditioning method is aimed at creating an efficient and safe system for handling, transport and disposal. The objective of this method is to use the same standardized containers for this waste category as for the legacy waste. The waste can be loaded into the containers either directly or by loading primary waste packages, such as drums or moulds, into the containers.

Description

The waste is packed into the new containers as described in section 5.3.2. The waste or waste packages are stabilized with grout in the container. The new container will provide the mechanical strength needed for safe transport and handling of the waste.

Prerequisite

Transport containers. A new set of transport containers needs to be designed for the new waste containers for SFL. The external dimensions and weights of these containers must comply with SKB's current transportation and handling system. In Pettersson (2013), it is suggested that two new transport containers be designed carrying one or two waste containers, respectively.

Consequences

Dose impact during handling. Since handling of the waste is expected to be more efficient using a common system for waste handling, the dose to personnel is likely to be lower than if the waste is handled without further conditioning.

Impact on long-term safety. This handling alternative is expected to have only a limited impact on the long-term safety of the repository.

Cost. The main cost is related to the unit cost of the new containers.

5.5 General methods

This section presents a method that is not restricted to a particular category of waste.

5.5.1 Addition of agents for chemically improved retention

The aim of this conditioning method is to prevent radionuclides from being transported out of the repository by stabilizing the waste in the repository. The method involves the addition of a large amount of active material to the repository that will chemically stabilize the waste. The material is co-deposited with the waste in the containers or caverns, or mixed into other components of the repository such as concrete walls etc.

This method is not restricted to a particular category of waste. However, different waste categories may require the use of different chemically active materials to ensure the most efficient retention of the radionuclides. In practice, it is likely that a few radionuclides will be targeted by this method, and retention materials designed for those particular radionuclides will be included in the repository.

A literature review of materials suitable for improving sorption is presented in Krall (2012). The review has placed a special emphasis on the adsorption of anions. Due to their high release and dose rates from the far field, Cl-36, Mo-93, C-14, and I-129 are of particular interest.

Description

This conditioning method is simple in the sense that it does not involve treatment of the waste form itself. A selected material used to improve retention is added either directly together with the waste or mixed with other repository components, such as grout or engineered barriers.

Materials that can improve the retention properties of the repository components can be added:

- To the waste containers (instead of using grout).
- To the grout inside the waste containers.
- To the space between the waste containers.
- To containers filled only with material for chemically improved retention.
- To the concrete or other materials in the engineered barriers.

Many types of materials with differing properties can be used to improve retention. Below is a list of possible retention mechanisms covered by this conditioning method.

Formation of a stable chemical compound. A chemically stable, insoluble compound is formed by a reaction between the radionuclides and a component present in the added material.

Chemisorption. A strong covalent bond is formed between the radionuclides and a surface group on the active material without the actual formation of a new compound.

Physisorption. A bond is formed between the radionuclide and the material without the formation of a chemical compound. This type of bond often relies on electrostatic forces rather than the sharing of electrons as in a covalent bond.

Trapping. The radionuclides are physically trapped inside voids in the active material without an actual bond being formed. This method can be regarded as a filtration process rather than a chemical or physical process. Zeolites can be used for this purpose.

Red-ox control. Ensuring that the Eh in the repository is maintained at a level where both Np and Tc as well as other radionuclides such as Pu and U are in their lower oxidation states.

pH control. The corrosion rate of the metallic waste forms and thus the release rate of the radionuclides can be dramatically reduced if it is ensured that the metals are in a passive state by the addition of a material which ensures a high pH in the repository. Concrete or other types of cementitious materials can be used for this purpose.

Prerequisites

Adding chemical compounds to the repository can be expected to be technically simple. However, extensive R&D efforts are foreseen, given the combination of different materials and the task of proving the stability of the added compounds over the very long time periods required. The most important prerequisites for the use of retention agents in the repository are outlined in this section.

Repository environment. The method requires that the active material remains stable for the entire lifetime of the repository. It also requires that the active component can be delivered at a sufficient rate to react with the radionuclides when needed.

Choice of material. It has to be ensured that the materials used to improve retention are not consumed by reactions with other types of materials present in the repository.

Consequences

Need for R&D. The use of retention materials can be expected to require rather extensive R&D efforts in order to identify suitable materials or combinations of materials for the different waste categories.

Dose impact during handling is expected to limited, since the method does not imply handling of the waste itself.

Impact on long-term safety. It is assumed that the addition of materials that improve the retention effect of the components in the repository will have a positive impact on long-term safety. The introduction of new materials into the repository will, however, increase the complexity of the analyzed system, making the safety assessment a more challenging task.

The cost of sorption materials will probably be limited since the volumes are small. The main cost is likely to be related to the R&D needed to verify the long-term effects of the materials and to take credit for these effects in the assessment of long-term safety.

6 Requirements and evaluation factors

6.1 Applicable laws and regulations

The external requirements have been identified in the following laws and conventions (introduced in section 1.1):

- Nuclear Activities Act (SFS 1984:3).
- Radiation Protection Act (SFS 1988:220).
- Environmental Code (SFS 1998:808).
- Work Environment Act (SFS 1977:1160).
- Planning and Building Act (SFS 2010:900).
- Transport of Dangerous Goods Act (SFS 2006:263).
- Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management (SÖ 1999:60).

The regulations issued by the Swedish Radiation Safety Authority based on the Nuclear Activities Act and the Radiation Protection Act are also taken into account in the study.

The selection of laws and regulations is not to be seen as a complete set of requirements on the repository system, and is made based on their judged relevance given the scope of the study and their possible influence on the repository concepts and the evaluation of the concepts.

6.2 Requirements on the SFL system and system components

The repository system includes the entire chain from the origin of the waste, via conditioning, storage, transportation to deposition in the repository. Some requirements apply to the entire system and while others apply only to a specific component of the system. The requirements are therefore divided into two levels: *level 1* comprising requirements that apply to the entire system and *level 2* comprising requirements that apply to specific components. Level 1 requirements are also applicable to the individual components. Figure 6-1 illustrates the relationship between external requirements (laws and regulations as well as owner requirements), and the derived level 1 and level 2 requirements.

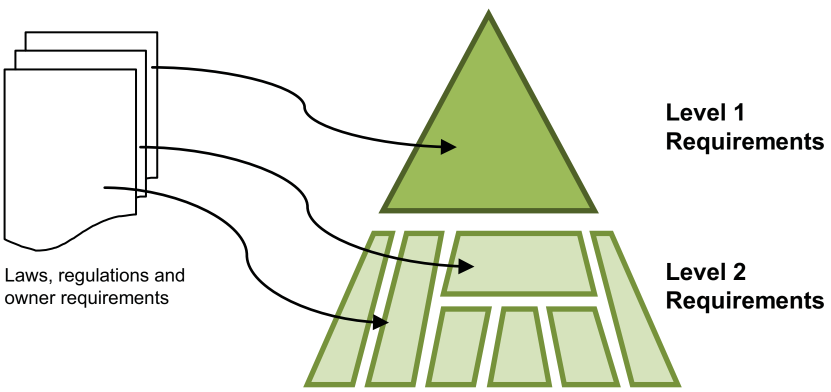


Figure 6-1. *Illustration of the relationship between external requirements (laws, regulations and owner requirements), level 1 requirements (applicable to the entire system) and level 2 requirements (applicable to specific components).*

Level 2 requirements are subdivided according to which component they are applicable to:

- Waste and containers.
- System for external transportation.
- Facilities in the repository system;
  - Facilities for conditioning of waste.
  - Storage facilities.
  - Repository facilities (prior to closure).
- Repository (the barrier system for long-term safety).

The scope of each component is defined by a general functional requirement. There are also requirements set by one component on another. These requirements typically arise at the interface between components, where the design of one component will affect the design of another. The requirements are listed in Appendix 1.

6.3 Definition of evaluation factors

Evaluation factors are formulated based on the derived requirements. By associating each requirement with at least one evaluation factor, the relationship between the requirements and the evaluation factors is made traceable as shown in Figure 6-2.

As a result of this association, requirements that are verifiable can be used directly to exclude concepts that do not fulfil a requirement. The evaluation factors are used in the evaluation of the remaining concepts. The requirements are thus used indirectly in the evaluation through the use of the evaluation factors. The associations between the requirements and the evaluation factors are shown in the list of requirements in Appendix 1.

The evaluation factors are grouped into the following evaluation categories: *Long-term safety*, *Environment and society*, *Technology* and *Cost and time*. The evaluation factors within each evaluation category are illustrated in Figure 6-3 and described below.

6.3.1 Long-term safety

Feasibility of making a post-closure safety assessment

The ability to provide a state-of-the-art safety assessment of the repository is evaluated by estimating the feasibility of making a post-closure safety assessment. This includes the level of process understanding and the how well these processes can be modelled. The assessment also includes how well the initial state of the repository can be defined, achieved and verified, in other words how well input data can be defined for the assessment of safety after closure.

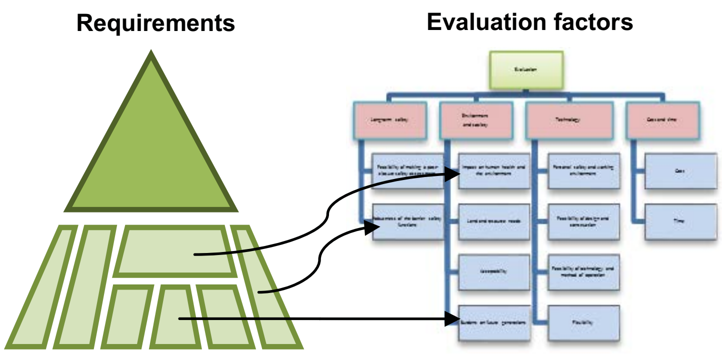


Figure 6-2. *Illustration of the relationship between derived requirements and evaluation factors.*

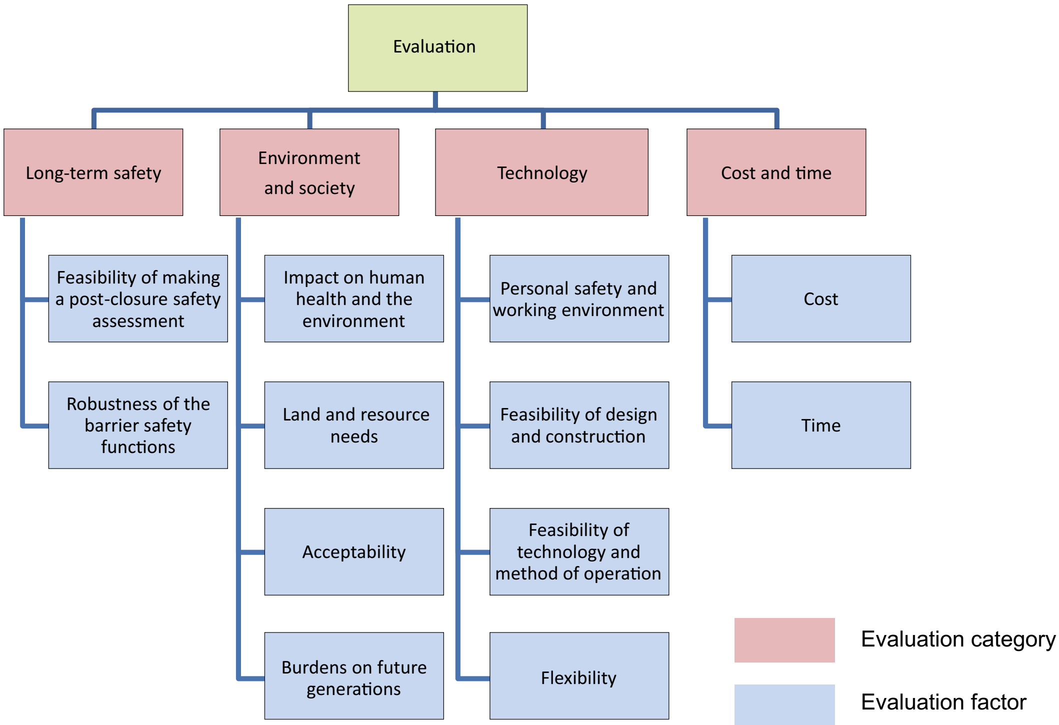


***Figure 6-3.** Illustration of the evaluation categories and evaluation factors.*

Robustness of the barrier safety functions

An evaluation of the robustness of the barrier safety functions is done by means of a qualitative assessment of the potential long-term safety of the barrier system. The evaluation of barrier safety functions is a complex issue. This qualitative assessment includes many different aspects such as effects of processes in the waste and barriers, chemical reactivity, and release rate of radionuclides from the waste. The assessment also includes the risk of unintentional intrusion in the future.

6.3.2 Environment and society

Impact on human health and the environment

The impact on human health and the environment is evaluated by means of an assessment of disturbances (noise and emissions to air, land and water, and visual impacts) caused during the construction and operation of the system facilities, including transportation. The evaluation factor also includes the use of such chemical products or biotechnical organisms that are likely to pose risks to human health or the environment.

Land and resource needs

Land and resource needs are estimated by means of an assessment of the consumption of raw materials and energy during construction and operation of the system facilities, including transportation. The need for raw materials that can be expected to be in short supply in the future is considered. The evaluation factor also includes the land area needed for, and the area affected by, the repository system and whether the system may give rise to some future restrictions on land use (construction prohibition zones, etc.)

Acceptability

Acceptance and support from society is necessary for a successful implementation of the repository system. It will also be an important factor in the site selection process for the system's facilities. Acceptability is related to how well a concept meets society's demands on resource efficiency, limited environmental impact, and pre- and post-closure safety.

Burdens on future generations

Burdens on future generations refers to the need for supervision and monitoring after closure of the repository. The possibility of post-closure retrieval of the waste is assessed, although SKB does not actively entertain this possibility when designing repositories. The possibility of implementing the concept without unnecessary time delay is included in this evaluation factor.

6.3.3 Technology

Personal safety and working environment

Personal safety and working environment includes an assessment of the risk and consequences of an accident that leads to release of radioactivity. The assessment includes the consequences for personnel, visitors and local residents. The evaluation factor also includes assessment of the risk of accidents during construction and operation, emergency conditions in the facilities, how accidents can be prevented, etc. The need for radiation protection and expected dose to personnel and visitors are also considered, as well as the ability to provide a good working environment in the system's facilities.

Feasibility of design and construction

The assessment of feasibility of design and construction includes the maturity of the technology and the simplicity of the design and construction of the facility. The robustness and reliability of the design during the operating period (in achieving the initial state) is assessed. The feasibility of quality control and maintenance and the ability to correct deficiencies in safety or impact on the initial conditions for the repository are considered. The feasibility of physical protection and nuclear safeguards is also included.

Feasibility of technology and method of operation

The assessment of the feasibility of the technology and method of operation includes the maturity of the technology and the feasibility of the method of operation. Reliability, the ability to rectify problems during operation, and robustness for unforeseen events during operation are considered. The assessment also includes the ability to carry out quality control of deposition, conditioning, etc and to correct deficiencies in safety or impact on the initial conditions.

Flexibility

Assessment of the evaluation factor "flexibility" is concerned with flexibility in design and method of operation. This includes the ability to adapt to changing volumes, waste types and packaging, and adaptation to waste that is physically, chemically or radiologically difficult to handle is also considered. The ability to adapt to different geological environments as well as the availability of different modes of transport is included.

6.3.4 Cost and time

Cost

Assessment of the cost of the project includes investment, operation and maintenance (including decommissioning and closure) as well as site selection and research and development. Economic risks are also included.

Time

This evaluation factor concerns an assessment of the probability that the programme can be implemented in accordance with the timetable. The risk of delays is also included in the assessment.

7 Repository concepts

This chapter presents the repository concepts that have been identified by the project as possible solutions for disposal of long-lived low and intermediate level waste. The workflow for idea generation based on the hierarchic levels of *strategy*, *concept*, and *safety function* is presented in Chapter 3. No concepts have been left out or censored in the creative process, so all identified concepts are accounted for in this chapter. Due to the nature of the creative process, concepts that have been found unsuitable in other evaluations have also been included. This means that concepts that have, for example, been deemed less suitable for the disposal of spent nuclear fuel in Sweden can also be presented as an option for disposal of long-lived low and intermediate level waste.

The chapter is divided into two sections. Repository concepts that have been found by an initial screening to violate Swedish law or international agreements, to be unrealistic or to be inappropriate for the problem at hand are described in brief in section 7.1. Reasons for the rejection of each concept are also presented. The remaining concepts are presented in section 7.2. The concept descriptions have sufficient detail to permit comparison and evaluation. These concept descriptions, together with additional information from the literature and investigations undertaken within the frame of this study, constitute the information base that is evaluated in Chapter 8.

7.1 Repository concepts rejected during initial screening

This section presents a brief description of the concepts which were rejected on legal grounds, or on the grounds of being unrealistic or inappropriate for the problem at hand, as well as a brief explanation of why they were rejected.

7.1.1 Elimination by partitioning and transmutation

Primary safety function

Limited inventory.

Method

The radionuclides are separated from the non-radioactive material in the waste. The radioactive material is transformed into stable isotopes and can then be released for unrestricted use together with the non-radioactive material. Secondary waste is conditioned and placed in a repository. A presentation of the current status in this field is found in Blomgren (2010).

Prerequisites

The radionuclides must be able to be separated from the inactive components of the waste in an efficient and environmentally sound process.

Judgement

Nuclear transmutation is a technique used to convert long-lived radionuclides into more short-lived or stable nuclides by nuclear-physical processes. Several processes have been studied for processing spent nuclear fuel to reduce the concentrations of long-lived nuclides. In practice, the only process that has been used thus far for transmutation on a large scale is irradiation with neutrons. Neutrons can split the nuclei in transuranic elements present in the spent fuel – e.g. neptunium, plutonium, and americium – so that they are transformed into other nuclides. However, the long-lived nuclides in the SFL waste are mainly activation products present in the structural components of the reactor, such as Cl-36 and Mo-93 in core components made of stainless steel.

Prior to transmutation by neutron-irradiation, the nuclides to be transmuted must be separated from other nuclides in the waste in order to avoid the formation of new activation products by the neutron-irradiation process. However, the concentrations of the long-lived activation products present in stainless steel components in particular are very low. It is not practically possible to extract the long-lived radionuclides in an efficient way.

Decision

The concept is rejected.

7.1.2 Sea dumping

Primary safety function

Dilution.

Method

Dumping of the waste in large rivers, large lakes or oceans where strong currents will ensure that the radionuclides leaching from the waste are diluted in large volumes of water.

Prerequisites

That sea dumping once again becomes an accepted method and is permitted under international laws and agreements.

Judgement

Deliberate disposal into the sea of wastes is a violation of international agreements to prevent pollution of the sea that have been ratified by Sweden (IMO 1974, 1996). This concept has no chance of passing an evaluation.

Decision

The concept is rejected.

7.1.3 Sub-seabed disposal

Primary safety function

Containment.

Method

Deposition of sealed containers in deep sea sediments at great depth. Waste canisters can be deposited in the sediments by different methods, e.g. using free-falling or guided penetrators that are launched from a ship.

Prerequisites

That sea dumping once again becomes an accepted method and is permitted under international laws and agreements.

Judgement

Disposal of wastes in the seabed or the subsoil is a violation of international agreements to prevent pollution of the sea that have been ratified by Sweden (IMO 1974, 1996). This concept has no chance of passing an evaluation.

Decision

The concept is rejected.

7.1.4 Disposal beneath large glaciers

Primary safety function

Dilution.

Method

Deposition of the waste beneath a large glacier where a significant water flow exists. The radionuclides leached from the waste are transported by the water flow and diluted on reaching the ocean.

Prerequisites

The required type of glacier does not exist in Sweden and this concept therefore requires a permit to export radioactive waste to other countries or continents with the required type of glaciers.

Judgement

There is no Swedish law prohibiting exports of radioactive waste in general, although exports to Antarctica is prohibited (SÖ 1984:5). However, exports of radioactive waste violate the notion of "national responsibility" that has been a guiding principle in Swedish nuclear policy for the past 25 years. The principle can be expressed as: Every country that has chosen to generate nuclear energy should itself take full responsibility for the residual radioactive material that arises within the country as a result of that process. This principle constitutes the moral ground for the Swedish prohibition against disposal of e.g. spent nuclear fuel from other countries in Sweden, as expressed in the Nuclear Activities Act (SFS 1984:3). The principle was first worded by the Swedish parliament in 1985 (NU1984/85:30) and has since gained recognition and widespread acceptance. The principle is expressed in the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management (IAEA 1997, SÖ 1999:60). It is also one of the general principles of the Council Directive 2011/70/EURATOM of 19 July 2011 establishing a Community framework for the responsible and safe management of spent fuel and radioactive waste (EU 2011). A prohibition of exports of radioactive waste has been proposed by the Swedish Radiation Safety Authority to meet the requirements of the Council Directive (SSM 2012).

Decision

The concept is rejected.

7.1.5 Containment beneath glaciers

Primary safety function

Containment.

Method

Deposition of encapsulated waste beneath a large glacier with permafrost conditions underneath. The waste in the canisters will remain largely unaffected over time, since there is no flow of water in the frozen ground.

Prerequisites

The concept requires a glacier that remains unaffected for the lifetime of the repository, i.e. hundreds of thousands of years.

Judgement

The evolution of ice sheets and permafrost varies with the climate. Climate change may occur over long and short timescales due to a variety of factors. It is not possible to show that a particular glacier will endure for a sufficiently long time. Furthermore, the discussion regarding export restrictions in section 7.1.4 is valid also for this concept. The concept thus has no chance of passing an evaluation.

Decision

The concept is rejected.

7.1.6 Launching into space

Primary safety function

Isolation.

Method

The waste is launched into space towards the sun or placed in an orbit around another planet by means of a rocket or a cannon.

Prerequisites

The method requires development of a powerful enough cannon or a cheap and efficient rocket launching system. It also requires that these systems be made safe and reliable enough to deliver the waste into the desired orbit and avoid accidents during launching.

Judgement

The concept was studied for disposal of high level waste in USA in the 1970s and 1980s (Rice et al. 1980). The concept was also included in the system analysis for disposal of the Swedish spent nuclear fuel undertaken by SKB (SKB 2000a).

The method requires the rocket or missile to enter the atmosphere or space over the territory of other nations. The method can thus not be strictly said to meet the requirement of disposal within the country, as discussed in section 7.1.4. Another inherent feature of the concept is the use of fuel to propel the rocket. The studies undertaken show very high energy consumption, since launching of large quantities of materials into space will require very large quantities of rocket fuel. Cost and the environmental impact are thus deemed high for this concept. The US studies in the 1970s and 1980s concluded that the concept could only be feasible if some sort of reprocessing of the spent nuclear fuel was included in the system, to reduce the volumes. Given the resources required, the strategy is judged to be unrealistic.

Radiological safety was assessed as part of the US evaluation of the concept and compared with the safety of geological disposal (Rice et al. 1982). The conclusion of the comparison was that the risks associated with launching into space are too high to justify the method.

Decision

The concept is rejected.

7.1.7 Monitored repository / surface repository

Primary safety function

Containment.

Method

To maintain the systems available today for storage, e.g. the central storage facility for spent nuclear fuel (Clab) and the BFA rock cavern at the Oskarshamn nuclear power plant site. All new long-lived low and intermediate level waste will be conditioned to fit the existing structure for the waste. The time for which the waste will be stored will vary depending on its radioactivity.

Prerequisites

Monitoring and supervision of the storage facility will be necessary for very long times.

Judgement

One of the requirements on the repository identified as part of this study is: The repository shall be designed so that no monitoring is required after closure. The requirement has been deduced from SÖ 1999:60, article 11 vii stating that appropriate steps shall be taken to avoid imposing undue burdens on future generations. The concept of a monitored geological repository or surface repository to dispose of long-lived waste does not meet this requirement.

The concept is only suitable for waste that can be released from regulatory control within a short period of time after the closure of the repository such as short-lived very low level waste.

Decision

The concept is rejected.

7.1.8 Deposition on a large number of separate sites

Primary safety function

Limited inventory.

Method

The waste is deposited on a large number of sites, only a very small amount of waste on each site. The layout of the repository at the different sites and the types of barriers in place (if any) may vary depending on the specific properties of the respective site. However, it is assumed that each site hosts a borehole disposal facility, which is operated directly from the ground surface as described in IAEA (2009b).

The borehole disposal facility concept was initially developed for the disposal of small amounts of waste with a fairly short half-life, such as disused radioactive sources, in countries where a deep geological repository is not available (IAEA 2009b). Typically, the diameters of the boreholes vary from a few tenths of a metre to a few metres, with a depth of about 15 metres or more. Owing to the small diameter of the boreholes, this concept is not suitable for large containers or components.

Prerequisites

Requires local acceptance at a large number of sites for construction of repositories.

Judgement

This concept will require a large number of sites to distribute the waste sufficiently to ensure long-term safety. According to IAEA (2009b, annex) a 30 metres deep borehole – appropriately sited and constructed according to best practice – has the potential to be used for disposal of waste holding around 1E+12 Bq of activity (both short-lived and long-lived nuclides). The total activity content of the long-lived low and intermediate level waste planned for disposal in SFL is estimated to be 2.1E+17 in 2075 (see Chapter 2). This means that around 200,000 boreholes would be needed to dispose of the SFL inventory. If a distance of 1 km between boreholes is assumed, the 200,000 boreholes will be distributed over almost half of Sweden.

The concept may be modified by increasing the disposal depth, which would probably reduce the number of boreholes needed. On the other hand, the SFL waste probably has a relatively higher content of long-lived nuclides compared to the case presented in IAEA (2009b, annex), which mainly involves disused radioactive sources. This fact will likely require that the waste be deposited at a greater depth anyway, to reduce the effects associated with the long time span required for disposal of long-lived waste. Even with a reduction, the total number of boreholes will be sizable.

The principle of local acceptance is important for the selection of both method and site. It is deemed unlikely that this method could gain acceptance from the general public.

Decision

The concept is rejected.

7.2 Repository concept descriptions

The concepts that passed the first screening are presented in this section. These remaining concepts are all based on the strategy of geological disposal. To allow comparison and evaluation, the concept descriptions have been developed to a certain level of detail. Given the conceptual level of the study, measures and dimensions should be treated as approximate and are subject to change in the future. The purpose is to facilitate the evaluation by permitting reasonable estimates of parameters related to long-term safety, technology, environment, and cost. The concept descriptions presented here, together with other technical documents, form the basis for the evaluation of the concepts in Chapter 8.

Methods for plugging and sealing of the repositories are described only briefly in this section. The reasons for this are twofold:

- In the case of concepts based on the construction of a set of engineered barriers in a rock cavern such as the Clay repository or the Concrete repository, it is assumed that the methods for sealing and closure will be very similar and therefore cannot be used to differentiate between different concepts. Some conceptual ideas regarding the design of plugs and backfill as well as sealing of the entire repository are outlined in Grahm et al. (2013, Chapter 12).
- For other more specific concepts such as the Deep borehole concept or WP-Cave, sealing and closure are almost inherent parts of the engineered barrier system and differ widely between the concepts. It is assumed in the evaluation that the methods used for sealing and closure in the different concepts will be commensurate with the respective concepts. Details on sealing and closure in these concepts are found in the references given in each section.

7.2.1 Clay repository

Overview

The Clay repository is a geological disposal concept aimed at limiting the flow of groundwater through the waste. This is achieved by the use of large amounts of a material with very low hydraulic conductivity, such as bentonite. The bentonite barrier makes diffusion the predominant transport process for radionuclides.

The Clay repository is situated in a rock cavern 300–500 metres below ground level in crystalline rock. A concrete structure is built in the rock cavern to host the waste containers. The purpose of the concrete structure is to provide radiation protection during the operating phase and to facilitate grouting of the waste containers. The concrete structure is built in a way that facilitates the backfill of bentonite also beneath the waste containers, and is not credited with a long-term barrier function. At closure, the free space between the concrete structure and the rock is backfilled with bentonite, see Figure 7-1.

Characteristics of the Clay repository

A cross-section view of the Clay repository is shown in Figure 7-1. The length of the repository has not yet been determined but it can be adjusted to accommodate the required disposal volume.

The Clay repository concept has the following characteristics:

- The structure hosting the waste containers is made of concrete with a thickness of about 0.5 metre and supported on granite pillars.
- Optionally, the concrete structure can be divided into separate compartments as in the existing 1BMA in SFR (see Figure 7-3 in section 7.2.2).
- Two metres of high-density bentonite blocks are placed beneath the concrete structure during construction or as backfill (a dry density of 1,600–1,700 kg/m^3 is assumed as a starting point).
- Two metres of high-density bentonite blocks are placed as backfill between the concrete structure and the rock walls.
- A 3–4 metres thick layer of high-density bentonite blocks is placed as backfill on top of the concrete structure.
- The dome-shaped part between the bentonite layer and the ceiling is filled with low-density bentonite pellets as backfill (a dry density of ~1,000 kg/m^3 is assumed as a starting point).

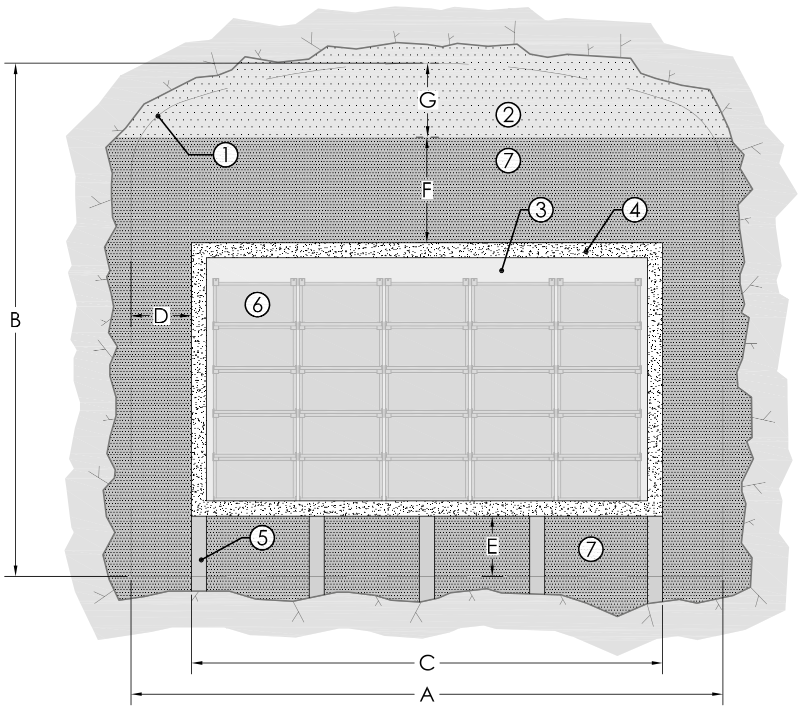


Figure 7-1. *Cross-sectional view of the Clay repository concept. Legend: 1) Theoretical tunnel contour. 2) Bentonite pellets. 3) Grout. 4) Concrete structure for the operating period (0.5 metre). 5) Granite pillars. 6) Waste containers. 7) Bentonite blocks. Approximate dimensions: A = 20 m, B = 17 m, C = 16 m, D = 2 m, E = 2 m, F = 3–4 m, G = 2–3 m.*

Safety functions

The primary safety function of this concept is ***retardation***.

This is achieved by the use of large amounts of a material with low hydraulic conductivity such as bentonite, which restricts the flow of water through the repository. The bentonite also provides high sorption capacity for many of the radionuclides present in the waste and has the ability to filter colloids.

The safety functions of the engineered barrier are:

- Low hydraulic conductivity in the bentonite.
- Low diffusivity in the bentonite.
- Strong sorption of radionuclides in the bentonite.
- Filtering of colloids in the bentonite.

In addition to this, the natural barrier – the bedrock – will contribute to the primary safety function retardation.

Operation and closure

During the operating period, the waste is emplaced in the concrete structure by means of an overhead crane and grouted if required. However, grouting can also be done at any other time prior to sealing and closure if that is found more suitable.

At closure, a concrete lid is placed on the concrete structure and the entire rock cavern is filled with bentonite. It is suggested that the top part of the cavern be filled with bentonite pellets and that bentonite blocks be placed beneath and at the sides of the concrete structure. The thickness of the clay barrier is a minimum of 2 metres.

Plugs are installed between the rock cavern and the adjacent areas.

Waste types suitable for this concept

The Clay repository is intended to handle all present and future long-lived low and intermediate level waste. However, it might be less suitable for iron and steel since the pH in the repository is likely to be roughly neutral, which is too low to guarantee that iron and steel will be in a passive state. The corrosion rate will be relatively high, resulting in a high release rate for radionuclides.

7.2.2 Concrete repository

Overview

The Concrete repository is a geological disposal concept aimed at limiting the flow of groundwater through the waste and limiting the diffusive transport of species to and from the waste. The main barrier in the concept is an extensive use of concrete, thus producing a matrix with low hydraulic conductivity and diffusion as the predominant transport process. The concrete also provides an alkaline environment in which the corrosion rate of the metals in the repository will be low. The repository is located in crystalline rock at a depth of 300–500 metres.

Two alternatives are possible for the Concrete repository: The full Concrete repository and the Concrete repository with a hydraulic cage.

The full Concrete repository involves the construction of a solid concrete monolith, see Figure 7-2 (top). This is achieved by waste conditioned in concrete, repository structures made of concrete, and the use of concrete for grouting and backfill. The concrete backfill fills the entire volume between the waste and the rock wall.

In the Concrete repository with a hydraulic cage, only the inner part of the concrete structure is filled with concrete, while the space between the concrete structure and the rock wall is filled with gravel, see Figure 7-2 (bottom).

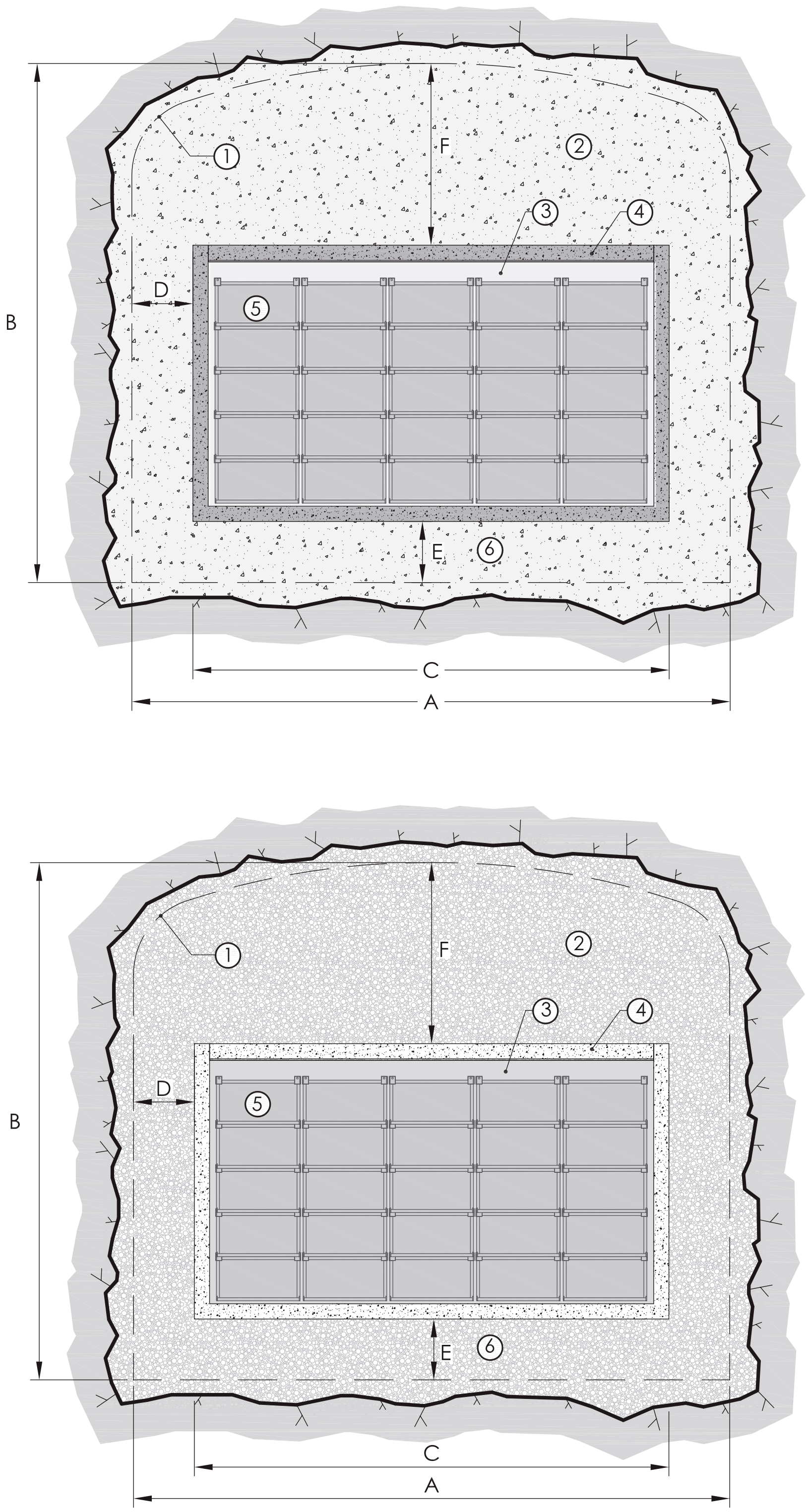


Figure 7-2. Cross-sectional views of the two alternative designs of the Concrete repository – the full Concrete repository (top) and the Concrete repository with a hydraulic cage (bottom). Legend: 1) Theoretical tunnel contour. 2) Concrete (top) or gravel (bottom). 3) Grout. 4) Reinforced concrete (0.5 m). 5) Waste containers. 6) Concrete (top) or gravel (bottom). Approximate dimensions: A = 20 m, B = 17 m, C = 16 m, D = 2 m, E = 2 m, F = 5–10 m.

Characteristics of the Concrete repository

A cross-sectional view of the Concrete repository is shown in Figure 7-2. The length of the repository has not yet been determined but it can be adjusted to accommodate the required disposal volume.

The Concrete repository concept has the following characteristics:

- The repository will be divided into separate compartments as in the existing 1BMA in SFR, see Figure 7-3. The size of each compartment will be approximately 10 metres × 15 metres. The structure is made of concrete with a thickness of about 0.5 metre.
- The space between the waste containers in the compartments will be filled with gas-permeable grout with a thickness of about 0.1 metre.
- The space between the concrete structure and the rock wall will be filled with concrete (the full Concrete repository) or gravel (the Concrete repository with a hydraulic cage) with a thickness of about 2 metres.
- The space above the concrete structure will be filled with concrete (the full Concrete repository) or gravel (the Concrete repository with a hydraulic cage) with a thickness of 5–10 metres.

Safety functions

The primary safety function of this concept is ***retardation***.

This is achieved by limiting the groundwater flow through the waste, making diffusion the predominant transport process for radionuclides. Furthermore, diffusive transport between the waste and the surrounding rock is limited by the use of a material in which the diffusive transport of radionuclides is very low, such as concrete. Also, the use of concrete will create an alkaline environment, resulting in low corrosion rates for steel. This will result in a limited release rate of activation products bound in metal, since most of the metallic waste will corrode only slowly.

The safety functions of the engineered barrier are:

- Low hydraulic conductivity in the concrete.
- Low diffusivity in the concrete.
- Strong sorption of radionuclides in the concrete.
- High-pH conditions provided by the concrete.

The natural barrier – the bedrock – will in addition to this contribute to the primary safety function retardation.

Figure 7-3. *The rock cavern for short-lived intermediate level waste, 1BMA in SFR.*

The Concrete repository with a hydraulic cage has an additional safety function:

- High hydraulic conductivity in the hydraulic cage formed by the gravel around the concrete structure.

The hydraulic cage is intended to limit the groundwater flow through the concrete structure, and to provide diffusive conditions for the transport of radionuclides through the concrete structure.

Operation and closure

During the operating period the waste is placed in the concrete structure by means of an overhead crane and grouted if required. However, grouting can also be done at closure.

At closure, the waste is grouted and a concrete lid is placed on the concrete structure, after which the entire repository is filled with concrete (or gravel). Plugs are installed between the repository rock cavern and the adjacent areas.

However, the design of the plug and backfill and sealing of the other areas are not covered in this report. Instead, please refer to Grahm et al. (2013, Chapter 12).

Waste types suitable for this concept

The Concrete repository is intended to handle all present and future long-lived low and intermediate level waste. The high pH of the concrete pore water provides passive conditions for iron and steel, which makes the Concrete repository especially well suited for metallic waste such as core components. The corrosion rate when alkaline anoxic conditions prevail is low, which means the release rate of radionuclides will also be low.

7.2.3 Deep boreholes

Overview

The Deep boreholes concept is a geological disposal concept aimed at isolating the waste from the biosphere. The concept is based on the assumption that groundwater conditions at great depths are stagnant. The reason for the stagnant conditions is that the groundwater has high salinity (and thereby also high density) and therefore tends not to mix with the lighter fresh water above. Groundwater movements that do occur at great depth are not believed to have any contact with the ground surface. This means that radionuclides from the deposited nuclear waste cannot be carried up to the surface by the groundwater.

The waste is placed in canisters surrounded by a bentonite buffer in crystalline rock at a depth between 3,000 and 5,000 metres below the ground surface, see Figure 7-4. The flow of groundwater through the waste and diffusive transport of species from the waste are initially limited by engineered barriers consisting of the steel canister, the bentonite buffer, and the concrete matrix. However, the rock is the main barrier for isolating the waste and preventing radionuclides from spreading to the biosphere in the long term. Due to the conditions at depth (high salinity and high temperature), together with the somewhat complicated deposition method and limited opportunities to inspect the deposited waste, the engineered barriers are expected to be of limited importance and the waste is assumed to be in contact with the groundwater immediately or shortly after deposition. The bedrock (along with the stagnant groundwater conditions) is considered to be the only safety feature of the concept that can guarantee long-term safety.

Characteristics of the Deep boreholes concept

The characteristics of the Deep boreholes concept differ markedly from those of many of the other geological concepts. The most obvious differences are of course those related to the shape of the repository, but also the fact that the main barrier function is based on the fact that the waste is deposited in an isolated hydraulic regime at a very great depth. Detailed information about the Deep boreholes concept can be found in SKB (1992, 2000b), Grundfelt and Wiborgh (2006), Grundfelt (2010, 2013), Marsic and Grundfelt (2013a, b) and Odén (2013). Furthermore, Sandia, one of the

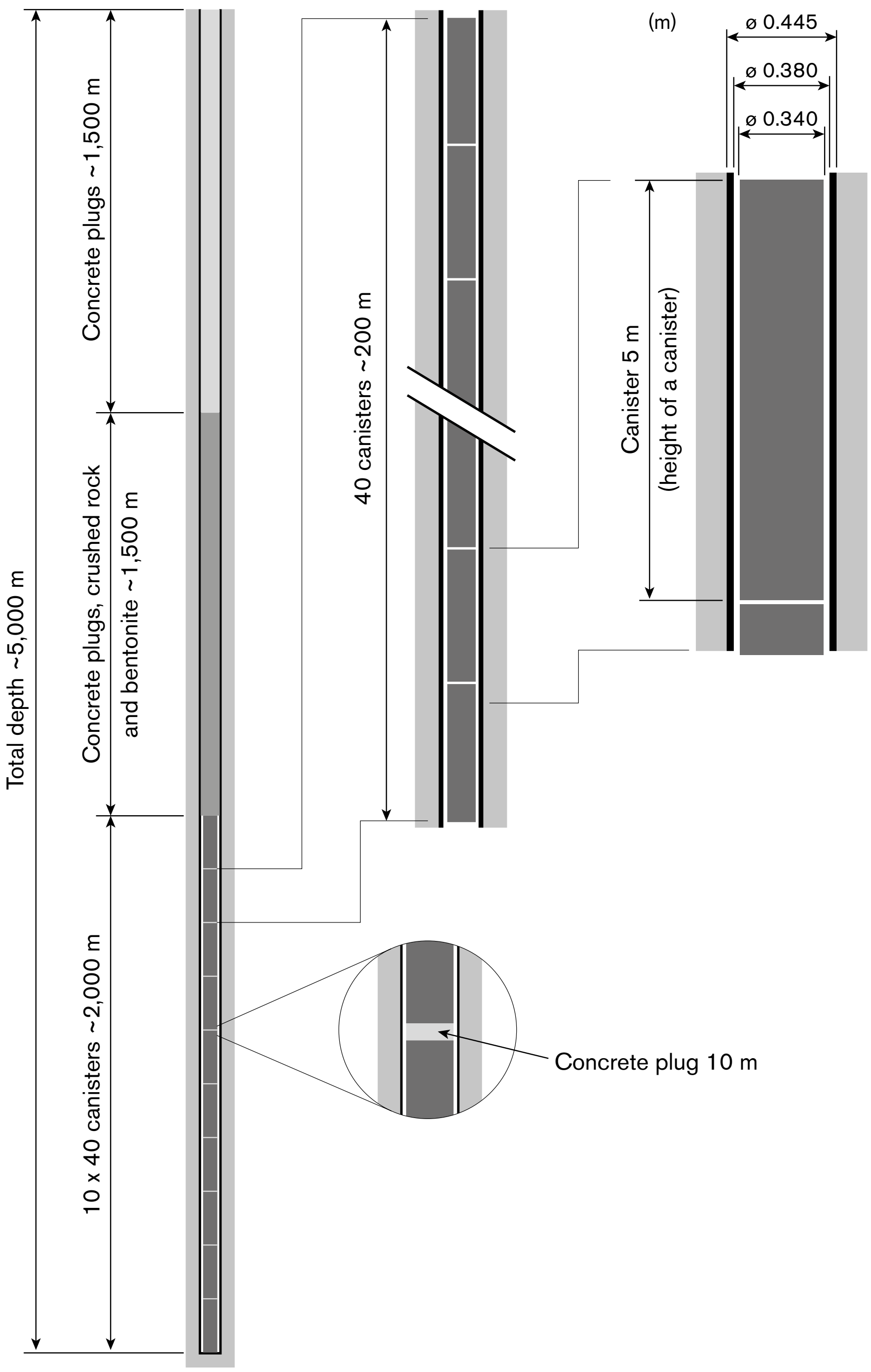


Figure 7-4. *Schematic illustration of the Deep boreholes concept, based on Arnold et al. (2011).*

US Department of Energy's energy laboratories, has published reports on disposal of radioactive waste in deep boreholes (Brady et al. 2009, Arnold et al. 2011).

The Deep boreholes concept for long-lived low and intermediate level waste is presumed to have the following characteristics:

- The entire Deep borehole repository consists of several 5,000-metre-deep boreholes with a diameter in the deposition zone of 0.445 metre.
- The waste is deposited at a depth of about 3,000–5,000 metres.
- The inside diameter of the casing is 0.380 metre.
- The outside diameter of the canisters is 0.340 metre.

- The height of the canister is 5 metres.
- The canisters are made of carbon steel with a thickness of 12 mm.
- Each borehole can hold approximately 150 m^3 of waste provided that the inside diameter of the canister is 0.316 metre.
- At least 100 deep boreholes will be required to host all long-lived low and intermediate level waste.
- Bentonite is used as buffer material.
- The waste is grouted in the canister.

Safety functions

The primary safety function of this concept is ***isolation***.

This is mainly achieved by the isolated hydraulic regime at the depth of disposal, where the bedrock and the highly saline groundwater are expected to provide stagnant groundwater conditions.

The safety functions of the concept are:

- Containment by the metal canister.
- Limited advective transport, due to stagnant groundwater.

Operation and closure

The boreholes are drilled from surface using stabilizing mud and stabilizing steel casing. The mud stabilizes the hole until the casing is in place.

Arnold et al. (2011) propose that the canisters should be put on top of each other and fastened together in a long string of 40 canisters (with a length of 200 metres) before being emplaced in the borehole. A plug can be installed approximately 0.5 metre above the top of the string of canisters after it has reached the bottom of the borehole. A 10-metre-long concrete plug is then placed in the borehole in order to form a stabile ground for the next string of canisters. This procedure is repeated 10 times until the repository area is filled with a total of 400 canisters.

When waste and plugs have been deposited in the deposition zone, a 10-metre-long concrete plug is installed in the borehole adjacent to the upper plug in the deposition zone. After this, the lining is removed and the section between 3,000 and 1,500 metres is sealed with a combination of concrete plugs, crushed rock and bentonite. In the upper part of the borehole, the lining is left in place and the borehole section is finally sealed with a number of concrete plugs.

Waste types suitable for this concept

All types of waste that fit into the disposal canister can be deposited in a deep borehole. However, this will require extensive segmentation of the core components and PWR pressure vessels as well as repacking and/or reconditioning of the waste currently stored in steel tanks or moulds. The intended isolating function of the concept ensures that the characteristics of the waste will have a limited impact on long-term safety.

7.2.4 Drained rock repository

Overview

The Drained rock repository concept is a geological disposal concept aimed at preventing the flow of groundwater through the waste. This is achieved by locating the repository above the groundwater table in a naturally drained rock domain. The concept was originally developed for long-term dry storage of spent nuclear fuel (Eggert et al. 1993). The concept has been adapted in this study to serve the purpose of a repository for long-lived low and intermediate level waste.

The concept consists of a rock cavern located above the groundwater table and with a minimum overburden of 200 metres of bedrock. The distance to the surface of the rock should be great enough to allow rain water to drain in the rock outside the repository cavern. Grout is injected into the bedrock in order to completely seal the bedrock. The cavern floor, which is inclined to permit water drainage by gravity, is covered with a layer of gravel. The space between the waste containers and the space between containers and cavern walls is also filled by gravel, creating a hydraulic cage. If needed, the gap between waste containers is filled with concrete with low hydraulic conductivity.

At closure the rock cavern is backfilled with a high-permeable material, such as coarse gravel.

Characteristics of the Drained rock repository

The Drained rock repository concept differs from many of the other geological concepts, since it is assumed to be located in rock above the groundwater table and since the repository will stay dry during the entire operating life of the repository. More information about the Drained rock repository can be found in Eggert et al. (1993), and SKB (2000a) and references therein.

The Drained rock repository for long-lived low and intermediate level waste is presumed to have the following characteristics (see also Figure 7-5):

- The repository is located above the groundwater table.
- The repository is covered with at least 200 metres of rock overburden.
- A number of deposition tunnels or rock caverns are excavated in the bedrock.
- Grout is injected into the bedrock surrounding the excavated zones in order to completely seal the bedrock.
- At closure the deposition tunnels and rock caverns are filled with gravel or crushed rock with a thickness of 2–5 metres.
- Optional engineered barriers can be installed in the excavated deposition tunnels and rock caverns in order to further improve the barrier system. For example, the deposition tunnels and rock caverns can be filled with concrete.

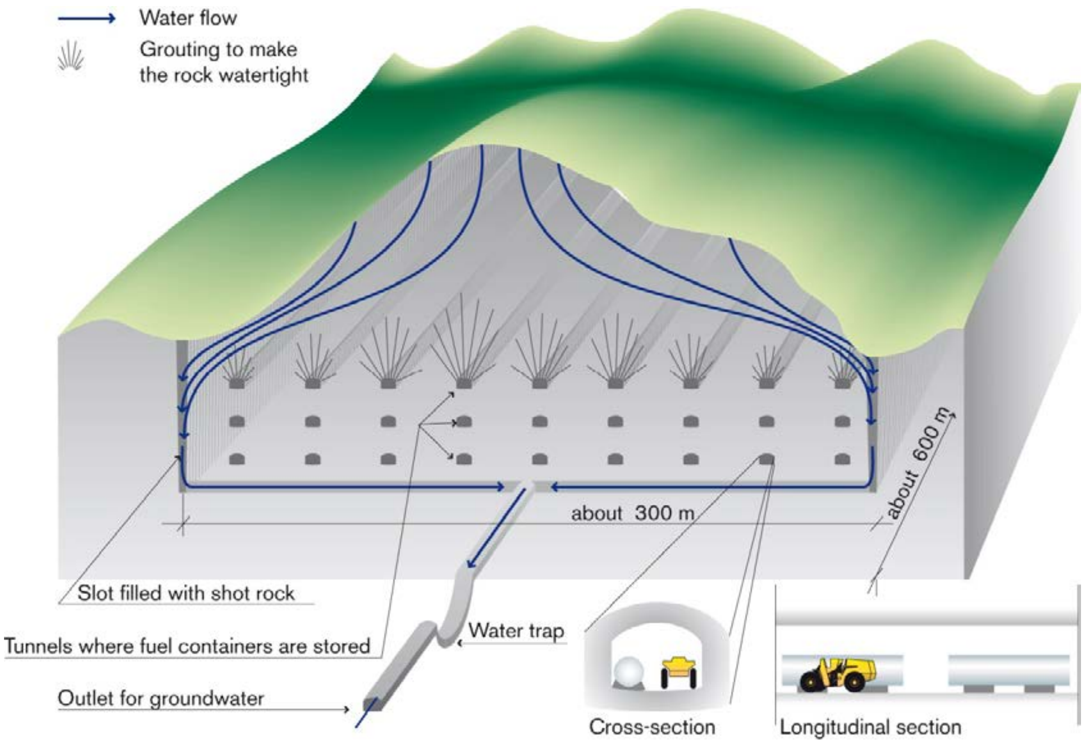


Figure 7-5. *Illustration of the Drained rock repository concept.*

Safety functions

The primary safety function of this concept is ***retardation***.

The fundamental purpose of the concept is to ensure dry conditions over the entire operating lifetime of the repository. This is achieved by locating the repository above the groundwater table and by surrounding the repository with a hydraulic cage consisting of a thick layer of gravel that diverts drainage water past the waste.

The safety functions of the concept are:

- Limited advective transport by the choice of a suitable location and injection of grout into the bedrock.
- High permeability in the hydraulic cage formed by the gravel around the waste.

Operation and closure

During the operating period, the waste is emplaced in the repository by means of an overhead crane or a forklift and grouted if required. However, grouting can also be done at the time of closure.

At closure, deposition tunnels and rock caverns are filled with gravel, crushed rock or large boulders.

Waste types suitable for this concept

The Drained rock repository is intended to handle all present and future long-lived low and intermediate level waste. The intended dry conditions make the concept rather insensitive to waste characteristics, since the dissolution and transport properties of the waste are of limited importance.

7.2.5 Gravel repository

Overview

The Gravel repository is a geological disposal concept with a single engineered barrier. The purpose of the engineered barrier is to direct the flow of groundwater *around* rather than *through* the waste containers. This is achieved by surrounding the waste with a material with high hydraulic conductivity such as gravel.

In the repository rock cavern, which is placed in crystalline rock at a depth of 300–500 metres, a base slab consisting of reinforced concrete is cast on a bed of gravel and/or crushed rock. Optional concrete walls can be built to provide radiation protection during the operating period of the repository.

The ceiling and walls in the cavern are lined with shotcrete. At closure, the free space between the waste containers and the rock is backfilled with a material with high hydraulic conductivity, such as gravel, to create a hydraulic cage around the waste.

Characteristics of the Gravel repository

A cross-sectional view of the Gravel repository is shown in Figure 7-6. The length of the rock cavern has not yet been determined and will be adjusted in order to fit the required disposal volume.

The Gravel repository will have the following characteristics:

- The repository is located in the bedrock at depth of 300–500 metres.
- The space between the waste containers and the bedrock on all sides is filled with a material with high hydraulic conductivity (10^{-5} metre per second or greater) such as gravel.
- The thickness of the layer of hydraulic material is between 2 and 10 metres.
- Optionally, a concrete structure can be erected in the repository to provide radiation shielding during the operating period of the repository.
- The space between the waste containers in the repository can be filled with gas-permeable grout for greater stability during backfill, sealing and saturation.
- Plugs are installed between the rock cavern and the transport tunnels.

Safety function

The primary safety function of this concept is ***retardation***.

This is achieved by a single engineered barrier with the following safety function:

- High hydraulic conductivity in the hydraulic cage formed by the gravel around the waste.

The hydraulic cage is designed to limit the groundwater flow through the waste containers and thus limit the transport of radionuclides. However, this requires that the hydraulic conductivity of the waste containers is much lower than the hydraulic conductivity of the cage. The integrity of the waste packages needs to be maintained for a sufficient period of time. The conditioning of the waste is therefore crucial in maintaining the safety function of the concept.

The natural barrier – the bedrock – will furthermore contribute to the primary safety function retardation.

Operation and closure

During the operating period the waste is placed on the concrete structure by means of a overhead crane or a fork lift and grouted if required. However, grouting can also be done at any other time prior to sealing and closure if that is found more suitable.

At closure, the entire rock cavern is backfilled with gravel. The rock cavern is then plugged.

Waste types suitable for this concept

The Gravel repository is intended to handle all present and future long-lived low and intermediate level waste. However, it may be less suitable for iron and steel since the pH in the repository is likely to be roughly neutral, which is too low to guarantee that iron and steel will be in a passive state. The corrosion rate will be relatively high, resulting in a high release rate for radionuclides.

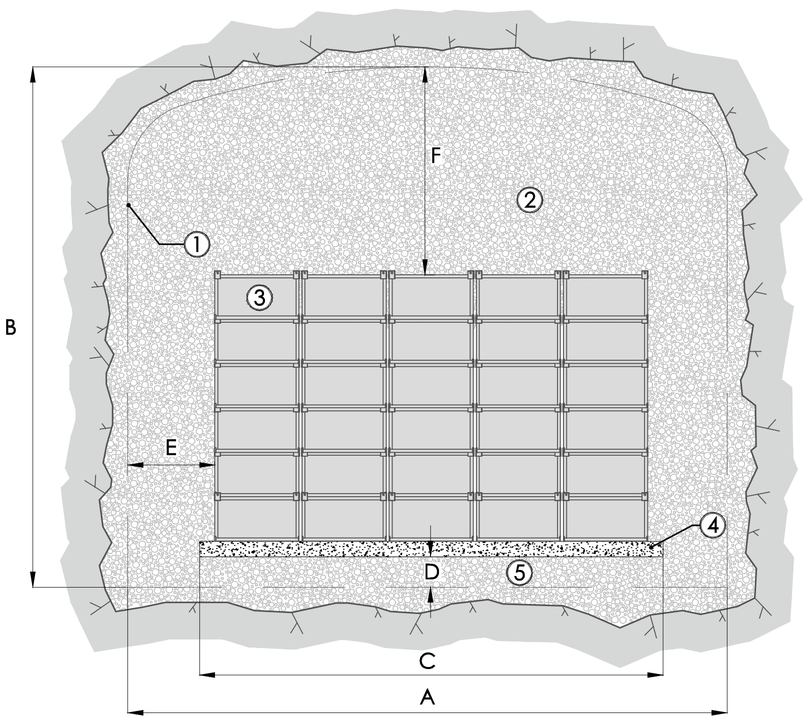


Figure 7-6. *Cross-sectional view of the Gravel repository concept. The waste is placed on a concrete base and the entire repository is then filled with gravel. Legend: 1) Theoretical tunnel contour. 2) Gravel. 3) Waste containers. 4) Concrete base slab. 5) Gravel or crushed rock. Approximate dimensions: A = 20 m, B = 17 m, C = 15 m, D = 1 m, E = 2–3 m, F = 5–10 m.*

7.2.6 KBS-3 for long-lived low and intermediate level waste

Overview

KBS-3 is SKB's method for disposal of spent nuclear fuel. KBS-3 is a geological disposal concept where the waste is placed in copper canisters with a cast-iron insert and deposited in bentonite-lined holes in rock tunnels at a depth of 300–500 metres below surface in crystalline rock, see Figure 7-7 and SKB (2007, 2011). The KBS-3 concept for long-lived low and intermediate level waste presented in this section is an adaptation of the basic concept to the specific characteristics of this particular waste. The KBS-3 concept for long-lived low and intermediate level waste would not need to be located at the same site as the repository for spent nuclear fuel.

Characteristics of the KBS-3 concept for long-lived low and intermediate level waste

The KBS-3 system for long-lived low and intermediate level waste will comprise several components which together form the engineered barrier system:

- A copper canister with or without a cast iron insert. The copper canister is 4.8 metres long with an outside diameter of 1.05 metres and walls with a thickness (not including the cast-iron insert) of 50 mm.
- A bentonite buffer surrounding the copper canister in the deposition hole.
- A tunnel backfill consisting of bentonite.
- Plugs between the deposition tunnels and the transport tunnels.
- Backfill material and plugs in the transport tunnels and ramps.

Safety functions

The primary safety functions of the KBS-3 concept are ***containment*** and ***retardation***.

This is achieved by a combination of barriers with the following safety functions:

- Containment provided by the copper canister's ability to withstand corrosion as well as isostatic and shear loads.
- Low hydraulic conductivity in the bentonite buffer and backfill.
- Low diffusivity in the bentonite buffer and backfill.
- Strong sorption of radionuclides in the bentonite buffer and backfill.

In addition to this, the natural barrier – the bedrock – will contribute to the primary safety function retardation.

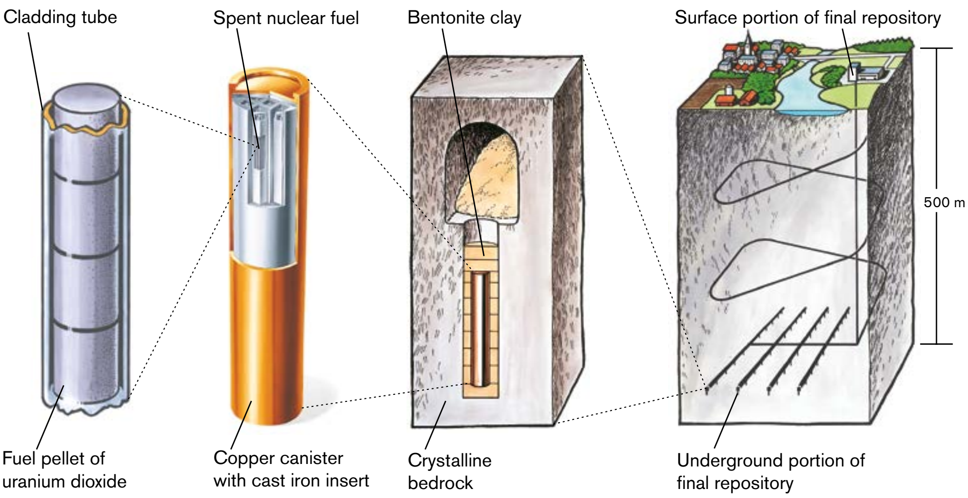


Figure 7-7. *A schematic illustration of the KBS-3 concept for disposal of spent nuclear fuel.*

Operation and closure

Operation and closure of a KBS-3 deposition tunnel comprise the following steps:

- Creation of deposition hole in the tunnel.
- Installation of bentonite buffer in the deposition hole.
- Emplacement of the copper canister.
- Installation of bentonite buffer on top of the copper canister.
- Installation of backfill material (bentonite blocks and pellets) in the deposition tunnel.
- Plugging of the deposition tunnel.

Waste types suitable for this concept

In this work it is assumed that the KBS-3 concept will handle all present and future long-lived low and intermediate level waste.

From a handling point of view, the copper canister design offers both opportunities and restrictions. From a handling point of view, the canister is considered to be suitable primarily for BWR control rods, which without segmentation can be emplaced in a copper canister, see section 5.2.1. Metallic waste can be melted and cast into a desired shape that fits into the copper canister, according to the method described in section 5.1.5. Waste currently stored in grouted drums, moulds or tanks is likely to require reconditioning in order to fit into the copper canister.

7.2.7 Super silo

Overview

The Super silo is a geological disposal concept aimed at limiting the flow of groundwater through the waste and also limiting the diffusive transport of species from the waste. This is achieved by the combination of a hydraulic cage and materials with low hydraulic conductivity and low diffusivity such as concrete and bentonite.

The concept consists of a cylindrical double-wall concrete silo placed in crystalline rock at a depth of 300–500 metres. The space between the concrete walls is filled with bentonite on the side and on top (at closure) and with a mixture of sand and bentonite at the cylinder base. The outer silo is placed on a bed of gravel and the space between the outer silo and the rock is also filled with gravel. At closure, a lid comprising a concrete-bentonite-concrete combination with the same properties as in the silo walls is placed on the silo. The top of the cavern is then backfilled with gravel.

Characteristics of the Super silo

Cross-sectional view of the Super silo is shown in Figure 7-8.

The Super silo will have the following characteristics:

- An inner cylinder with separate shafts for deposition of the waste containers made of concrete, with or without reinforcement, with a thickness of 0.5–1 meter.
- An outer cylinder made of reinforced concrete with a thickness of 0.5–1 metre.
- The space between the concrete cylinders – approximately 1–2 metres wide – is filled with high-density bentonite (a dry density of ~1,500 kg/m^3 is used as a starting point).
- The space between the outer cylinder and the rock wall is filled with a material with high hydraulic conductivity such as gravel or crushed rock.
- The outer concrete cylinder is placed on a bed of gravel or crushed rock with a thickness of about 2 metres.
- The space between the waste containers and walls of the shafts is filled with gas-permeable grout.
- The dome-shaped volume on top of the concrete structures will be filled with gravel at closure of the repository.
- The dimensions of the concrete cylinder will be adjusted according to the properties of the bedrock and the total amount of waste.

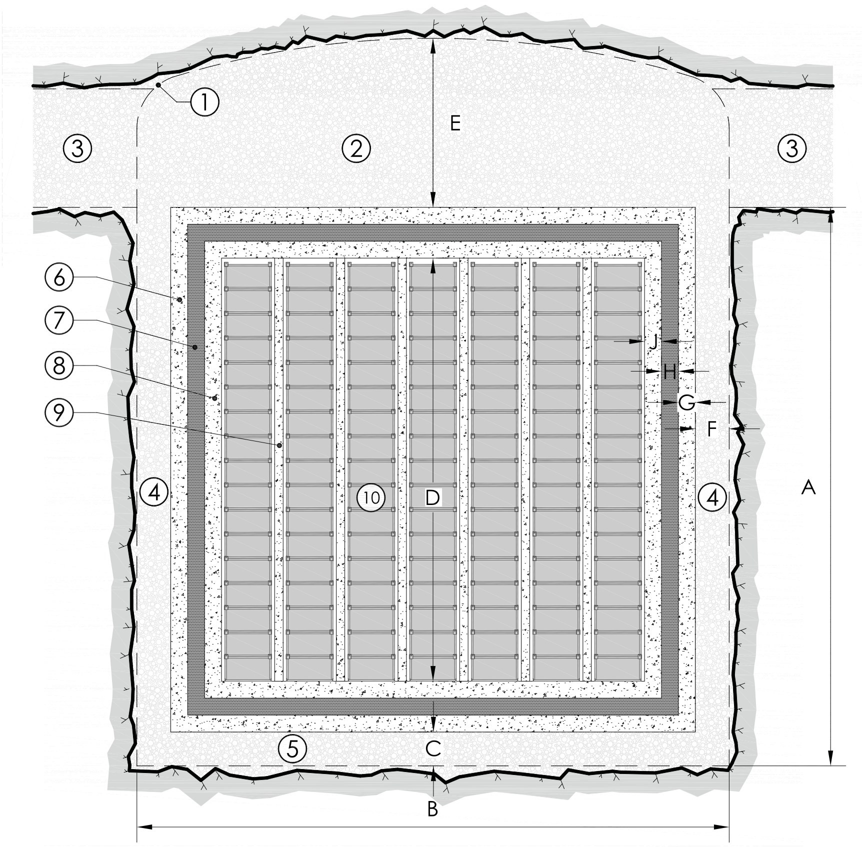


Figure 7-8. *Cross-sectional view of the Super silo concept. Legend: 1) Theoretical rock cavern contour. 2, 3, 4, 5) Gravel or crushed rock. 6) Reinforced concrete. 7) Bentonite blocks. 8) Concrete. 9) Concrete shaft walls (0.5 m). 10) Waste containers. Approximate dimensions: A = 33 m, B = 35 m, C = 2 m, D = 25 m, E = 5–10 m, F = 2 m, G = 1 m, H = 1 m, J = 1 m.*

Safety functions

The primary safety function of this concept is ***retardation***.

This is achieved by a combination of engineered barriers with the following safety functions:

- Low hydraulic conductivity in both the concrete cylinders and the bentonite liner between the concrete cylinders.
- Low diffusivity in both the concrete cylinders and the bentonite liner between the concrete cylinders.
- Strong sorption of radionuclides mainly in the concrete cylinders but also in the bentonite liner between the concrete cylinders.
- High hydraulic conductivity in the hydraulic cage formed by the gravel around the outer concrete cylinder.
- Filtering of colloids in the bentonite liner.

The natural barrier – the bedrock – will furthermore contribute to the primary safety function retardation.

Operation and closure

During the operating period, the waste is emplaced in the concrete structure by means of an overhead crane and grouted.

At closure, a concrete lid is placed on the concrete structure, after which the entire repository is filled with bentonite. Plugs are installed between the repository rock cavern and the adjacent areas.

Types of waste suitable for this concept

The Super silo is intended to handle all present and future long-lived low and intermediate level waste. The silo interior can be adapted to waste containers of different size.

7.2.8 WP-Cave

Overview

WP-Cave is a geological disposal concept aimed at limiting the flow of groundwater through the waste and also limiting the diffusive transport of species from the waste. This is achieved by the combination of a hydraulic cage and a material with low hydraulic conductivity such as bentonite.

WP-Cave is designed as an egg-shaped underground structure for storage and disposal of nuclear waste. A typical WP-Cave for disposal of high level waste canisters has a bentonite barrier with a height of about 300 metres and a diameter of about 100 metres at mid-height, see Figure 7-9. In the existing design, the top part of the cave is located at a depth of about 200 metres, but this can be adjusted (Skagius and Svemar 1989, SKB 1989).

The repository part has a compact layout and is surrounded by two engineered barriers. The innermost barrier is a 5-metre wide shield with low hydraulic conductivity, consisting of pure bentonite clay or a mixture of bentonite clay and sand/crushed gravel. The outer barrier is a hydraulic cage surrounding the shield, which initially drains the repository rock mass and later diverts the groundwater flow past the repository. In this way an initial dry supervision period can be obtained. After closure of the repository and subsequent water saturation, stagnant conditions are established inside the bentonite barrier, preventing the disposed waste from migrating to the geosphere.

The interior part, inside the bentonite barrier, is designed according to the waste packages to be disposed of.

Characteristics of WP-Cave

WP-Cave is an egg-shaped underground structure for storage and disposal of nuclear waste, see Figure 7-9. It has the following characteristics:

- The repository has two engineered barriers.
- The innermost barrier is a 5-metres wide shield with low hydraulic conductivity consisting of pure bentonite clay or a mixture of bentonite clay and sand/crushed gravel.
- The inner barrier is surrounded by a hydraulic cage comprising a large number of holes drilled in a suitable pattern, which functions as an outer barrier.
- The layout of the interior, inside the bentonite barrier, is designed according to the waste packages to be disposed of. For low and intermediate level waste, the interior of the repository can comprise tunnels or a number of small rock caverns, similar to those used in the Concrete repository (presented in section 7.2.2).
- The dimensions of a WP-Cave repository for disposal of low and intermediate level waste can be adjusted to the amount of waste and the properties of the bedrock on the repository site. However, a suggested size of the bentonite barrier is a height of about 300 metres and a diameter of about 100 metres at mid-height.
- A transport shaft is located in the centre of the repository.
- The deposition tunnels are backfilled with gravel and plugged.

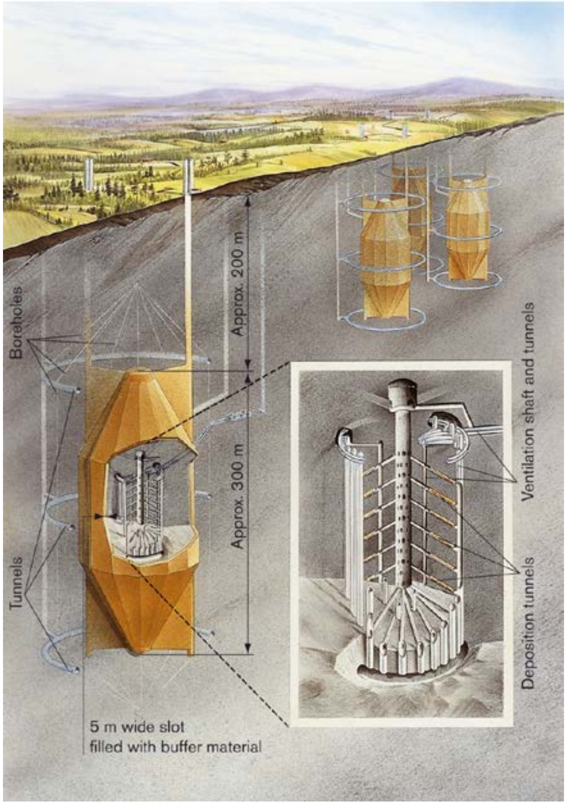


Figure 7-9. *The WP-Cave system for disposal of high-level waste. For disposal of low and intermediate level waste, the interior of the cave can be designed according to the waste packages to be disposed of.*

Safety functions

The primary safety function of this concept is ***retardation***.

This is achieved by the use of two engineered barriers. The barriers have the following safety functions:

- Low hydraulic conductivity in the bentonite barrier.
- Low diffusivity in the bentonite barrier.
- Strong sorption of radionuclides in the bentonite barrier.
- High permeability in the hydraulic cage formed by the drilled borehole structure.

In addition to this, the natural barrier – the bedrock – will contribute to the primary safety function retardation.

Operation and closure

During operation, the waste is lowered to the desired level in the WP-Cave by means of a large elevator in the central shaft. At the desired level in the repository, the waste container is lifted out of the waste transport container by a purpose-built vehicle and placed in its final destination in the repository. The waste transport container then returns to the surface.

When the repository has been filled with waste it is backfilled with crushed rock and finely ground rock in order to create a well-defined chemical environment and also to stabilize the containers in the deposition channels (Skagius and Svemar 1989).

An alternative approach (not considered in this study since it is not part of the original WP-Cave concept) is to fill the entire repository with grout. This would be beneficial from the point of view of reducing the rate of metal corrosion.

Types of waste suitable for this concept

The WP-Cave concept is intended to handle all present and future long-lived low and intermediate level waste.

7.2.9 No action alternative

Overview

The no action concept represents the alternative of *not* building SFL. This in turn entails the use of existing and planned repositories (SFR and the planned repository for spent nuclear fuel) for disposal of all present and future long-lived low and intermediate level waste.

It should be noted that this concept is not identical to the case where SFL is built on the same site as an existing or planned repository.

Characteristics of the no action alternative

SFL is not designed and built. The long-lived low and intermediate level waste is distributed to other repositories. The methods for waste management and disposal, as well as the release mechanisms for radionuclides, are assumed to be the same as for the existing and planned repositories.

Safety functions

The primary safety functions are those of the existing and planned repositories: ***containment*** and ***retardation*** (KBS-3), and ***limited inventory*** and ***retardation*** (SFR).

This is achieved by the use of the barriers in KBS-3 (SKB 2007, 2011) and SFR (SKB 2008).

Operation and closure

The operation and closure of the planned repository for spent nuclear fuel and SFR will be conducted according to the plans for the respective repository.

Types of waste suitable for this concept

The most important prerequisite is that all the waste currently planned for disposal in SFL needs to be suitable for disposal in SFR or the planned repository for spent nuclear fuel. Since the main criteria for disposal in SFR is the absence of significant amounts of long-lived nuclides, and basically all the SFL waste has significant amounts of long-lived nuclides, the planned repository for spent nuclear fuel will have to accommodate most of the SFL waste.

As mentioned in section 7.2.6, the copper canister can be considered to be a suitable container primarily for BWR control rods. The BWR control rods can without segmentation be emplaced in a canister, see section 5.2.1. Metallic waste can be melted and cast into a desired shape that fits into the copper canister, according to the method described in section 5.1.5. Waste currently stored in grouted drums, moulds or tanks is likely to require reconditioning in order to fit into the copper canister.

8 Evaluation of repository concepts

This chapter presents an evaluation of the proposed repository concepts. The evaluation is done in two steps: a primary assessment of the concepts in question, followed by an expanded assessment of the concepts identified as viable repository concepts in the first step. The first step eliminates those concepts whose characteristics do not satisfy the absolute requirements. This step also excludes concepts from further analysis if a qualified judgement of the fundamental characteristics of the concepts deems it difficult or unlikely for the concept to meet the requirements. In step 2, the expanded assessment, the remaining concepts are further developed and compared with each other using the evaluation factors. For each evaluation factor, the characteristics of the remaining concepts are summarized and appraised. The assessment focuses on the differences between the concepts and an evaluation of those differences.

In evaluating concepts, the entire chain from the origin of the waste, conditioning, research and development, construction, operation and disposal should be taken into consideration. In practice, the scope of the evaluation is narrower since the process focus on the *differences* between the proposed concepts. For example, as no site has yet been selected for the future SFL, the requirements on the system for external transportation will not distinguish between the concepts in the evaluation.

8.1 Step 1 – Primary assessment

The primary assessment focuses on the most important characteristics of the concepts. These characteristics are accounted for in the following section.

8.1.1 Clay repository

Overview

The characteristics of the Clay repository are presented in section 7.2.1. The main safety function depends on the use of massive amounts of a material with low hydraulic conductivity, such as bentonite, which would be installed between the waste containers and the bedrock. It was proposed that the repository be located in crystalline bedrock at a depth of 300–500 metres.

Evaluation of long-term safety

The long-term safety of the Clay repository depends on the use of large amounts of bentonite, which restricts the advective flow of groundwater through the repository and consequently the transport rate of radionuclides from the repository. For the repository to be safe over its entire lifetime, the properties of the bentonite must be maintained for a sufficiently long period of time.

The evolution of the properties of bentonite over time is considered to be well understood, owing to the extensive research conducted over the years, especially in connection with the development of the KBS-3 method. However, bentonite is adversely affected by concrete, which is present both in the operational structures in the repository and in the already conditioned waste. The long-term effects of the interaction between concrete and bentonite need further studies.

Evaluation of technological aspects

The most important issue concerns the handling of the bentonite. If bentonite is installed during the construction of the cavern, the bentonite must be protected from interactions with groundwater to avoid premature and uneven swelling. The backfilling process, in which large amounts of bentonite will be installed, requires careful planning already during the design of the cavern. Other aspects of importance concern how to handle the gas that is formed in the repository as a consequence of chemical and physical processes in the waste and containers.

However, these issues are considered possible to solve by available methods and do not pose a threat to the feasibility of building the Clay repository.

Conclusion and decision

The Clay repository exhibits potential as a concept, where the use of massive amounts of bentonite should reduce the flow of groundwater in the vicinity of the waste and thus limit the transport of radionuclides. However, further studies of the long-term safety of the concepts are needed to gain more knowledge of, for example, how gas production in the waste affects the engineered barrier. The technical aspects of designing and building the repository also need further investigation, especially the practical aspects of backfilling and closure.

It has been decided that the concept will be further analyzed and evaluated.

8.1.2 Concrete repository

Overview

The characteristics of the Concrete repository are presented in section 7.2.2. The main safety function relies on the use of massive amounts of a material with a low hydraulic conductivity and high sorption capacity for radionuclides such as concrete, which would be installed between the waste containers and the bedrock. It is proposed that the repository be located in crystalline bedrock at a depth of 300–500 metres.

Evaluation of long-term safety

The long-term safety of the Concrete repository depends on the use of large amounts of concrete, which restricts the advective flow of groundwater through the repository and consequently the transport rate of radionuclides from the repository. Concrete acts as a low-diffusivity medium and also provides high sorption capacity for many radionuclides. The use of concrete creates an alkaline environment which is beneficial for the corrosion rate of steel in particular. This is of special relevance for the core components and reactor internals from the nuclear power plants, which are made of steel and stainless steel.

The ability to maintain the safety functions of the concept over time is related to the degradation rate of the beneficial properties of the concrete. For the assessment of long-term safety, the degradation rate of the beneficial properties needs to be known and the processes well understood. The processes related to chemical and physical degradation of concrete are rather well understood, although the time scale for these processes in the repository environment requires further study.

Evaluation of technological aspects

The design of the Concrete repository has much in common with that of the existing rock cavern 1BMA in SFR. Design and construction of the Concrete repository are therefore not expected to pose any major problems.

The most important concern regarding the concrete barriers – operational structures and backfill – is how to control the formation of cracks in the concrete during construction. Another aspect of importance for the integrity of the concrete is the use of reinforcement bars in the concrete operational structures as well as the use of form ties during the construction work. These issues are all related to the importance of limiting the advective flow of water through the concrete barrier.

The handling of gas formed in the repository as a consequence of chemical and physical processes in the waste and containers needs to be carefully considered. Consideration must also be given to how to limit detrimental chemical reactions with cement components and species present in the groundwater such as chloride, sulphate and carbonate.

Conclusion and decision

The Concrete repository exhibits potential as a concept. The use of concrete should limit the transport of radionuclides by virtue of its low hydraulic conductivity, low diffusivity and beneficial sorption properties. Further studies of the long-term safety of the concept are needed in order to gain more knowledge of e.g. how the properties of the barrier evolve over time. The technical aspects of building the repository also need further investigation, especially the processes of backfilling and closure to achieve a well-defined initial state.

The Concrete repository with a hydraulic cage offers a blend of the safety functions of the full Concrete repository and the Gravel repository, and can thus be regarded as a combination of the two concepts. At this stage, it is probably advisable to explore the potential of individual barriers rather than working on a larger number of barrier combinations.

It has been decided that the full Concrete repository concept will be further analyzed and evaluated. The Concrete repository with a hydraulic cage will not be studied further as a separate concept. The barrier components of concrete and the hydraulic cage will be evaluated within the full Concrete repository and Gravel repository concepts, respectively. The combination of a concrete barrier and a hydraulic cage may yet be considered in the future.

8.1.3 Deep boreholes

Overview

The characteristics of the Deep boreholes concept are presented in section 7.2.3. The long-term safety of the repository depends on the deposition of the waste in a stable groundwater regime at a depth between 3,000 and 5,000 metres below the ground surface. According to the most recent design, the waste is placed in canisters with an inside diameter of 0.316 metre and the diameter of the borehole is 0.445 metre. In the proposed design, the canisters are made of 12 mm thick steel. Very little is known today about the actual geological, hydrogeological and geochemical conditions at the great depths in question. However, it is expected that the groundwater beneath flat landscapes at depths of one thousand metres and below has high salinity and hence high density, resulting in essentially stagnant groundwater conditions.

Evaluation of long-term safety

As mentioned above, the long-term safety of the repository depends primarily on the waste being deposited in a stable groundwater regime that has no contact with the groundwater at shallow depth. For that reason, the most critical issue when it comes to proving the long-term safety of the Deep boreholes concept is an understanding of the interactions of different groundwater regimes, the stability of the groundwater at very great depth and the influence of the borehole itself on the interactions between groundwater in the different regimes.

Due to the great depth, detailed characterization of the rock is very difficult. Characterization of the rock has to be done by measurements in a pilot hole. Knowledge of the surrounding rock volumes can therefore never be as good for the Deep boreholes concept as for a repository built in the uppermost kilometre of the bedrock.

Our limited knowledge of the conditions at the depths in question makes it difficult to evaluate the long-term safety of the Deep boreholes concept, and thus it will be difficult to verify that the concept fulfils the requirement N2-156 *Possible to analyze safety after closure*. It is difficult to predict what the consequences of future glaciations or earthquakes would be for the safety of a Deep boreholes repository.

The aggressive environment at great depth (high salinity, high pressure and high temperature) makes it doubtful that even very sophisticated materials will remain intact in the long term. It is therefore uncertain whether the engineered barriers, the waste container and the buffer are sufficiently stable to ensure isolation and retardation of radionuclides.

Evaluation of technological aspects

With today's technology it is possible to drill deep boreholes down to a depth of at least five kilometres in hard crystalline rock, with a diameter at the bottom of the holes of about 0.445 metre (Odén 2013). However, due to the requirement that the groundwater regimes must not be mixed, it will be essential to prevent mixing during drilling by using drilling slurry with sufficiently high density.

One critical and challenging handling step in the Deep boreholes concept is the emplacement of the waste. Accidents can happen during emplacement in deep boreholes with consequences that cannot be remedied. For example, a canister can get trapped in the hole and break before it has reached disposal depth. As a consequence, a damaged canister may get stuck in a location with flowing groundwater, without being surrounded by a protective buffer.

At present, control of the deposition process in deep boreholes is associated with great uncertainties and challenges. These uncertainties and challenges will make it difficult to verify that the concept fulfils the requirements N1-26 *Possible to maintain, check, test* and N1-49 *Possible to correct*. Another issue of importance is to avoid the intermixing of the different groundwater regimes during emplacement of the canisters.

Another aspect of the Deep boreholes concept is that a very large fraction of the waste requires conditioning in order to fit in deep boreholes. All waste will need extensive segmentation to fit into the relatively small canisters, except for BWR control rods which can be fit into the canisters without segmentation. Containers currently used for storage of the long-lived low and intermediate level waste – steel tanks, drums, moulds, etc – are not adapted for disposal in deep boreholes. This affects the feasibility of the concept to provide an effective solution, and thus fulfilling requirement N1-42 *Cost-effective*.

Progress and developments in the field of deep borehole drilling techniques are brought about by other activities such as petroleum extraction and geothermal energy production. A number of deep boreholes have been drilled around the world, which means that the drilling technology is partly available and that basic estimates can be made of the costs of drilling deep boreholes – see for example SKBdoc 1176359.

Finally, very little is currently known about the actual geological, hydrogeological and geochemical conditions at the great depths in question. This knowledge needs to be improved. Investigation techniques that can be used at these depths are also limited, which may cause problems in site investigations and characterization of the holes. If unfavourable conditions are encountered in a borehole, there is the risk that an entire deposition hole, with its capacity of 400 deposited canisters, cannot be used.

Evaluation of environmental aspects

Each borehole can accommodate approximately 150 m^3 of waste according to the concept description (see section 7.2.3), which means at least 100 deep boreholes will be required to accommodate all long-lived low and intermediate level waste. The drilling will result in approximately 80,000 m^3 of rock mud. Unlike boulders and gravel, rock mud is not suitable for reuse and has to be disposed of in a safe manner.

Evaluation of aspects related to cost and time

The approximate cost of drilling deep boreholes for all the SFL waste is estimated to be almost SEK 20 billion, not including costs associated with research and development, site investigations, or conditioning and disposal of the waste. This rough estimate means that the cost of the Deep boreholes concept will be at least 10 times higher than that of deep geological repository concepts based on rock caverns or silos. This significant difference in cost affects the judged possibility of the concept to fulfil the requirement N1-42 *Cost-effective*.

Conclusions and decision

Substantial efforts are required to enhance the current body of knowledge and develop the technologies needed to build, operate and close a Deep boreholes repository. The main argument against the concept is the difficulty of *proving* long-term safety. Although technology development is making progress, there is a great risk that it will still not be possible to show that the long-term safety can be guaranteed after closure, due mainly to the difficulties characterizing the rock and the hydrogeological conditions at the great depths in question.

In addition, significant pre-conditioning efforts are needed to fit the waste into the small space of a borehole. The concept is not considered a cost-effective alternative for the waste in question.

Based on the evaluation of the Deep boreholes concept presented above, it is concluded that disposal in Deep boreholes is not a realistic alternative for the disposal of long-lived low and intermediate level waste and it will not be studied further.

8.1.4 Drained rock repository

Overview

The characteristics of the Drained rock repository are presented in section 7.2.4. The long-term safety of the repository depends on that the repository remaining in a dry and drained rock domain throughout the life of the repository. This is achieved by locating the repository in a rock domain above the groundwater table and ensuring that the repository is covered by bedrock with a thickness of at least 200 metres.

Evaluation of long-term safety

The long-term safety of the Drained rock repository depends on ensuring that the bedrock in which the repository is located will remain dry through the entire lifetime of the repository. The most critical aspect for the long-term safety of the repository is thus to be able to show that the repository will remain drained throughout its lifetime.

Considering solely the transgression of seas or lakes, it would probably not be impossible to find a site for a repository somewhere inland in Sweden where dry conditions in terms of groundwater flow can be maintained in a Drained rock repository. However, it is not possible to rule out the possible existence of continental ice sheets and glacial conditions at any site in Sweden during the next 100,000 years. The most common situation at the bottom of an ice sheet is bottom melting, resulting in fully saturated groundwater conditions in the bedrock below the ice. Significantly increased groundwater flows can also be expected during certain periods of future glaciations.

Accordingly, it is not possible to demonstrate that a repository in Sweden according to the Drained rock concept would remain dry during such expected glaciations. On the contrary, current knowledge about the properties of continental ice sheets and icing processes in Scandinavia suggest that an initially dry repository would probably fill up with water during some period of a future ice age. It is thus not possible to verify that the concept fulfils the requirement N2-162 *Appropriately sited*.

Evaluation of technological aspects

The construction of a repository according to the Drained rock repository concept is considered a challenging task, but one which can be accomplished using currently known technology.

Conclusions and decision

The concept requires a site with very specific characteristics in terms of topography and rock conditions to ensure long-term safety after closure. It is not possible to demonstrate the existence of a site with these properties in Sweden.

It is concluded that disposal in a Drained rock repository is not a realistic alternative for the disposal of the Swedish long-lived low and intermediate level waste and the concept will not be studied further.

8.1.5 Gravel repository

Overview

The characteristics of the Gravel repository are presented in section 7.2.5. The safety function depends on the use of large amounts of a material with high hydraulic conductivity that forms a hydraulic cage around the waste containers. The groundwater flow through the repository rock cavern is directed through the cage. With this method there would be no driving force for water to flow through the waste and the interactions between the waste and the groundwater would thus be limited. It is proposed that the repository be located in crystalline bedrock at a depth of 300–500 metres.

Evaluation of long-term safety

The long-term safety of the Gravel repository depends on the use of large amounts of gravel that forms a hydraulic cage around the waste, through which the groundwater flow is directed. The function of the cage is dependent on the differences in hydraulic conductivity between the host rock, the gravel and the waste containers. Changes of the hydraulic conductivity of either the gravel or the waste containers will impact the intended function.

The processes that govern the evolution of the hydrological properties over time need to be controlled. For example, it must be assumed that the influx of fracture minerals or growth of organic matter will alter the hydraulic properties of the gravel with time. These processes could change the groundwater flow through the repository, redirecting more groundwater away from the hydraulic cage and through the stack of waste containers. Moreover, to ensure the function of the hydraulic cage, the integrity of the waste containers has to be maintained for a sufficiently long time. If the function of the hydraulic cage is maintained, the main release mechanism for radioactive material from the waste to the flowing groundwater will be by diffusion through the waste containers.

Evaluation of technological aspects

It is not expected that the construction of a repository based on the Gravel repository concept will pose any technical challenges apart from the conventional challenges related to underground work.

Conclusions and decision

The Gravel repository exhibits potential as a concept, with a straightforward design that may be advantageous for the construction of the repository as well as for the assessment of long-term safety. The hydraulic cage can also be combined with dense barriers. Further studies on the long-term safety of the concept are needed to gain more knowledge concerning the function of the hydraulic cage.

It has been decided that the Gravel repository concept will be further analysed and evaluated.

8.1.6 KBS-3 for long-lived low and intermediate level waste

Overview

The characteristics of the KBS-3 concept are described in section 7.2.6. The KBS-3 concept is a well-defined method for disposal of spent nuclear fuel and has been thoroughly studied over the past 30 years; see section 7.2.6 and references therein. The waste is placed in copper canisters with a cast-iron insert and deposited in bentonite-filled holes in tunnels at a depth of 300–500 metres below ground.

Evaluation of long-term safety

The long-term safety of the KBS-3 concept depends on the stability of the copper canisters that isolate the waste from interactions with the groundwater throughout the operating lifetime of the repository.

The KBS-3 concept has been developed as a method for disposal of the high-level spent nuclear fuel and has shown its potential as a disposal method for such waste in recent safety assessments (SKB 2011). The concept can be expected to work well for other less radiotoxic types of waste as well. However, it should be noted that the concept has been developed specifically for spent nuclear fuel, which is a well-defined ceramic form of waste. The long-lived low and intermediate level waste consists of different kinds of waste with varying properties, and it is yet to be established that the waste is compatible with the materials used in the KBS-3 concept. For example, it is not certain that the copper canisters are able to withstand forces acting from the inside caused by gas pressure from the degradation of the waste or corrosive liquids. However, the KBS-3 would probably be a suitable method for metallic waste forms such as BWR control rods. The PWR control rods are currently planned to be deposited together with the spent nuclear fuel in the Spent Fuel Repository.

Evaluation of technological aspects

The technologies for building the repository and disposing of copper canisters are known. The long-lived low and intermediate level waste in question consists of different kinds of waste, for example core components, control rods and drums filled with a mixture of waste and grout. Some drums contain waste in the form of liquids as well as mercury, organic material and very small amounts of plutonium and uranium. It is therefore expected that efforts will be made to develop conditioning methods for the waste, to adapt the waste so that it will not adversely affect the copper canister, and to make the most efficient use of the canister volume. Due to the varied content of the waste, it is expected that various kinds of conditioning will be required. All conditioning is expected to add to the dose burden of the personnel and must be carefully weighed against the expected improvement of long-term radiological safety. From a handling perspective, the KBS-3 method might be suitable for a small fraction of the SFL waste, for example the control rods; see section 5.2.1.

Evaluation of environmental aspects

In this evaluation it is assumed that the proposed concept will handle all types of present and future long-lived low and intermediate level waste. Implementation of this solution requires extensive use of non-renewable resources (such as copper or bentonite) per volume of waste. The activity concentration and the radiotoxicity of the low and intermediate level waste are also much lower than the concentration in the high level waste, for which the KBS-3 method was originally designed. Large amounts of energy are also needed for production of the copper canisters. Furthermore, the solution requires a much bigger excavated rock volume per volume of disposed waste than concepts using rock caverns or silos. Long-lived low and intermediate level waste, apart from the BWR control rods, will require extensive conditioning to fit the canister volume, and this can be expected to contribute to air emissions, effluents and increased energy consumption. Altogether, the possibility of the concept to fulfil the requirements N1-30 *Limited impact on the environment*, N1-31 *Resource-effective* and N1-42 *Cost-effective* is judged to be limited, when evaluated for disposal of long-lived low and intermediate level waste.

Conclusions and decision

It is concluded that the KBS-3 method is not an efficient alternative for the disposal of long-lived low and intermediate level waste. Despite excellent long-term safety properties, the resources needed to achieve this – copper, bentonite, energy, rock volume – are disproportionate to the benefits of using the KBS-3 method. The concept is therefore not being studied further for disposal of long-lived low and intermediate level waste. However, an exception could be made for the BWR control rods, which constitute a small fraction of the long-lived low and intermediate level waste and present properties suitable for disposal according to the KBS-3 method.

8.1.7 Super silo

Overview

The characteristics of the Super silo are described in section 7.2.7. The main components of this concept are a cylindrical double-wall concrete silo, a layer of bentonite between these silos and a layer of gravel that acts as a hydraulic cage between the outer silo and the bedrock.

Evaluation of long-term safety

The long-term safety of the Super silo depends on the use of three different barriers: a double-wall concrete silo that provides sorption and slow diffusive transport of the radionuclides, a layer of bentonite between the silo walls that further reduces the flow of groundwater through the engineered barrier system, and a layer of gravel between the outer concrete silo wall and the rock wall that acts as a hydraulic cage through which most of the groundwater entering the rock cavern will be transported. This engineered barrier system reduces interaction between the groundwater and the waste components as well as transport of radionuclides out of the repository.

The degradation rate of the barriers is crucial in the assessment of the long-term function of repository. From a material perspective, the issues are same as those identified for the Clay repository, section 8.1.1, the Concrete repository, section 8.1.2, and the Gravel repository, section 8.1.5. Beside these, the physical, mechanical and chemical interactions between the different components of the engineered barrier system also have to be considered. The degradation of the swelling properties of the bentonite due to interactions with high-pH leachate from the concrete, and crack formation in the outer concrete structure due to the swelling of the bentonite, are examples of processes that have been identified as being critical for the long-term integrity of the repository.

Evaluation of technological aspects

Apart from the technological challenges identified for the Clay repository (section 8.1.1) and the Concrete repository (section 8.1.2), the most challenging task in the construction of the Super silo will be casting of the double-walled concrete structure. It is considered possible to accomplish this by e.g. slip form casting, as for the existing silo in SFR. Although regarded as a demanding operation, casting is not considered to be so difficult that construction of the Super silo is jeopardized.

Conclusions and decision

The Super silo exhibits potential as a concept, with several engineered barriers contributing to the long-term safety of the repository. However, further studies of the long-term safety of the concept are needed to accumulate knowledge concerning the function of the individual barriers, interactions between barriers, etc. The technical aspects of building the Super silo also require further investigation.

It has been decided that the Super silo concept will be further analyzed and evaluated.

8.1.8 WP-Cave

Overview

The characteristics of the WP-Cave concept are described in section 7.2.8. The concept is an egg-shaped underground facility where the deposition area consists of multi-level tunnels emanating from a central shaft, like the spokes of a wheel. The area is surrounded by two engineered barriers. The inner barrier is a shield, consisting of either pure bentonite clay or a mixture of bentonite clay and sand/crushed gravel. Outside the shield is a hydraulic cage comprising a system of drilled tunnels and holes that divert the groundwater around the deposition area. The design of the repository is intended to keep the repository dry during operation. After sealing, the repository fills with water and a stable chemical environment is established in the bentonite barrier.

Evaluation of long-term safety

The concept was evaluated as a repository for spent nuclear fuel by SKB in the late 1980s (Skagius and Svemar 1989). The evaluation included a comparison with the KBS-3 system for disposal of spent nuclear fuel. The study concluded that the concept is associated with difficulties demonstrating that long-term safety is maintained after closure. However, some of these uncertainties stem from the fact that WP-Cave was evaluated as a repository for spent nuclear fuel, which generates heat. These uncertainties are thus not relevant to the study of the concept as a repository for long-lived low and intermediate level waste.

The purpose of the hydraulic cage is to reduce the hydraulic gradient over the repository and thereby divert the groundwater flow around the repository and away from the bentonite barrier. The function of the hydraulic cage is difficult to evaluate. A major uncertainty lies in the interaction between the flow paths in the cage and the flow paths formed by the fracture system in the rock at the selected site. Another uncertainty is how long the holes will remain open and thus sustain the intended function. Furthermore, it is considered unfavourable that the rapid flow paths provided by the bored tunnel systems will also allow any radioactivity released from the repository that passes the bentonite barrier to enter the biosphere via the fastest transport channel that reaches the cage. The sensitivity of the concept to existing rapid flow paths in the bedrock limits the options in the site-selection process. The use of WP-Cave requires model development on these and other areas where the body of data is currently incomplete. It will be challenging to verify the fulfilment of requirement N2-156 *Possible to analyze safety after closure*.

Evaluation of technological aspects

Major uncertainties exist regarding the feasibility of building a WP-Cave repository. The principle design of the WP-Cave repository includes a very large rock volume that is supposed to rest on a thick bed of bentonite. It has to be proven that the bentonite is stable enough to prevent the rock volume from moving around in the excavated rock cavern. Further, it has to be ensured that the rock volume, in which the tunnels for waste disposal will be constructed, remains stable when the stresses are relieved after separation from the surrounding bedrock.

Large amounts of bentonite clay have to be installed in a rock volume where the water flow is not negligible. Several issues that need to be addressed have been identified, including the risk of piping and erosion of the bentonite clay in the inner barrier.

Furthermore, the question of how to respond to an accident in the repository has to be considered, if the only transportation path down to the repository is via the shaft.

Assessment of whether a specific site will be suitable must consider both the need for rock volumes of appropriate quality and the requirements created by the predetermined geometric configuration of the WP-Cave repository. The picture of the occurrence of horizontal or near-horizontal major fracture in the Swedish bedrock has changed. Previously it was believed that these fractures were rare, but geological surveys in various parts of the country show that the average distance between them in many places is 200–300 metres. As a result, it may be harder than previously thought to find a site for a WP-Cave repository, which requires a vertical extent of several hundred metres. This affects the possibility of the concept to fulfil the requirement N2-162 *Appropriately sited*.

Evaluation of environmental aspects

The WP-Cave concept requires the excavation of large quantities of rock as well as the use of a large quantity of backfill. In the comparison between WP-Cave and the KBS-3 repository (Skagius and Svemar 1989), WP-Cave was concluded to entail larger quantities of excavated rock and to require greater quantities of backfill than the KBS-3 repository. Both the excavation of a large quantity of rock and the handling of rock masses have a negative impact on environment. More energy is consumed and transport emissions are greater. It will be difficult to show that the concept fulfils the requirements N1-30 *Limited impact on the environment* and N1-31 *Resource-effective.*

Evaluation of aspects related to cost and time

The cost of a complete system for management and disposal of spent nuclear fuel based on the WP-Cave concept has been calculated to be considerably higher than that of a KBS-3 repository (SKB 1989). The maturity of the concept is also low, posing a significant risk that both the cost and the time associated with this solution may increase. The high risk level in terms of time impacts the judged possibility of the concept to fulfil requirement N1-38 *Timetable*.

The resources need to achieve a repository for SFL waste according to the WP-Cave concept are disproportionate. Significant safety advantages should be required in order to compensate for a considerably more expensive system. The fulfilment of requirements N1-31 *Resource-effective* and N1-42 *Cost-effective* will be difficult to show.

Conclusions and decision

It is concluded that the WP-Cave repository is not a realistic alternative for the disposal of long-lived low and intermediate level waste. There are several great uncertainties that require model development, particularly in order to show that long-term safety is maintained after closure.

It has been decided that the WP-Cave concept will not be further analyzed.

8.1.9 No action alternative

Overview

The no action alternative concept represents the alternative of *not* building SFL (see section 7.2.9). This in turn entails the use of existing and planned repositories (SFR and the planned repository for spent nuclear fuel) for disposal of all present and future long-lived low and intermediate level waste.

Evaluation of long-term safety

The most important prerequisite is that all the waste currently planned for disposal in SFL needs to be suitable for disposal in SFR or the planned repository for spent nuclear fuel. Since the main criterion for disposal in SFR is the absence of significant amounts of long-lived nuclides, and basically all the SFL waste has significant amounts of long-lived nuclides, the planned repository for spent nuclear fuel will have to accommodate most of the SFL waste. There may be minor fractions of waste that can be disposed of in SFR, but for large fractions of the waste, such as the core components and PWR reactor pressure vessels, the content of long-lived nuclides is simply too high for SFR. The evaluation of the no action alternative thus follows the evaluation of the KBS-3 concept for long-lived low and intermediate level waste reported in section 8.1.6.

The assessment of the long-term safety aspects of the KBS-3 concept for long-lived low and intermediate level waste concludes that the KBS-3 concept has been developed specifically for the spent nuclear fuel, which is a well-defined ceramic form of waste. However, the KBS-3 concept would probably also be suitable for metallic waste forms such as BWR control rods. Today the PWR control rods are planned to be deposited together with the spent nuclear fuel in the repository for spent nuclear fuel.

Evaluation of technological aspects

The assessment of the technological aspects of the KBS-3 concept for long-lived low and intermediate level waste is reported in section 8.1.6. It is concluded that although the KBS-3 method is developed to a high level, the conditioning of the current SFL waste to fit into the KBS-3 system will probably require further development. From a handling perspective, the KBS-3 method might be suitable for a small fraction of the SFL waste, for example the control rods; see section 5.2.1.

Evaluation of environmental aspects

The assessment of the environmental aspects of the KBS-3 concept for long-lived low and intermediate level waste is reported in section 8.1.6. The assessment concludes that the environmental impact is high relative to the radiotoxicity of the waste. The possibility of the concept to fulfil the requirements N1-30 *Limited impact on the environment*, N1-31 *Resource-effective* and N1-42 *Cost-effective* is judged to be limited.

Conclusions and decision

Since the no action alternative in practice means that all the long-lived low and intermediate level waste currently planned for SFL will be disposed of the planned repository for spent nuclear fuel, the conclusions from the assessment of the KBS-3 concept for long-lived low and intermediate level waste (in section 8.1.6) are also applicable to this concept.

It is thus concluded that the no action alternative is not an efficient alternative for the disposal of long-lived low and intermediate level waste and it is not being studied further. However, future events may still change the final handling of particular fractions of the waste. As discussed, BWR control rods are suitable from a handling perspective for disposal in a copper canister.

8.2 Assessment, investigation and development of remaining concepts

A primary assessment of the nine repository concept candidates for the future SFL was presented in section 8.1. Five concepts were excluded from further analysis since fundamental characteristics of the concepts deemed it difficult or unlikely for the concepts to meet the requirements. The remaining four concepts will be evaluated further in order to find the most suitable repository concept for the long-lived low and intermediate level waste.

To support the further evaluation process, the concept descriptions need to be further developed and defined in some cases. The process of assessment – investigation – development has been described in section 3.4. It should be noted that the result of development iterations at this stage of the study should be seen as further refinement of a concept and not as a new concept. The development work is done within the framework of the original concept.

The primary assessment revealed the need to further investigate the properties of the barriers, how they evolve over time and the influence of this degradation on the long-term safety of the repository. The primary assessment also indicated a need to further define the technical aspects of the concepts to permit technical evaluation. Based on the primary assessment of the concepts, investigations have been conducted to identify "show-stoppers" and to reveal differences between concepts. The results of these investigations are published in two separate reports concerning investigations related to the long-term safety of the concepts and investigations concerning technology:

- Evins L Z, 2013. Progress report on Evaluation of Long term Safety of Proposed SFL Concepts. SKB R-13-41, Svensk Kärnbränslehantering AB.
- Grahm P, Luterkort D, Mårtensson P, Nilsson F, Nyblad B, Oxfall M, Stojanovic B, 2013. SFL Concept Study – Technical design and evaluation of potential repository concepts for long-lived low and intermediate level waste. SKB R-13-24, Svensk Kärnbränslehantering AB.

If the concepts require development and refinement to permit evaluation, this is described in the reports. The reports also give a detailed account of the evaluation of the concepts for the evaluation category *Long-term safety* and the evaluation categories *Technology* and *Cost and time*, respectively.

8.3 Step 2 – Expanded assessment

In step 2 of the evaluation, the remaining concepts are evaluated and compared with each other using the evaluation factors presented in section 6.3. For each factor, the characteristics of the remaining concepts are summarized and appraised. The analysis focuses on the differences between the concepts and an evaluation of those differences.

The expanded assessment concerns the four remaining concepts:

- The Concrete repository.
- The Clay repository.
- The Gravel repository.
- The Super silo.

The same parameters are assumed for all four concepts:

- same waste (quantity, type, inventory),
- same containers,
- same conditioning methods, and
- same characteristics and prerequisites for the "virtual" location.

Since all remaining concepts are geological concepts located deep down in the bedrock, it should be noted that the main safety function for all four concepts is retardation by 300–500 metres of bedrock. The safety functions of the natural barrier are:

- To provide chemically favourable conditions.
- To provide favourable hydrological and transport conditions.
- To provide mechanically stable conditions.
- To provide thermally favourable conditions.

The concepts are evaluated and compared using the evaluation factors derived in Chapter 6. The outline of the following section is based on the four evaluation categories into which the evaluation factors are grouped (see section 6.3):

Long-term safety

Evaluation factor 1 – Feasibility of making a post-closure safety assessment.

Evaluation factor 2 – Robustness of the barrier safety functions.

Environment and society

Evaluation factor 3 – Impact on human health and the environment.

Evaluation factor 4 – Land and resource needs.

Evaluation factor 5 – Acceptability.

Evaluation factor 6 – Burdens on future generations.

Technology

Evaluation factor 7 – Personal safety and working environment.

Evaluation factor 8 – Feasibility of design and construction.

Evaluation factor 9 – Feasibility of technology and method of operation.

Evaluation factor 10 – Flexibility.

Cost and time

Evaluation factor 11 – Cost.

Evaluation factor 12 – Time.

No formal weighting factors have been assigned to the individual evaluation factors, and the number of evaluation factors within each category should not be taken as an indication of the relative weight of the category. Each factor matters and needs to be considered to give the full picture of a certain concept, but considerations related to the safety of the repository, both pre- and post-closure, are certainly those that carry the greatest weight.

8.3.1 Evaluation factor 1 – Feasibility of making a post-closure safety assessment

As described in section 6.3.1, this evaluation factor includes the level of process understanding and how well these processes can be modelled. It also includes an assessment of how easy it would be to define, achieve and verify the initial state. It is linked to the requirements N2-41 *Possible to analyze waste*, and N2-156 *Possible to analyze safety after closure* (see Appendix 1).

Clay repository

The safety functions of the bentonite barrier are low groundwater flow, strong sorption and low diffusivity. One additional safety function is filtering of colloids by the bentonite. The processes that will be important to understand and to model are mainly sorption, diffusion and processes affecting permeability. A number of relevant processes expected to occur in bentonite have been identified through previous work (SKB 2010b):

- Water uptake and transport for unsaturated conditions.
- Water transport for saturated conditions.
- Piping/erosion.
- Swelling/mass redistribution.
- Liquefaction.
- Freezing.
- Water transport/Two-phase flow/Mixing.
- Dissolution/Precipitation.
- Speciation and sorption, including ion exchange.
- Diffusion/Diffusive transport of species.
- Alteration of montmorillonite.
- Bentonite erosion.
- Colloid formation and transport.

In general, much research has been conducted on these bentonite processes, primarily in connection with the development of the KBS-3 system, and process understanding is advanced for this material. There is also substantial experience available for modelling of these processes (SKB 2010b, Åkesson et al. 2010, Neretnieks and Moreno 2013). There are, however, some areas in which further research is needed. These are further described in section 10.2. One example, related to the proposed design of the Clay repository, is the effect of inhomogeneous water saturation and swelling during the saturation phase on the overall long-term permeability of the bentonite. Another example of an area which requires a better understanding is cement-bentonite interaction (Neretnieks and Moreno 2013). The SFL legacy waste contains cement, as will the structures inside the vault in the Clay repository. Therefore, it needs to be investigated how detrimental this interaction will be on the bentonite properties, and if this effect warrants the use of so-called low-pH cement. In the low-pH cement, the porewater pH will be less than 11, which would have less detrimental effect than conventional Portland cement (SKB 2013a).

When it comes to achieving the defined initial state, it is expected that there may be difficulties in verifying that the initial state has in fact been achieved in the full volume of bentonite.

In summary, both process understanding and modelling of bentonite in a repository setting are at an advanced stage. There are, however, some remaining uncertainties related to verification of the initial state and the homogenization processes (i.e. bentonite swelling model) that may affect the feasibility of performing the safety assessment for the Clay repository.

Concrete repository

The safety functions of the Concrete repository are low groundwater flow, low diffusivity, strong sorption, and high-pH conditions. The concrete will raise the pH of the groundwater in contact with the waste, which will in turn allow a passivating layer to be developed on the metallic parts of the waste, primarily steel and Zr. This creates a chemical environment that slows down waste dissolution. The four safety functions mentioned above are based upon knowledge of a number of processes, such as:

- Water transport.
- Freezing.
- Fracturing.
- Dissolution/Precipitation.
- Speciation and sorption.
- Colloid formation and transport.

In addition, the safety functions are globally affected by concrete degradation. In general, enough is known about these properties of pristine concrete to model radionuclide transport through the materials (Neretnieks and Moreno 2013). However, there is more work to be done on the properties of degraded concrete. The planned research is further described in section 10.2. Identifying the initial state is assumed to be relatively straightforward; however, verifying that the initial state has been achieved might pose some problems, and development of a suitable technique for this purpose is required.

In summary, some major uncertainties remain for the concrete repository relating to the degradation processes and verification of the initial state. However, concrete is a widely used building material whose properties are well known, and the relevant processes in pristine concrete are deemed to be well enough understood for the modelling required in a safety assessment.

Gravel repository

The one safety function of the gravel repository is high hydraulic conductivity. Relevant processes expected to affect this property are:

- Clogging.
- Freezing.
- Water transport/Two-phase flow/Mixing.
- Dissolution/Precipitation.
- Colloid formation and transport.
- Microbial processes.

This material is used in various structures and for various purposes. In general, many of the relevant processes are deemed to be sufficiently well understood. There is, however, a scarcity of relevant experiments and a lack of long-term studies to assess the evolution of hydraulic conductivity over time. Assuming no significant changes occur in the hydraulic properties over time, the modelling is expected to be relatively straightforward (Neretnieks and Moreno 2013). There are basic models that can be used, but in order to perform reactive transport modelling for this material, specific site data will be required, as well as some model development.

No specific difficulties have been identified in achieving and verifying the initial state.

Modelling the gravel in terms of initial hydraulic properties is expected to be relatively straightforward. However, the properties of the material are expected to change with time, and little knowledge exists concerning how these changes affect hydraulic conductivity, due to a lack of long-term studies of clogging and specific site data.

Super silo

The safety functions of the Super silo include the safety functions assigned to the Clay repository (section 7.2.1), the Concrete repository (section 7.2.2) and the Gravel repository (section 7.2.5). Thus, the concrete and bentonite are both expected to provide low groundwater flow, low diffusivity and strong sorption. The gravel around the silo is expected to function as a hydraulic cage. However, the combination of materials adds complexity, as does the proposed double concrete wall. It is likely that the barriers will affect each other, and these interactions needs to be taken into account in the models used in a safety assessment. The Super silo therefore requires both further research and model development, for example regarding bentonite-concrete interaction (Neretnieks and Moreno 2013). Uncertainties also exist concerning the initial state due to the envisaged method of waste and barrier emplacement. Thus, the Super silo is judged to require substantial modelling and research concerning the physical and chemical interaction between bentonite and concrete.

8.3.2 Evaluation factor 2 – Robustness of the barrier safety functions

As described in section 6.3.1, this evaluation factor includes a qualitative assessment of many different aspects of the barrier safety functions. Since the comparison performed here relates mainly to the safety functions of the suggested engineered barriers, some aspects of this evaluation factor are considered more relevant than others. The most relevant are:

- The potential of the engineered barriers to achieve and maintain the defined safety functions.
- The effects of processes in the barriers which may affect the safety functions of other barriers.
- The effect of the repository environment on the long-term durability of the engineered barriers.

Aspects that also need to be considered include those directly related to the waste and reactions within the waste package:

- Release rate of radionuclides from the waste.
- Chemical reactivity of the waste.
- Processes in the waste which changes the properties of the waste package.

In addition, retention in the natural barrier, effects of backfilling and closure on the safety functions of the barriers, and the risk of inadvertent intrusion are also included but are considered less important for the comparison performed here.

For this evaluation factor, an important aspect to keep in mind is the time frame, which is determined by the radionuclide inventory. Due to the long-lived character of the radionuclides in the SFL waste, the safety assessment needs to cover at least 100,000 years, or the time for a glacial cycle. This should be considered when evaluating the robustness of the barrier safety functions.

This evaluation factor is related to the following requirements: N1-113 *Safety during handling and final disposal*, N1-19 *Best available technology and optimization*, N1-17 *Robust design*, N1-18 *Reliably constructed*, N2-47 *Resistance after closure*, N2-132 *Contain, prevent or retard*, N2-162 *Appropriately sited*, N2-152 *Multi-Barrier Principle*, N2-117 *Long-term durable barriers*, N2-95 *Consider inadvertent intrusion* (see Appendix 1).

Clay repository

The thick layer of bentonite surrounding the waste in the Clay repository is expected to provide a low groundwater flow, low diffusivity and strong sorption. In addition, bentonite filters colloids, which limits colloid-facilitated transport of radionuclides. Colloids may form when organic matter in the waste is broken down. Some radionuclides can sorb on colloids, which may affect the transport of radionuclides in the barriers. These safety functions are expected to be maintained for as long as the bentonite is in place and intact. The processes which may affect the bentonite's safety functions are listed above (section 8.3.1). The robustness of these safety functions have recently been thoroughly evaluated (SKB 2011).

A comparison between the initial radionuclide transport capacity of a the Concrete repository and the Clay repository shows that, initially, hydraulic conductivity is higher for bentonite, and retardation (from diffusion and sorption) is not as strong as in concrete (Neretnieks and Moreno 2013). Note, however, that this relates to an initial, pristine concrete monolith, and in evaluating robustness, one needs to take into account the potential degradation of the properties which provide the safety functions. It has been shown that one threat to the bentonite is diluted groundwater, which may cause bentonite erosion (SKB 2011). However, the large quantities of bentonite which will be present in the Clay repository will counteract the effect of bentonite erosion. Another threat to the robustness of the bentonite safety functions is the interaction between the concrete structural components in the core of the cavern, since it is known that bentonite is prone to react with products formed from concrete degradation (Huertas et al. 2005, SKB 2010b, Neretnieks and Moreno 2013). The cement in the waste may also have a detrimental effect. The bentonite-concrete interaction is expected to affect properties related to radionuclide sorption and diffusion through the bentonite.

In general, however, the bentonite is expected to be robust, and safety functions are expected to be maintained, for a long period of time.

Concrete repository

As stated above, the safety functions of a concrete repository are low groundwater flow, low diffusivity, strong sorption, and high-pH conditions. These are affected by a number of processes. The concrete can be expected to degrade over time, affecting all safety functions. However, the concrete monolith can be assumed to be initially pristine, and a qualitative assessment of the proposed safety functions can be obtained from a modelling exercise. This has been done by Neretnieks and Moreno (2013), with the conclusion that a pristine concrete monolith should show excellent performance with regard to low hydraulic conductivity, slow diffusion and high sorption.

Since this evaluation factor intends to elucidate how robust the proposed safety functions are, a qualitative assessment of the long-term performance of the concrete is required. First, the processes affecting the safety functions need to be identified. All processes relate in one way or another to degradation of the concrete. This includes fracturing as well as dissolution and precipitation. Fracturing can be caused by changes in the waste packages (e.g. expansion) or by rock movements. Dissolution and precipitation can cause changes in chemical properties affecting sorption, as well as hydraulic properties. Considering the chemical differences between concrete and the surrounding natural environment, concrete degradation is inevitable. The question is how fast it will proceed, and how long in this process the safety functions are upheld. Leaching of the alkali and earth-alkali hydroxides of the concrete will also affect the surrounding chemical environment in the natural barrier, i.e. the hydrogeochemical properties of the rock system. There will be a high-pH plume in the groundwater which has interacted with the concrete. Considering the vast amounts of concrete which will be available, the pH can be expected to be buffered for a long period of time, providing a high-pH environment. This should be a robust safety function for the steel-based waste.

In summary, the barrier safety functions of the Concrete repository are expected to be initially very robust, as indicated by Neretnieks and Moreno (2013). There are important uncertainties concerning the fracturing and degradation of the concrete over time. This could affect hydraulic and chemical properties important for radionuclide transport. Further research is required in order to evaluate how water flow, diffusion and sorption will be affected. However, since the concrete will most likely buffer pH for a long time, the safety function of a highly alkaline environment is expected to be sustained for a substantial amount of time after closure. Thus, all safety functions are initially expected to be strong, but over time the most robust safety function may be the alkaline environment. This is an important consideration for the metallic waste.

Gravel repository

The Gravel repository exhibits one safety function, namely the high hydraulic conductivity of the gravel. The most relevant processes that can affect this safety function are clogging, dissolution and precipitation, all of which will change the hydraulic properties of the gravel.

Even if it were to be shown that the hydraulic properties of the gravel fill can be maintained for a long time, the safety function should first be shown to be sound and robust. Neretnieks and Moreno (2013) performed a modelling exercise to evaluate the transport capacity of a tunnel in which the waste containers are surrounded on all sides by gravel. In this model, there is nothing but the waste containers themselves to divert the seeping groundwater around the waste, thereby minimising the groundwater flow through the waste. Nuclides will, in this model, escape mainly by diffusion, but also partly by flow, from the waste to the water flowing in the gravel. The result of this exercise shows that using only gravel around the waste is not sufficient (Neretnieks and Moreno 2013). This assessment is based on simple model calculations involving no additional protection of the waste other than the lower hydraulic conductivity of the waste package than the surrounding gravel (Neretnieks and Moreno 2013).

In a possible case where additional protective features are considered, such as a dense barrier between the waste and the gravel, the hydraulic cage may still contribute to safety by diverting the groundwater flow. However, great uncertainties remain regarding how long the safety function will be sustained. Some studies suggest clogging of crushed rock is faster than for sand (Sternö 2005), and the performance of the hydraulic cage is probably very site-specific. The chemical reactions governing dissolution and precipitation need to be specified. It should also be noted that if the groundwater flow is high, the concentration of radionuclides at the waste-gravel interface will be very low, potentially affecting the diffusion rate of radionuclides from the waste to the high-flow cage.

In summary, considering high hydraulic conductivity as a sole safety function is deemed to be insufficient. Even if the Gravel repository concept were improved to avoid the negative effects of the high groundwater flow, the robustness of the safety function of a high hydraulic conductivity is questionable.

Super silo

The safety functions of the Super silo are low hydraulic conductivity, low diffusivity, and strong sorption in both concrete and bentonite. The gravel surrounding the silo is expected to function as a hydraulic cage, with high hydraulic conductivity. The bentonite also filters colloids. The processes affecting these safety functions have been described above (section 8.3.1).

The main idea is that using these multiple engineered barriers will reinforce the retardation of radionuclides seeping from the waste. This idea is in general sound; however, there are several uncertainties and issues that need to be addressed in the concept as it is defined. The main concern is the proposed double concrete wall. Since the bentonite is negatively affected by concrete, it will in such a design be attacked from both sides, which may cause loss of the bentonite safety functions. If the design instead involves bentonite outside of a concrete wall, the bentonite may retain the safety functions sufficiently so that low hydraulic conductivity can be retained, lowering groundwater flow through the waste even in the case of a severely fractured concrete wall (Neretnieks and Moreno 2013). If low-pH cement is used, the negative effect of concrete degradation on bentonite may be limited; however, these issues require further research and model development. The shape of the silo may have a beneficial effect on the likelihood of avoiding severe fracturing in the concrete.

In summary, the suggested design of the Super silo needs to be modified in order to improve the robustness of the safety functions of both concrete and bentonite. The Super silo has potential, and with a modified design and pristine barriers, the retention properties are strong. This is, however, dependent on the effects of the bentonite-concrete interaction.

8.3.3 Evaluation factor 3 – Impact on human health and the environment

First of all it should be noted that all four remaining concepts are deep geological repositories. Furthermore, and as mentioned in the beginning of section 8.3, the same assumptions (e.g. same waste and same prerequisite for the "virtual" location) are made for all four concepts. This considerably narrows the scope for differences between the concepts in environmental factors. Therefore, differences between concepts in terms of impact on human health and the environment are few and rather small.

One aspect that varies between concepts is the volume of the repository. The volume of the repository is in turn related to a number of environmental impacts:

- Blasting of shafts and underground openings as well as driving of tunnels require the use of explosives containing nitrogen that will end up in effluents such as drainage water. The larger the volume of the repository the more nitrogen will be released with effluents.
- Variations in the volume of the repository also mean that the amount of rock spoil from the underground construction of the repository will vary. The amount of rock spoil directly influences both on-site and off-site haulage with increased noise levels (both on site and along transport roads) and increased atmospheric emissions as a result.
- The volume of the repository is also correlated to impact on groundwater such as groundwater lowering. All underground structures below the groundwater table influence the groundwater level, and a larger volume of the repository will induce greater groundwater lowering or at least a greater risk for groundwater lowering.

The total volume of the underground part of the repository and hence the amount of extracted rock spoil vary between the concepts, but the differences are small and there is a high degree of uncertainty attached to the figures. It is also important to recall that a major part of the total volume of the underground part of the repository is due to the ramp and the shaft. This means that the volume required for the disposal of the waste is relatively small compared to the total volume of the underground part of the repository (including shaft and ramp): 15–25 percent depending on concept, assuming a repository depth of 500 metres.

Regarding environmental risks, it can be noted that the Concrete repository may cause a high-pH plume in the groundwater which has interacted with the concrete. The Gravel repository may be more vulnerable to groundwater contamination, as it will be in direct contact with groundwater. However, these issues are mainly relevant after closure of the repository and will therefore be considered in the assessment of long-term safety.

8.3.4 Evaluation factor 4 – Land and resource needs

As mentioned earlier, the four remaining concepts are deep geological repositories which, for the purpose of the evaluation, share identical parameters (e.g. same waste and same prerequisite for the “virtual” location).

This means that the surface part as well as the underground part of the four concepts will have many similarities in terms of land and resource needs.

Land use and material needs during construction of the surface part of the facility should be very similar for all four concepts, whereas the amount and type of materials needed during construction, operation and closure/decommissioning of the underground part of the repository differ. The guiding principles are to limit the amounts of material and resources used and to consider how and where those materials are produced.

Variations in the volume of the underground part of the repository will also affect the amount of sealant/grouting, chemicals and backfill materials. As mentioned earlier, the volume of the repository varies between concepts, but this difference has been considered to have limited significance for environmental impact.

Besides the quantities of materials needed during construction, operation and closure of the repository, it is the types of materials used that lead to variations in the environmental effects that can be expected. A significant difference concerns the type of backfill material used to fill in the rock caverns, tunnels and shaft of the underground part of the repository. In fact, in the case of the Clay repository concept it is assumed that bentonite is used as the main backfilling material, whereas in the case of the Gravel repository it is crushed rock that is mainly used for backfill. In the case of the Concrete repository, it is assumed that concrete will be used to fill the repository, while in the case of the Super silo concept it is proposed that sand, gravel, bentonite, and concrete could be used to fill in the different parts of the repository.

Bentonite is a finite resource that is produced far away from Sweden. The use of large amounts of bentonite will therefore lead to increased transportation needs over long distances. In contrast, crushed rock used for the Gravel repository can be produced locally in most parts of the country or even reused from the excavation of the repository if storage volumes are made available during the operating phase. Concrete, which will be used in large quantities for the concrete concept, can be produced on site but will require transportation of raw materials. Furthermore, concrete production consumes considerable quantities of energy and is a source for carbon dioxide exhausts.

8.3.5 Evaluation factor 5 – Acceptability

Acceptability refers to the proposed concept(s) being understood and accepted by local residents, politicians, officials and non-governmental organizations at both the local and national levels. The acceptability of the concept is based on SKB being able to gain confidence and credibility for the solutions that are proposed for the SFL repository.

The principle of a hydraulic cage as a barrier seems to be more difficult to communicate than the "dense" barriers that are proposed for the other concepts. For a novice, it may not be credible that a large volume of gravel saturated with groundwater would be enough of a barrier to prevent radioactive materials from reaching the surface.

The Super silo is based on several engineered barriers that may give it more credibility, but that concept may also be more difficult to explain and understand.

8.3.6 Evaluation factor 6 – Burdens on future generations

Burdens on future generations can be viewed as requirements on monitoring and possible retrievability of the deposited waste.

All the remaining repository concepts are based on the assumption that neither monitoring nor maintenance is required after closure.

With respect to retrieval of the waste, it can be noted that it is easier to remove dry bentonite than concrete. But if the bentonite is water-saturated, waste retrieval is much more complicated. Retrieval from the Gravel repository is probably the easiest of the four concepts, whereas retrieval from the Super silo is probably the most difficult due to the complex structure of the silo.

8.3.7 Evaluation factor 7 – Personal safety and working environment

This evaluation factor includes occupational safety, the probability and consequences of an accident involving the release of radioactive substances, the need for radiation protection, and the ability to provide a good working environment. The most important conclusions from the evaluation (Grahm et al. 2013) are summarized in this section.

The risk of fire must be prevented in a similar way for all concepts, and it is considered to be no significant difference in the probability or consequence between the concepts.

Clay repository

The design requires a concrete structure on pillars in which the waste is emplaced. This structure acts as a radiation shield during operation. The waste is handled by remote overhead crane, so low doses to personnel are expected. No parts of the engineered barrier system will be detrimental to a good working environment. In the case of the Clay repository, the most serious risk is dropping a waste container. If this happens above the concrete structure, the situation can be handled relatively easily.

Concrete repository

The concept offers good opportunities for high safety and acceptable working conditions. The concrete structures will provide sufficient protection against gamma radiation. The waste is handled by a remote overhead crane, so low doses to personnel are expected. No parts of the engineered barrier system will be detrimental to a good working environment. The most serious risk is dropping a waste container. If this happens above the concrete structure, the situation can be handled relatively easily.

Gravel repository

Among the four remaining concepts, the Gravel repository stands out as the concept that offers the poorest personal safety and working conditions, since there are no high-density barriers and thus no radiation shielding during operation and backfilling of the repository. This would be even more problematic in the event of an accident. However, this problem can be solved relatively easily, for example by erecting precast (and removable) concrete beams around the waste. One identified risk is that a complicated decontamination situation could arise if a radioactive leak occurs into the gravel backfill. A large volume of gravel would then have to be checked and removed as radioactive waste.

Super silo

This concept offers good potential for high safety and acceptable working conditions. Compared with the other three concepts, however, the Super silo is associated with a higher risk of accidents during the complicated construction phase.

During the operating phase, the concrete structures will provide sufficient protection against gamma radiation. The waste is handled remotely by overhead crane, so low doses to personnel are expected. No parts of the engineered barrier system will be detrimental to a good working environment. In the case of the Super silo as well, the most serious risk is dropping a waste container. If this happens above the concrete structure, the situation can be handled relatively easily.

8.3.8 Evaluation factor 8 – Feasibility of design and construction

This evaluation factor includes technical maturity, simplicity in design and construction, resistance to degradation during the operating phase, feasibility of quality control, maintenance, feasibility of physical protection and safeguards. The most important conclusions from the evaluation (Grahm et al. 2013) are summarized in this section.

The Gravel and Concrete repositories are considered the easiest to design and build, whereas the Super silo is by far the most challenging. The Clay repository is ranked as the second most difficult, due to the challenge of installing bentonite beneath the stack of waste containers, and rapid bentonite emplacement is a key factor. The technical feasibility of each repository concept is explained below in simple terms.

Clay repository

The feasibility of a Clay repository is judged to be good, and it bear certain similarities to the Concrete repository in terms of design and construction of the supporting structures. There are, however, technical issues that have to be resolved. For instance, the foundation and bentonite installation are examples of new designs in need of development.

Backfilling is expected to be rather complicated, since backfilling with bentonite needs to be carried out swiftly and without interruption once started. A great advantage of backfilling with bentonite immediately prior to sealing and closure of the facility is that there will be no time for any chemical or mechanical degradation of the bentonite to occur. However, this requires a design where the base slab is placed on pillars or the like.

As discussed by Grahm et al. (2013), an alternative to the pillar type of foundation would be to build a cradle in which bentonite blocks are put in place prior to the operating phase. The concrete base slab is then placed on the bentonite blocks. It is vital that the bentonite beneath the base slab can be protected from free water and humidity during the entire operating phase. The design of the base slab, the cradle and the lid has to guarantee this. The mechanical properties of the bentonite blocks beneath the concrete slab has to be investigated to ensure that it will be stable enough.

Quality control of the bentonite is very important for the function of the concept. Robust and quick systems for quality control need to be developed. The high mass of bentonite should ensure a reasonable safety margin. The maintenance of a Clay repository during the operating phase is expected to be easy.

Concrete repository

Reinforced concrete structures for SFL can be fabricated by using well-known technology and materials. Lessons can be learned from the existing SFR and the planned extension of SFR. Unreinforced concrete structures are much less well-understood, but have been studied within the SFR extension project for the 2BMA rock cavern. Standard concrete is generally well-understood. The effect of additives, such as fly ash or other mineralogical additives, is less well-known. This is particularly true when it comes to the long-term properties of the material.

Difficulties may arise if it is decided that formwork struts cannot be used. Casting of a tall structure without the use of formwork struts will require special precautions and measures. To avoid joints, concrete structure should be cast all at once. This is particularly important if it is decided that reinforcement cannot be used due to the great potential for crack formation during the hardening of the concrete.

Building a Concrete repository should in general be rather straightforward. Concrete structures with an expected service life of about 100 years can be built in a marine environment according to well-known standards and using well-known technology. For successful results, drainage systems and waterproofing membranes must be installed and function satisfactorily during the entire operating time. Requirements on the climate in the repository need to be considered when the climate control systems are designed.

It is not expected that the materials used in the engineered barriers will adversely affect the waste containers during the operating phase of the repository. Maintenance during the operating phase is deemed feasible. All parts of the structure can be checked during the operating period of the repository.

The use of concrete or grout as backfill material can nevertheless be complicated due to the large amounts of material required. The backfilling phase of the Concrete repository will be challenging with respect to logistics, quality control and heat generation in the backfill with possible subsequent crack formation.

Gravel repository

The Gravel repository is judged to offer high feasibility in design, construction and quality control. Design and construction of the repository are expected to be easy, provided that a suitable site with very few conductive fractures can be found.

If the design of the Gravel repository is modified so that backfilling with gravel can be done immediately prior to sealing and closure of the facility, the operating phase will not contribute to degradation of the hydraulic safety function. However, this will require the base slab to be placed on pillars. Otherwise, clogging of the gravel medium may become a problem already during the operating phase.

Super silo

The design of the Super silo has been assessed by Grahm et al. (2013) with the result that the original design is judged to be inappropriate. The high hydrostatic pressure, as well as the high rock tensions at the planned SFL repository depth, may make it impossible to build a Super silo with the proposed dimensions. It has been shown by finite-element calculations that the radius of the outer cylindrical concrete wall cannot exceed 7 metres if it is to remain intact after closure under the load of the hydrostatic pressure (5 MPa at 500 metres depth). Furthermore, the use of high-quality bentonite in the intermediary barrier should be avoided if possible, to reduce the load on the concrete walls from the bentonite swelling pressure. If simplifications are made in the design (e.g. omitting the outer concrete silo), the technical feasibility of the concept can be improved. This problem could also be remedied if several small silos are built instead. A major drawback is that the available disposal volume in each silo decreases dramatically.

The double-walled silo is a complex structure, which is expected to entail complicated installation. Slip-form casting is a well-known technology and experience is available from a large number of projects. Even though slip-form casting is a well-known technology, it will be very complex to execute due to the limited space outside the formwork and other complicated logistics at this great depth. The slip-form casting method does not require struts, which is an advantage compared to the Concrete repository. However, slip-form casting puts heavy demands on the logistics during construction in terms of e.g. material deliveries. Slip-form casting will also require extensive use of reinforcement in order to provide support during the casting process. What is said about concrete materials for the Concrete repository also holds true for the Super silo concept.

As regards bentonite installation, there are still a number of open technical questions, for example the loadbearing capacity of bentonite blocks as well as bentonite installation and quality control. Casting a large concrete structure on a bed of bentonite or a sand/bentonite mixture is not a well-known technology.

Interaction between bentonite and concrete will add extra complexity to the construction work. Chemical and mechanical interactions between the concrete structures and the bentonite need to be taken into consideration when designing the repository, formulating the concrete recipe and determining the quantity of bentonite.

Swelling of the bentonite has to be avoided until the hydrostatic pressure on the outer silo is fully evolved (several years after closure). This will require late installation of bentonite or drainage and other types of protection, such as a waterproofing membrane, during the operating phase.

The bentonite at the bottom of the silo needs to be installed before the start of the operating phase. Otherwise there is a risk that the bentonite blocks will deteriorate during the operating phase, leading to settlement of the inner silo. The design can be improved, e.g. by replacing the bentonite blocks with a mixture of crushed rock and bentonite.

It will be difficult to check the status of the engineered barrier system once deposition of the waste has started. Quality control will be challenging, e.g. to ensure high reliability in control of the bentonite density.

8.3.9 Evaluation factor 9 – Feasibility of technology and method of operation

This evaluation factor includes the feasibility of the method, the maturity of the technology, safety during operation, and opportunities for quality control of deposited waste. The most important conclusions from the evaluation (Grahm et al. 2013) are summarized in this section.

All four repository concepts will use a remote-controlled overhead crane for emplacement of the waste in the repository. Good safety during waste disposal is thus ensured by all concepts. It will be easy to verify that the waste containers are being deposited in the predetermined positions. It will also be possible to correct and retrieve waste containers, with the possible exception of the Super silo.

Grouting during the operating phase is required for the Super silo alone in order to stabilize the stack of waste containers, whereas in the case of the other concepts, grouting is optional or can be done at any time in a campaign prior to sealing and closure of the repository. Mixing of the grout and the grouting process adds complexity to the operation of the Super silo compared to the other three concepts. In addition, verifying the outcome of the grouting process in the depth of a silo structure is a challenging task.

8.3.10 Evaluation factor 10 – Flexibility

This evaluation factor includes flexibility of technology and method of operation and flexibility of design and construction. The most important conclusions from the evaluation (Grahm et al. 2013) are summarized in this section.

All concepts, except the Super silo, are considered to be flexible in the sense that they can handle waste containers of many different sizes as long as they can be handled by the overhead crane or an alternative handling device such as a forklift. In order for the waste containers to be handled with a forklift, they must be deposited in the outermost section of the repository. However, the design of the Super silo makes deposition of containers with dimensions larger than the standard size impossible.

Once the repository has been put into operation, the repository volume cannot easily be increased in any of the concepts. It will still be possible to increase repository volume, but this requires excavation and construction of a separate rock cavern or silo. Increasing the volume of a rock cavern in which waste has already been deposited is considered inadvisable.

In summary, the Super silo is considered to be the least flexible alternative, while the other three concepts are rather similar in this respect.

8.3.11 Evaluation factor 11 – Cost

This evaluation factor includes investment costs and the costs of operation and maintenance, as well as the costs of research and development. The most important conclusions from the evaluation (Grahm et al. 2013) are summarized in this section.

It is estimated that the costs for the SFL site investigations, construction of the above-ground facilities and the rock work to reach repository level will be very similar for all four repository concepts. The difference in the required rock excavation volume for the deposition area is small between the concepts, and the rock excavation volume for the deposition area is only a fraction of the total rock volume for the SFL repository. Consequently, differences in the layout of the deposition area are of minor importance in the perspective of the total cost of a repository.

Furthermore, all supporting systems such as electricity, ventilation, physical protection, etc are independent of the choice of concept. The main requirements on these systems originate from requirements on a good working environment. In addition, the cost of operation and maintenance will be quite normal and is not expected to differ significantly between the concepts. Final sealing and closure of the SFL facility will be relatively uncomplicated and similar for the repository concepts.

The differences in cost will instead be dependent on the choice of barrier material and construction technology. With this in mind, the Super silo will be the most expensive concept, mainly due to the complexity of its design and construction. In terms of material consumption, the Gravel repository will be the least expensive alternative, in part because excavated rock can be reused as a backfilling material. The Clay repository and the Concrete repository are considered to be equally expensive in terms of investment costs.

Costs associated with research and development of materials, such formulating the concrete recipe and determining construction methods, should also be considered, even though they are small in comparison with the investment cost. Here again, it is expected that the cost of development of methods and materials for the Super silo will be higher than for any of the other concepts.

The Super silo also poses the highest project risks related to cost, due to the complexity of its design and construction and the required R&D.

8.3.12 Evaluation factor 12 – Time

This evaluation factor assesses the probability that the programme can be implemented in accordance with the timetable. The most important conclusions from the evaluation (Grahm et al. 2013) are summarized in this section.

Considering that the SFL repository is planned to be put into operation in around 2045, it is expected that development of materials and technology for all four concepts can be done in keeping with the current timetable. Backfilling is even farther off in the future, so there will be ample time to resolve any remaining issues in this area.

A Clay repository or a Concrete repository is a well-known quantity for SKB, since experience with such construction exists in the organization. The Super silo, on the other hand, is once again expected to require the greatest resources in terms of technical development. It is judged that the development of materials and construction methods for the Super silo will be more time-consuming than for the other three concepts, even though this factor is not considered to be critical.

The Super silo also poses the highest project risks related to the timetable, due to the complexity of its design and construction and the required R&D.

8.4 Summary of evaluation of repository concepts

The most important findings from the expanded assessment are summarized below. The summary is follows the four evaluation categories: *long-term safety, environment and society, technology*, and *cost and time*.

8.4.1 Long-term safety

The *Clay repository* is expected to have less ability to retard the release of radionuclides compared to the Concrete repository initially. However, it is considered easier to achieve the defined initial state and also to achieve an understanding of the degradation processes in the Clay repository compared to the Concrete repository. The Clay repository is considered more robust over time compared to the Concrete repository, with a lower expected degradation rate. The chemical environment is affected by the concrete structures used during the operating phase, which may have detrimental effects on the swelling properties of the clay.

The *Concrete repository* is expected initially to have good ability to retard the release of radionuclides by limiting advective flow and providing strong sorption and a low diffusion rate, allowing radionuclides to decay during transport through the barrier. The chemical environment in the repository is greatly affected by the concrete, creating a high-pH environment. This is a major advantage since it affects the corrosion rate for the metallic waste, and a corrosion-rate-limited source term can be used in the safety assessment. The uncertainties associated with the concept are mainly related to the rate and manner in which the barrier functions degrade.

The *Gravel repository* is based on the idea of a hydraulic cage, which offers limited potential to control the outflow of radionuclides. Diffusion through the waste volume is driven by the concentration gradient created towards the waste surface. The radionuclides that reach the surface are transported immediately with the fastest flow path. This means that the Gravel repository will probably not offer sufficient safety, but the concept may still be of interest if combined with another concept.

The *Super silo* is expected to have good ability to retard the release of radionuclides by limiting advective flow and providing strong sorption and a low diffusion rate, allowing radionuclides to decay during transport through the barrier system. However, the concept entails many uncertainties related to both its initial state and its long-term evolution. The chemical and mechanical effects of the interaction between barriers need further studies. It will require a lot of R&D to achieve an adequate understanding of the processes governing the evolution of the concept.

8.4.2 Environment and society

The difference in the environmental impact between the concepts is relatively small, since they are all geological repositories. For example, the rock excavation volumes are largely determined by the chosen repository depth, since the disposal volume is relatively small. The volume of the deposition area accounts for only 15–25 percent of the total rock excavation volume at a disposal depth of 500 metres. Transportation, which will comprise a significant part of SFL's future environmental impact, is largely determined by the choice of site and is not concept-specific. The *Gravel repository* is considered to be the solution that has the least environmental impact due to the reuse of crushed rock and small haulage needs. However, this concept will be challenging to communicate and create acceptance for. It may be difficult to argue for a repository that does not have a dense barrier.

8.4.3 Technology

The *Clay repository* and the *Concrete repository* both have relatively good technical maturity, and fairly straightforward technology development is expected to overcome the remaining challenges.

The *Gravel repository* is considered to be the easiest concept to design and build. One weakness is the absence of radiation protection during the operating phase, but this can probably be resolved relatively easily.

The *Super silo* is a very complex design that requires more extensive technical work to be built. The silo in the proposed design and size cannot withstand the loads on the outer concrete cylinder during the water saturation phase and subsequently when the bentonite swells. A significant reduction of the radius of the cylinder is needed to save the basic design, but at the cost of a very large reduction in available disposal volume. A number of silos would need to be built to achieve sufficient disposal volume. Bentonite clay with less pronounced swelling properties can probably be used between the concrete cylinders to reduce stresses on the outer concrete cylinder caused by the swelling clay, but the load from the groundwater during the water saturation phase is still a tough design criterion to meet. The uncertainties and risks associated with the development of the Super silo are considered to be the highest of any of the studied alternatives. The technical feasibility of the concept can be improved if simplifications are made in the design (e.g. omitting the outer concrete silo).

The differences in operating characteristics and flexibility between the four concepts mainly relates to the silo form of a Super silo repository.

8.4.4 Cost and time

The cost and time required to develop and build a specific concept is closely related to the complexity of the design. The *Gravel repository* is considered to be the least expensive concept and the *Super silo* the most expensive, both to develop and to build. Operating costs are expected to be similar between the four alternatives. The project risks associated are also related to the complexity of the design. The *Gravel repository* is thus associated with the lowest risks and the *Super silo* with the highest risks.

9 Conclusions

The evaluation of repository concepts presented in Chapter 8 shows that the Clay repository and the Concrete repository are the most promising concepts among the ones evaluated. They both exhibit potential to maintain radiological safety in the long term, in combination with feasible technical design and the use of mature technology.

The long-lived low and intermediate level waste comprises several waste fractions with different properties. To ensure long-term safety, different repository solutions may be considered for the various waste fractions. A large fraction of the waste consists of segmented components of steel and stainless steel, which will suffer from corrosion in the repository. The release mechanism for the activated radionuclides bound in the steel can be controlled by controlling the corrosion process. Since these radionuclides have been shown to have a high impact on long-term safety (SKB 1999a, b), control of the corrosion process is vital. The Concrete repository concept exhibits properties that are considered beneficial for this waste fraction. The concrete in the barrier will create an alkaline environment in the repository section, reducing the corrosion rate of the steel and thus limiting the release rate of radionuclides.

Another large fraction of the waste is already conditioned. For this waste fraction, corrosion is not considered the main release mechanism. The impact of this waste on long-term safety can be limited by limiting the transport rate for radionuclides. The groundwater flow can be limited by the use of a dense barrier, and diffusion will be the main transport mechanism for radionuclides. Slow diffusion and high sorption capability in the barrier will allow for important radionuclides to decay during transport. The Clay repository concept exhibits properties that are considered beneficial for this waste fraction. The clay in the barrier will contribute to low hydraulic conductivity, slow diffusivity and strong sorption. When the creation of an alkaline environment in the repository is of limited importance, the Clay repository concept is considered a more robust solution in the long term.

There are a variety of conditioning methods, and their effects have to be evaluated for each fraction of the waste at hand. The effect of conditioning on properties related to transport and handling, long-term safety etc has to be weighed against the environmental and economical impact of the procedure and the expected dose to personnel. Caution should be exercised before extensive treatment and conditioning methods are proposed. It is furthermore vital to consider the entire management chain for the waste, including conditioning methods, when comparing potential solutions for management and disposal.

Based on the evaluation in Chapter 8 a complete system for management and disposal of the long-lived low and intermediate level waste has been devised. The SFL system includes the waste and containers, a system for external transportation, facilities for storage and conditioning of the waste, the repository facilities (prior to closure) and the repository itself. Section 9.1 provides an overview of the proposed system and its components. This system is proposed for further assessment of long-term safety.

Section 9.2 presents a more detailed description of the repository itself and the barriers designed to ensure the long-term safety of the repository. The rationale for the proposal is presented.

Additional elements that may be added to the repository system have been identified and are summarized in section 9.3. The elements are conditioning methods that can be added to the proposed concept to provide additional or strengthened safety functions in the repository.

The consequences of the proposal are outlined in section 9.4, including the flexibility of the system to embrace disposal of whole components.

9.1 Proposed SFL system for further assessment of long-term safety

The proposed system for disposal of the Swedish long-lived low and intermediate level waste consists of the following components:

Waste and containers

The waste is separated into two fractions:

- metallic waste from the nuclear power plants, and
- waste from AB SVAFO and Studsvik Nuclear AB.

The metallic waste from the nuclear power plants, including BWR control rods, is segmented and placed in steel tanks and stabilized in the tanks with grout.

The waste from AB SVAFO and Studsvik Nuclear AB is placed in containers designed for SFL (described in section 5.3.2) and stabilized in the containers with grout.

System for external transportation

The system for external transportation is based on SKB's current system for external transportation. Future planning, including selection of the repository site, will decide which additional parts are needed to fully enable the transportation of SFL waste, e.g. waste transport containers.

Facilities for conditioning of waste

To implement the proposed solution up to three conditioning facilities will be built. The conditioning facilities will:

- segment BWR control rods and load the segments into steel tanks,
- stabilize metallic waste in steel tanks with grout, and
- load waste from AB SVAFO and Studsvik Nuclear AB into SFL containers.

The loading station for waste from AB SVAFO and Studsvik Nuclear AB should be located at the Studsvik site, since it already hosts this waste fraction. The location of the other facilities will be subject to further analysis, where co-location with the loading station is one of the options to be considered.

Storage facilities

Storage of SFL waste is managed by making use of existing storage capacity at the nuclear power plant sites and in Studsvik, and planned storage capacity in SFR (see section 4.4). Additional storage facilities may be added in the future.

Repository and repository facilities

SFL is designed as a deep geological repository with two different sections:

- one section for the metallic waste from the nuclear power plants, and
- one section for the waste from AB SVAFO and Studsvik Nuclear AB.

The repository section for the metallic waste is designed with a concrete barrier. The waste is segmented, after which the parts are deposited in steel tanks (see section 5.1.2) and stabilized with grout. This section of the repository is backfilled with concrete, which acts as a barrier against hydraulic flow and contributes to a low diffusion rate and high sorption of many radionuclides. The concrete in the barrier will create an alkaline environment in the repository section, reducing the corrosion rate of the steel and thus limiting the release rate of radionuclides.

The repository section for the waste from AB SVAFO and Studsvik Nuclear AB is designed with a bentonite barrier. The waste is deposited in containers designed for SFL (see section 5.3.2) and stabilized with grout. These containers are emplaced in the repository. The section is backfilled with bentonite, but there will be no bentonite in the repository during the operating phase. The bentonite acts as a barrier by limiting the hydraulic flow and making diffusion the dominating transport mechanism for radionuclides through the bentonite. Bentonite clay also has the ability to effectively filter colloids.

9.2 Proposed repository and barrier system

The proposed design includes two separate sections, each of which will accommodate one fraction of the waste. The design of each repository section is adapted to the specific properties of the waste in the individual section. This section provides a more detailed description of the barriers designed to ensure the long-term safety of the repository and explains the intended function of the barriers.

9.2.1 Repository section for the metallic waste

The repository section for the metallic waste contains the metallic waste from the nuclear power plants – mainly neutron-irradiated components such as BWR control rods, core components and PWR pressure vessels. This part of the inventory comprises approximately 3,000 tonnes of steel and stainless steel, corresponding to approximately 5,000 m^3 of disposal volume, and contains more than 99 percent of the estimated activity of the full SFL inventory. The nuclide inventory also contains most of the activation products that were identified in the preliminary safety assessment in 1999 (SKB 1999a, b) as contributing significantly to the dose burden of the critical group, e.g. Cl-36 and Mo-93. These activation products are mainly induced activity bound in the steel and stainless steel and to a lesser extent surface contamination.

This section of SFL will be fully backfilled with concrete and plugged at the end of the operating phase. The primary safety function of the repository section is *retardation*. The design principles of the barriers for long-term safety are summarized in Table 9-1, which shows the key elements of this repository section with respect to long-term safety, and how the elements are supposed to contribute to the primary safety function.

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 (SKB 2013a). 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 the barrier. The concrete has a high sorption coefficient for many radionuclides (although not for the anions formed by chlorine and molybdenum). The diffusion rate through concrete is low for many important radionuclides. In the pristine concrete, the hydraulic conductivity is also very low, which makes diffusion the dominant transport mechanism for radionuclides through the concrete.

Table 9-1. Design principles of the barriers for long-term safety for the repository section for metallic waste.

Element	Safety functions
Waste and containers	– Limited release rate – Strong sorption (grout and corrosion products)
Concrete barrier	– Low hydraulic conductivity – Low diffusivity – Strong sorption – High-pH conditions

Since most of the activity is bound in the metal, it is possible to postulate a dissolution rate – equal to the corrosion rate – as the source term in the assessment of long-term safety. Considering the large amounts of concrete present in the repository section, the pH can be expected to be buffered for a long period of time, providing a high pH environment. This should be a robust safety function for the steel and stainless steel in the waste. The amount of concrete backfill will be conservatively calculated to guarantee its buffering capacity for as long as needed.

Concrete is a well-known material for construction of underground structures and is used for example in SFR. Limited R&D efforts are foreseen to address remaining issues regarding the environmental impact of the concrete structures. Acceptance is expected to be obtained for the proposed barrier solution.

The concrete repository section can be built using known and mature materials and methods. The repository section can be constructed either as a cavern or as a silo, offering flexibility to design the repository based on local conditions. There is also flexibility in the choice to build one single large cavern or silo or several small ones. Functionality and radiation protection during the operating phase are considered fully adequate. Grouting of the waste is considered a mature technology and may be implemented in both cavern and silo. An overhead crane, running on tracks mounted on separate pillars, may be used for deposition of the containers.

If a *cavern* is chosen as the preferred geometry of the repository, the conceptual design of the repository section for the metallic waste follows the design of the *Concrete repository*, as described in section 7.2.2. The cross-sectional layout of the repository section for the metallic waste designed as a cavern is shown in Figure 9-1. The length of the repository is subject to detailed design, but should, given the current estimate of the disposal volume, be approximately 75 metres, including unloading area.

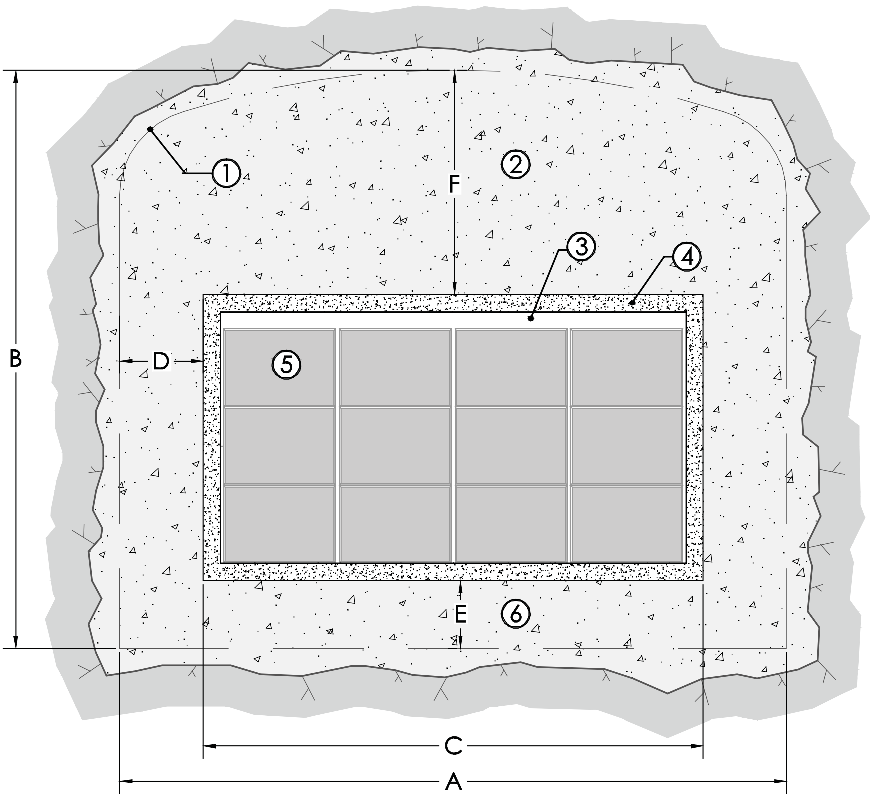


Figure 9-1. *Schematic cross-sectional layout of the repository section for the metallic waste designed as a cavern. Legend: 1) Theoretical tunnel contour. 2) Concrete. 3) Grout. 4) Reinforced concrete (0.5 m). 5) Steel tank. 6) Concrete. Approximate dimensions: A = 20 m, B = 17 m, C = 15 m, D = 2–3 m, E = 2 m, F = 5–10 m.*

The R&D efforts needed prior to designing and building the repository section are judged relatively limited since proven technology will be used. R&D will be necessary to gain a better understanding of the initial state and the long-term degradation of the engineered barriers and to describe the initial state and the processes in a safety assessment. The relatively well-defined source term and the straightforward design of the engineered barrier provide good premises for achieving a good understanding of the processes in the repository.

The building cost of the repository section itself is believed to be on a par with other concrete-based repository sections, such as the rock cavern for intermediate level waste (2BMA) in the planned extension of SFR, although with a larger quantity of concrete. There are few big project risks associated with the proposed solution, which thus should be possible to implement on time and budget.

9.2.2 Repository section for the waste from SVAFO and Studsvik Nuclear

The repository section for the waste from AB SVAFO and Studsvik Nuclear AB contains the legacy waste from the early research in the Swedish nuclear programmes and the long-lived waste from hospitals, industry and research. This section of the inventory has a total disposal volume of approximately 11,000 m^3 and contains less than 1 percent of the estimated activity of the full SFL inventory. This part of the inventory also contains chemotoxic waste comprising cadmium, lead, beryllium, and mercury.

This section of SFL will be fully backfilled with bentonite at the end of the operating phase and plugged. The bentonite backfill forms the engineered barrier for this repository section. The primary safety function of the repository section is *retardation*. The design principles of the barriers for long-term safety are summarized in Table 9-2, showing the key element of this repository section with respect to the long-term safety, and how the element is supposed to contribute to the primary safety function.

The bentonite permits only a small advective flow, which makes diffusion the dominant transport mechanism for radionuclides through the bentonite. The initial state of the barrier is relatively easily described, in part because it is an established research field, in part due to the planned late installation of 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 and also exhibits self-healing properties. Bentonite is a natural material, and its long-term properties are well characterized 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 to the metallic waste. Actinides, together with mixed organic material, could form different types of actinide-carrying colloids that could be transported out of the repository. All in all, this makes bentonite a very robust barrier.

Bentonite has also been well characterized from an environmental viewpoint in the work carried out in SKB's other repository projects. Limited R&D efforts are foreseen to address remaining issues regarding environmental impact. Acceptance is expected to be obtained for this barrier solution as well.

Table 9-2. Design principles of the engineered barrier for long-term safety for the repository section for waste from AB SVAFO and Studsvik Nuclear AB.

Element	Safety functions
Bentonite barrier	– Low hydraulic conductivity – Low diffusivity – Strong sorption – Filtering colloids

A repository section based on massive use of bentonite is deemed possible to design and construct both as a cavern and as a silo. The new containers developed for the legacy waste are designed to give the waste the mechanical stability required by a technical solution based on swelling clay. Grouting between containers can be done in the same fashion as in the silo in SFR today. Some structures made of concrete will be needed in the operating phase. The waste containers need to be placed on a flat surface with support so they can be stacked to the desired height. The concrete structures are also necessary to provide adequate radiation protection. An overhead crane, running on tracks mounted on separate pillars can be used for deposition of the containers.

If a *cavern* is chosen as the preferred geometry of the repository, the conceptual design of the repository section for the waste from AB SVAFO and Studsvik Nuclear AB follows the design of the *Clay repository*, as described in section 7.2.1. The cross-sectional layout of the repository section for the waste from AB SVAFO and Studsvik Nuclear AB designed as a cavern is shown in Figure 9-2. The concrete structures intended to provide support for the waste containers and act as radiation protection during the operating phase are indicated. The length of the repository is subject to detailed design, but should, given the current estimate of the disposal volume, be approximately 150 metres, including unloading area.

A slightly greater effort is required for technical development to design and build a clay-based repository compared to a concrete-based one. In return, the R&D efforts required to understand the initial state and the degradation of the engineered barrier are deemed smaller. The source term comprises the main uncertainty connected with this repository section. The nuclide and material inventory of the legacy waste is not fully characterized and current and future demands on the waste contribute to the project uncertainties. The straightforward design of the engineered barrier provides good prospects for achieving a good understanding of the processes in the repository. The project risks are deemed somewhat higher compared to the concrete repository section, partly due to a new construction, partly due to the source term.

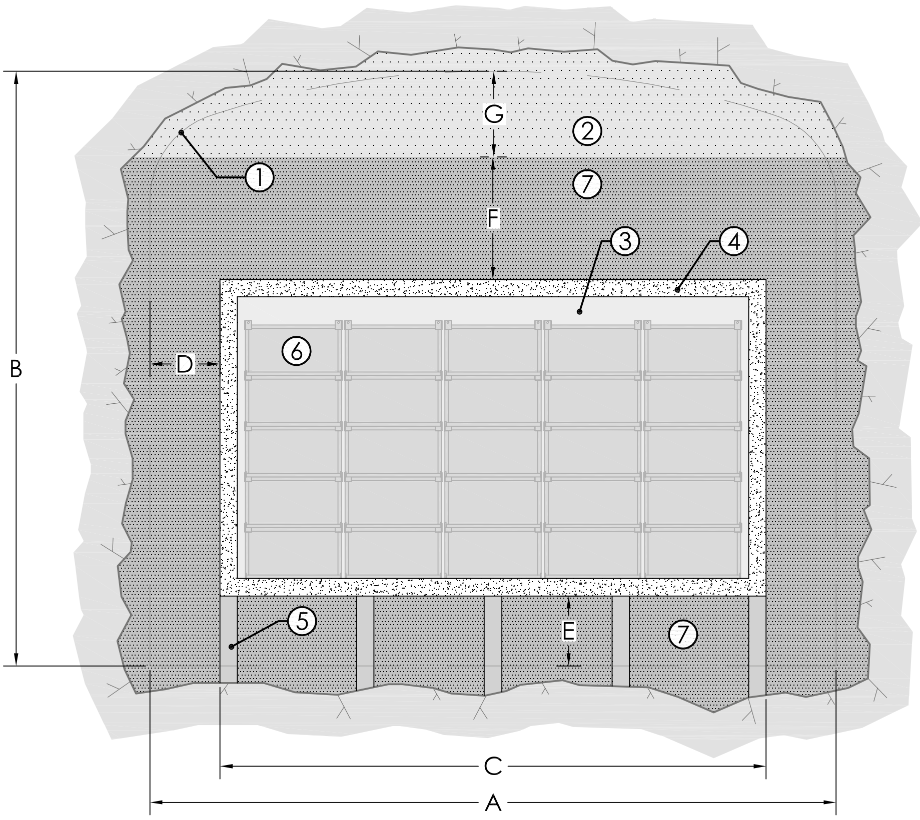


Figure 9-2. *Schematic cross-sectional layout of the repository section for the waste from AB SVAFO and Studsvik Nuclear AB. Legend: 1) Theoretical tunnel contour. 2) Bentonite pellets. 3) Grout. 4) Concrete structure for the operating phase (0.5 m). 5) Granite pillars. 6) Waste container. 7) Bentonite blocks. Approximate dimensions: A = 20 m, B = 17 m, C = 16 m, D = 2 m, E = 2 m, F = 3–4 m, G = 2–3 m.*

9.2.3 General characteristic of the repository

Disposal depth

The repository will be located deep in the bedrock, at an appropriate depth. It is not within the scope of the concept study to establish the depth of the future SFL. However, certain factors that need to be considered in the future analysis can readily be identified:

- Isolation from future human interaction.
- Hydrogeological conditions.
- Future glaciations.
- Constructability.

The risk of future human intrusion decreases with increasing disposal depth. The hydrogeological conditions are very site-specific, and experience shows that hydraulic transmissivity can vary with depth. This will influence the release rate of radionuclides to the geosphere. Freezing has a strong detrimental impact on concrete. To avoid freezing the repository need to be established at a depth where it will not be adversely affected by permafrost during future glacial cycles. The ability to build underground structures is affected by the mechanical properties of the host rock, e.g. stress, and is thus also site-specific.

Considering these factors and the time frames associated with the long-lived waste, the current assumption is that SFL needs to be located at a depth similar to the depth of the repository for spent nuclear fuel. Local conditions and future site-specific safety analyses will determine the exact repository depth.

Geometric configuration of disposal rooms

The geometric configuration of the disposal rooms – cavern or silo – will be decided on when the site is known. Both options are deemed possible to design and build. In the forthcoming evaluation of the long-term safety, the geometric configuration is of limited concern.

Interaction between repository sections

The two repository sections are located at a suitable distance from each other, with the section for the metallic waste placed "downstream" of the section for waste from AB SVAFO and Studsvik Nuclear AB. This is to limit the influence of the alkaline plume from the concrete on the bentonite. Again, local conditions and future site-specific safety analyses will determine the exact distance.

Means of access

The central area at repository depth is accessed via a single ramp. A shaft is built for transportation of personnel and as an emergency escape route. This solution has been proposed for the repository for spent nuclear fuel and is presented mainly as a feasible solution for SFL as well and a starting point for further analyses. The means of access has no impact on the evaluation of repository concepts at this stage. The various options need to be conceived to further analyze the consequences of disposal of large components without segmentation, i.e. PWR pressure vessels. There will also be further input on the matter once the site has been selected.

Backfill and closing

The central area, ramp, shafts etc will be backfilled and plugged at the end of the operating phase. The entrances to the repository will be sealed in a way that prevents intrusion and ensures that post-closure surveillance will not be needed. The details regarding backfill and closure of the repository will be worked out in the future. Knowledge from ongoing development projects associated with both the repository for spent nuclear fuel and SFR will serve as basis for this work.

9.3 Possible additional elements in the system

Possible additional elements include solutions that have not been rejected by the study and can be utilized to strengthen the repository from a long-term safety perspective. These components are not part of the original proposal for reasons such as low maturity, high cost or high expected dose burden during operation. Should concerns for the long-term safety warrant increased R&D efforts, cost or dose burden during operation, these additional components may be added to the proposal.

9.3.1 Possible additional elements in the repository section for metallic waste

The following possible additional elements have been identified in the section for the metallic waste:

- Melting of the metallic waste to well-defined ingots, which are deposited (see section 5.1.5).
- Use of long-term durable containers (see section 5.1.4).
- Addition of agent to chemically improve retention (see section 5.5).

The original proposal is that the metallic waste is segmented and deposited in steel tanks and stabilized in the tanks with grout. If subsequent assessments of long-term safety show a need for further measures to meet the regulatory risk criterion, it is suggested that these solutions be considered for further studies. Both melting and long-term durable container can, according to this study, only be warranted by explicit needs to improve the long-term safety of the repository, since they are deemed both costly and time-consuming. Melting is also associated with increased dose to personnel. Melting of the metallic waste requires a decay time of at least 50 years to allow handling in a melting facility, as well as a new melting facility. The use of a sealed container would add the safety function *containment* to the repository, but would in return entail stricter quality control of the welding process, extensive R&D costs, a new conditioning facility for loading, stabilization and welding, and a very high production cost per unit. The use of additional material to chemically improve retention may be an attractive route to increase the long-term safety of the repository, but requires extensive R&D efforts to become reality.

9.3.2 Possible additional elements in the repository section for waste from SVAFO and Studsvik

The following possible additional elements have been identified for the waste from AB SVAFO and Studsvik Nuclear AB:

- Retrieval and sorting of legacy waste (see section 5.3.3).
- Vitrification or similar (see section 5.3.4).

These elements apply primarily to the legacy waste, which already is conditioned today by grouting. The original proposal only suggests the use of SFL containers to achieve rational handling and transportation, and apart from that as little handling of the waste as possible. If it is not possible to manage remaining uncertainties concerning the composition of the legacy waste in future safety assessments, for example by further processing of existing data, sorting of the waste may be needed to divide the waste into appropriate fractions. More advanced reconditioning plants are considered expensive and time-consuming, involving construction of new facilities etc.

9.4 Consequences

9.4.1 Requirements on separation of waste fractions

The waste categories that need to be separated in the proposed solution – metallic waste and waste from AB SVAFO and Studsvik Nuclear AB – are more or less already separated, since they come from different waste producers. The two waste categories will be handled in different sets of containers, differing in geometric shape, and the risk of confusing those containers is minimal.

9.4.2 Site requirements

Since the proposed repository depends on the primary safety function *retardation*, detailed requirements on the properties and conditions of the site is expected from future safety assessments, for example requirements on the mechanical, chemical and hydrological conditions at repository depth. A site with low hydraulic conductivity is beneficial in terms of both degradation rate and release rate.

9.4.3 Potential for stepwise construction

The proposed repository design does not impose any restrictions on the pace or the order in which the repository sections are built. The repository sections can be built at the same time, or in sequence.

9.4.4 Flexibility in the design

As of now, Ringhals AB is planning to deposit intact PWR pressure vessels in SFL (including reactor internals). In the proposed system for further analysis of the long-term safety it is assumed that PWR pressure vessels are segmented and placed in steel tanks in the same fashion as the other metallic waste. This proposal should not be seen as a decision on the details of the repository or the handling of the waste, but rather a base design to be used as a starting point for further work.

The next step, the safety assessment, will determine the conditions under which the current repository concept can provide for safety in the long-term. Following the safety assessment it will be possible to develop the base design, by for example adding repository sections with equivalent barrier solutions or by using other handling alternatives for the waste. A detailed analysis of the entire handling chain for the PWR pressure vessels need to be conducted to investigate the technical possibilities for the handling and disposal of whole tanks, and highlight the consequences in all steps in terms of dose to personnel and cost.

If deposition of whole reactor pressure vessels in SFL is preferred over segmentation, then the repository space where whole pressure vessels will be emplaced can inherit the same barrier solution as the solution developed for the metallic waste.

10 Proposed research and development programme

The proposed SFL system includes engineered barrier systems for the two repository sections based on concrete and bentonite. As concluded in the evaluation, repository concepts based on concrete and bentonite both require research and development work both to build them and to support future safety assessments.

Grahm et al. (2013) have identified research needs related to the development of materials and methods of construction of the engineered barrier system, including backfill and plugs. Besides identifying work of direct importance for the repository concepts, this study has also identified other areas which will require future attention such as the design of waste containers, conditioning methods and technical systems for the repository facilities. Research needed to provide a scientific basis for future safety evaluation and safety assessments have been identified in Evins (2013).

The following sections present plans for research and development of importance to the SFL system. The design and construction of the engineered barrier system and waste containers as well as other technology-related issues are dealt with in section 10.1, while the scientific research programme related to safety assessments is presented in section 10.2. The plans are only preliminary at this stage and will be presented in greater detail when the evaluation of the long-term safety of these concepts has been concluded.

A summary of the research and development conducted by SKB, related to both SFL and other repositories, is presented in the RD&D programme (SKB 2013a).

10.1 Technology development

Grahm et al. (2013) have addressed a number of technical challenges related to choice of material, design and construction of the Clay repository and the Concrete repository. Examples of factors of importance are the composition of the concrete in the engineered barriers in the Concrete repository and the method of installation of the bentonite in the Clay repository. The programme for technology development for the Concrete repository and the Clay repository is briefly outlined in the following sections.

Concrete repository

The long-term safety of the Concrete repository depends on the fact that the dominant transport mechanism for the radionuclides is diffusion in a stagnant medium, i.e. groundwater in the pore volume in the concrete. With diffusion as the dominant mechanism, the active contact area in the engineered barriers will be great and hence the capacity for sorption of radionuclides will also be great. If cracks are formed in the barriers, the radionuclides will instead be transported through the cracks in a flowing medium, groundwater, and the contact area will be dramatically reduced, as will the sorption capacity of the engineered barrier system.

For this reason, the focus when developing the composition of the concrete and the method for casting the concrete structures in the repository is to limit the formation of cracks and other imperfections such as joints etc during construction, and to prevent cracks from occurring during the post-closure period.

Composition of the concrete

The most important issue in formulating the composition of the concrete is to minimize shrinkage of the material, and several approaches can be used alone or in combination to achieve this. The most important factors are the water/cement ratio, the type and particle size of the aggregate, mineral additives such as fly ash, and the amount of cement in the concrete. The programme for development of concrete for the engineered barriers in SFL will need to include all these factors to ensure that the concrete will meet the requirements specific for SFL.

Construction methods

Construction of the engineered barriers includes aspects such as choice of reinforcement, formwork design and material, and casting strategy. Properly executed construction can reduce imperfections in the structure such as joints and formwork struts and improve the long-term properties of the engineered barriers.

The choice of construction method is between traditional casting and slip-form casting. Slip-form casting was used for casting of the silo in SFR, whereas traditional casting was used for the other concrete structures in SFR. Historically, traditional casting has required the use of formwork struts but has also led to the formation of joints between different sections of the concrete structure. In slip-form casting, no form struts are used and joint formation is reduced owing to the continuously moving casting process. Careful evaluation is needed to decide which of these methods is most suitable for the future SFL.

Reinforcement of the concrete structures

In order to withstand tensile forces, concrete structures are reinforced using steel bars which are embedded in the concrete structure. Over the very long post-closure life of a repository for nuclear waste, the reinforcing bars will corrode and channels or cracks will form in the concrete. This could compromise the functional properties of the engineered barrier, potentially affecting the release rate of radionuclides.

Different methods can be used to avoid this. The most obvious is to refrain from using any reinforcement at all, at the risk of compromising the tensile strength of the engineered barriers. Other options involve the use of alternative materials such as mineral or plastic fibres. The future development programme will decide which of these methods will ultimately be used.

Grout and method for grouting of the waste containers

All space between the waste containers, as well as between the waste containers and the walls of the repository, will be filled with a cementitious grout such as is used today in the silo in SFR. The purpose is to stabilize the waste container stack, reduce the void volume in the repository and provide support to the concrete structures by backfilling and saturation of the repository. The detailed properties of this grout and the method used for grouting have to be determined during the design phase, but aspects of importance are compressive strength and gas permeability as well as rheological properties.

Backfilling of the repository

The Concrete repository will be backfilled with concrete or grout. Different methods can be used to achieve backfill with the desired properties. The interface between rock and concrete is a recognized challenge. Will a shrinkage gap arise and should it then be filled with something? Basically, two different approaches can be identified for the backfilling process: filling the entire repository with wet grout, or filling most of the repository with pre-cast concrete blocks and then filling the space between them with wet grout. The future development programme needs to investigate the details of the methods and which method is the most suitable.

The logistics of transporting large quantities of concrete for backfill also needs to be examined.

Plugs

Plugs will be installed at selected locations in the repository. However, in this section only plugs located between the Concrete repository rock cavern and the adjacent transport tunnel are considered.

The design of the plug between the repository rock cavern and the adjacent tunnel must take into account the fact that the entire repository section is filled with concrete or grout that needs no mechanical support but that serves as a medium for gas transport. These and other issues related to the function of the plug will serve as premises for the design specifications for the plug. The programme for design and construction of plugs in the Concrete repository will be formulated during the design phase.

Clay repository

The long-term safety of the Clay repository depends on the fact that the waste is surrounded by a material with very low hydraulic conductivity and that interaction between the waste and the groundwater is limited during the entire operating lifetime of the repository. In order to achieve this, great care has to be taken in choosing the type and form of the bentonite as well as the method and time for installation of the bentonite.

Type of bentonite

The choice of bentonite quality will affect the swelling pressure and the long-term properties of the material. The first task will be to identify the requirements and translate them into desired bentonite properties, such as dry density, swelling pressure and hydraulic conductivity.

Form of bentonite

Depending on the required properties of the bentonite, blocks or pellets or a combination thereof can be used in the Clay repository. The form of the bentonite will affect the method of installation as well as the final swelling pressure of the material.

Installation of bentonite

The method used for installation of the bentonite in the repository will be dependent on the form of the bentonite, i.e. blocks, pellets or a combination thereof. Methods for installation of blocks are being developed for backfilling of the deposition tunnels in the Spent Fuel Repository, and equipment will probably be available by the time of backfilling and closure of the Clay repository.

The most complicated task will be the installation of the bentonite underneath the waste containers. The proposed design for the repository implies installation of the bentonite during backfilling of the repository, to ensure a well-defined initial state. However, also the option of installing the bentonite during construction of the repository will be further evaluated. Both options have their benefits and drawbacks, and the choice of method influences the design of the repository and vice versa. The question of which method will be used must be evaluated carefully already during the design of the repository.

Plugs

Plugs will be installed at selected locations in the repository. However, this section considers only plugs located between the Clay repository rock cavern and the adjacent transport tunnels.

Since the entire repository section is filled with bentonite, the most crucial design criterion for the plug is the ability to provide strong enough mechanical support to keep the bentonite in place. This and other functions of the plug will serve as premises for the design specifications of the plug. Extensive work on plugs, including full-scale experiments, is already being pursued as a part of the development work for the Spent Fuel Repository, and the technical programme for SFL will benefit from this work.

Waste containers

The low and intermediate level waste destined for disposal in SFL is currently being stored in many different types of containers such as drums, moulds and steel tanks, but also in open boxes and other containers. In order to provide an efficient and safe transportation system but also to facilitate handling in the repository, there is a need for a system of uniform containers in which all waste, including the containers being used today, can be placed.

A new set of waste containers is proposed in Pettersson (2013). All waste containers will have the same lateral dimensions but will differ in height, wall thickness and internal dimensions. This will enable all waste containers to be handled with the same equipment in the repository and transported in the same waste transport containers between the different nuclear facilities and the repository.

Preliminary, containers for waste packed in drums and moulds will be made of corrugated steel, as described in section 5.3.2. These types of containers can be manufactured using standardized materials and methods and will require very little development work.

If the long-term safety of the repository requires the container to provide an additional barrier function, a long-term durable container made of 100 mm thick steel plates joined by means of a special welding method can be used (see details in section 5.1.4). However, welding of 100 mm thick steel plates is a very complex task requiring extensive development work on automatic welding procedures for thick materials, inspection methods etc.

Handling of special waste

Certain categories of waste have been identified as requiring further attention:

- An overall evaluation is needed to examine the benefits and consequences of disposing of PWR pressure vessels in one piece in SFL. This option must be compared with the option of segmenting and packing the reactor pressure vessels in waste containers.
- The method for segmentation and packing of BWR control rods in containers suited for SFL needs further development. Today, most of the used control rods are stored in Clab. A study is needed of the feasibility of carrying out segmentation and packing in or adjacent to Clab.
- Management of spent core instrumentation from the nuclear power plants needs to be studied further to include packing in the appropriate waste containers for disposal.

Technical systems

Development and design of processes, technical systems and equipment will be carried out according to well established routines and standards. The development work must be carried out methodically in order to ensure that all requirements are identified, satisfied, verified and validated. Design of technical systems must also take into account human capabilities and limitations (Human Factors Engineering). This is an important part of the development of new processes and technologies to ensure safe and reliable operation.

The initial steps for SFL will be to prepare a work plan for integration of Human Factor Engineering in the design of the facility while at the same time start studying processes and production systems in order to optimize activities, work flow, logistics, functions and requirements.

Furthermore, some strategic issues need to be resolved at an early stage. Examples include the level of automation of handling equipment, means of access (via ramp or shaft) and identification of bottlenecks in the logistics.

A preliminary design of the repository will be prepared with layout and description of technical systems, both on the surface and underground.

10.2 Research and safety analysis

The safety assessment of any geological disposal facility for nuclear waste is based on scientific knowledge gained from research experiments, modelling and field studies. The safety assessment needs to be kept up to date with the latest international developments and take into account new information and knowledge resulting from continuous research efforts. Research is also required to determine and reduce the level of uncertainty associated with the safety assessment. A stable and long-term research programme needs to be linked closely to the safety assessment for the purpose of supporting the safety case and providing confidence in the conclusions of the assessment. Knowledge gained from studies both in the laboratory and in the field must be combined and incorporated into the modelling of the repository system.

Thus, it is expected that research requirements will continue to emerge as the work with the SFL safety assessment continues. With this in mind, the current report aims only to provide a short description of the areas which would need attention in the future work and general research needs identified up until now. How these needs will be met is outside the scope of this report. A specific research programme, identifying more detailed research efforts, can be expected in the coming years.

The following text is organized in two sections. The first concerns the scientific research needs identified for the development of the man-made part: the repository with the engineered barriers. The second concerns the natural environment: the surrounding bedrock (natural barrier) and the surface system. Changes in climate will affect both the natural and the engineered environment, so a fundamental cornerstone of the research programme is climate studies.

The repository

In the repository, waste will be emplaced in a central structure surrounded by one or more engineered barriers that contribute to safety. The research programme will therefore be concerned with the long-term performance of the waste package and the engineered barriers: concrete, bentonite and, possibly, crushed rock (gravel). In addition, research may be required to improve the waste form by conditioning. The main safety function of the engineered barrier system is *retardation* of released radionuclides. This implies that all features, events and processes which may affect the barriers with respect to this main safety function need to be identified. The main concern revolves around the interaction between the released radionuclides and the barrier material over the time period covered by the safety assessment. Thus, research efforts are needed for establishing and modelling chemical behaviour of the SFL-specific key radionuclides in the expected repository environment.

The research programme aimed at studying the long-term performance of concrete in a repository environment is described in the RD&D programme 2013 (SKB 2013a) and is also a vital part of the safety assessment of the planned extension of SFR.

A detailed understanding of the mechanisms and time scales associated with the chemical and mechanical degradation of the concrete engineered barriers is of key importance for the safety assessment. An understanding of the mechanisms behind individual chemical processes such as leaching, carbonation and sulphate attack exists today, but the detailed effects of these processes, as well as the effects of combinations of them, are less well understood, which is also true of the time scales associated with these processes. Finally, the effects of these processes on the mechanical properties of the concrete over time also require further investigation.

Several research projects are currently under way aimed at improving our understanding of these processes. In the project "Chemical and mechanical properties of aged cement", specimens are being prepared by means of an electrochemical leaching method. In this way, the portlandite and some of the CSH gel can be leached out of the specimen. The mechanical properties of the specimens can then be studied as a function of the degree of leaching, and an estimated time scale can be established for the development of the mechanical properties of the concrete engineered barriers.

Further studies needed of the evolution of the properties of the concrete engineered barriers concern the influence of interactions between the concrete and species resulting from the degradation of the different waste forms. Experiments focusing on these issues were installed in the Äspö HRL as a part of the project "Concrete and Clay" and will be retrieved and analyzed at regular intervals to chart the progress of the degradation process over time.

SKB is also participating in the international research project Long term Cement Studies, LCS, coordinated by the Swiss organisation NAGRA. The main efforts in this project are focused on increasing the understanding of processes involving the interactions between ground water, cementitious materials and minerals in the bedrock. Of processes of particular interest within this project are precipitation of secondary minerals in cracks and mineral alterations of the bedrock due to interactions with the high-pH cement pore water. This is done through a combination of experiments – both in the Grimsel URL and in different laboratories – and thermodynamic modelling of observations made in natural analogues such as the Maqarin natural analogue in Jordan.

Finally, even though our current understanding of permafrost suggests that locating SFL at a depth of 300–500 metres below the ground surface will be sufficient to avoid freezing due to permafrost development in the future, detailed knowledge is required concerning the effects of freezing of concrete in a repository environment. This is due to uncertainties regarding the site, especially at what depth permafrost can be expected over the time covered by the safety assessment. The effects of freezing on the properties of concrete need to be described to be able to handle for example alternative scenarios in the safety assessment.

Issues relating to bentonite as a barrier in a repository are being studied extensively as a part of the long-term research programme for the spent nuclear fuel repository, as well as for the SFR facility (the silo in SFR). Certain aspects of bentonite are crucial to understand and describe for the assessment of the SFL repository concepts. The physical properties of the buffer, in particular its swelling pressure and low hydraulic conductivity, depend on the affinity of montmorillonite for water. Any process that influences these properties therefore needs to be thoroughly understood. One process which stands out as being of primary concern for the proposed SFL concepts is the cement-bentonite interaction. Cement and bentonite are not fully compatible with each other. The research programme therefore includes studies involving the interaction between cement and bentonite. One example is the project "Concrete and Clay", which is a part of SKB's long-term experimental programme. During 2014, five different experiments will be emplaced in the bedrock in Äspö. The purpose of these is to study the interaction between concrete and different types of bentonite in the presence of elements and compounds similar to what can be found in the waste aimed for SFL. Other activities identified as relevant for cement-bentonite interaction are further development and testing of coupled models, natural analogue studies, and field tests. Currently SKB is taking part in the process of starting a new international project on Cement Barrier Materials, CEBAMA. The project, which currently is in the process of detailing its research activities, will focus on the interaction of cement with other materials in a repository environment, as well as the radionuclide retention and thermodynamics.

The proposed SFL concepts include massive bentonite barriers, in which the water saturation process will be both slow and difficult to model. For the needs of the future evaluation of long-term safety of SFL, it will be necessary to describe the saturation process. This may require additional development and testing of existing models describing the thermal, hydraulic and mechanical properties of the bentonite during the saturation process. The research programme must also include the effects of the swelling bentonite on the waste packages and other barriers in the repository. Other identified issues concern gas and water transport in a fully saturated bentonite, where diffusion will dominate radionuclide transport through the barrier. An important factor to consider is the effect of the gas production by the corrosion of metallic waste. Model development is required to describe radionuclide transport in both aqueous and gas phases. One main safety function of the bentonite is sorption of radionuclides. However, due to the different radionuclides in SFL compared with the spent nuclear fuel repository, a better understanding is needed of the sorption of these key radionuclides in the bentonite barrier. The colloid filtering capacity of the bentonite will likely need to be tested for the specific conditions expected in SFL.

Gravel, or crushed rock, can function as a hydraulic cage, thereby diverting the groundwater flow around the waste rather than through it. It has been shown (section 8.3.2, Neretnieks and Moreno 2013) that using only the gravel as a barrier is not sufficient for SFL. However, gravel or crushed rock can be used as an additional feature to enhance the safety function of another barrier. In this case, an evaluation of the long-term performance of this type of barrier is necessary; several issues and uncertainties have been identified as relevant for the research programme, and a brief description follows here. Dissolution and precipitation are expected to occur in the gravel. These chemical processes are influenced by the chemistry of the groundwater and the composition of the gravel, as well as the flow of groundwater. The reactive surface area in a volume of crushed rock is expected to be large, and the chemical reactions that will occur are expected to both dissolve the gravel and precipitate new minerals in the pore space. If gravel is used as an additional feature to enhance the safety function of another barrier, the expected chemical reactions and ensuing changes in hydraulic conductivity need to be described and modelled. It would also be needed to study the physical process of clogging by biofilms, suspended matter and colloids. If the hydraulic properties of the gravel are of central importance to the proposed safety function, then research on the effects of "clogging" must include geochemistry and hydrogeology, both inside and outside the repository. Certain aspects of these processes are very site-specific, and when site data is available it can be used in reactive transport modelling.

The natural environment

The bedrock surrounding the engineered part of the repository system will contribute to the main safety function *retardation* and will also provide a geologically *stable environment* for the repository. However, though quite similar in principle, the safety assessment done for the spent nuclear fuel repository (SKB 2011) is not directly applicable to SFL. Many important factors differ such as layout, bedrock properties, tectonic setting, etc. Based on a preliminary safety assessment and feedback from it in terms of knowledge gaps, we anticipate that it will be necessary to initiate a dedicated R&D programme targeting the issues specific to SFL, details of which will emerge at a later stage of the SFL project. Naturally, the site description of SFL will need to include findings from ongoing research projects such as DFN (Discrete Fracture Network) methodology, uncertainty propagation, sampling strategies, etc. The aim of the DFN work is to improve the treatment of the connectivity of the fracture system in the conceptual model.

Groundwater flow and chemical reactions occurring in the water-filled fractures of the bedrock need to be understood for the purpose of the safety assessment modelling. Since certain key radionuclides in SFL, in contrast to the spent fuel repository, are transported as anionic species, for example C-14, Cl-36 and Mo-93, the behaviour of anionic species in the groundwater is of specific importance for SFL, and needs to be elucidated, for example with diffusion experiments with rock samples from the selected site. It has previously been observed (Chapman et al. 2000) that small changes in the sorption of key radionuclides have a large effect on calculated dose rates, which illuminates the importance of reducing the uncertainties regarding behaviour of anionic species for the purpose of radionuclide transport modelling. In this context, a specific concern for SFL is the anionic exclusion effect, which tends to limit the diffusion of trace anions into rock. Radionuclide transport modelling tools need to be refined so that the data on matrix porewaters, for example at Forsmark, can be adequately described and then further tested with appropriate laboratory experiments. Laboratory studies for this purpose need to be planned and started as a part of the research programme, with SFL-specific radionuclides and rock samples from candidate sites. Previous modelling of the transport of groundwater components (e.g. oxygen, sulfides, colloids, salt) has identified a need for better integration of geochemistry and hydrogeology. These kinds of studies will be of particular interest for SFL as the chemical environment in the repository and the transport of anionic radionuclides from it may be affected. Model development is under way in various projects concerning e.g. Connectflow and DarcyTools. For SFL, development work needs to include modelling of the flow paths, the flow field, the advective travel time of simulated particles and flow-related transport resistance.

The SFL waste contains many radionuclides and other elements that differ from the key radionuclides in the planned repository for spent nuclear fuel. The release of these species contributes significantly to the environmental risk posed by SFL, and their behaviour in the surface system and biosphere needs to be understood. The research programme therefore includes studies of transport and accumulation of key radionuclides in the surface system. One example is Cl-36, a mobile radionuclide with a relatively long half-life. Traditionally, this nuclide is expected to be transported at a fast rate, since it does not sorb well on mineral surfaces and it is impeded from rock matrix diffusion by anion exclusion. It was also identified as a main problem in the previous assessment of SFL (SKB 1999a, b). However, chloride has been shown to be more reactive than previously thought, and the amount of organically bound chlorine is higher than the amount of mobile chlorine. Biological processes in soil have been singled out as being important for retention of chlorine (Bastviken et al. 2013). These studies are just getting started and considerable research efforts are required in order to describe and understand these processes. SKB is therefore planning a study focused on chlorine in which patterns observed at the already well-characterized sites are linked to processes in terrestrial soil and vegetation. Other parts of the research programme involve radium (Ra), carbon (C), iodine (I), caesium (Cs), nickel (Ni), and molybdenum (Mo). Most of these are biologically active elements. Thus, biological processes affect their distribution and speciation in the surface system. Information gained from these studies will be used in the radionuclide modelling activities. Development of the method used for calculating dose is needed for SFL. The method, including the models it will use, must be developed and tested, and the results must be summarized and evaluated.

For SFL, important questions revolve around the evolution of the ice sheet and permafrost during a transition to a colder climate, as well as modelling a transition to warmer climate (global warming scenario). The question of maximum ice sheet thickness is important for the choice and performance of the repository concept. The thickness of the ice sheet will determine the maximum hydrostatic pressure, which will in turn affect the engineered barriers. This parameter is important for the evolution of the repository in the future evaluation of the long-term safety of the chosen concept. Therefore, a model study, performed within the spent nuclear fuel programme, of how ice thickness evolved above Forsmark during past glaciations will provide the necessary information via a combination of numerical ice sheet simulations and climate simulations. These are being done for the maximum Saalian ice configuration 140,000 years before present, i.e. for the largest Fennoscandian glaciation of the Quaternary. It is necessary to describe a climate case with deep permafrost, since both concrete and possibly also bentonite may be adversely affected by freezing. The repository needs to be located at a depth where the risk of freezing due to permafrost development is small. This actual depth is site-dependent, but a study of expected variations of maximum freezing depth for various possible repository sites is planned as a basis for discussion of the repository depth. In addition, future periods of permafrost will affect groundwater flow patterns and chemistry. This will need to be analysed for the site selected for the SFL repository.

Climate modelling activities concern the description of two possible climate scenarios: global warming and extended global warming (SKB 2013b). An extended period of warm climate may affect the groundwater chemistry, which in turn may affect the chosen barrier. For example, bentonite is sensitive to groundwater of low salinity, and an analysis is therefore planned of how prolonged temperate conditions affect groundwater chemistry.

Finally, for the continued work of evaluating the long-term safety of the SFL repository concept, there is a great need for strengthening the coordination (both conceptually and in modelling terms) between the different disciplines in studies of the natural system. One fundamental difference between the repository for spent nuclear fuel and the SFL repository is connected to the main safety function, which for SFL is retardation, not containment. For SFL there is also a different inventory, with different key radionuclides, which affects the modelling approach. The work that needs to be done involves enhancing the connection in the model between climate change, landscape evolution, hydrogeology, water chemistry, radionuclide transport, and dose calculations. In order to succeed with this challenge, it is suggested that new information and knowledge resulting from the research performed in the Greenland Analogue Project (Harper et al. 2011) is applied in the future SFL studies. Glacial and periglacial conditions prevail for a large part of the time that has to be covered by the safety assessment, and with the help of these recent results, the hydrogeological and geochemical analyses of the glacial and the periglacial periods will be improved. These research efforts have already shown to be valuable by reducing uncertainty in model boundary conditions.

References

SKB's (Svensk Kärnbränslehantering AB) publications can be found at www.skb.se/publications. References to SKB's unpublished documents are listed separately at the end of the reference list. Unpublished documents will be submitted upon request to document@skb.se.

Andra, 2005. Dossier 2005 Argile: Evaluation of the feasibility of a geological repository in an argillaceous formation. Châtenay-Malabry: Agence nationale pour la gestion des déchets radioactifs.

Arnold B W, Brady P V, Bauer S J, Herrick C, Pye S, Finger J, 2011. Reference design and operations for deep borehole disposal of high level radioactive waste. Sandia Report SAND2011-6749, Sandia National Laboratories, Albuquerque, New Mexico.

Bastviken D, Svensson T, Sandén P, Kylin H, 2013. Chlorine cycling and fates of ^{36}Cl in terrestrial environments. SKB TR-13-26, Svensk Kärnbränslehantering AB.

Blomgren J (ed), Karlsson F, Pomp S, Aneheim E, Ekberg C, Fermvik A, Skarnemark G, Wallenius J Zakova J, Grenthe I, Szabó Z, 2010. Partitioning and transmutation. Current developments – 2010. A report from the Swedish reference group for P&T-research. SKB TR-10-35, Svensk Kärnbränslehantering AB.

Brady P V, Arnold B W, Freeze G A, Swift P N, Bauer S J, Kanney J L, Rechard R P, Stein J S, 2009. Deep borehole disposal of high-level radioactive waste. Sandia Report SAND2009-4401, Sandia National Laboratories, Albuquerque, New Mexico.

Bundesamt für Strahlenschutz, 2013. Jahresbericht 2012. Salzgitter: Bundesamt für Strahlenschutz.

Chapman N, Apted M, Glasser F, Kessler J, Voss C, 2000. Djupförvar för långlivat låg- och medelaktivt avfall i Sverige (SFL 3-5). En internationell expertgranskning av SKB:s preliminära säkerhetsanalys. SKI Rapport 00:54, Statens kärnkraftinspektion (Swedish Nuclear Power Inspectorate). (In Swedish.)

Eggert U, Johansson A, Kvamsdal O, 1993. DRD-metoden: en kort presentation. Gnosjö Service Tryckeri AB. (In Swedish.)

Ekenborg F, 2012. Röntgenprojektet – första upplagan. SVAFO report S-11-55:1, AB SVAFO. (In Swedish.)

Ene D, 2012. ESS Preliminary waste management plan. ESS-0003144 Rev No 5, European Spallation Source ESS AB, Sweden.

EU, 2009. Council Directive 2009/71/EURATOM of 25 June 2009 establishing a Community framework for the nuclear safety of nuclear installations. Brussels: Council of the European Union.

EU, 2011. Council Directive 2011/70/EURATOM of 19 July 2011 establishing a Community framework for the responsible and safe management of spent fuel and radioactive waste. Brussels: Council of the European Union.

Evins L Z, 2013. Progress report on evaluation of long term safety of proposed SFL concepts. SKB R-13-41, Svensk Kärnbränslehantering AB.

Grahm P, Luterkort D, Mårtensson P, Nilsson F, Nyblad B, Oxfall M, Stojanovic B, 2013. SFL Concept Study – Technical design and evaluation of potential repository concepts for long-lived low and intermediate level waste. SKB R-13-24, Svensk Kärnbränslehantering AB.

Grundfelt B, 2010. Jämförelse mellan KBS-3-metoden och deponering i djupa borrhål för slutligt omhändertagande av använt kärnbränsle. SKB R-10-13, Svensk Kärnbränslehantering AB. (In Swedish.)

Grundfelt B, 2013. Radiological consequences of accidents during disposal of spent nuclear fuel in a deep borehole. SKB P-13-13, Svensk Kärnbränslehantering AB.

Grundfelt B, Wiborgh M, 2006. Djupa borrhål – Status och analys av konsekvenserna vid användning i Sverige. SKB R-06-58, Svensk Kärnbränslehantering AB. (In Swedish.)

Harper J, Hubbard A, Ruskeeniemi T, Claesson Liljedahl L, Lehtinen A, Booth A, Brinkerhoff D, Drake H, Dow C, Doyle S, Engström J, Fitzpatrick A, Frape S, Henkemans E, Humphrey N, Johnson J, Jones G, Joughin I, Klint KE, Kukkonen I, Kulessa B, Landowski C, Lindbäck K, Makahnouk M, Meierbachtol T, Pere T, Pedersen K, Pettersson R, Pimentel S, Quincey D, Tullborg E-L, van As D, 2011. The Greenland Analogue Project. Yearly report 2010. SKB R-11-23, Svensk Kärnbränslehantering AB.

Herschend B, 2013. Long-lived intermediate level waste from Swedish nuclear power plants: Reference inventory. SKB R-13-17, Svensk Kärnbränslehantering AB.

Huertas F J, Rozalen M L, Garcia-Palma S, Iriarte I, Linares J, 2005. Dissolution kinetics of bentonite under alkaline conditions. In Michau N (ed). ECOCLAY II: effects of cement on clay barrier performance – phase II. EUR 21921, European Commission.

Huutoniemi T, Larsson A, Blank E, 2012. Melting of metallic intermediate level waste. SKB R-12-07, Svensk Kärnbränslehantering AB.

IAEA, 1997. Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management. Vienna: IAEA.

IAEA, 2009a. Classification of radioactive waste: General safety guide. Vienna: IAEA (Safety Standard series GSG-1)

IAEA, 2009b. Borehole disposal facilities for radioactive waste: Specific Safety Guide. Vienna: IAEA (Safety Standard series SSG-1)

IAEA, 2011. Disposal of radioactive waste: Specific safety requirements. Vienna: IAEA. (Safety Standard series SSR-5)

IMO, 1974. Convention on the Prevention of Marine Pollution by Dumping of Wastes and Other Matter. London: International Maritime Organization.

IMO, 1996. 1996 Protocol to the Convention on the Prevention of Marine Pollution by Dumping of Wastes and Other Matter, 1972. London: International Maritime Organization.

Krall L, 2012. High sorption materials for SFL – A literature review. SKB R-12-10, Svensk Kärnbränslehantering AB.

Marsic N, Grundfelt B, 2013a. Modelling of thermally driven groundwater flow in a facility for disposal of spent nuclear fuel in deep boreholes. SKB P-13-10, Svensk Kärnbränslehantering AB.

Marsic N, Grundfelt B, 2013b. Review of geoscientific data of relevance to disposal of spent nuclear fuel in deep boreholes in crystalline rock. SKB P-13-12, Svensk Kärnbränslehantering AB.

Neretnieks I, Moreno L, 2013. Comparison of different SFL design alternatives. SKB R-13-23, Svensk Kärnbränslehantering AB.

NU 1984/85:30. Näringsutskottets betänkande 1984/85:30 om energipolitik. Stockholm: Riksdagen. (In Swedish.)

Odén A, 2013. Förutsättningar för borrning av och deponering i djupa borrhål. SKB P-13-08, Svensk Kärnbränslehantering AB. (In Swedish.)

OPG, 2011. OPG's deep geologic repository for low and intermediate level waste. Preliminary safety report. 00216-SR-01320-00001 R000, Ontario Power Generation, Canada.

Pettersson S, 2013. Feasibility study of waste containers and handling equipment for SFL. SKB R-13-07, Svensk Kärnbränslehantering AB.

Rice E E, Miller N E, Yates K R, Martin W E, Friedlander A L, 1980. Analysis of nuclear waste disposal in space – Phase III. Columbus, OH: Battelle Columbus Laboratories.

Rice E E, Denning R S, Friedlander A L, Priest C C, 1982. Preliminary risk benefit assessment for nuclear waste disposal in space. 33rd Congress of the International Astronautical Federation, Paris, 27 September – 2 October 1982, Paper 82–234.

SFS 1977:1160. Arbetsmiljölag ("Work environment act"). Stockholm: Riksdagen.

SFS 1984:3. Lagen om kärnteknisk verksamhet ("Nuclear activities act"). Stockholm: Riksdagen.

SFS 1988:220. Strålskyddslagen ("Radiation protection act"). Stockholm: Riksdagen.

SFS 1988:1597. Lag om finansiering av hanteringen av visst radioaktivt avfall m.m. ("Act on the financing of the management of certain radioactive waste etc.") Stockholm: Riksdagen.

SFS 1998:808. Miljöbalk ("Environmental code"). Stockholm: Riksdagen.

SFS 2006:263. Lag om transport av farligt gods ("Transport of dangerous goods act"). Stockholm: Riksdagen.

SFS 2010:900. Plan- och bygglag ("Planning and building act"). Stockholm: Riksdagen.

Skagius K, Svemar C, 1989. Performance and safety analysis of WP-Cave concept. SKB TR 89-26, Svensk Kärnbränslehantering AB.

SKB, 1982. Radioactive waste management plan. Plan 82. Part 1: General. SKBF/KBS TR 82-09, Svensk Kärnbränsleförsörjning AB.

SKB, 1989. WP-Cave – assessment of feasibility, safety and development potential. SKB TR 89-20, Svenska Kärnbränslehantering AB.

SKB, 1992. Project on Alternative Systems Study (PASS). Final report. SKB TR 93-04, Svensk Kärnbränslehantering AB.

SKB, 1993. Plan 93. Cost for management of radioactive waste from nuclear power production. SKB TR 93-28, Svensk Kärnbränslehantering AB.

SKB, 1999a. Djupförvar för långlivat låg- och medelaktivt avfall. Preliminär säkerhetsanalys. SKB R-99-59, Svensk Kärnbränslehantering AB. (In Swedish.)

SKB, 1999b. Deep repository for long-lived low- and intermediate-level waste: Preliminary safety assessment. SKB TR-99-28, Svensk Kärnbränslehantering AB.

SKB, 1999c. Deep repository for spent nuclear fuel. SR 97 – Post-closure safety. Main report – Vol. I, Vol. II and Summary. SKB TR-99-06, Svensk Kärnbränslehantering AB.

SKB, 2000a. Systemanalys. Val av strategi och system för omhändertagande av använt kärnbränsle. SKB R-00-32, Svensk Kärnbränslehantering AB. (In Swedish.)

SKB, 2000b. Förvarsalternativet djupa borrhål. Innehåll och omfattning av FUD-program som krävs för jämförelse med KBS-3-metoden. SKB R-00-28, Svensk Kärnbränslehantering AB. (In Swedish.)

SKB, 2007. Långsiktig säkerhet för slutförvar för använt kärnbränsle vid Forsmark och Laxemar – en första värdering. Förenklad svensk sammanfattning av säkerhetsanalysen SR-Can. SKB R-07-24, Svensk Kärnbränslehantering AB. (In Swedish.)

SKB, 2008. Safety analysis SFR 1. Long-term safety. SKB R-08-130, Svensk Kärnbränslehantering AB.

SKB, 2010a. RD&D programme 2010. Programme for research, development and demonstration of methods for the management and disposal of nuclear waste. SKB TR-10-63, Svensk Kärnbränslehantering AB.

SKB, 2010b. Buffer, backfill and closure process report for the safety assessment SR-Site. SKB TR-10-47, Svensk Kärnbränslehantering AB.

SKB, 2011. Long-term safety for the final repository for spent nuclear fuel at Forsmark. Main report of the SR-Site project. SKB TR-11-01, Svensk Kärnbränslehantering AB.

SKB, 2013a. RD&D programme 2013. Programme for research, development and demonstration of methods for the management and disposal of nuclear waste. SKB TR-13-18, Svensk Kärnbränslehantering AB.

SKB, 2013b. Climate and climate-related issues for the safety assessment SR-PSU. SKB TR-13-05, Svensk Kärnbränslehantering AB.

SKI and SSI, 2001. SKI:s och SSI:s gemensamma granskning av SKB:s preliminära säkerhetsanalys för slutförvar av långlivat låg- och medelaktivt avfall. Granskningsrapport. SKI Rapport 01:14, Statens kärnkraftsinspektion (Swedish Nuclear Power Inspectorate), SSI-rapport 2001:10, Statens strålskyddsinstitut (Swedish Radiation Protection Institute). (In Swedish.)

SSM, 2008. The Swedish Radiation Safety Authority's regulations and general advice concerning the protection of human health and the environment in connection with the final management of spent nuclear fuel and nuclear waste. Stockholm: Strålsäkerhetsmyndigheten (Swedish Radiation Safety Authority). (SSMFS 2008:37)

SSM, 2012. Genomförande av rådets direktiv 2011/70/Euratom av den 19 juli 2011 om inrättandet av ett gemenskapsramverk för ansvarsfull och säker hantering av använt kärnbränsle och radioaktivt avfall. Dnr: SSM2012-1246. Stockholm: Strålsäkerhetsmyndigheten (Swedish Radiation Safety Authority). (In Swedish.)

SÖ 1984:5. Antarktisfördraget (The Antarctic treaty) Washington, 1 December 1959. Stockholm: Utrikesdepartementet.

SÖ 1999:60. Konvention om säkerheten vid hantering av använt kärnbränsle och om säkerheten vid hantering av radioaktivt avfall. (Joint convention on the safety of spent fuel management and on the safety of radioactive waste management.) Vienna, 5 September 1997. Stockholm: Utrikesdepartementet.

Sternö E, 2005. Bergkross som filtermaterial vid vattenbehandling. Master thesis. Royal Institute of Technology, Stockholm, Sweden. (In Swedish.)

U.S. DOE, 2012. Waste isolation pilot plant – documented safety analysis. DOE/WIPP 07-3372, REV. 3a, U.S. Department of Energy Carlsbad Field Office, Carlsbad, NM.

Wiborgh M (ed), 1995. Prestudy of final disposal of long-lived low and intermediate level waste. SKB TR 95-03, Svensk Kärnbränslehantering AB.

Åkesson M, Kristensson O, Börgesson L, Dueck A, Hernelind J, 2010. THM modelling of buffer, backfill and other system components. Critical processes and scenarios. SKB TR-10-11, Svensk Kärnbränslehantering AB

Unpublished documents

SKBdoc id, version	Title	Issuer, year
1176359 ver 1.0	Deep boreholes with large diameter in crystalline rock	IDEAS Ing.-Büro A. Sperber, 2007
1416968 ver 1.0	Low and intermediate level waste in SFL3–5: Reference inventory	SKB, 2009

Appendix 1

Requirements

The requirements listed below are functional and feature requirements on the SFL system and its components. The requirements are SKB's interpretation of what functions and features a system for disposal of long-lived low and intermediate level waste should have from an ownership perspective and also in response to the requirements of various regulations.

1 Level 1 requirements

The requirements at this level apply to the system as a whole or are common to all components of the system.

1.1 General functional requirements

ID	Requirement	Reference to regulatory requirements	Reference to evaluation factors
N1-47	**Disposal of long-lived low and intermediate level waste** The repository system shall enable and provide a system for disposal of long-lived low and intermediate level waste.		

1.2 Nuclear safety and radiation protection

ID	Requirement	Reference to regulatory requirements	Reference to evaluation factors
N1-113	**Safety during handling and disposal** The repository system shall be designed in order to guarantee safety during handling and disposal of the waste.	**SSMFS 2008:37** 3§ **SSMFS 2008:51** 2 kap 1§	1.2 Robustness of the barrier safety functions 3.1 Personal safety and working environment
N1-19	**Best available technology and optimization** Among technically feasible alternative designs, techniques and measures, those alternatives which in the short term best restrict radiation doses to humans and which in the long term are judged to offer the best protective capability shall be selected, as far as is reasonable. The radiation protection in the repository system shall be optimized and technology choices should, as far as is reasonable, be based on the best available technology.	**MB (1998:808)** 2 kap 3§ **SÖ 1999:60** Artikel 11, iii) **SÖ 1999:60** Artikel 11, v) **SÖ 1999:60** Artikel 12, ii) **SÖ 1999:60** Artikel 14, i) **SÖ 1999:60** Artikel 24, 1 i) **SÖ 1999:60** Artikel 24, 2 i) **SÖ 1999:60** Artikel 24, 2 ii) **SSMFS 2008:21** 6§ **SSMFS 2008:23** 3§ **SSMFS 2008:23** 4§ **SSMFS 2008:26** 4§ **SSMFS 2008:37** 3§ **SSMFS 2008:37** 4§ **SSMFS 2008:37** 6§ **SSMFS 2008:51** 2 kap 1§ **PBF (2011:338)** 3 kap 9§ **SSMFS 2008:1** 6 kap 1§	1.2 Robustness of the barrier safety functions 3.1 Personal safety and working environment

ID	Requirement	Reference to regulatory requirements	Reference to evaluation factors
N1-20	**Protect personnel and visitors from radiation** Radiation doses to personnel and other individuals, caused by the repository system design and its activities shall be as low as reasonably achievable.	**SÖ 1999:60** Artikel 11, v) **SÖ 1999:60** Artikel 14, i) **SÖ 1999:60** Artikel 24,1 i) **SÖ 1999:60** Artikel 24, 1 ii) **SÖ 1999:60** Artikel 24, 2 ii) **SSMFS 2008:23** 3§ **SSMFS 2008:23** 4§ **SSMFS 2008:26** 4§ **SSMFS 2008:37** 3§ **SSMFS 2008:51** 2 kap 1§ **SSMFS 2008:1** 6 kap 1§	3.1 Personal safety and working environment
N1-17	**Robust design** Systems, components and devices needed for the safety of the repository system shall be resistant to malfunctions and features, events and processes that could have a detrimental impact on their functions.	**SSMFS 2008:1** 2 kap 1§ **SSMFS 2008:1** 3 kap 1§ **SSMFS 2008:1** 3 kap 4§ **SSMFS 2008:21** 5§	3.2 Feasibility of design and construction 1.2 Robustness of the barrier safety functions 3.3 Feasibility of technology and method of operation
N1-18	**Reliably constructed** Systems, components and devices needed for the safety of the repository system shall be built in a reliable manner.	**MB (1998:808)** 2 kap 3§ **PBL (2010:900)** 8 kap 19§ **SÖ 1999:60** Artikel 14, iv) **SSMFS 2008:1** 3 kap 1§ **SSMFS 2008:1** 3 kap 2§ **SSMFS 2008:1** 3 kap 4§ **PBF (2011:338)** 3 kap 7§	3.2 Feasibility of design and construction 1.2 Robustness of the barrier safety functions 3.3 Feasibility of technology and method of operation
N1-48	**Proven technology** Systems, components and devices needed for the safety of the repository system shall be designed and built with proven or evaluated technology.	**MB (1998:808)** 2 kap 3§ **SÖ 1999:60** Artikel 14, iv) **SSMFS 2008:1** 3 kap 2§	3.2 Feasibility of design and construction 3.3 Feasibility of technology and method of operation
N1-26	**Possible to maintain, check, test** Systems, components and devices needed for the safety of the repository system shall be designed so that they are possible to maintain, check and test during construction and operation. Properties required to obtain the initial state of the repository shall be possible to demonstrate.	**MB (1998:808)** 2 kap 3§ **PBL (2010:900)** 8 kap 19§ **SÖ 1999:60** Artikel 14, iv) **SÖ 1999:60** Artikel 16, i) **SSMFS 2008:1** 2 kap 1§ **SSMFS 2008:1** 3 kap 1§ **SSMFS 2008:1** 3 kap 4§ **SSMFS 2008:1** 5 kap 3§ **PBF (2011:338)** 3 kap 7§	3.3 Feasibility of technology and method of operation 3.2 Feasibility of design and construction
N1-49	**Possible to correct** Identified deficiencies in the systems, components and devices needed for the safety of the disposal system or other properties affecting the initial state of the repository shall be possible to correct during the construction and operating phases.	**SÖ 1999:60** Artikel 16, i) **SSMFS 2008:1** 2 kap 1§ **SSMFS 2008:1** 5 kap 3§	3.2 Feasibility of design and construction 3.3 Feasibility of technology and method of operation

1.3 Environmental impact (non-radiological)

ID	Requirement	Reference to regulatory requirements	Reference to evaluation factors
N1-29	**Best available technology** In order to prevent, hinder or counteract damage or harm to human health or the environment caused by non-radiological emissions from operations in the repository system causes, the best available technology shall be used.	**MB (1998:808)** 2 kap 3§	

ID	Requirement	Reference to regulatory requirements	Reference to evaluation factors
N1-30	**Limited impact on the environment** The impact of the repository system on the environment (non-radiological emissions to water and air, noise, etc.) shall be as limited as far as is reasonably possible.	**MB (1998:808)** 2 kap 3§ **PBL (2010:900)** 2 kap 3§ **PBL (2010:900)** 2 kap 9§ **PBL (2010:900)** 8 kap 19§ **SÖ 1999:60** Artikel 11, v) **SÖ 1999:60** Artikel 15, i) **SÖ 1999:60** Artikel 15, ii) **PBF (2011:338)** 3 kap 9§ **PBF (2011:338)** 3 kap 13§	2.1 Impact on human health and the environment 2.2 Land and resource needs
N1-31	**Resource-effective** The repository system shall be designed to ensure sustainable use of raw materials and utilize opportunities for reuse and recycling.	**MB (1998:808)** 2 kap 5§ **PBL (2010:900)** 2 kap 3§ **PBL (2010:900)** 2 kap 6§ **PBL (2010:900)** 8 kap 19§ **PBF (2011:338)** 3 kap 14§	2.2 Land and resource needs
N1-32	**Environmentally-friendly product choices** Products with no or little impact on the environment shall be chosen and used in the construction and operation of the repository system as far as is reasonably possible.	**MB (1998:808)** 2 kap 4§ **PBL (2010:900)** 8 kap 19§ **SÖ 1999:60** Artikel 11, v)	2.1 Impact on human health and the environment

1.4 Other

ID	Requirement	Reference to regulatory requirements	Reference to evaluation factors
N1-37	**Capacity** The repository system shall be designed for all Swedish long-lived low and intermediate level waste from the nuclear power programme, SVAFO and Studsvik. Uncertainties shall be considered.		3.4 Flexibility
N1-38	**Timetable** The repository system shall be designed, built, operated and closed in accordance with the timetable.	**SÖ 1999:60** Artikel 11, vii)	4.2 Time 2.4 Burdens on future generations
N1-42	**Cost-effective** The repository system shall be designed, built and operated so that the entire chain, from conditioning of the waste to closure of the repository, is cost-effective.	**SÖ 1999:60** Artikel 12, ii)	4.1 Cost

2 Level 2 requirements

The requirements at this level apply to different parts of the system as indicated in the headings.

2.1 Waste and containers

2.1.1 General functional requirements

ID	Requirement	Reference to regulatory requirements	Reference to evaluation factors
N2-154	**Enable handling, transport and disposal of SFL-waste** Waste containers shall enable handling, transport and disposal of the SFL waste.		

2.1.2 Properties requirements

ID	Requirement	Reference to regulatory requirements	Reference to evaluation factors
N2-41	**Possible to analyze waste** It must be possible to determine the characteristics of the waste at disposal and closure (initial state of the waste) and to analyze its development.	**SSMFS 2008:1** 6 kap 9§	1.1 Feasibility of making a post-closure safety assessment
N2-47	**Resistance after closure** In case of any damage to the engineered barriers after closure, the properties of the waste shall as far as possible delay the release of radionuclides.	**MB (1998:808)** 2 kap 3§ **SÖ 1999:60** Artikel 24, 1 i) **SSMFS 2008:1** 2 kap 1§ **SSMFS 2008:37** 3§	1.2 Robustness of the barrier safety functions

2.1.3 Requirements from connecting system components

ID	Requirement	Reference to regulatory requirements	Reference to evaluation factors
N2-145	**Adaptation to system for external transportation and handling equipment** Waste containers shall be designed to be handled within the system's facilities and the system for external transportation.	**SÖ 1999:60** Artikel 11, iii)	

2.2 System for external transportation

2.2.1 General functional requirements

ID	Requirement	Reference to regulatory requirements	Reference to evaluation factors
N2-146	**Provide transportation of SFL waste** The system for external transportation shall provide transport of SFL waste from the waste producer to the system's facilities as well as between the system's facilities.		

2.2.2 Properties requirements

ID	Requirement	Reference to regulatory requirements	Reference to evaluation factors
N2-50	**The waste shall be transported safely** It shall be possible to transport the waste in an environmentally and occupationally safe manner.	**SÖ 1999:60** Artikel 11, v) **SSMFS 2008:51** 2 kap 1§ **SFS 2006:263** 1§	3.1 Personal safety and working environment 3.3 Feasibility of the technique and method of operation
N2-130	**Good working environment** It shall be possible to offer a good working environment during external transportation.	**AML (1997:1160)** 1 kap 1§ **AML (1997:1160)** 2 kap 4§ **SSMFS 2008:1** 3 kap 3§	3.1 Personal safety and working environment
N2-57	**Well-protected and guarded** It shall be possible to transport the waste with good physical protection.	**SÖ 1999:60** Artikel 14, i) **SSMFS 2008:1** 2 kap 11§ **SSMFS 2008:12** **SFS 2006:263** 1§	3.2 Feasibility of design and construction

2.2.3 Requirements from connecting system components

ID	Requirement	Reference to regulatory requirements	Reference to evaluation factors
N2-148	**Adapting to waste and containers** The system for external transportation shall be designed to handle the SFL waste and its containers.	**SÖ 1999:60** Artikel 11, iii)	

2.3 Facilities of the repository system

2.3.1 Common requirements on the facilities of the repository system

2.3.1.1 Property requirements

ID	Requirement	Reference to regulatory requirements	Reference to evaluation factors
N2-99	**Multi-Barrier Principle** Facilities of the repository system shall have a system of barriers that in one or more ways help to contain, prevent or retard the release of radioactive substances, either directly or indirectly, by protecting the other barriers in the barrier system.	**SSMFS 2008:1** 2 kap 1§ **SSMFS 2008:1** 6 kap 1§	
N2-90	**Accessible for Safeguards** Safeguardclassified material that is placed in the facilities of the repository system shall be accessible for inspection.	**SSMFS 2008:3** **SSMFS 2008:1** 6 kap 2§	3.2 Feasibility of design and construction
N2-91	**Well-protected and guarded** Facilities of the repository system shall be well protected and guarded from intrusion and unlawful handling of nuclear material or nuclear waste.	**SÖ 1999:60** Artikel 14, i) **SSMFS 2008:1** 2 kap 11§ **SSMFS 2008:12** **SSMFS 2008:1** 6 kap 1§	3.2 Feasibility of design and construction
N2-133	**Appropriately sited** Facilities of the repository system shall be sited so that the facilities can be built, operated and closed with a minimum of damage and harm to human health and the environment.	**MB (1998:808)** 2 kap 6§ **PBL (2010:900)** 2 kap 2§ **PBL (2010:900)** 2 kap 5§ **PBL (2010:900)** 2 kap 6§ **PBL (2010:900)** 2 kap 8§ **PBL (2010:900)** 2 kap 9§ **SÖ 1999:60** Artikel 13, 1 i) **SÖ 1999:60** Artikel 13, 2)	2.2 Land and resource needs
N2-109	**Acceptability** Location of the repository system's facilities shall be based on the voluntary participation of the municipalities involved. This means that facilities must obtain acceptance with regard to e.g. environmental impact and safety.		2.3 Acceptability
N2-77	**Good working environment** It shall be possible to offer a good working environment during construction and operation of the repository system's facilities.	**AML (1997:1160)** 1 kap 1§ **AML (1997:1160)** 2 kap 4§ **PBL (2010:900)** 8 kap 19§ **SSMFS 2008:1** 3 kap 3§ **PBF (2011:338)** 3 kap 9§ **PBF (2011:338)** 3 kap 10§ **PBF (2011:338)** 3 kap 13§ **PBF (2011:338)** 3 kap 14§	3.1 Personal safety and working environment

ID	Requirement	Reference to regulatory requirements	Reference to evaluation factors
N2-76	**Radiation protection during operation** The repository system's facilities, equipment and procedures for the handling of waste shall be designed so that radiation doses to personnel and the environment are limited.	**MB (1998:808)** 2 kap 3§ **SÖ 1999:60** Artikel 11, v) **SÖ 1999:60** Artikel 14, i) **SÖ 1999:60** Artikel 24, 1 i) **SÖ 1999:60** Artikel 24, 1 ii) **SÖ 1999:60** Artikel 24, 2 ii) **SSMFS 2008:23** 3§ **SSMFS 2008:23** 4§ **SSMFS 2008:26** 4§ **SSMFS 2008:37** 3§ **SSMFS 2008:51** 2 kap 1§ **PBF (2011:338)** 3 kap 9§ **SSMFS 2008:1** 6 kap 1§	3.1 Personal safety and working environment
N2-131	**Safe operations** Nuclear activities in the repository system's facilities must be safe.	**SÖ 1999:60** Artikel 11, i) **SÖ 1999:60** Artikel 14, i) **SÖ 1999:60** Artikel 24, 1 iii) **SSMFS 2008:1** 2 kap 1§ **SSMFS 2008:1** 2 kap 2§ **SSMFS 2008:1** 3 kap 1§ **SSMFS 2008:1** 3 kap 3§ **SSMFS 2008:1** 5 kap 3§ **SSMFS 2008:1** 6 kap 1§	3.1 Personal safety and working environment
N2-93	**Safe handling of waste packages** Waste packages shall be transported, deposited and otherwise handled in a safe way in the repository system's facilities.	**SSMFS 2008:1** 2 kap 1§ **SSMFS 2008:1** 6 kap 1§	3.1 Personal safety and working environment
N2-75	**Good safety at work** The repository system's facilities shall offer good occupational safety conditions during construction and operation.	**AML (1997:1160)** 1 kap 1§ **AML (1997:1160)** 2 kap 4§ **PBF (2011:338)** 3 kap 10§	3.1 Personal safety and working environment
N2-94	**Minimize the impact of accidents** The repository system's facilities shall be provided with equipment and facilities to minimize the impact of fire and other accidents.	**SSMFS 2008:1** 2 kap 1§ **PBF (2011:338)** 3 kap 8§	3.1 Personal safety and working environment
N2-134	**Safe for users** The repository system's facilities shall be safe for people who are working or for any other reason are present in or near the facilities.	**MB (1998:808)** 2 kap 3§ **AML (1997:1160)** 1 kap 1§ **AML (1997:1160)** 2 kap 4§ **PBL (2010:900)** 8 kap 19§ **SÖ 1999:60** Artikel 11, v) **SSMFS 2008:1** 2 kap 1§ **PBF (2011:338)** 3 kap 8§ **PBF (2011:338)** 3 kap 9§ **PBF (2011:338)** 3 kap 10§ **PBF (2011:338)** 3 kap 13§	3.1 Personal safety and working environment
N2-42	**Possible to analyze safety** The operational safety of the repository system's facilities shall be possible to analyze and report in accordance with the regulatory requirements.	**SÖ 1999:60** Artikel 15, i) **SSMFS 2008:1** 4 kap 1§	

2.3.1.2 Requirements from connecting system components

ID	Requirement	Reference to regulatory requirements	Reference to evaluation factors
N2-138	**Adaptation to waste and container** The repository system's facilities shall be designed to handle the SFL waste and its containers.	**SÖ 1999:60** Artikel 11, iii) **SSMFS 2008:1** 6 kap 1§	

2.3.2 Requirements on the facilities for conditioning of waste

2.3.2.1 General functional requirements

ID	Requirement	Reference to regulatory requirements	Reference to evaluation factors
N2-142	**Provide for the conditioning of the SFL waste** The facilities for conditioning of waste shall provide for the conditioning of the SFL waste.		

2.3.3 Requirements on the storage facilities

2.3.3.1 General functional requirements

ID	Requirement	Reference to regulatory requirements	Reference to evaluation factors
N2-135	**Provide for storage of the SFL waste** The facilities for storage shall provide for the storage of the SFL waste.		

2.3.3.2 Property requirements

ID	Requirement	Reference to regulatory requirements	Reference to evaluation factors
N2-158	**Passive safety functions** The facility shall utilize passive safety functions, as far as is reasonably possible.	**SSMFS 2008:1** 6 kap 2§	

2.3.3.3 Requirements from connecting system components

ID	Requirement	Reference to regulatory requirements	Reference to evaluation factors
N2-72	**Provide an environment that does not adversely affect the waste** The waste container must not be adversely affected during storage.	**SÖ 1999:60** Artikel 11, iii) **SSMFS 2008:1** 6 kap 2§	3.2 Feasibility of design and construction

2.3.4 Requirements on the repository facilities (prior to closure)

2.3.4.1 General functional requirements

ID	Requirement	Reference to regulatory requirements	Reference to evaluation factors
N2-136	**Accommodate and allow disposal of SFL waste** The repository facility shall accommodate and facilitate disposal of the SFL waste.		

2.4 Requirements on the repository and its barrier system for long-term safety

2.4.1 General functional requirements

ID	Requirement	Reference to regulatory requirements	Reference to evaluation factors
N2-132	**Contain, prevent or retard** The repository shall contain, prevent or retard the dispersion of radioactive substances so that the ionizing radiation does not cause harm to human health or the environment.	**MB (1998:808)** 2 kap 3§ **SÖ 1999:60** Artikel 11, vi) **SSMFS 2008:21** 3§ **SSMFS 2008:37** 3§ **SSMFS 2008:37** 5§ **SSMFS 2008:37** 6§	1.2 Robustness of the barrier safety functions

2.4.2 Properties requirements

ID	Requirement	Reference to regulatory requirements	Reference to evaluation factors
N2-162	**Appropriately sited** The repository shall be located at a site that has suitable conditions for a long-term safe disposal of the SFL waste.		1.2 Robustness of the barrier safety functions 2.2 Land and resource needs
N2-152	**Multi-Barrier Principle** The barrier system for long-term safety shall have multiple barriers that in one or more ways help to contain, prevent or retard the release of radioactive substances, either directly or indirectly, by protecting the other barriers in the barrier system.	**SSMFS 2008:1** 2 kap 1§ **SSMFS 2008:21** 3§ **SSMFS 2008:21** 7§	1.2 Robustness of the barrier safety functions
N2-21	**Passive barriers** After closure of the repository the barriers shall be passive.	**SÖ 1999:60** Artikel 11, vii) **SSMFS 2008:21** 2§	
N2-117	**Long-term durable barriers** The repository system of barriers and barrier functions shall provide protection against harmful effects of radiation for such a long time that the repository with its waste does not cause harm to human health or the environment.	**SÖ 1999:60** Artikel 11, vi) **SÖ 1999:60** Artikel 11, vii) **SSMFS 2008:21** 5§ **SSMFS 2008:21** 10§ **SSMFS 2008:37** 5§ **SSMFS 2008:37** 6§	1.2 Robustness of the barrier safety functions
N2-156	**Possible to analyze safety after closure** The safety of the repository after closure shall be possible to analyze and report in accordance with the regulatory requirements. It shall also be possible to analyze the properties of the facility and the closure and their impact on the barriers. The safety assessment for the post-closure period shall cover as long a time as the barrier functions are needed, but at least ten thousand years.	**SÖ 1999:60** Artikel 11, iii) **SÖ 1999:60** Artikel 13, 1 ii) **SÖ 1999:60** Artikel 15, ii) **SSMFS 2008:1** 4 kap 1§ **SSMFS 2008:21** 9§ **SSMFS 2008:21** 10§ **SSMFS 2008:21** 11§ **SSMFS 2008:37** 4§ **SSMFS 2008:37** 5§ **SSMFS 2008:37** 9§ **SSMFS 2008:37** 10§ **SSMFS 2008:37** 11§ **SSMFS 2008:37** 12§	1.1 Feasibility of making a post-closure safety assessment
N2-129	**Primarily maintain the barrier functions** Measures taken to facilitate access, surveillance or retrieval of disposed waste, or to impede intrusion, shall not be detrimental to the safety of the repository.	**SÖ 1999:60** Artikel 11, vi) **SSMFS 2008:37** 8§	

ID	Requirement	Reference to regulatory requirements	Reference to evaluation factors
N2-95	**Consider inadvertent intrusion** Inadvertent intrusion shall be considered in the design of the repository so that the repository site, after closure of the repository, can be utilized in such a manner that the freedom of choice, needs and aspirations of future generations are restricted as little as possible.	**SÖ 1999:60** Artikel 11, vi) **SÖ 1999:60** Artikel 11, vii) **SSMFS 2008:37** 9§	1.2 Robustness of the barrier safety functions
N2-96	**Closure** The repository shall be designed so that no monitoring is required after closure.	**SÖ 1999:60** Artikel 11, vii) **SÖ 1999:60** Artikel 14, iii)	2.4 Burdens on future generations

Appendix 2

Terms and definitions

Term	Definition
ALARA	As Low As Reasonably Achievable International radiation protection principle that all exposures from the use of radiation sources should be "as low as reasonable achievable, economic and social factors being taken into account".
barrier	physical confinement of radioactive substances *Source*: SSMFS 2008:1 *Remark*: An engineered (man-made) or natural part of repository that has barrier function.
conditioning	treatment of the SFL waste, in order to meet requirements on the properties of the waste, including the packaging for transportation and disposal
engineered barrier	barriers in the repository, which are engineered by man *Source* : SSMFS 2008:21 General Guidance *Remark:* Examples of engineered barriers are containers for nuclear waste, concrete and backfill materials of clay, sand or concrete.
facility for conditioning of waste	one or more facilities necessary for the conditioning of the SFL waste
storage facility	one or more facilities required for (interim) storage of SFL waste prior to conditioning or waste packages prior to disposal in the SFL
natural barrier	barrier not engineered by man, e.g. the rock
repository	structure in which SFL waste is emplaced for disposal. Includes the barriers for long-term safety.
repository facility	facility at which the repository (SFL) is built *Remark*: Includes all buildings and technical systems required for the operation of the repository.
repository system	facilities (e.g. for conditioning, storage and disposal), transportation system, etc and the activities required to implement the disposal of long-lived low and intermediate level radioactive waste in SFL
rock caverns	large openings blasted in rock, e.g. caverns, vaults, chambers and niches
safety function	role by which a repository component contributes to safety
SFL	repository for long-lived waste
SFL waste	long-lived low and intermediate level radioactive waste from the Swedish nuclear power programme, SVAFO and Studsvik that will be disposed of in SFL *Remark*: Includes the waste package
system for external transportation	equipment needed to transport the SFL waste
technical systems	technical installations and mobile equipment that are required in order to carry out the activities in the repository facilities *Remark*: Technical installations comprise, during construction and operation, permanently installed systems for supply, communication, safety, drainage, ventilation, etc. Mobile equipment consists of machinery, vehicles, etc.

Appendix 3

Abbreviations

ALARA	As low as reasonably achievable
ATB	Radiation-shielded transport cask for intermediate level waste.
ATB 1T	A new cask for transport of long-lived low and intermediate level waste in steel tanks.
BAT	Best available technology.
BMA	Rock cavern for intermediate level waste in SFR.
BWR	Boiling water reactor.
CSH	Calcium silicate hydrate.
Clab	Central storage facility for spent nuclear fuel in Oskarshamn.
IAEA	International Atomic Energy Agency.
MB	Swedish Environmental Code.
NPP	Nuclear power plant.
PWR	Pressurized water reactor.
R&D	Research and Development.
RD&D	Research, Development and Demonstration.
RPV	Reactor pressure vessel.
SFL	Repository for long-lived waste.
SFR	Repository for short-lived radioactive waste in Forsmark.
SFS	Swedish Code of Statutes.
SKI	Swedish Nuclear Power Inspectorate.
SSI	Swedish Radiation Protection Institute.
SSM	Swedish Radiation Safety Authority.
SSMFS	Regulations of the Swedish Radiation Safety Authority.
SVAFO	AB SVAFO – which is owned by Ringhals AB, Forsmarks Kraftgrupp AB, OKG AB and Barsebäck Kraft AB – is a member of the Vattenfall Group. Its activities include management and treatment of waste from previous research and development activities at e.g. Studsvik, for the purpose of disposal.
SÖ	Sweden's international agreements.
1BMA	Existing BMA in SFR.
2BMA	Additional BMA in the planned extension of SFR.