Report

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Low and intermediate level waste in SFR

Reference inventory for waste 2013

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Svensk Kärnbränslehantering AB

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

The original report, dated August 2015, was found to contain editorial errors which have been corrected in this updated version.

Summary

The purpose of this report is to describe the waste that is expected to be deposited in SFR at the closure of the facility in 2075. The report serves as a basis for the design of the extension of SFR, as well as the dispersion and dose calculations which are presented in the assessment of the long-term safety of SFR.

The waste inventory is calculated for the low- and intermediate-level waste that arises from operation and decommissioning of the Swedish nuclear facilities and is expected to be managed in SFR. The inventory consists of waste from the nuclear power companies BKAB, FKA, OKG and RAB as well as from the nuclear facility Clab/Clink and from the nuclear activity being conducted by SNAB and Svafo.

The report provides information on number of packages, disposal volume, material content and radionuclide content for existing and future waste types. Based on given assumptions risks and uncertainties in the estimated inventory are discussed.

The inventory for operational waste has been calculated with the aid of the report and forecasting tool Triumf NG and the inventory for decommissioning waste has been estimated based on decommissioning studies and forecast data from the waste suppliers.

List of definitions

BKAB	Barsebäck kraft AB.
BLA	Waste vault for low-level waste (existing waste vault 1BLA and planned waste vaults XBLA).
BMA	Waste vault for intermediate-level waste (existing waste vault 1BMA and planned waste vaults XBMA).
BRT	Waste vault for reactor pressure vessels (planned).
BTF	Waste vault for concrete tanks (existing waste vaults 1BTF and 2BTF).
Clab	The interim storage facility for spent nuclear fuel, situated in Oskarshamn.
Clink	Central facility for interim storage and encapsulation of the spent nuclear fuel (Clab and the encapsulation plant, where the encapsulation plant is planned to be built).
Corrosion surface	The area of metals that may be exposed to corrosion. Metal surfaces in contact with bitumen are not included in the definition for corrosion surface.
Disposal volume	Total volume of the waste package, including the volume around the package that is occupied at deposition.
ESS	European Spallation Source (planned).
FKA	Forsmarks kraftgrupp AB.
Inventory	In this document this means the number of packages, volumes, material quantities and radionuclide content.
OKG	Oskarshamn kraftgrupp AB.
PSU	Project SFR Extension.
RAB	Ringhals AB.
Ranstad	The Ranstad site, mining and uranium plant, mining concluded 1969.
SFL	Final repository for long-lived low- and intermediate-level waste (planned).
SFR	Final repository for low- and intermediate-level waste, situated in Forsmark.
Silo	Silo, repository for intermediate-level waste.
SKB	Swedish Nuclear Fuel and Waste Management Company (Svensk kärnbränslehantering AB)
SNAB	Studsvik Nuclear AB.
SSM	Swedish Radiation Safety Authority (Strålsäkerhetsmyndigheten).
Svafo	AB SVAFO.
TRIUMF	Database for the deposited waste in SFR.
Triumf NG	Report and forecasting tool for low- and intermediate-level waste.
Waste type description	Safety report for the current waste type. For each waste type description, the waste container, waste category, type of treatment and final disposal site for the waste is set out.
Waste category	The material composition of the waste, given in code form.
Waste packaging	A packaging in which waste is placed.
Waste description	A waste description is a simpler/stripped down variant of a waste type description. These can be used when for example only a few individual waste packages is concerned.

Waste form	Refers to the physical and chemical form after treatment and/or conditioning.
Waste package	The waste form and its packaging, constituting a unit for treatment, transportation, short-term or long-term storage.
Waste type	Classification and grouping of waste. The waste types are described in a waste type description or waste description.
Ågesta	Nuclear power plant in Ågesta, decomissioned since 1974.

Contents

1 Introduction

1.1 Background

SFR has been in operation since 1988. When the regulatory authorities gave their licence for operation, they required that the safety analysis report for the repository should be updated at least every 10 years. A preliminary safety analysis was established in 1983 and the safety analysis for commissioning was finished in 1987. An update report was published in 1991.

In 1997, project SAFE was initiated to make a thorough update of the safety analysis. A detailed prognosis of the waste in SFR, including materials and radionuclide content, was developed to be used as a basis for radionuclide transport and dose calculations (Riggare and Johansson 2001). An updated safety analysis for SFR was submitted to the regulatory authorities in June 2001.

Prior to the submission of an updated safety analysis for SFR to the authorities in 2008, SAR-08, an update of the previous reference inventory for waste was made (Almkvist and Gordon 2007).

The present report is an update of the 2007 reference inventory. Two important differences between the reports are that this report also includes information on decommissioning waste and the planned extension of SFR. The report constitutes a basis for PSU, SKB's project for the extension of SFR. A best estimate and uncertainties of the waste that will be deposited in SFR is given.

Even though the report gives exact numbers, it should be kept in mind that it is only a prognosis. The assumptions made are sometimes greatly simplified due to the uncertainties concerning the future waste.

The purpose of this report is to describe the waste and the waste packages that will be deposited in SFR at closure of the facility in 2075. The waste inventory given here is aimed to be used as a basis for e.g. designing the capacity of SFR and in the radionuclide transport and dose calculations.

Data for operational waste is extracted from the report and forecasting tool Triumf NG. Migration of data from the TRIUMF database in SFR to Triumf NG takes place once per year, with a breakpoint at the end of the year. This report is based on the turn of the year 2012/2013, i.e. all waste disposed after 2012-12-31 is regarded as forecasted.

For decommissioning waste the information given is only an estimate of content and amounts in the decommissioning waste. The forecasts for decommissioning waste will be more detailed as the decommissioning approaches.

1.2 SFR

SFR is situated in Forsmark in north-eastern Uppland, close to the Forsmark nuclear power plant. The waste vaults are located in the bedrock, about 60 m beneath the seabed and 1 km from the coast. The underground parts of the repository are reached by two tunnels.

The existing SFR is designed for final disposal of low- and intermediate-level operational waste from the Swedish nuclear power plants and Clab as well as SNAB and Svafo.

Today's facility is licensed to contain a radioactivity of a total of 10^{16} Bq. The disposal capcity for waste in the existing parts of the facility is a total of about 60,000 m^3 of which nearly 35,000 m^3 are utilised as of 2012-12-31.

SFR is today divided into four types of waste vaults:

- Silo.
- Waste vault for intermediate-level waste (1BMA).
- Waste vaults for concrete tanks (1–2BTF).
- Waste vault for low-level waste (1BLA).

The waste vaults are linked via a system of tunnels. Figure 1-1 shows the design of the existing SFR.

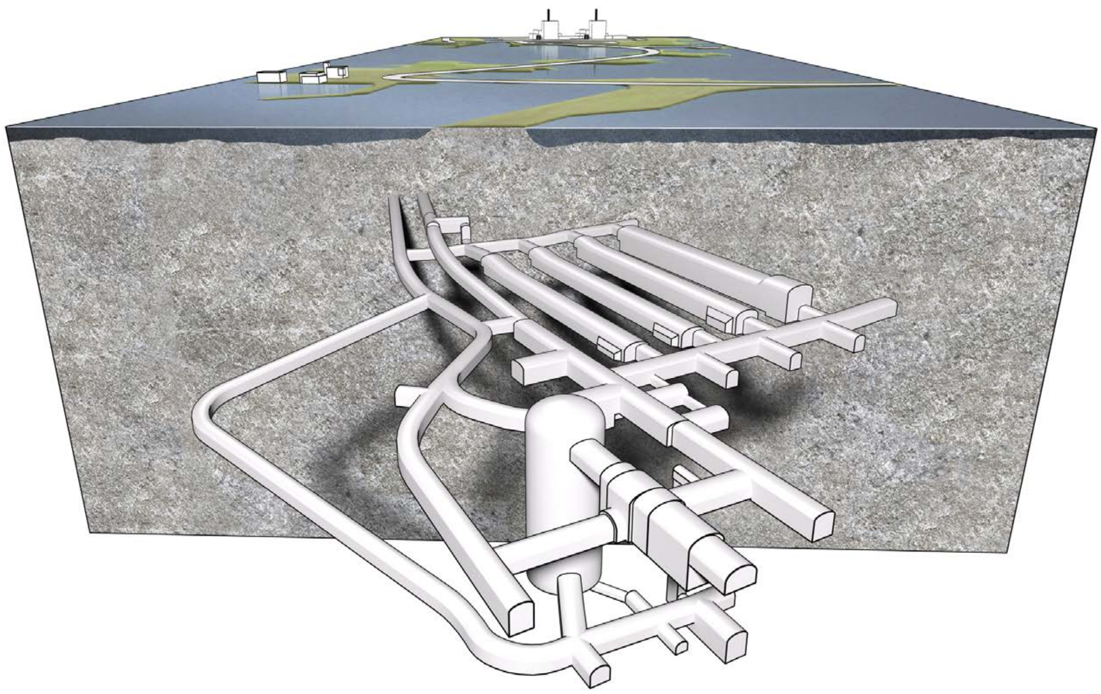

Figure 1-1. The existing SFR.

In order for SFR to be able to receive all operational and decommissioning waste that is expected to arise, an extension of SFR is planned. The details regarding design and extent of the extension have not yet been fully determined, but building of the following types of waste vault is planned:

- Waste vault for reactor pressure vessels (BRT).
- Waste vaults for intermediate-level waste (BMA).
- Waste vaults for low-level waste (BLA).

1.2.1 Silo

The Silo consists of a cylindrical concrete structure with vertical shafts of different sizes, the shafts are separated by concrete walls. The largest shafts are square with sides of 2.5 m. The Silo vault is about 70 m high, of which about 50 m is intended for waste. It has a diameter of about 30 m. The outer walls are made of reinforced concrete and have a thickness of approximately 0.8 m. The gap of about 1.2 m between the outer walls and the surrounding rock is backfilled with bentonite. The 1 m thick reinforced concrete floor in the bottom of the Silo is founded on a bed of a sand and bentonite mixture. To hinder leakage of drainage water in the shafts a waterproofing membrane is installed in the roof above the Silo. The waterproofing membrane will be removed prior to the closure of SFR.

In the Silo intermediate-level waste from the nuclear power plants, Clab, SNAB and Svafo is deposited. The waste consists mostly of ion exchange resins solidified in cement or bitumen, but concrete-embedded trash and scrap metal also occurs. Concrete or steel moulds and steel drums placed on drum trays are used as waste packaging.

The waste packages are normally placed in the shafts with four moulds, or 16 drums placed on four drum trays, at a time. After certain deposited levels the space between the waste packages is backfilled with a cement based grout. At the top of the shafts concrete lids are placed, which may be lifted away temporarily during deposition.

The Silo is designed for disposal of the majority of the radioactivity that is disposed of in SFR and thereby has the most extensive barriers. The function of the barriers is to delay the radionuclide release after closure of the repository.

All handling of the waste is done with remote-controlled control and surveillance equipment. Figure 1-2 shows a schematic drawing of the Silo.

Figure 1-2. Schematic drawing of the Silo.

1.2.2 Waste vault for intermediate-level waste (1BMA)

The waste vault is about 160 m long, 19.5 m wide and has a height of 16.5 m. The concrete structure in the waste vault consists of 13 large compartments and 2 small compartments. Each compartment is built as a box with concrete walls. The large compartments have a dimension of 9.9×14.8×7.3 m and the small compartments have a dimension of 5.0×7.2×7.3 m. In the rock vault roof, there is a waterproofing membrane installed. This is to hinder leakage of drainage water into the compartments. The waterproofing membrane will be removed prior to the closure of SFR.

In 1BMA, intermediate-level waste from the nuclear power plants, Clab, SNAB and Svafo is deposited. The waste consists of cement- or bitumen-solidified ion exchange resins, evaporator concentrates and sludge as well as concrete-embedded trash and scrap metal. Concrete or steel moulds and steel drums placed on drum trays or drums placed in drum boxes are used as waste packaging.

The waste is stacked on the concrete floor in a way that enables the concrete moulds to provide support for the concrete lids. When a compartment has been filled with waste a prefabricated concrete lid is placed on top. After the lid has been put in place a layer of concrete is also cast on top of the lid. There is also the possibility of grouting the space between the waste packages in a compartment.

Figure 1-3 shows a schematic drawing of 1BMA.

1.2.3 Waste vaults for concrete tanks (1–2BTF)

There are two waste vaults for concrete tanks, 1BTF and 2BTF, which are similar in design. The waste vaults are about 160 m long, 15 m wide and have a height of 9.5 m.

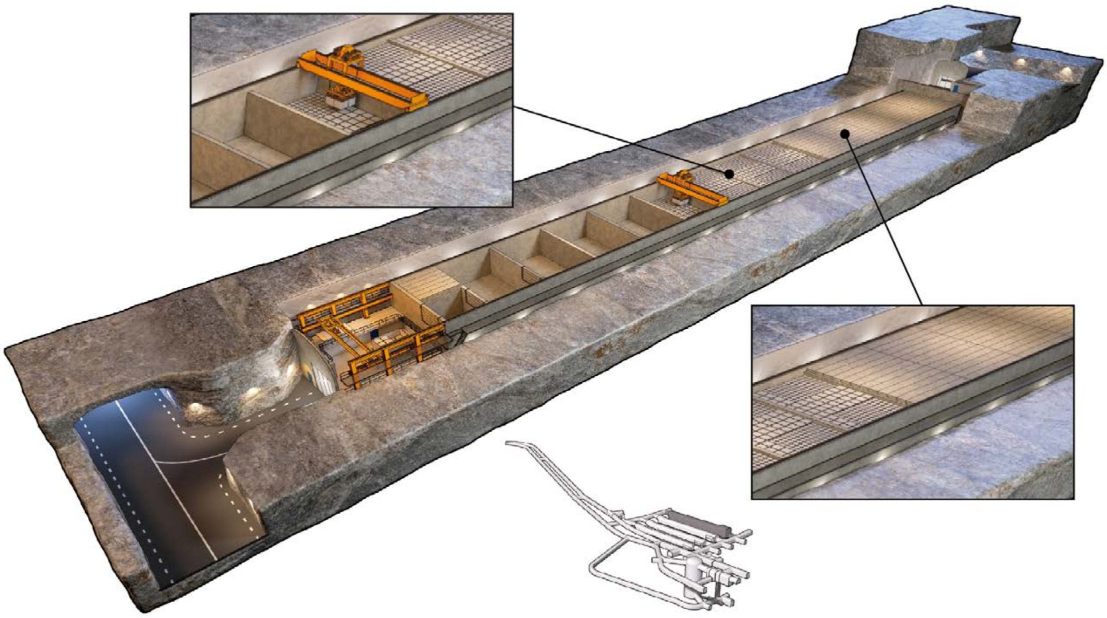

Figure 1-3. Schematic drawing of 1BMA.

Intermediate-level waste from the nuclear power plants, Clab, SNAB and Svafo is deposited in 1–2BTF. The waste consists largely of dewatered ion exchange resins in concrete tanks. In 1BTF, there are, besides concrete tanks, steel drums containing ashes and concrete moulds containing cement-solidified ion exchange resins. Some miscellaneous waste is also deposited in 1–2BTF, such as a reactor pressure vessel lid.

The concrete tanks, with a volume of about 10 m^3 each, are arranged four side-by-side and stacked two high with a middle layer of steel in order to facilitate handling. A concrete lid for protection against radiation is placed on top of the concrete tanks as the stacking is finished.

The steel drums with ashes in 1BTF are placed lying down in the longitudinal direction of the repository and concrete moulds are placed across the waste vault as support. Concrete tanks are located along the rock walls and act as stabilising walls for the ash drums. To further stabilise the drums they are progressively grouted with concrete.

Figure 1-4 shows a schematic drawing of 1BTF and Figure 1-5 shows a schematic drawing of 2BTF.

1.2.4 Waste vault for low-level waste (1BLA)

The waste vault is about 160 m long, 15 m wide and has a height of 13 m. The waste vault has a simple structure with a concrete floor on which containers are placed. Since the commissioning of the repository, corrugated sheet has been placed as a roof over the waste to minimize the moisture from drainage water on the containers. The sheet will be removed prior to the closure of the repository.

Low-level waste from the nuclear power plants, Clab, SNAB and Svafo is deposited in 1BLA. The waste deposited in 1BLA consists mainly of low-level trash and scrap metal waste placed in ISO containers. Part of the waste in the containers is also placed inside inner packaging such as steel drums and bales. The containers are arranged side-by-side in two rows and stacked three full-height or six half-height containers on top of each other.

Figure 1-6 shows a schematic drawing of 1BLA.

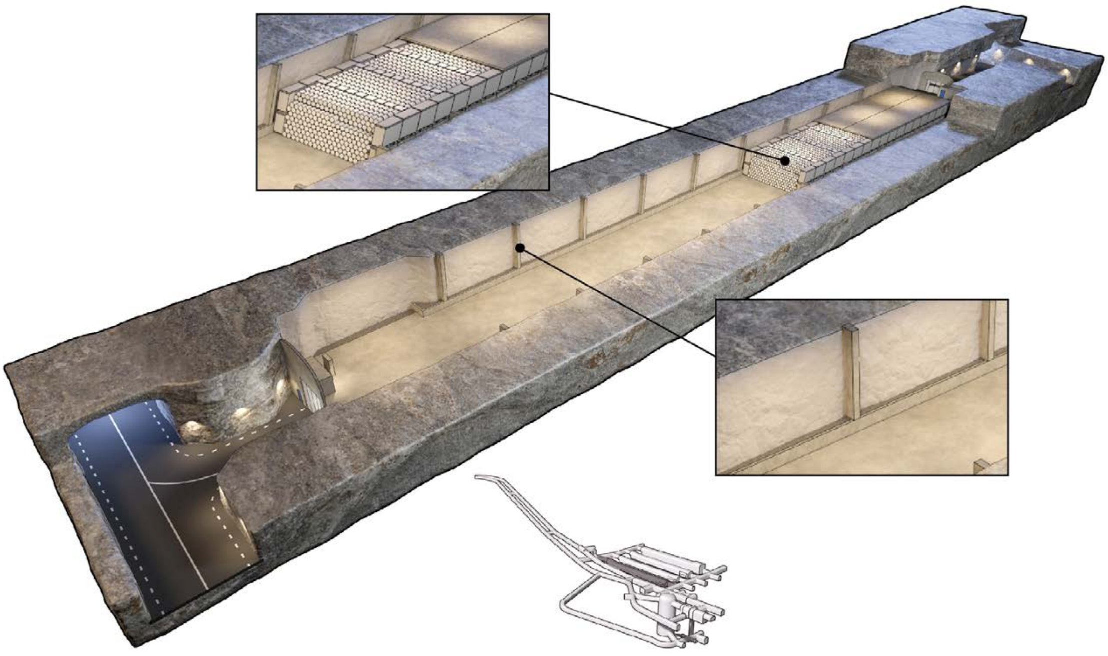

Figure 1-4. *Schematic drawing of 1BTF.*

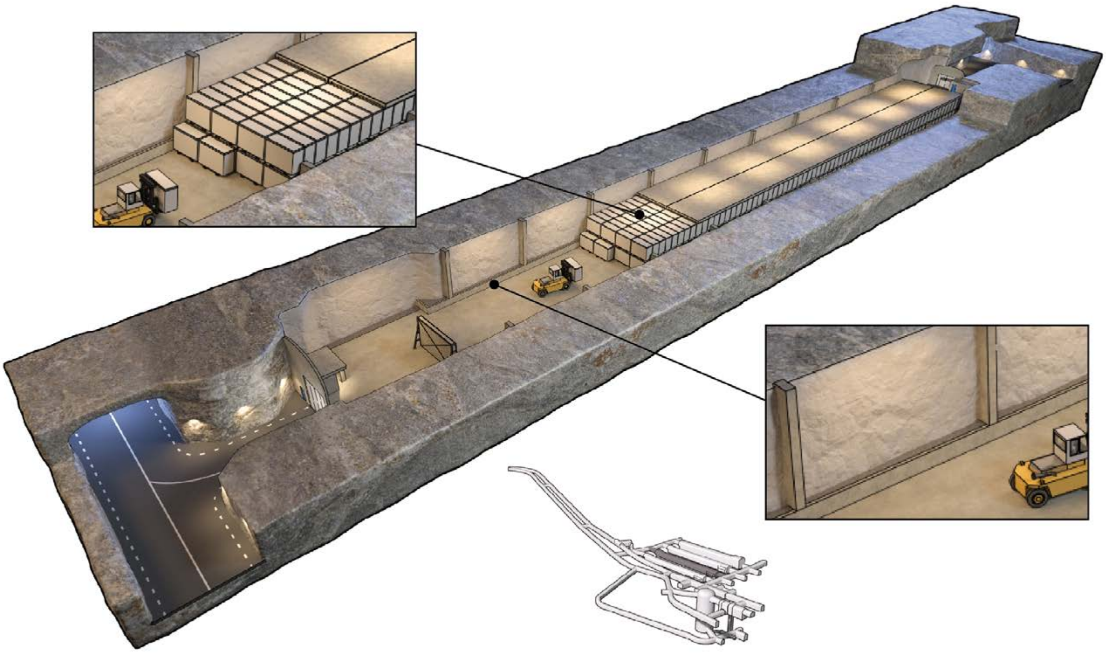

Figure 1-5. *Schematic drawing of 2BTF.*

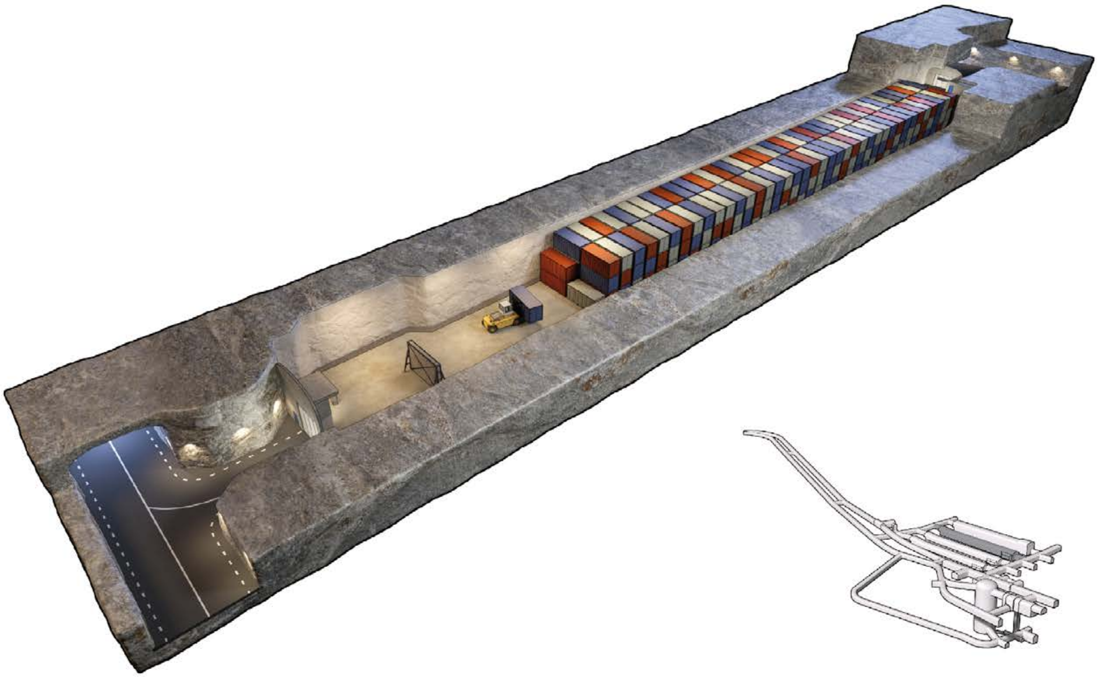

Figure 1-6. *Schematic drawing of 1BLA.*

1.2.5 Waste vault for reactor pressure vessels in the extended SFR (BRT)

The construction of a waste vault for reactor pressure vessels is planned for the extension of SFR. Intact BWR pressure vessels are planned to be deposited in this waste vault.

1.2.6 Waste vaults for intermediate-level waste in the extended SFR (XBMA)

The construction of one or several waste vaults similar to the existing 1BMA is planned for the extension of SFR. The waste vaults will in this report be referred to as XBMA.

The plan is to deposit intermediate-level waste, mostly in the form of trash and scrap metal, concrete and sand in concrete and steel moulds or tetramoulds.

1.2.7 Waste vaults for low-level waste in the extended SFR (XBLA)

The construction of a number of waste vaults similar to the existing 1BLA is planned for the extension of SFR. The waste vaults will in this report be referred to as XBLA.

The plan is to deposit waste similar to that deposited in 1BLA, i.e. mostly low-level waste in the form of trash and scrap metal placed in ISO containers. Waste in the form of sand, soil, gravel, asphalt and concrete will also be deposited here.

1.3 Premises

The waste inventory given in this report is calculated for low- and intermediate-level waste that is expected to arise from operation and decommissioning of the Swedish nuclear facilities.

The reactors at FKA and OKG, and reactor R3 and R4 at RAB are expected to have an operating time of 60 years, while the reactors R1 and R2 at RAB are expected to operate for 50 years. For BKAB shutdown operation is calculated to continue until 2020 before decommissioning is initiated. The facilities Clab/Clink, where nuclear fuel is stored and treated before final disposal, are assumed to be in operation until the year 2070. The operations at SNAB and Svafo are assumed to continue until 2040 and 2045 respectively. The waste from SNAB and Svafo includes waste from industries, research and medical care. Operational waste from Ågesta is deposited via SNAB and Svafo but decommissioning waste from Ågesta is presented independently in this report. Waste from Ranstad is deposited via Svafo. Potential waste from the planned nuclear facility ESS (European Spallation Source) is not included in the present report.

Operational waste includes waste already deposited and a prognosis for future waste. The determination of the inventory is based on knowledge and experiences on existing waste supplemented with prognoses given by each waste supplier. The inventory for decommissioning waste is based on decommissioning studies and prognoses from the waste suppliers. These studies are based on inventories and facility blueprints. A smaller fraction of secondary waste is also assumed to arise during decomissioning, similar to the secondary waste that arises during normal operation. The prognoses for decommissioning waste given by SNAB and Svafo are lacking information on radioactivity content, which also applies to this report.

1.3.1 Operational waste

Existing waste in SFR

Waste that is already deposited is assumed not to be relocated.

Forecasted waste to SFR

Forecasted waste is assumed to be deposited as stated in the waste type description. Operational waste in the extension is assumed to be distributed in the same way as in the existing SFR, i.e. waste that is deposited in 1BMA and 1BLA will be deposited in the equivalent repository parts in the extension. The distribution of waste between the existing and extended SFR is, however, not yet established.

The waste emplacement strategy will consider long-term safety requirements.

Distribution SFR/SFL

The operational waste that will be deposited in SFL consists mainly of reactor internals with high radioactivity and a high proportion of long-lived nuclides.

Near-surface repositories and free release

The wastes that presently are managed in near-surface repositories or released for unrestricted use are assumed to also be so in the future.

Clab storage canisters

Based on analyses and smear tests, the fuel storage canisters used in Clab are judged to be cleared for free release. They are therefore not included in this report.

Packing degree

The packing degree of the operational waste is well-known and based on long experience in waste management. In the forecasts for future operational waste, today's packing degrees have been assumed, as optimisations are carried out on several occasions.

1.3.2 Decommissioning waste

Distribution SFR/SFL

Decommissioning studies show that the decommissioning waste expected to be deposited in SFL consists of long-lived near-core waste (< 1 m from the core), e.g. core grids and core instrumentation. BWR pressure vessels are planned to be deposited in SFR and PWR pressure vessels are planned to be placed in SFL. An exception is that all systems that contain > 10^{10} Bq C-14 are intended to be deposited in SFL. From decommissioning waste data, it is shown that the contents of C-14 will be high in a few systems. This waste would in a small volume contribute with high C-14 activity in SFR. For this reason, and because C-14 in the previous safety assessments has been shown to be a risk dominant nuclide in SFR, the plan is to deposit these wastes in SFL.

Intact reactor pressure vessels

As a prerequisite for this report it has been assumed that the reactor pressure vessels will be intact when deposited, i.e. they will not be segmented at decommissioning.

Distribution in repository parts

For decommissioning wastes, systems that are calculated to have a specific radioactivity below 10^6 Bq/kg and do not require any specific radiation shielding will be deposited in BLA. Systems with higher specific radioactivity content will be deposited in BMA. BWR reactor pressure vessels are deposited in BRT. Ion exchange resins from system decontamination prior to decommissioning are deposited in the Silo. The distribution of waste between the existing and extended SFR is not yet established.

The waste emplacement strategy will consider long-term safety requirements.

Interim storage

This report does not account for possible interim storage of long-lived low- and intermediate-level decommissioning waste in the extended SFR.

Packing degree

The packing degrees used in the decommissioning studies are used to estimate the amount of decommissioning waste in SFR. The densities used are 1.5 tonnes/m^3 for concrete and 1.1 tonnes/m^3 for scrap, which are taken from national and international studies and previous projects. The limitation in total weight for different waste packaging will also determine how much waste can be accommodated in a waste package.

System decontamination

System decontamination has been carried out on surfaces for a number of systems as stated in the decommissioning studies. The decontamination factor is set to 10, i.e. the radioactivity on the components is reduced by 90% after system decontamination and the radioactivity is instead relocated to ion exchange resins.

Near-surface repositories and free release

The decommissioning plans for the nuclear power plants include shallow land disposal of very low-level and short-lived decommissioning wastes as an alternative to final disposal in SFR. In this report all decommissioning waste that has not been cleared for free release is assumed to be deposited of in SFR.

1.4 Work and review process

1.4.1 Operational waste

Each waste package is supplied with a waste data file from the waste supplier. The file contains information about the package according to SSM's requirements for registration of radioactive waste. This file transfers information to the TRIUMF database at SFR. From TRIUMF, data of the deposited packages is migrated annually to the report and forecasting tool Triumf NG.

For material calculations, Triumf NG is supplemented with information from waste type descriptions of the material content in different waste types. For radioactivity calculations, Triumf NG is supplemented with non-package-specific radioactivity data from measurements and calculations performed by the waste suppliers and Studsvik ALARA Engineering. In Triumf NG, there are also additional calculation methods for determination of difficult-to-measure nuclides and for compilation of the nuclide content in SFR.

Assumptions about interim-stored and forecasted operational waste have been given by each waste supplier. Information about waste type, packaging type, number of waste packages and year of deposition are estimated.

The inventory of operational waste has been calculated in the report and forecasting tool Triumf NG v1.0.1.3. Triumf NG is a verified calculation program.

1.4.2 Decommissioning waste

The inventory of the decommissioning waste is based on finalised decommissioning studies. The basis for the studies is the inventories and radioactivity calculations of the nuclear power plants, which partly can be found in the relevant decommissioning studies produced by the waste suppliers and partly have been carried out by SKB. The radioactivity calculations have been reviewed by the the waste suppliers and SKB, and the final results for the reactors have been compared with each other as an assessment of plausibility.

Data from inventories and radioactivity calculations have been compiled in Excel. These compilations have been reviewed based on the assumption that data in specified references has been applied correctly. The amount of waste has been compared on reactor level and type of waste packaging. Review of the radioactivity has been made reactor- and system-wise regarding the total system radioactivity. The distribution of radioactivity per repository part has been checked based on which packaging type the system is placed in, according to the decommissioning studies.

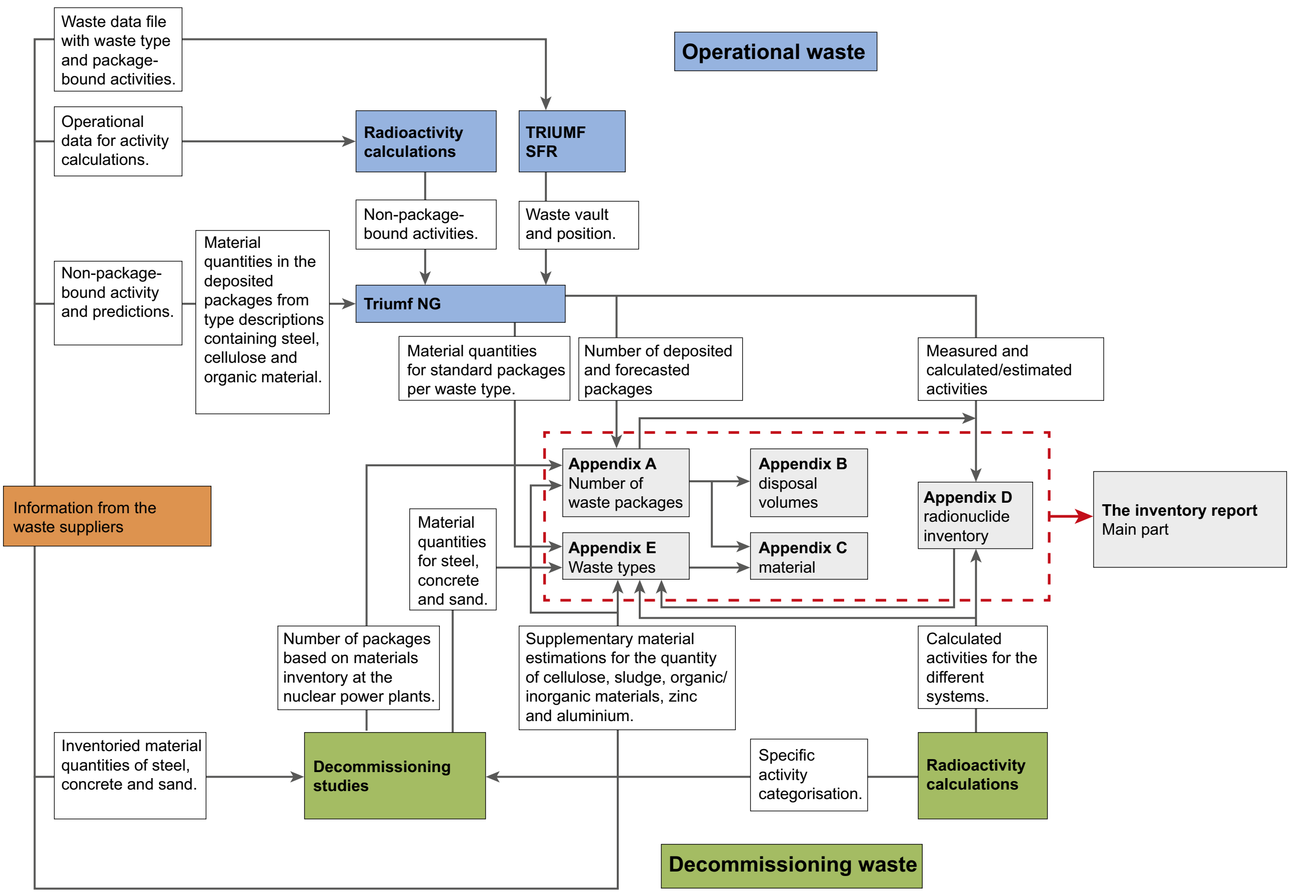

Figure 1-7. Process image, which shows the preparation of an inventory report.

1.5 Report structure

This report contains a main part and five appendices.

The main part consists of a description of existing and planned repository parts, general information on waste, waste packaging and waste forms and a summary of all waste, both existing and future, that is disposed or will be disposed in SFR. The main part also includes information on quality assurance and uncertainties.

The appendices consist of:

- Appendix A – Number of waste packages.
- Appendix B – Waste disposal volumes.
- Appendix C – Materials in the waste.
- Appendix D – Radionuclide inventory.
- Appendix E – Description of waste types.

Appendices A–D describe calculations of the inventory in SFR. Appendix E describes each waste type and average values of material and radionuclide content in each waste type. The material content is an estimated average and the radionuclide content is a calculated mean value at the closure of SFR on 2075-12-31.

2 Quality assurance

This chapter explains how to ensure that the right waste is placed in the right packaging and in the right waste vault.

A detailed documentation of the waste fulfils the following purposes:

- It should serve as a basis for safety-related assessments of the adequacy of the waste for final disposal, i.e. provide a sufficiently detailed description of the waste to serve as a basis for future assessments regarding the safety of the final repository.
- It should constitute support for the monitoring or review of the operations in the final repository and serve as a basis for e.g. decisions on transports and other planned activities.
- It should provide support for decisions on rectifications in conjunction with unplanned events.
- It imposes requirements on the waste suppliers.

The nuclear facilities that generate waste are responsible for manufacturing and interim storage of the waste. SKB is responsible for the transportation, deposition and final disposal of the waste. The waste supplier, i.e. the nuclear facility where the waste is generated always has the overall responsibility for the waste produced.

2.1 Existing waste in SFR

The systems used to ensure that the right waste has been placed in the correct packaging and in the correct waste vault are both the governing documents "Waste handling manual for low- and intermediate-level waste" and waste type descriptions, plus registries and databases for the waste, and waste audits on site. The systems are described in detail in Section 2.1.1–2.1.4. Section 2.1.5 gives a schematic account of waste handling, from the origin of the waste to the deposition of the waste package, with respect to quality assurance.

2.1.1 The Waste handling manual for low- and intermediate-level waste

The Waste handling manual is a key document for the administrative management of low- and intermediate-level wastes that is intended for final disposal.

The Waste handling manual provides directives on what information and other data a waste type description should contain, and presents the requirements the waste must meet.

Acceptance criteria

The Waste handling manual provides acceptance criteria that must be met by waste packages intended for disposal in SFR. The criteria takes into account requirements on mechanical, physical, radiological and chemical properties.

Waste codes

A system for registration and reporting of radioactive waste was developed at the end of the 1970s. The system is based on the use of codes so the waste packages can be easily defined.

New codes have been added to the system and an adaptation has been made to present-day conditions. The codes apply to SFR, but can also apply to near-surface repositories and future decommissioning repositories for both short-lived and long-lived low- and intermediate-level waste. The interface between waste suppliers and SKB with common codes is currently well established via SFR. The codes are provided in the Waste handling manual.

Detailed codes are available for the following:

- Containment and packaging type (e.g. 612, Half-height container).
- Waste category (e.g. 310, Evaporator concentrate PWR).
- Type of treatment (e.g. 70, embedded in concrete).
- Waste type (e.g. 12, Container with trash and scrap metal for deposition in 1BLA).

2.1.2 Waste type descriptions

Each waste type deposited in SFR should, before deposition is initiated, have an approved waste type description that describes the entire handling process from manufacturing to the final disposal of the waste. Waste type descriptions have been used since the end of the 1980s with the aim of documenting the waste that is deposited in SFR. Waste type descriptions are also included as a part of the safety analysis report for both SFR and for the different nuclear facilities.

A general account of the entire waste handling process is given in the waste type description, as well as a detailed account of the waste type properties and characteristics, including waste categories, packaging types, treatment types, etc. Moreover, how the waste fulfils the specified requirements during the entire handling process is shown through an account of manufacturing data, results of investigations, calculations and control measures.

In order to identify which requirements are of particular importance for the different waste types, it is applicable to analyse the waste package in each step of the handling process, including final disposal. Such an analysis leads to the question of which stipulated requirements, based on the system in which the waste package is included, are of greatest importance. The stipulated requirements are an important basis for the total assessment of the waste type adequacy and for setting limit values.

A description of control measures is specified for waste packaging, waste form and waste packages. For waste packaging, the waste supplier needs to make sure that the manufacturer has a sufficient programme for quality control. Inspection of the waste form is done mainly by monitoring the manufacturing, where the manufacturing process includes both technical and administrative procedures that influence the properties of the waste form. For waste packages, the waste supplier should carry out an inspection of the finished package with respect to determination of nuclide content and dose rates.

Review and account of waste type descriptions for the final repository

Development and review of waste type descriptions is an iterative process between the waste supplier and SKB. The safety review carried out by the waste suppliers focuses on manufacturing and further handling until transport to the final repository. SKB's safety review is mainly focused on safety for transport and deposition and on the long-term safety of the final repository. Before waste packages of a specific waste type can be deposited, SSM must approve the waste type description and give consent to SKB that transport and deposition is permissible.

2.1.3 Waste audits

SKB is conducting quality audits with respect to waste management in the nuclear facilities, according to a continuous four-year plan, i.e. each facility gets an audit visit every four years.

Each nuclear facility must have procedures and instructions for sorting and placement of waste. The type of waste that are to be placed in specific packaging is well defined. For waste solidification there are specific solidification recipes that must be followed.

The audits aim to determine whether the current waste type descriptions and the waste supplier's own procedures and instructions for waste management are followed and whether the supplier's control, management and documentation of the waste process is in order, including procedures for safety audit.

2.1.4 Waste registry

All treated nuclear waste stored at the nuclear power plants, Clab, SNAB and Svafo should, in accordance with stipulated requirements from the Swedish Radiation Safety Authority, be registered in a waste registry. Each package is registered individually with a unique identity and with information on general properties, package type, manufacturing date, measured surface dose rate and nuclide content measured with respect to gamma radiation and with calculated estimates of alpha and beta radiation. In some cases measurements of alpha and beta radiation are carried out linked to waste streams, which then can be linked to specific waste types. Difficult-to-measure nuclides are calculated either separately based on correlation with so-called key nuclides (Co-60, Cs-137, Pu-239/240), or with sampling in waste streams for determination and distribution at repository level. Furthermore, the registry states the location where the package is disposed.

Before transport to SFR a waste data file is produced. This file contains waste data for each individual package in the transport. The information is added to SKB's waste database in connection with placement in the repository. The database is used to produce reports and information on the waste deposited in SFR.

2.1.5 Quality assurance of the handling process

Figure 2-1 gives a schematic illustration of the handling process for waste, from origin to deposition. Control points (CP) are also indicated for the different steps.

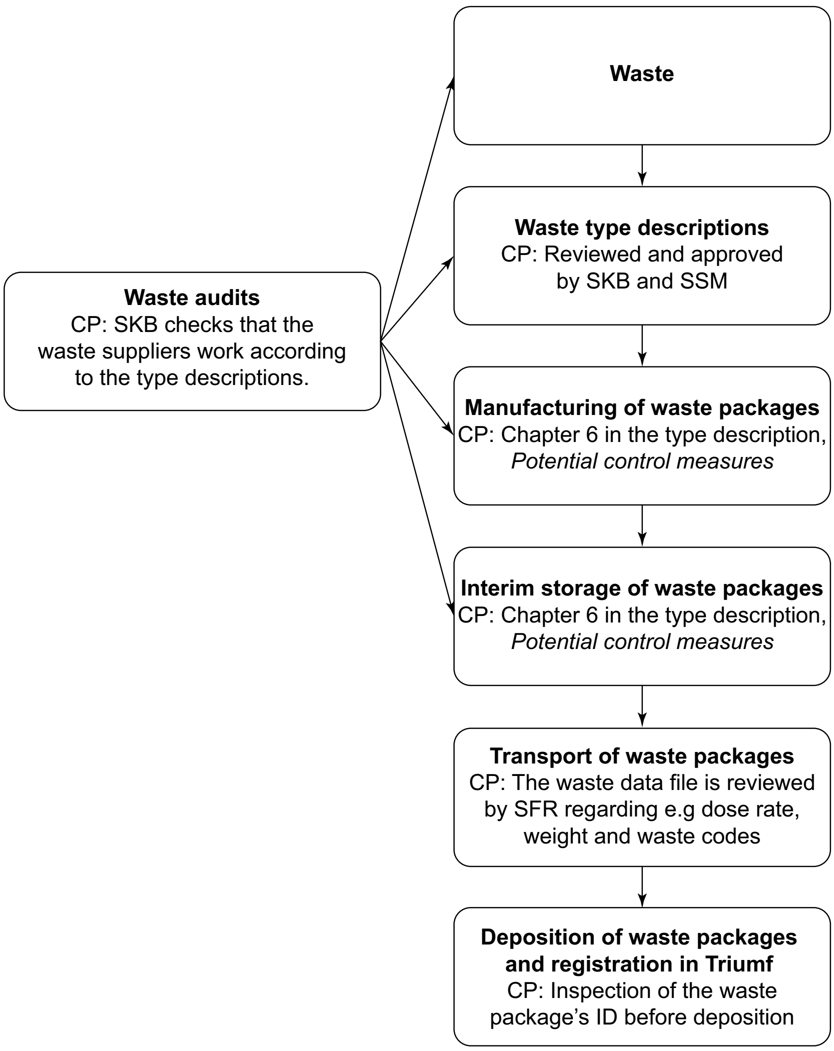

Figure 2-1. *Schematic illustration of quality assurance for the handling process for waste, from origin to deposition. (CP = control point.)*

Waste

When a new waste type occurs, the supplier must develop a waste type description. This is done with guidance from the Waste handling manual, using its waste codes for packaging, waste and treatment.

Waste type description

An approved waste type description requires that the waste type, the manufacturing, the handling and potential control measures are described, controlled and approved.

Manufacturing of waste packages

Manufacturing is carried out as described in the waste type description. The waste type description may also include instructions for e.g. solidification recipes or sorting of the waste with respect to the dose rate.

Interim storage of waste packages

Interim storage is done as described in the waste type description. The waste type description may also include instructions for e.g. transport to the interim storage.

Transport of waste packages

Before transport of waste packages, a transport message (TRAM) and a waste data file are produced. TRAM and the waste data file are reviewed against the acceptance criteria including a control against specified approved codes for deposition.

Deposition of waste packages and registration in TRIUMF

When the waste package is lifted out of the transport container at SFR, the ID number is inspected, and checked against the TRAM number and the waste data file that was sent with the package. After deposition, the exact deposition position of the waste package is specified in the TRIUMF database. The information in the waste data file is also entered into TRIUMF.

Waste audits

SKB regularly audits the waste suppliers with respect to their waste management at all stages.

2.2 Future waste in the existing SFR and the planned extension

The system that will be used to ensure that the right waste is placed in the correct packaging and in the correct waste vault in the existing SFR and in the planned extension of SFR are the same as those used today. This includes the Waste handling manual for low- and intermediate-level waste, waste type descriptions for different waste types, waste audits and waste registries.

2.2.1 The Waste handling manual for low- and intermediate-level waste

The Waste handling manual, including acceptance criteria and waste codes, will continue to be valid. The document is updated continually and will, if necessary, be expanded with new waste codes for the extension of SFR. The work of preparing acceptance criteria for waste to the extension of SFR is in progress.

2.2.2 Waste type descriptions

Existing waste type descriptions will be used when possible and new waste type descriptions are developed when needed.

2.2.3 Waste audits

Waste audits will continue to be carried out every four years.

2.2.4 Waste registries

Waste registries will continue to be used by the nuclear facilities and SKB. The computer program may be re-programmed but the same waste data will be registered and already existing data will be preserved.

3 The waste in SFR – in general

3.1 Waste types in SFR

The first commercial Swedish nuclear power plant (Oskarshamn 1) has been in operation since the early 1970s. Before reactor O1 was in operation, radioactive waste arose from research and from the first research reactors. The Swedish system for final disposal of waste was not available at the time of commissioning of the nuclear power plants and as a consequence, different plants have used different treatment methods and different types of packaging for waste, even though the waste has similar characteristics. To obtain a systematic classification of waste management, different waste types have been defined and a code system has been developed. The system is used in e.g. data transfer between the nuclear power plants and SFR.

The code system consists of a letter that states which nuclear facility that manufactures the waste, and two numbers that provide information on the type of waste, the treatment method used, which type of packaging the waste is allowed to be contained in, and in what part of SFR the waste will be deposited. An additional number (added after ":") can be used to provide information on properties that differentiate the waste from other waste of the same type. The meaning of the additional number is defined for each waste type, e.g. for B.05:2, the additional number ": 2" means that it contains defective drums placed in a drum box. The only additional number that has the same meaning generally is ":9 ", which indicates that the waste is of an older type. An older type here refers to waste that was produced before 1988, i.e. before SFR was commissioned.

For example, the code R.01:9 means that the waste type is produced by Ringhals, that it consists of cement-solidified ion exchange resins mounted in a concrete mould, that the waste will be deposited in the waste vault for intermediate-level waste (1BMA) and that it is an older type of waste. In Table 3-1 and Table 3-2, explanations are given for the different abbreviations used in the code system today.

For each waste type there should be a waste type description. The waste type description is a document with descriptions of the origin of the waste, its treatment process, possible interim storage, transport to and handling at SFR and final disposal in SFR. The document also contains defined requirements for each waste type regarding chemical, physical, radiological and mechanical properties, and how these requirements are fulfilled. Waste type descriptions are written by the waste supplier and approved first by SKB and then by SSM. Before this document is approved by SSM, no waste of the type in question is allowed be deposited in SFR.

For the decommissioning waste there are at present no established waste types. In this report, an assumption has, however, been made about which waste types will be produced at decomissioning. The assumption is based on decommissioning studies of the nuclear power plants, Clink and Ågesta, data for decommissioning waste from SNAB and Svafo, and a number of compiled waste descriptions for decommissioning waste. For most of the decommissioning waste it is possible to adopt the code system that exists for operational waste today. In addition to the abbreviations given in Table 3-1, Å is used for decommissioning waste from Ågesta and V for decommissioning waste from Svafo. In Table 3-3, explanations are given for the abbreviations that will be used for the decommissioning waste. To be able to distinguish easily between operational and decommissioning waste, the letter D (for decommissioning) has been added after each waste type concerning decommissioning waste in the present report.

For decommissioning waste where no similar waste types exist today, like waste packages containing sand or asphalt and intact BWR pressure vessels, new designations for waste types have been adopted. In Table 3-3, "A" stands for asphalt, "C" for concrete, "S" for sand, "4K" for the new packaging tetramould and "BWR" for BWR pressure vessels. Tetramould is the only waste packaging not currently used.

It should be pointed out that the waste types ending in the letter ":D ", i.e. the entirety of Table 3-3, are only assumed and used to facilitate the account of waste in this report.

Table 3-1. Abbreviations for nuclear facilities in the code system.

Abbreviation	Nuclear facility
B	Barsebäck
C	Clab (in the future Clink)
F	Forsmark
O	Oskarshamn
R	Ringhals
S	Studsvik/Svafo

Table 3-2. Abbreviations for waste types in the code system.

Waste type	Waste vault	Waste	Waste packaging	Treatment
01	1BMA	Ion exchange resin	Concrete mould	Cement solidification
02	Silo	Ion exchange resin	Concrete mould	Cement solidification
04	Silo	Ion exchange resin	Steel drum	Cement solidification
05	1BMA	Ion exchange resin	Steel drum	Bitumen solidification
06	Silo	Ion exchange resin	Steel drum	Bitumen solidification
07	BTF	Low-level ion exchange resin	Concrete tank	Dewatering
10	1BMA	Sludge	Concrete mould	Cement solidification
11	Silo	Sludge and ion exchange resin	Steel mould	Cement solidification
12	1BLA	Trash and scrap metal	Container	–
13	BTF	Ash drum	Steel drum	Embedment in concrete
14	1BLA	Trash and scrap metal	Steel drum in container	Embedment in concrete
15	1BMA	Ion exchange resin	Steel mould	Cement solidification
16	Silo	Ion exchange resin	Steel mould	Cement solidification
17	1BMA	Ion exchange resin	Steel mould	Bitumen solidification
18	Silo	Ion exchange resin	Steel mould	Bitumen solidification
20	1BLA	Ion exchange resin	Steel drum in container	Bitumen solidification
21	1BMA	Trash and scrap metal	Steel drum	Embedment in concrete
23	1BMA	Trash and scrap metal	Steel/concrete mould	Embedment in concrete
24	Silo	Trash and scrap metal	Steel/concrete mould	Embedment in concrete
29	1BMA	Evaporator concentrate	Concrete mould	Cement solidification
99	–	Miscellaneous waste	–	–

Table 3-3. Abbreviations for assumed waste types for decommissioning waste.

Waste type	Waste vault	Waste	Waste packaging	Treatment
02:D	Silo	Ion exchange resin	Steel mould	Cement solidification
12:D	XBLA	Trash and scrap metal	Container	–
12A:D	XBLA	Asphalt, gravel, soil	Container	–
12C:D	XBLA	Concrete	Container	–
12S:D	XBLA	Sand	Container	–
16:D	Silo	Ion exchange resin	Steel mould	Cement solidification
18:D	Silo	Ion exchange resin	Steel mould	Bitumen solidification
23:D	XBMA	Trash and scrap metal	Concrete and steel mould	Embedment in concrete
4K23:D	XBMA	Trash and scrap metal	Tetramould of steel	Embedment in concrete
4K23C:D	XBMA	Concrete	Tetramould of steel	Embedment in concrete
4K23S:D	XBMA	Sand	Tetramould of steel	Embedment in concrete
25:D	XBMA	Ashes	Steel drum	Embedment in concrete
BWR:D	BRT	Reactor pressure vessel	–	–

3.2 Waste packaging and treatment methods

As described in previous sections, there are many different waste types in SFR, but the variants of the geometry of the waste packaging and the treatment methods for waste are limited.

The waste packaging for operational waste in the existing SFR consist of five different types:

- Steel drums. Standard 200-litre drums. The dimensions of the drums vary, but they are approximately 90 cm high and have a diameter of about 60 cm. In 1BMA and the Silo, drums are handled four by four placed on a drum tray or in a drum box. Both types are custom made for the deposition system. In BTF the drums are handled one by one. In BLA steel drums are only found as inner packaging in containers.
- Concrete moulds. Concrete cubes with sides of 1.2 m. The walls normally have a thickness of 10 cm, but variants of 25 cm and 35 cm also exist. The thickness of the bottom and lid is normally at least 10 cm. The moulds are deposited mainly in 1BMA and the Silo. Moulds with low dose rate are used to build stabilising walls in 1BTF.
- Steel moulds. Steel cubes with the same outside dimensions as concrete moulds, but only 5 or 6 mm thick walls. The steel moulds have room for more waste than those in concrete, but provide less radiation protection. The moulds are deposited in 1BMA and the Silo.
- Concrete tanks. The tanks have a length of 3.3 m, a width of 1.3 m and a height of 2.3 m. The walls are 15 cm thick. The tank has a dewatering system in the bottom of the tank which is connected to a vacuum line in the lid. The tanks are used for deposition of waste in 1BTF and 2BTF.
- ISO containers. Standard containers with the dimensions 3.0×2.4×1.3 m (10 feet half-height container), 3.0×2.4×2.6 m (10 feet full-height container), 6.1×2.5×1.3 m (20 feet half-height container) or with the dimensions 6.1×2.5×2.6 m (20 feet full-height container). The containers may contain waste placed in inner packaging such as drums, plastic bags or boxes. The containers may also contain scrap metal without an inner packaging. The containers are used for deposition in 1BLA.

It is assumed that the following waste packaging will be used for decommissioning waste:

- Concrete moulds. Concrete cubes with sides of 1.2 m. The moulds will be deposited in the Silo and BMA.
- Steel moulds. Steel cubes with sides of 1.2 m. The moulds will be deposited in the Silo and BMA
- Tetramoulds. Steel packaging with the dimensions 2.4×2.4×1.2 m, i.e. the same outer volume as four moulds. The tetramoulds will be deposited in BMA.
- Steel drums. Standard 200-litre drums. Positioned four by four on a drum tray. The drums will be deposited in BMA.
- ISO containers. Standard containers with the dimensions 6.1×2.5×1.3 m (20 feet half-height container). The containers will be deposited in BLA.

For the intact BWR pressure vessels no waste packaging will be used.

Figure 3-1 shows the waste packaging that are used in the existing SFR and the tetramould that will be added for decommissioning waste.

The different treatment methods for waste are:

- Cement solidification. Ion exchange resins, sludge or evaporator concentrates are mixed with cement in drums or moulds.
- Embedment in concrete. Trash and scrap metal are placed in moulds or steel drums and concrete is poured over it. Waste types 13 and 14 differ somewhat, as the waste is placed in 100-litre drums, which are then placed inside standard 200-litre drums and concrete is poured in the space between the drums.
- Bitumen solidification. Ion exchange resins or evaporator concentrates are dried and mixed with bitumen and then poured into steel moulds. Bitumen solidification has also been used in steel drums, but this treatment method is no longer used.

- Dewatering. Wet ion exchange resins or sludge are pumped into a concrete tank. Water is then led away in the bottom of the tank by vacuum suction.
- No treatment. Some waste is deposited without treatment, directly in containers.

No further treatment methods than those given above have been assumed for the decommissioning waste. Embedment in concrete and no treatment are the treatment methods that will mainly be used for the decommissioning waste. For decontamination waste to the Silo, both cement and bitumen solidification is assumed.

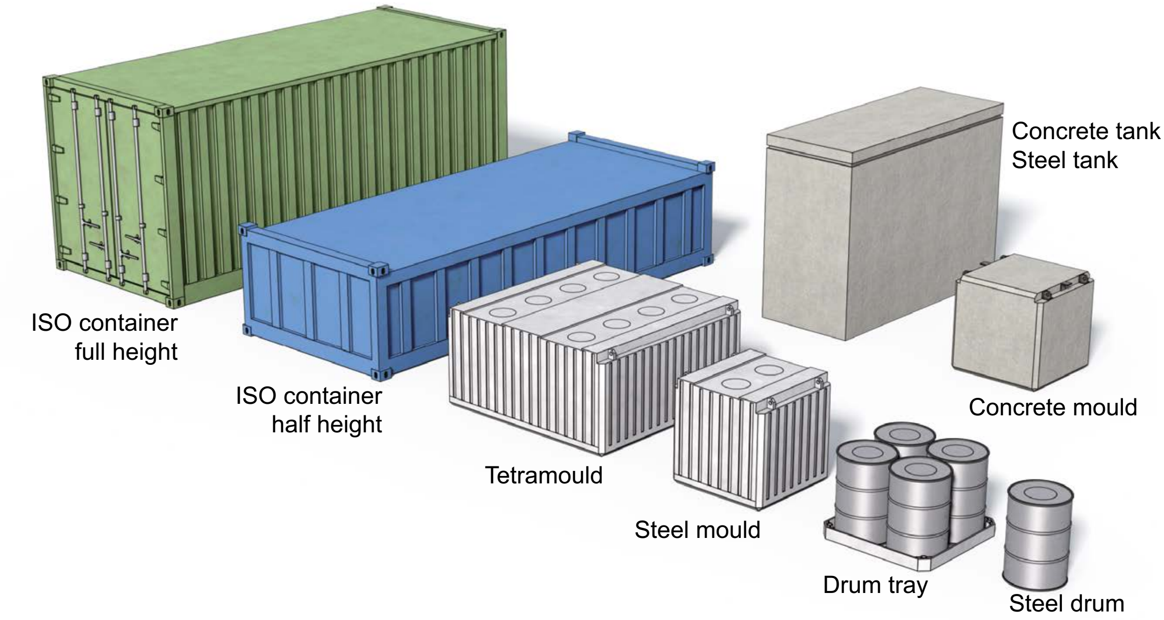

Figure 3-1. *Waste packaging in SFR.*

4 Description of the waste in the Silo

4.1 Waste in the Silo

Intermediate-level waste is deposited in the Silo. The waste contains bitumen- and cement-solidified ion exchange resins and smaller quantities of concrete-embedded trash and scrap metal. A small amount of cement-solidified sludge is also included. The waste is deposited in concrete or steel moulds and steel drums on drum trays.

4.2 Amount of waste in the Silo

The number of waste packages that are expected to be deposited in the Silo is presented per waste type in Table 4-1. How the number of waste packages has been estimated is described in Appendix A. The table also presents the total disposal volume per waste type. The volume calculations are described in Appendix B.

The total disposal volume is expected to be 16,459 m^3. The disposal capacity of the Silo is 17,740 m^3.

The estimated quantity of materials expected to be deposited in the Silo is presented in Table 4-2. The amounts include materials in the waste form and the packaging of the waste packages. The surface of metals that might be exposed for corrosion, and the estimated void in the waste packages, are also given. Material calculations are described in Appendix C. Mean values for the material content in each waste type are presented in Appendix E.

Table 4-1. Number of packages of different waste types in the Silo at the closure of SFR in 2075.

Waste type	Waste packaging	Number of waste packages in SFR	Disposal volume [m^3]
B.04	Steel drum	768	249
B.06	Steel drum	1,776	575
C.02	Concrete mould	1,361	2,352
C.16:D	Steel mould	7	12
C.24	Concrete mould	350	605
F.18	Steel mould	804	1,389
F.18:D	Steel mould	21	36
O.02/O.02:9	Concrete mould	1,944	3,359
O.16:D	Steel mould	28	48
O.24	Steel mould	204	353
R.02/R.02:9	Concrete mould	371	641
R.02:D	Steel mould	42	73
R.16	Steel mould	2,839	4,906
R.24	Steel mould	60	104
S.04	Steel drum	452	146
S.11	Steel mould	106	183
S.24	Concrete or steel mould	826	1,427

Table 4-2. Summed quantities of material in the Silo at closure of SFR in 2075.

Material	Weight [kg]	Corrosion surface [m²]	Void [m³]
Aluminium/zinc	8.26E+03	1.22E+03	–
Concrete	1.17E+07	–	–
Bitumen	1.06E+06	–	–
Cellulose	1.80E+04	–	–
Cement	1.22E+07	–	–
Filter aids	1.01E+04	–	–
Ion exchange resin	3.31E+06	–	–
Iron/steel	4.93E+06	2.21E+05	–
Sludge	3.53E+04	–	–
Other inorganic material	1.07E+06	–	–
Other organic material	5.31E+04	–	–
Void	–	–	2.14E+03

4.3 Radionuclide inventory in the Silo

The inventory of radionuclides in the Silo at closure of SFR is presented in Table 4-3. The calculation method is described in Appendix D. Mean values of the radioactivity in each waste type are presented in Appendix E.

The Silo is calculated to contain a total of 6.71×10^{14} Bq at the closure of SFR on 2075-12-31. It is about 69% of the total radionuclide inventory in SFR at the time of closure.

4.4 Waste placement in the Silo

Distribution of waste packages will be carried out based on the main principle that bitumen solidified waste is placed in the centre of the Silo.

Table 4-3. Total radioactivity of different radionuclides in the Silo at closure of SFR in 2075.

Nuclide	Radioactivity [Bq]	Nuclide	Radioactivity [Bq]	Nuclide	Radioactivity [Bq]
H-3	8.97E+09	Cd-113m	9.58E+09	Np-237	5.36E+08
Be-10	9.89E+05	In-115	0.00E+00	Pu-238	7.29E+10
C-14 org	7.55E+11	Sn-126	2.05E+08	Pu-239	1.70E+10
C-14 inorg	2.72E+12	Sb-125	1.32E+11	Pu-240	2.39E+10
C-14 ind	0.00E+00	I-129	9.84E+08	Pu-241	3.07E+11
Cl-36	8.94E+08	Cs-134	2.20E+11	Pu-242	1.23E+08
Ca-41	0.00E+00	Cs-135	4.47E+09	Am-241	2.32E+13
Fe-55	2.73E+12	Cs-137	5.97E+13	Am-242m	3.22E+08
Co-60	1.29E+13	Ba-133	6.16E+08	Am-243	1.59E+09
Ni-59	6.86E+12	Pm-147	3.58E+11	Cm-243	1.89E+08
Ni-63	5.47E+14	Sm-151	4.63E+11	Cm-244	9.26E+09
Se-79	1.05E+09	Eu-152	8.64E+08	Cm-245	1.49E+07
Sr-90	3.61E+12	Eu-154	5.24E+11	Cm-246	4.29E+06
Zr-93	4.48E+09	Eu-155	9.96E+10		
Nb-93m	9.31E+12	Ho-166m	6.82E+09		
Nb-94	8.65E+10	U-232	6.20E+05		
Mo-93	9.46E+09	U-234	3.58E+07		
Tc-99	5.00E+10	U-235	1.42E+07		
Pd-107	2.75E+08	U-236	1.58E+07		
Ag-108m	2.30E+11	U-238	3.28E+07		

5 Description of the waste in BRT

5.1 Waste in BRT

BRT will be designed to host the BWR pressure vessels from the Swedish nuclear power plants. The waste consists of iron/steel.

5.2 Amount of waste in BRT

The number of reactor pressure vessels (RPVs) that is expected to be deposited in BRT is presented per waste type in Table 5-1. The table also presents the total disposal volume per waste type. Assumptions and calculations are described in Appendix A and B.

The total disposal volume is expected to be 8,765 m^3.

The estimated quantity of material expected to be deposited in BRT is presented in Table 5-2. The surface of metals that might be exposed for corrosion, and the estimated void in the RPVs, are also given. Material calculations are described in Appendix C. Material content for the different reactor pressure vessels are presented in Appendix E.

Table 5-1. Number of packages for different waste types in BRT at closure of SFR in 2075.

Waste type	Waste	Number of waste packages in SFR	Disposal volume [m^3]
B.BWR:D	Intact reactor pressure vessel	2	1,720
F.BWR:D	Intact reactor pressure vessel	3	3,570
O.BWR:D	Intact reactor pressure vessel	3	2,625
R.BWR:D	Intact reactor pressure vessel	1	850

Table 5-2. Summed quantities of materials in BRT at the closure of SFR in 2075.

Material	Weight [kg]	Corrosion surface [m2]	Void [m3]
Iron/steel	5.55E+06	7.24E+03	–
Void	–	–	4.67E+03

5.3 Radionuclide inventory in BRT

The inventory of radionuclides in BRT at the closure of SFR is presented in Table 5-3. The calculation method is described in Appendix D. The radioactivity for the different reactor pressure vessels is presented in Appendix E.

BRT is calculated to contain a total of 1.59×10^{13} Bq at the closure of SFR on 2075-12-31. This is about 1.6% of the total radionuclide inventory in SFR at closure.

Table 5-3. Total radioactivity of different radionuclides in BRT at closure of SFR in 2075.

Nuclide	Radioactivity [Bq]
H-3	0.00E+00
Be-10	0.00E+00
C-14 org	0.00E+00
C-14 inorg	0.00E+00
C-14 ind	1.02E+10
Cl-36	7.20E+06
Ca-41	0.00E+00
Fe-55	1.49E+10
Co-60	1.93E+11
Ni-59	1.60E+11
Ni-63	1.44E+13
Se-79	0.00E+00
Sr-90	2.32E+10
Zr-93	1.84E+08
Nb-93m	1.06E+12
Nb-94	7.94E+09
Mo-93	2.99E+09
Tc-99	4.50E+08
Pd-107	0.00E+00
Ag-108m	1.62E+09
Cd-113m	0.00E+00
In-115	0.00E+00
Sn-126	7.52E+05
Sb-125	1.34E+07
I-129	0.00E+00
Cs-134	0.00E+00
Cs-135	0.00E+00
Cs-137	0.00E+00
Ba-133	0.00E+00
Pm-147	1.37E+06
Sm-151	3.42E+08
Eu-152	5.41E+05
Eu-154	9.26E+07
Eu-155	2.40E+06
Ho-166m	8.00E+03
U-232	6.86E+03
U-234	0.00E+00
U-235	1.49E+01
U-236	3.92E+05
U-238	0.00E+00
Np-237	4.70E+05
Pu-238	2.71E+09
Pu-239	4.16E+08
Pu-240	5.93E+08
Pu-241	9.06E+09
Pu-242	3.11E+06
Am-241	1.99E+09
Am-242m	1.32E+07
Am-243	4.14E+07
Cm-243	6.38E+06
Cm-244	6.74E+08
Cm-245	6.82E+05
Cm-246	2.24E+05

6 Description of the waste in BMA

6.1 Waste in BMA

The waste vault 1BMA is designed for intermediate-level waste with a lower dose rate than waste contained in the Silo, or waste that somehow is not suitable for the Silo. The waste consists mainly of bitumen- or cement-solidified ion exchange resins and concrete-embedded trash and scrap metal. A small amount of bitumen- or cement-solidified evaporator concentrates and sludge is also deposited. All waste in 1BMA is packaged in concrete and steel moulds or steel drums on drum trays or in drum boxes.

The extended repository part, XBMA, will be designed for intermediate-level waste in the form of trash, scrap metal and concrete. In addition, smaller quantities of sludge and ion exchange resins may be deposited. The waste in BMA will normally be packaged in concrete or steel moulds and tetramoulds. Steel drums placed on drum trays may also be used as packaging.

6.2 Amount of waste in BMA

The number of waste packages that are expected to be deposited in BMA is presented per waste type in Table 6-1. How the number of waste packages has been estimated is presented in Appendix A. The table also presents the total disposal volume per waste type. Volume calculations are described in Appendix B.

The total disposal volume is expected to be 24,791 m^3.The disposal capacity for 1BMA is 13,090 m^3.

The estimated quantity of material expected to be deposited in BMA is presented in Table 6-2. The amounts include materials in the waste form and the packaging of the waste packages. The surface of metals that might be exposed for corrosion, and the estimated void in the waste packages, are also given. Material calculations are described in Appendix C. Mean values for the material content of each waste type are presented in Appendix E.

6.3 Radionuclide inventory in BMA

The inventory of radionuclides in BMA at the closure of SFR is presented in Table 6-3. The calculation method is described in Appendix D. Mean values of the radioactivity in each waste type are presented in Appendix E.

BMA is calculated to contain a total of 2.74×10^{14} Bq at the closure of SFR on 2075-12-31. It is about 28% of the total radionuclide inventory in SFR at closure.

6.4 Waste placement in 1BMA

Table 6-4 shows the distribution of the different waste types in the compartments in 1BMA, for waste deposited before 2012-12-31. Compartments 1–5, compartments 7 and 9 are full and therefore sealed.

Table 6-1. Number of packages of different waste types in BMA at the closure of SFR in 2075.

Waste type	Waste packaging	Number of waste packages in SFR	Disposal volume [m³]
B.05/B.05:9	Steel drum	3,360	1,089
B.05:2	Drum box	224	387
B.23	Steel mould	33	57
B.23:D	Steel mould	608	1,051
C.01:9	Concrete mould	68	118
C.23	Concrete mould	161	278
C.4K23:D	Tetramould	3	21
F.05:1/F.05:2	Steel drum	1,712	555
F15	Steel mould	11	19
F.17/F.17:1	Steel mould	1,382	2,388
F.23	Concrete or steel mould	527	911
F.4K23:D	Tetramould	237	1,638
F.4K23C:D	Tetramould	70	484
F.99:1	Steel mould	2	3
O.01:9	Concrete mould	675	1,166
O.23/O.23:9	Concrete mould	609	1,052
O.4K23:D	Tetramould	198	1,369
O.4K23C:D	Tetramould	82	567
O.4K23S:D	Tetramould	15	104
R.01/R.01:9	Concrete mould	1,689	2,919
R.10	Concrete mould	121	209
R.15	Steel mould	254	439
R.23	Concrete or steel mould	606	1,047
R.23:D	Steel mould	153	264
R.4K23:D	Tetramould	314	2,170
R.4K23C:D	Tetramould	149	1,030
R.29	Concrete mould	380	657
S.21	Steel drum	488	158
S.23	Concrete mould	718	1,241
S.23:D	Concrete mould	164	283
S.25:D	Steel drum	2,384	772
Å.4K23:D	Tetramould	45	311
Å.4K23C:D	Tetramould	5	35

Table 6-2. Summed quantities of material in BMA at the closure of SFR in 2075.

Material	Weight [kg]	Corrosion surface [m²]	Void [m³]
Aluminium/zinc	2.77E+04	4.07E+03	–
Ashes	1.51E+05	–	–
Concrete	2.58E+07	–	–
Bitumen	1.93E+06	–	–
Cellulose	1.51E+05	–	–
Cement	4.84E+06	–	–
Filter aids	8.37E+04	–	–
Evaporator concentrate	4.34E+05	–	–
Ion exchange resin	2.13E+06	–	–
Iron/steel	1.21E+07	5.51E+05	–
Sand	1.06E+05	–	–
Sludge	1.03E+05	–	–
Other inorganic material	1.16E+05	–	–
Other organic material	3.55E+05	–	–
Void	–	–	4.33E+03

Table 6-3. Total radioactivity of different radionuclides in BMA at the closure of SFR in 2075.

Nuclide	Radioactivity [Bq]
H-3	3.31E+12
Be-10	2.43E+05
C-14 org	1.51E+11
C-14 inorg	1.91E+12
C-14 ind	5.09E+09
Cl-36	5.36E+08
Ca-41	1.56E+10
Fe-55	1.58E+11
Co-60	2.39E+12
Ni-59	3.05E+12
Ni-63	2.39E+14
Se-79	2.17E+08
Sr-90	9.09E+11
Zr-93	1.43E+09
Nb-93m	1.31E+13
Nb-94	9.49E+10
Mo-93	5.02E+09
Tc-99	7.64E+09
Pd-107	2.60E+09
Ag-108m	6.01E+10
Cd-113m	8.91E+08
In-115	3.13E+05
Sn-126	4.37E+07
Sb-125	3.06E+08
I-129	1.54E+08
Cs-134	3.70E+08
Cs-135	8.95E+08
Cs-137	9.05E+12
Ba-133	1.92E+08
Pm-147	7.78E+08
Sm-151	1.18E+11
Eu-152	1.33E+11
Eu-154	3.01E+10
Eu-155	1.40E+09
Ho-166m	1.93E+09
U-232	2.35E+05
U-234	9.71E+06
U-235	3.08E+06
U-236	8.65E+06
U-238	7.18E+06
Np-237	3.49E+07
Pu-238	5.17E+10
Pu-239	9.55E+09
Pu-240	1.31E+10
Pu-241	1.90E+11
Pu-242	7.02E+07
Am-241	7.04E+10
Am-242m	2.28E+08
Am-243	8.64E+08
Cm-243	1.21E+08
Cm-244	1.14E+10
Cm-245	1.21E+07
Cm-246	3.87E+06

Table 6-4. Number of packages per waste type deposited in the different compartments in 1BMA until 2012-12-31.

Waste type	Waste packaging	Matrix	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	In total
B.05/B.05:9	Drum on drum tray	Bitumen			1,168		2,000	192										3,360
B.05:2	Drum in drum box	Bitumen		382	270		96	144										892
C.01:9	Concrete mould	Cement							2			5						7
C.01:9	Concrete mould masonite	Cement				20			10	1	21	9						61
C.23	Concrete mould	Concrete				1				12	30							43
F.05:1	Drum on drum tray	Bitumen		1,454														1,454
F.05:2	Drum on drum tray	Bitumen		258														258
F.15	Steel mould	Cement				8		3										11
F.17	Steel mould	Bitumen			144		8	247				16						415
F.17:1	Steel mould	Bitumen					20	12										32
F.23	Concrete mould	Concrete				49			2	4		2						57
F.23	Steel mould	Concrete				15			10	88		38						151
F.99:1	Steel mould	–						2										2
O.01:9	Concrete mould	Cement				11			209	10	28	134						392
O.01:9	Concrete mould masonite	Cement				45			43	19	156	15						278
O.23/O.23:9	Concrete mould	Concrete				35			36	134	137	113						455
R.01/R.01:9	Concrete mould	Cement	576	148	144	144	144	144	144	146	88			8				1,686
R.10	Concrete mould	Cement								36	48							84
R.15	Steel mould	Cement				124												124
R.23	Concrete mould	Concrete				124			120	38	52			4				338
R.23	Steel mould	Concrete								80	16							96
In total	**–**	**–**	**576 ***	**2,242 ***	**1,726 ***	**576 ***	**2,268 ***	**744**	**576 ***	**568**	**576 ***	**332**	**0**	**12**	**0**	**0**	**0**	**10,196**

*Compartments 1–5, 7 and 9 are full

7 Description of the waste in BTF

7.1 Waste in BTF

1BTF is designed mainly to host dewatered ion exchange resins, but cement-solidified ion exchange resins and concrete embedded ashes are also deposited. All waste in 1BTF is disposed in concrete tanks, concrete moulds or in steel drums.

2BTF is designed mainly to host dewatered ion exchange resins. All waste in 2BTF is disposed in concrete tanks.

7.2 Amount of waste in BTF

The number of waste packages expected to be deposited in BTF are presented per waste type in Table 7-1. How the number of waste packages has been estimated is presented in Appendix A. The table also presents the total disposal volume per waste type. The volume calculations are described in Appendix B.

The total disposal volume is estimated to be 14,513 m^3. The disposal capacity of BTF is 15,310 m^3.

The estimated quantity of materials expected to be deposited in BTF is presented in Table 7-2. The amounts include materials in the waste form and the packaging of the waste packages. The surface of metals that might be exposed for corrosion, and the estimated void in the waste packages, are also given. Material calculations are described in Appendix C. Mean values of the material content in each waste type are presented in Appendix E.

Table 7-1. Number of packages of different waste types in BTF at the closure of SFR in 2075.

Waste type	Waste packaging	Number of waste packages in SFR	Disposal volume [m^3]
B.07/B.07:9	Concrete tank	232	2,320
F.99:2	Steel box	18	180
O.01:9	Concrete mould	28	48
O.07/O.07:9	Concrete tank	890	8,900
O.99:1	Cortén box	40	135
R.01/R.01:9	Concrete mould	91	157
R.10	Concrete mould	4	7
R.23	Concrete mould	21	36
R.99:1	Reactor pressure vessel lid	1	100
S.13	Steel drum	8,116	2,630

Table 7-2. Summed quantities of material in BTF at the closure of SFR in 2075.

Material	Weight [kg]	Corrosion surface [m^2]	Void [m^3]
Aluminium/zinc	5.28E+04	7.81E+03	–
Ashes	5.15E+05	–	–
Concrete	1.44E+07	–	–
Cellulose	1.05E+03	–	–
Cement	2.37E+05	–	–
Filter aids	2.04E+05	–	–
Ion exchange resin	1.25E+06	–	–
Iron/steel	3.11E+06	1.17E+05	–
Sludge	6.90E+04	–	–
Other organic material	1.32E+05	–	–
Void	–	–	1.15E+03

7.3 Radionuclide inventory in BTF

The inventory of radionuclides in BTF at closure of SFR is presented in Table 7-3. The calculation method is described in Appendix D. Mean values of the radioactivity in each waste type are presented in Appendix E.

BTF is calculated to contain a total of 6.37×10^{12} Bq at the closure of SFR on 2075-12-31. It is about 0.7% of the total radionuclide inventory in SFR at closure.

Table 7-3. Total radioactivity of different radionuclides in BTF at closure of SFR in 2075.

Nuclide	Radioactivity [Bq]
H-3	1.75E+08
Be-10	3.85E+04
C-14 org	1.59E+10
C-14 inorg	4.58E+11
C-14 ind	0.00E+00
Cl-36	3.10E+07
Ca-41	0.00E+00
Fe-55	1.98E+08
Co-60	4.04E+10
Ni-59	7.13E+10
Ni-63	4.30E+12
Se-79	3.11E+07
Sr-90	9.24E+10
Zr-93	6.43E+07
Nb-93m	3.78E+09
Nb-94	6.66E+08
Mo-93	2.45E+08
Tc-99	2.85E+09
Pd-107	7.78E+06
Ag-108m	3.72E+09
Cd-113m	1.40E+08
In-115	0.00E+00
Sn-126	3.89E+06
Sb-125	1.78E+07
I-129	3.29E+07
Cs-134	1.59E+05
Cs-135	1.21E+08
Cs-137	1.34E+12
Ba-133	1.02E+07
Pm-147	8.41E+06
Sm-151	1.27E+10
Eu-152	6.85E+07
Eu-154	3.78E+09
Eu-155	1.08E+08
Ho-166m	2.47E+08
U-232	2.29E+04
U-234	1.44E+06
U-235	1.85E+07
U-236	7.57E+05
U-238	1.73E+06
Np-237	3.04E+06
Pu-238	2.55E+09
Pu-239	6.58E+08
Pu-240	7.84E+08
Pu-241	9.72E+09
Pu-242	4.32E+06
Am-241	7.97E+09
Am-242m	1.05E+07
Am-243	5.03E+07
Cm-243	4.24E+06
Cm-244	2.96E+08
Cm-245	4.30E+05
Cm-246	1.14E+05

8 Description of the waste in BLA

8.1 Waste in BLA

1BLA is designed to host low-level trash and scrap metal waste. Some concrete-embedded trash and scrap metal in steel drums and bitumen-solidified ion exchange resins in steel drums are also deposited. All waste is contained in ISO containers. As mentioned in previous chapters, the containers may contain various smaller inner packaging such as steel drums or steel boxes.

XBLA will be designed for low-level waste in the form of trash and scrap metal, sand, concrete, gravel, soil and asphalt. The waste in XBLA will be deposited in ISO containers. In the containers, the waste may be placed in inner packaging such as steel drums or steel boxes.

8.2 Amount of waste in BLA

The number of waste packages expected to be deposited in BLA is presented per waste type in Table 8-1. How the number of waste packages has been estimated is presented in Appendix A. The table also presents the total disposal volume per waste type. The volume calculations are described in Appendix B.

The total disposal volume is expected to be 90,230 m^3. The disposal capacity of 1BLA is 14,280 m^3.

Table 8-1. Number of packages of different waste types in BLA at the closure of SFR in 2075.

Waste type	Waste packaging	Number of waste packages in SFR	Disposal volume [m^3]
B.12/B.12:1	Container 20 m^3	193	3,860
B.12	Container 40 m^3	61	2,440
B.12:D	Container 20 m^3	297	5,940
B.12C:D	Container 20 m^3	389	7,780
B.12S:D	Container 20 m^3	190	3,800
B.20	Container 20 m^3	12	240
C.12:D	Container 20 m^3	11	220
C.12C:D	Container 20 m^3	7	140
F.12	Container 10 m^3	27	270
F.12	Container 20 m^3	43	860
F.12:D	Container 20 m^3	529	10,580
F.12C:D	Container 20 m^3	152	3,040
F.12S:D	Container 20 m^3	53	1,060
F.20	Container 20 m^3	15	300
O.12/O.12: 1	Container 20 m^3	81	1,620
O.12	Container 40 m^3	10	400
O.12:D	Container 20 m^3	457	9,140
O.12C:D	Container 20 m^3	160	3,200
O.12S:D	Container 20 m^3	37	740
O.99:3	Container 40 m^3	5	200
R.12/R.12: 1	Container 20 m^3	33	660
R.12	Container 40 m^3	118	4,720
R.12:D	Container 20 m^3	389	7,780
R.12C:D	Container 20 m^3	60	1,200
R.12S:D	Container 20 m^3	32	640
S.12	Container 20 m^3	260	5,200
S.12:D	Container 20 m^3	63	1,260
S.12C:D	Container 20 m^3	26	520
S.14	Container 20 m^3	87	1,740
V.12:D	Container 20 m^3	82	1,640
V.12A:D	Container 20 m^3	200	4,000
V.12C:D	Container 20 m^3	227	4,540
Å.12:D	Container 20 m^3	10	200
Å.12C:D	Container 20 m^3	15	300

The estimated quantity of materials expected to be deposited in BLA is presented in Table 8-2. The amounts include materials in the waste form and the packaging of the waste packages. The surface of metals that might be exposed for corrosion, and the estimated void in the waste packages, are also given. Material calculations are described in Appendix C. Mean values for the material content of each waste type are presented in Appendix E.

Table 8-2. Summed quantities of material in BLA at closure of SFR in 2075.

Material	Weight [kg]	Corrosion surface [m^2]	Void [m^3]
Aluminium/zinc	1.33E+05	1.97E+04	–
Asphalt, gravel, soil	3.60E+06	–	–
Concrete	1.81E+07	–	–
Bitumen	1.18E+05	–	–
Cellulose	6.66E+05	–	–
Cement	7.50E+04	–	–
Evaporator concentrate	2.70E+02	–	–
Ion exchange resin	9.74E+04	–	–
Iron/steel	3.89E+07	2.08E+06	–
Sand	5.26E+06	–	–
Sludge	7.25E+02	–	–
Other inorganic material	4.35E+05	–	–
Other organic material	3.50E+06	–	–
Void	–	–	3.90E+04

8.3 Radionuclide inventory in BLA

The inventory of radionuclides in BLA at closure of SFR is presented in Table 8-3. The calculation method is described in Appendix D. Mean values of the radioactivity in each waste type are presented in Appendix E.

BLA is calculated to contain a total of 2.39×10^{12} Bq at closure of SFR on 2075-12-31. This is about 0.2% of the total radionuclide inventory in SFR at closure.

Table 8-3. Total radioactivity of different radionuclides in BLA at closure of SFR in 2075.

Nuclide	Radioactivity [Bq]	Nuclide	Radioactivity [Bq]	Nuclide	Radioactivity [Bq]
H-3	1.95E+11	Pd-107	1.82E+06	U-234	5.71E+05
Be-10	1.91E+03	Ag-108m	1.72E+09	U-235	6.21E+08
C-14 org	3.04E+08	Cd-113m	8.09E+06	U-236	2.46E+05
C-14 inorg	4.95E+09	In-115	0.00E+00	U-238	9.10E+08
C-14 ind	1.19E+09	Sn-126	7.98E+06	Np-237	3.28E+05
Cl-36	6.77E+07	Sb-125	4.94E+06	Pu-238	1.87E+09
Ca-41	3.91E+09	I-129	2.37E+06	Pu-239	3.43E+08
Fe-55	4.54E+08	Cs-134	1.41E+06	Pu-240	3.63E+08
Co-60	2.70E+10	Cs-135	1.79E+08	Pu-241	7.03E+09
Ni-59	1.55E+10	Cs-137	5.14E+11	Pu-242	2.11E+06
Ni-63	1.43E+12	Ba-133	1.28E+07	Am-241	2.46E+09
Se-79	6.34E+06	Pm-147	1.49E+06	Am-242m	5.86E+06
Sr-90	2.47E+10	Sm-151	6.06E+09	Am-243	2.26E+07
Zr-93	3.06E+07	Eu-152	1.74E+10	Cm-243	4.16E+06
Nb-93m	1.34E+11	Eu-154	3.07E+08	Cm-244	3.34E+08
Nb-94	1.01E+09	Eu-155	1.32E+07	Cm-245	2.58E+05
Mo-93	9.18E+07	Ho-166m	9.45E+07	Cm-246	7.66E+04
Tc-99	2.35E+09	U-232	1.17E+04		

9 Uncertainties

The quantity of waste in SFR, including the content of different materials and radionuclides in the waste, is associated by numerous uncertainties, both the waste that is already deposited as well as the waste that is expected to arise. These uncertainties must be considered when making quantitative assessments, in particular regarding the dimensioning of an extended SFR and for long-term safety. Some uncertainties are difficult to foresee and impossible to quantify. Changes in laws and political decisions can result in changed premises and provide other waste quantities than those that have been estimated at the present time. Other changed premises that may be of importance for the forecasted waste quantities, with regard to volume, material and radioactivity, are changed operational conditions. For example, the extent of future fuel damage, higher enrichment of U-235 in the nuclear fuel and power increases may have a great impact. The closure of today's reactors could occur earlier or later than currently assumed and a decision on new nuclear power plants cannot be entirely ruled out.

In this section, the most significant uncertainties are presented in general terms. In the appendices A-D, uncertainties concerning the number of waste packages, disposal volumes, material content and radionuclide inventory for both deposited and future waste are discussed in more detail.

The forecasted waste production for operational waste is presented in the form of a minimum and a maximum range per waste type, and forms the main contribution to the uncertainty in estimating the number of operational waste packages. The forecast is based on operating experience and knowledge concerning how waste production has fluctuated in previous years. For decommissioning waste, the minimum and maximum range for the number of packages is based on the uncertainty in the inventoried waste amount presented in decommissioning studies. For waste from BKAB, however, no additional uncertainty allowance is made, as the quantities from BKAB are already estimated conservatively. The future waste quantities will also be affected by the extent of post treatments. This mainly concerns the decommissioning waste, as there are yet no established procedures for the handling of decommissioning waste. Examples of this are what packing degree can ultimately be attained and whether further volume reductions can be achieved by melting of low-level process systems. The waste quantities in SFR are also determined by how the work with clearance levels for free release, and the use of near-surface repositories, will proceed in the future. The possibility of depositing the very low-level decommissioning waste in near-surface repositories instead of in SFR would greatly reduce the waste volume to BLA. Material content in operational waste is based on the knowledge of today's waste. Waste packages with ion exchange resins have a well-defined material content while packages containing trash and scrap metal can be more variable. For the operational waste, however, there are waste type descriptions that must be followed. The decommissioning waste consists mainly of scrap metal, concrete and sand. The scrap metal is assumed to be iron/steel, but small quantities of other materials can occur. During decomissioning, secondary waste may also arise. In the present report, estimated quantities of secondary waste have been added to the material quantities inventoried in the decommissioning studies. The secondary waste has a higher degree of uncertainty than the inventoried material.

The radionuclide inventory has been estimated based on different nuclide-specific measurements, calculations and correlations with key nuclides. The uncertainty of these methods thereby affects the total inventory. Furthermore, the radioactivity in the decommissioning waste from SNAB and Svafo has not been estimated and is therefore missing from the compilation of the radionuclide inventory. This results in the inventory being, to some extent, underestimated, although the radioactivity quantities in these facilities are likely to be lower than those that arise during decommissioning of the nuclear power plants.

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1393449 ver 1.0	Klor-36 – Uppskattning av aktivitet i driftavfall från svenska LWR . (In Swedish.)	ALARA Engineering, 2005
1393496 ver 1.0	Mo-93, Tc-99 och Cs-135: Uppskattning av aktivitet i driftavfall från svenska LWR, Clab och Studsvik. (In Swedish.)	ALARA Engineering, 2010
1393796 ver 1.0	Measurement of 14C in process water from CLAB and estimation of the accumulated amount in spent ion exchange resins	Lund University, 2007
1400742 ver 1.0	Kontroll av SFR-avfall. Delprojekt mätteknik. (In Swedish.)	Gammadata mätteknik AB, 1990
1403739 ver 1.0	Decomissioning cost analysis for Barsebäck nuclear station. Document S33-1567-002, rev 0	Griffiths G M, Garrett T J, Cloutier W A, Adler J J, TLG Services, Inc. 2008

Appendix A

Number of waste packages

A1 Background and purpose

The number of deposited and forecasted waste packages serves as a basis for the volume, material and radioactivity calculations in Appendices B–E. The purpose of the compilation of the number of waste packages is to provide a prognosis for what type of short-lived low and intermediate-level waste that will arise during the operation and decommissioning of the Swedish nuclear facilities, and to describe the main uncertainties.

A2 Operational waste

A2.1 Available information sources

Information on the number of deposited and forecasted waste packages in SFR can be found in the report and forecasting tool Triumf NG. The input to Triumf NG is based on the following sources:

- Data on the deposited waste packages, such as waste type and waste vault, are annually migrated to Triumf NG from the TRIUMF database at SFR. Waste package information is transferred from the waste suppliers with a waste data file in to TRIUMF before the deposition in SFR.
- The forecasts for future waste packages are based on the knowledge and experience of waste management. Forecast data have been provided by each waste supplier.

A2.2 Assumptions

Existing waste in SFR is waste deposited until the end of December 2012. The forecast includes waste that will be disposed of from January 1st, 2013.

Operational waste delivered from the Swedish nuclear power companies BKAB, FKA, OKG and RAB is deposited in SFR, as well as waste from the nuclear facilities at SNAB, Svafo and Clab/Clink. Any operational waste from the planned nuclear facility ESS (European Spallation Source) is not included in the present report.

For an estimate of future operational waste in SFR, all waste currently categorised as suitable for SFR, i.e short-lived low and intermediate-level waste, has been included. The forecasted waste is assumed to follow existing or future waste type descriptions. It is also assumed that the type of waste that today can get released for unrestricted use, or waste that can be placed in near-surface repositories, or is classified as SFL waste, will not be disposed of in SFR in the future.

In the forecast data from the waste suppliers an estimated annual production of waste packages per waste type is given, as well as the number of waste packages that currently are interim-stored at the waste suppliers, both complete packages and waste still untreated. The forecast for operational waste extends to the expected final year for each waste supplier, see Table A2-1. Depending on how the prognosis has been designed, the final year may be the year when the last operational waste package is expected to be produced, or it can be a calculated average for when the reactors at a site are shut down, which is the case for FKA and OKG. In this report it is assumed that the production of waste will go on until the end of December in the given end year.

Table A2-1. Forecasted final year for production of operational waste.

Waste supplier	Forecasted final year of waste production
BKAB	2020
FKA	2042
OKG	2037
RAB	2044
Clink	2070
SNAB	2040
Svafo	2045

At BKAB the last reactor was shut down in 2005. Service operation is now being conducted until the decommissioning starts. Waste from decontamination that is generated during service operation is treated as operational waste. For the other nuclear power plants, the time between shutdown and decommissioning is expected to be shorter than for BKAB, and the decontamination is instead regarded as decommissioning waste.

For the nuclear power plants in operation, the reactors at FKA and OKG are expected to have an operating time of 60 years. At RAB, reactors R1 and R2 are expected to be operated for 50 years, while R3 and R4 will be in operation for 60 years.

It is, in present report, assumed that interim-stored waste is deposited in 2013 and forecasted waste in the same year as the waste arise. Interim-stored and forecasted waste packages are assumed to be placed in the types of waste vaults stated in the corresponding type descriptions.

A2.3 Calculation methods

Calculation of the number of waste packages is made in the report and forecasting tool Triumf NG v1.0.1.3. In Triumf NG, deposited, interim-stored and forecasted waste packages are registered per waste type and per packaging code, as well as per waste vault.

A2.4 Results

Table A2-2 presents the number of packages per waste type in each waste vault at closure of SFR. The table also presents deposited, interim-stored and forecasted waste packages separately.

In the table, waste types of different variants or with different packaging, with the exception of miscellaneous wastes with the numbering 99, have been combined. Waste content, conditioning material and packaging material may therefore vary within the reported waste type. The differences in the different variants are presented in Appendix E. Containers may have different sizes, in which case the disposal volume is given in the table, otherwise standard measures are applied following Appendix B.

Waste from Clab has previously been designated waste types with the letter O, but now generally has its own waste types with the letter C. Here the deposited packages from Clab are described with the letter C, even if they previously have been deposited as waste types with the letter O.

A3 Decommissioning waste

A3.1 Available information sources

Information on the number of waste packages arising during the decommissioning of the nuclear power plants and facilities at BKAB, FKA, OKG, RAB, Ågesta and Clink is presented in decommissioning studies (Jönsson 2013, Anunti et al. 2013, Larsson et al. 2013, Hansson et al. 2013, Lindow 2012, Edelborg et al. 2014). Westinghouse has conducted the decommissioning studies for FKA, OKG and Clink, and TLG Services has conducted the decommissioning studies for BKAB and Ågesta, and delivered data as a basis for RAB. The decommissioning studies are based on the following inventories:

- The BWR reactors B1 and B2 are sister reactors. Each reactor has a thermal effect of 1,800 MW_t and an electrical effect of 615 MW_e. An inventory has been carried out on each unit.
- The BWR reactors F1 and F2 are sister reactors. The reactors have a thermal effect of 2,928 MW_t and an electrical effect of 987 MW_e and 1,000 MW_e respectively. A complete inventory has been carried out on each unit.
- The BWR reactor O1 has a thermal effect of 1,375 MW_t and electrical effect of 491 MW_e. A complete inventory has been carried out on the unit.
- The BWR reactor O2 is a sister reactor to the reactors B1 and B2. O2 has a thermal effect of 1,800 MW_t and an electrical effect of 620 MW_e. The decommissioning study is based on the inventory that has been carried out of B1 and B2 with correction and review by OKG.

Table A2-2. Number of waste packages with operational waste that are expected to be deposited in SFR.

Waste vault	Waste type	Waste packaging	Deposited 2012-12-31	Interim- stored 2012-12-31	Prognosis 2013–2075	The total number of operational waste packages 2075-12-31
Silo	B.04	Steel drums	96	672	0	768
	B.06	Steel drums	1,776	0	0	1,776
	C.02	Concrete mould	150	251	960	1,361
	C.24	Concrete mould	0	54	296	350
	F.18	Steel mould	264	90	450	804
	O.02/O.02:9	Concrete mould	874	445	625	1,944
	O.24	Steel mould	0	204	0	204
	R.02/R.02:9	Concrete mould	348	23	0	371
	R.16	Steel mould	1,164	349	1,326	2,839
	R.24	Steel mould	0	15	45	60
	S.04	Steel drums	32	22	398	452
	S.11	Steel mould	96	10	0	106
	S.24	Concrete or steel mould	0	17	809	826
BMA	B.05/B.05:9	Steel drums	3,360	0	0	3,360
	B.05:2	Drum box	224	0	0	224
	B.23	Steel mould	0	25	8	33
	C.01:9	Concrete mould	68	0	0	68
	C.23	Concrete mould	43	15	103	161
	F.05:1/F.05:2	Steel drums	1,712	0	0	1,712
	F15	Steel mould	11	0	0	11
	F.17/F.17:1	Steel mould	447	185	750	1,382
	F.23	Concrete or steel mould	208	19	300	527
	F.99:1	Steel mould	2	0	0	2
	O.01:9	Concrete mould	670	5	0	675
	O.23/O.23:9	Concrete mould	455	29	125	609
	R.01/R.01:9	Concrete mould	1,686	3	0	1,689
	R.10	Concrete mould	84	5	32	121
	R.15	Steel mould	124	50	80	254
	R.23	Concrete or steel mould	434	54	118	606
	R.29	Concrete mould	0	0	380	380
	S.21	Steel drums	0	488	0	488
	S.23	Concrete mould	0	0	718	718
BTF	B.07/B.07:9	Concrete tank	216	12	4	232
	F.99:2	Steel box	18	0	0	18
	O.01:9	Concrete mould	28	0	0	28
	O.07/O.07:9	Concrete tank	537	3	350	890
	O.99:1	Cortén box	0	40	0	40
	R.01/R.01:9	Concrete mould	91	0	0	91
	R.10	Concrete mould	4	0	0	4
	R.23	Concrete mould	21	0	0	21
	R.99:1	Reactor pressure vessel lid	1	0	0	1
	S.13	Steel drums	4,800	1,711	1,605	8,116
BLA	B.12/B.12:1	Container 20 m^3	193	0	0	193
	B.12	Container 40 m^3	33	12	16	61
	B.20	Container 20 m^3	12	0	0	12
	F.12	Container 10 m^3	0	21	6	27
	F.12	Container 20 m^3	24	4	15	43
	F.20	Container 20 m^3	15	0	0	15
	O.12/O.12:1	Container 20 m^3	1	30	50	81
	O.12	Container 40 m^3	5	5	0	10
	O.99:3	Container 40 m^3	0	5	0	5
	R.12/R.12:1	Container 20 m^3	26	7	0	33
	R.12	Container 40 m^3	44	21	53	118
	S.12	Container 20 m^3	0	50	210	260
	S.14	Container 20 m^3	75	12	0	87

- The BWR reactor O3 is a sister reactor to the reactor F3. O3 has a thermal effect of 3,900 MW_t and electrical effect of 1,465 MW_e. The decommissioning study is based on previous reference studies, which have been completed after revision in 2006, and based on an analysis of the differences with F3.
- The BWR reactor F3 is a sister reactor to the reactor O3. F3 has a thermal effect of 3,300 MW_t and an electrical effect of 1,170 MW_e. The decommissioning study is based on O3's reference study, completed with inventory and correction by FKA.
- The BWR reactor R1 has a thermal effect of 2,500 MW_t and an electrical effect of 843 MW_e. An inventory has been carried out on the unit.
- The PWR reactors R2, R3, R4 have thermal effects of 2,652 MW_t, 2,992 MW_t, 2,775 MW_t and electrical effects of 870 MW_e, 1,040 MW_e, 907 MW_e. The decommissioning study is based on data from sister PWR reactors in the USA.

Information for the inventory work has been based on documentation (assembly drawings, data sheets and FSAR) or derived from 3D models of the different units. In cases where there are weaknesses in the supporting material, assumptions and extrapolation have been applied. Engineering-related assessments have also been made where the available information is insufficient.

Input data for the number of packages from SNAB and Svafo comes from the waste suppliers' own forecasts. The forecasts are more simple than the decommissioning studies. Waste quantities assumed to be generated during the decommissioning of the different facilities at the plants, such as buildings, units for waste handling and the research reactor S-R2, are included in the forecasts.

A3.2 Assumptions

Decommissioning waste will be disposed of by the Swedish nuclear facilities at BKAB, FKA, OKG, RAB, Ågesta, Clink, SNAB and Svafo (including Ranstad). Possible decommissioning waste from the planned nuclear facility ESS is not included in the present report.

The decommissioning waste that is disposed of in SFR is low and intermediate-level short-lived waste. Long-lived near-core (< 1 m from the core) waste are assumed to be disposed of in SFL. In the present report, it is assumed that waste with high contents of C-14 (systems > 10^{10} Bq) is also disposed of in SFL, regardless of what is specified in the decommissioning studies. The BWR reactor pressure vessels will be intact when disposed of in SFR while the PWR reactor pressure vessels are planned to be disposed of in SFL due to high radioactivity content. Regarding the lower activity limit, all waste that cannot be released for unrestricted use (> 500 Bq/kg) is assumed to be deposited in SFR. No near-surface repositories are expected to be used for decommissioning waste. For BKAB, an amount of waste < 500 Bq/kg is also included, in accordance with the decommissioning study.

Low-level waste is deposited in BLA and intermediate-level waste is deposited in BMA. Certain systems will, according to the decommissioning studies, be decontaminated before decommissioning and the ion-exchange resins from the decontamination are assumed to be deposited in the silo. The BWR reactor pressure vessels are deposited in BRT.

Currently, there are no valid waste type descriptions for decommissioning waste, however, there are preliminary waste type descriptions. The code system available today for operational waste is also applied in this report to decommissioning waste, but in order to be able to distinguish between operational and decommissioning waste, the letter D has been added after each waste type with decommissioning waste. The first letter in the code denotes the plant from which the waste comes, i.e B, F, O and R for the nuclear power plants and C for Clink, S for SNAB, V for Svafo and Å for Ågesta.

As a principle, decommissioning waste for BLA is referred to by the number 12 and decommissioning waste for BMA by the number 23, based on operational waste types X.12 and X.23 (where X can be B, C, O, R or S) which is trash and scrap metal in containers and moulds for the aforementioned waste vaults. Scrap metal and secondary waste from decommissioning, i.e "trash and scrap metal", are consequently referred to by the code 12:D or 23:D. Packages with concrete, sand and contaminated soil from decommissioning will also be delivered to BLA and BMA. There are no such waste types for operational waste. For concrete waste, consequently, the letter C ("concrete") is added yielding the codes 12C:D or 23C:D. In the same way, the letter S denotes sand and the letter A denotes soil and asphalt. Furthermore, for the decommissioning waste, tetramould, corresponding to four moulds, has been added as a packaging type in BMA. In order to distinguish the waste packaged in tetramould from the waste packaged in ordinary mould, the tetramould is marked by 4K, e.g 4K23:D. The tetramould may be replaced by four ordinary moulds with only a marginal effect on transport and deposition. An exception from the numbers 12 and 23 in BMA is ash drums from SNAB which are marked S.25:D according to the operational waste type that SNAB consider to be equivalent to the ash drums in the support document for the forecast (S.25).

In addition to 12, 23 and 25, there are waste types C.16:D, F.18:D, O.16:D, R.02:D and X.BWR:D. X.BWR:D has been created for the reactor pressure vessels, and the other waste types are decontamination waste, with a denomination based on operational waste types containing ion exchange resins to the silo.

The waste types reported for the decommissioning waste are only notational for this report to facilitate reading.

A3.3 Calculation methods

The number of steel moulds and tetramoulds with scrap metal, concrete and sand is stated in the decommissioning studies for BKAB, FKA, OKG, RAB, Ågesta and Clink. These will be deposited in BMA. For BKAB, a number of BFA-packagings with internal parts are also specified. BFApackagings are not assumed to be used as packaging in SFR and for this reason this waste material, in the present report, is instead expected to be emplaced in steel moulds, and deposited in BMA. Furthermore, the number of ISO containers (20-foot half height) with scrap metal, concrete and sand is specified in the decommissioning studies. These will be deposited in BLA. In addition to this, a number of steel moulds with decontamination waste to be delivered to the silo are specified, and intact BWR reactor pressure vessels that will be deposited in BRT.

The disposal volume assumed to be generated from waste from other nuclear facilities, including reactor S-R2, is specified in the forecast data from SNAB and Svafo. SNAB have also specified which waste types, like those of operational waste (S.12, S.23 and S.25), the decommissioning waste are assumed to correspond to. For the sake of clarity concerning the waste types assumed for decommissioning waste from the nuclear power plants and Clink, the waste from the forecast data for SNAB and Svafo is also categorised in the categories scrap metal and concrete. For Svafo, the category asphalt is also added when Svafo reports asphalt, soil and gravel from decontamination around the facilities. The number of waste packages for different waste types is then calculated based on the disposal volume for each waste category and type of packaging, which are specified as concrete moulds, ISO containers (20-foot half height) and steel drums.

Secondary waste assumed to occur during decommissioning, such as e.g. clothing, cloths and tools, is presented in the decommissioning studies for BKAB, RAB and Ågesta. For other waste suppliers, volume estimates of secondary waste are missing. This is compensated for in this report by assuming additional packages in accordance with the volumes presented for RAB unit R1. In the decommissioning study for RAB, 25 ISO containers with secondary waste from reactor R1 are presented. This number is assumed to also apply to each reactor at FKA and OKG, as well as to Svafo whomanages the decommissioning of the research reactor S-R2. This volume of secondary waste corresponds to about 11% of the total volume of inventoried BMA and BLA waste from each facility. For Clink and SNAB, 2 and 14 ISO containers with secondary waste, respectively, have been added, corresponding to 11% of the total volume specified in the forecast data for those facilities.

A3.4 Results

The number of forecasted waste packages in the different waste vaults at the closure of SFR is presented in Table A3-1.

Waste content, conditioning material and packaging material may vary in the reported waste type. The differences in the different variants are presented in Appendix E. Containers may have different sizes why the disposal volume for them are given in the table, otherwise standard measures according to Appendix B apply.

Table A3-1. Number of waste packages with decommissioning waste expected to be deposited in SFR.

Waste vault	Waste type	Waste packaging	Number of decommissioning waste packages 2075-12-31
Silo	C.16:D	Steel mould	7
	F.18:D	Steel mould	21
	O.16:D	Steel mould	28
	R.02:D	Steel mould	42
BRT	B.BWR:D	Reactor pressure vessel	2
	F.BWR:D	Reactor pressure vessel	3
	O.BWR:D	Reactor pressure vessel	3
	R.BWR:D	Reactor pressure vessel	1
BMA	B.23:D	Steel mould	608
	C.4K23:D	Tetramould	3
	F.4K23:D	Tetramould	237
	F.4K23C:D	Tetramould	70
	O.4K23:D	Tetramould	198
	O.4K23C:D	Tetramould	82
	O.4K23S:D	Tetramould	15
	R.23:D	Steel mould	153
	R.4K23:D	Tetramould	314
	R.4K23C:D	Tetramould	149
	S.23:D	Concrete mould	164
	S.25:D	Steel drums	2,384
	Å.4K23:D	Tetramould	45
	Å.4K23C:D	Tetramould	5
BLA	B.12:D	Container 20 m^3	297
	B.12C:D	Container 20 m^3	389
	B.12S:D	Container 20 m^3	190
	C.12:D	Container 20 m^3	11
	C.12C:D	Container 20 m^3	7
	F.12:D	Container 20 m^3	529
	F.12C:D	Container 20 m^3	152
	F.12S:D	Container 20 m^3	53
	O.12:D	Container 20 m^3	457
	O.12C:D	Container 20 m^3	160
	O.12S:D	Container 20 m^3	37
	R.12:D	Container 20 m^3	389
	R.12C:D	Container 20 m^3	60
	R.12S:D	Container 20 m^3	32
	S.12:D	Container 20 m^3	63
	S.12C:D	Container 20 m^3	26
	V.12:D	Container 20 m^3	82
	V.12A:D	Container 20 m^3	200
	V.12C:D	Container 20 m^3	227
	Å.12:D	Container 20 m^3	10
	Å.12C:D	Container 20 m^3	15

A4 Total number of waste packages

Table A4-1 presents the total number of packages with operational and decommissioning waste, per waste type, expected to be deposited in SFR at the closure of the repository.

In the table, waste types of different variants or with different types of packaging, excluding miscellaneous waste types numbered by 99, have been combined. Waste content, conditioning material and packaging material may therefore vary within the reported waste type. The differences in the different variants are presented in Appendix E. Containers may have different sizes why the disposal volume for them are given in the table, otherwise standard measures apply according to Appendix B.

Table A4-1. Number of waste packages with operational and decommissioning waste expected to be deposited in SFR.

Waste vault	Waste type	Waste packaging	Number of operational and decommissioning waste packages 2075-12-31
Silo	B.04	Steel drums	768
	B.06	Steel drums	1,776
	C.02	Concrete mould	1,361
	C.16:D	Steel mould	7
	C.24	Concrete mould	350
	F.18	Steel mould	804
	F.18:D	Steel mould	21
	O.02/O.02:9	Concrete mould	1,944
	O.16:D	Steel mould	28
	O.24	Steel mould	204
	R.02/R.02:9	Concrete mould	371
	R.02:D	Steel mould	42
	R.16	Steel mould	2,839
	R.24	Steel mould	60
	S.04	Steel drums	452
	S.11	Steel mould	106
	S.24	Concrete or steel mould	826
BRT	B.BWR:D	Reactor pressure vessel	2
	F.BWR:D	Reactor pressure vessel	3
	O.BWR:D	Reactor pressure vessel	3
	R.BWR:D	Reactor pressure vessel	1
BMA	B.05/B.05:9	Steel drums	3,360
	B.05:2	Drum box	224
	B.23	Steel mould	33
	B.23:D	Steel mould	608
	C.01:9	Concrete mould	68
	C.23	Concrete mould	161
	C.4K23:D	Tetramould	3
	F.05:1/F.05:2	Steel drums	1,712
	F15	Steel mould	11
	F.17/F.17:1	Steel mould	1,382
	F.23	Concrete or steel mould	527
	F.4K23:D	Tetramould	237
	F.4K23C:D	Tetramould	70
	F.99:1	Steel mould	2
	O.01:9	Concrete mould	675
	O.23/O.23:9	Concrete mould	609
	O.4K23:D	Tetramould	198
	O.4K23C:D	Tetramould	82
	O.4K23S:D	Tetramould	15
	R.01/R.01:9	Concrete mould	1,689
	R.10	Concrete mould	121
	R.15	Steel mould	254
	R.23	Concrete or steel mould	606
	R.23:D	Steel mould	153
	R.4K23:D	Tetramould	314
	R.4K23C:D	Tetramould	149
	R.29	Concrete mould	380
	S.21	Steel drums	488
	S.23	Concrete mould	718
	S.23:D	Concrete mould	164
	S.25:D	Steel drums	2,384
	Å.4K23:D	Tetramould	45
	Å.4K23C:D	Tetramould	5

Table A4-1. Continued.

Waste vault	Waste type	Waste packaging	Number of operational and decommissioning waste packages 2075-12-31
BTF	B.07/B.07:9	Concrete tank	232
	F.99:2	Steel box	18
	O.01:9	Concrete mould	28
	O.07/O.07:9	Concrete tank	890
	O.99:1	Cortén box	40
	R.01/R.01:9	Concrete mould	91
	R.10	Concrete mould	4
	R.23	Concrete mould	21
	R.99:1	Reactor pressure vessel lid	1
	S.13	Steel drums	8,116
BLA	B.12/B.12:1	Container 20 m^3	193
	B.12	Container 40 m^3	61
	B.12:D	Container 20 m^3	297
	B.12C:D	Container 20 m^3	389
	B.12S:D	Container 20 m^3	190
	B.20	Container 20 m^3	12
	C.12:D	Container 20 m^3	11
	C.12C:D	Container 20 m^3	7
	F.12	Container 10 m^3	27
	F.12	Container 20 m^3	43
	F.12:D	Container 20 m^3	529
	F.12C:D	Container 20 m^3	152
	F.12S:D	Container 20 m^3	53
	F.20	Container 20 m^3	15
	O.12/O.12:1	Container 20 m^3	81
	O.12	Container 40 m^3	10
	O.12:D	Container 20 m^3	457
	O.12C:D	Container 20 m^3	160
	O.12S:D	Container 20 m^3	37
	O.99:3	Container 40 m^3	5
	R.12/R.12:1	Container 20 m^3	33
	R.12	Container 40 m^3	118
	R.12:D	Container 20 m^3	389
	R.12C:D	Container 20 m^3	60
	R.12S:D	Container 20 m^3	32
	S.12	Container 20 m^3	260
	S.12:D	Container 20 m^3	63
	S.12C:D	Container 20 m^3	26
	S.14	Container 20 m^3	87
	V.12:D	Container 20 m^3	82
	V.12A:D	Container 20 m^3	200
	V.12C:D	Container 20 m^3	227
	Å.12:D	Container 20 m^3	10
	Å.12C:D	Container 20 m^3	15

A5 Uncertainties

The future waste will include both operational waste and decommissioning waste from the nuclear power plants, Clink, SNAB, Svafo and Ågesta. In order to determine the disposal capacity of the extension of SFR, quantitative assessments need to be carried out for the amount of waste that will occur. In the following sections, any uncertainties and risks that might be of importance for the estimated amount of future waste will be discussed.

In Section A5.1, uncertainties that concern only operational waste are presented, and Section A5.2 presents uncertainties related to decommissioning waste. Sections A5.3–A5.5 treats uncertainties that are general for both operational and decommissioning waste.

A5.1 Operational waste

For operational waste, there is both existing and forecasted waste. The deposited waste in SFR is well documented in the database Triumf at SFR. For interim-stored waste there are both ready-made waste packages and untreated waste. There is some uncertainty for the waste that has not been ready-made waste packages yet, but the greatest uncertainty is, of course, in the yet unproduced waste.

The estimates of the number of packages serve as a basis for the calculations of volume, material and radioactivity content, see Appendix B, C and D, why those calculations will also be affected by the uncertainties mentioned in this appendix.

A5.1.1 Forecasted waste package production

The forecasted waste production for operational waste has been estimated based on the experience of the waste produced so far, with assumptions on the future operation of the nuclear facilities. Each waste supplier has estimated an annual production of waste packages per waste type, and the number of packages interim-stored at each plant. There has also been an opportunity to specify an uncertainty interval for the minimum and maximum number of forecasted packages per waste type. The result of this uncertainty interval is presented in Table A5-1, together with the number of forecasted packages given in Section A2.4, referred to as the "expected" number of waste packages.

The forecasted amounts of operational waste have been estimated based on the history of the operation of the facilities and experience of the handling and production of waste. The packing degree in the operational waste is well-known and is based on long experience within waste management. In the forecasts for future operational waste today's packing degrees have been assumed since several rounds of optimisations have been carried out.

It is possible that new waste types may be produced in the future. This uncertainty may affect a certain waste type but should not substantially affect the total amount of waste.

Deviations from the forecasted amounts of operational waste will of course occur, but the deviations within a facility and also between facilities should neutralise each other to some extent in the long term. The forecasted final year of waste production at each facility can be brought forward or postponed, implying that the amount of waste may deviate more or less from the forecasts.

At the closure of the nuclear reactors, waste from a few years of shutdown and service operation before decommissioning starts, will occur. No individual quantity estimates have been done for the service operation due to the fact that the time for this is relatively short compared to the total years of operation and only a small amount of new waste is produced when the reactor is shut down. The waste quantities from the service operation can be assumed to be included in the estimated uncertainty range of the plants, where e.g. revisions, planned as well as unplanned, have been considered. The waste from system decontamination that takes place after the shutdown is furthermore included in the decommissioning waste for each reactor, except for B1 and B2 where system decontaminations are already done and included in the operational waste.

So far, the deposited volume of waste for SFR has been less than previously forecasted. This is the result of new methods and requirements and less fuel damage with less uranium dissolution, which might be a continuing trend.

Table A5-1. Number of packages with operational waste that are planned to be deposited in SFR, including min/max interval.

Waste vault	Waste type	Waste packaging	Minimum number of waste packages 2075-12-31	The expected number of waste packages 2075-12-31	Maximum number of waste packages 2075-12-31
Silo	B.04	Steel drums	668	768	868
	B.06	Steel drums	1,776	1,776	1,776
	C.02	Cncrete mould	1,241	1,361	1,361
	C.24	Concrete mould	168	350	350
	F.18	Steel mould	444	804	804
	O.02/O.02:9	Concrete mould	1,874	1,944	2,624
	O.24	Steel mould	204	204	204
	R.02/R.02:9	Concrete mould	371	371	371
	R.16	Steel mould	2,834	2,839	2,894
	R.24	Steel mould	52	60	63
	S.04	Steel drums	447	452	457
	S.11	Steel mould	106	106	106
	S.24	Concrete or steel mould	786	826	897
BMA	B.05/B.05:9	Steel drums	3,360	3,360	3,360
	B.05:2	Drum box	224	224	224
	B.23	Steel mould	28	33	38
	C.01:9	Concrete mould	68	68	68
	C.23	Concrete mould	115	161	168
	F.05:1/F.05:2	Steel drums	1,712	1,712	1,712
	F15	Steel mould	11	11	11
	F.17/F.17:1	Steel mould	1,232	1,382	1,532
	F.23	Concrete or steel mould	377	527	527
	F.99:1	Steel mould	2	2	2
	O.01:9	Concrete mould	675	675	675
	O.23/O.23:9	Concrete mould	584	609	734
	R.01/R.01:9	Concrete mould	1,689	1,689	1,689
	R.10	Concrete mould	121	121	127
	R.15	Steel mould	246	254	262
	R.23	Concrete or steel mould	599	606	619
	R.29	Concrete mould	360	380	410
	S.21	Steel drums	488	488	488
	S.23	Concrete mould	668	718	768
BTF	B.07/B.07:9	Concrete tank	232	232	232
	F.99:2	Steel box	18	18	18
	O.01:9	Concrete mould	28	28	28
	O.07/O.07:9	Concrete tank	840	890	940
	O.99:1	Cortén box	40	40	40
	R.01/R.01:9	Concrete mould	91	91	91
	R.10	Concrete mould	4	4	4
	R.23	Concrete mould	21	21	21
	R.99:1	Reactor pressure vessel lid	1	1	1
	S.13	Steel drums	8,076	8,116	8,156
BLA	B.12/B.12:1	Container 20 m^3	193	193	193
	B.12	Container 40 m^3	53	61	69
	B.20	Container 20 m^3	12	12	12
	F.12	Container 10 m^3	21	27	81
	F.12	Container 20 m^3	34	43	88
	F.20	Container 20 m^3	15	15	15
	O.12/O.12:1	Container 20 m^3	56	81	156
	O.12	Container 40 m^3	10	10	10
	O.99:3	Container 40 m^3	5	5	5
	R.12/R.12:1	Container 20 m^3	33	33	33
	R.12	Container 40 m^3	118	118	135
	S.12	Container 20 m^3	190	260	330
	S.14	Container 20 m^3	87	87	87

A5.1.2 Clab canisters

Based on analyses and sampling, the fuel storage canisters used in Clab are judged to get released for unrestricted use. The volume of these is thus not included in the present report. However, some post-processing is required for clearance to be possible. Decontamination/washing takes place upon withdrawal from the storage basin in order to get rid of particular radioactivity that has gathered in the bottom of the storage canister. The additional quantity of ion exchange resins that the treatment might generate is not taken into account in the present inventory compilation. If needed, the material might be melted. The quantity of waste for SFR that might be added would only include the secondary waste that appears during melting, which has been estimated at 140 m^3 or about seven half-height containers. The secondary waste from any melting is, however, not included in the present report.

A5.1.3 S.13

Future deposition of drums of waste type S.13 in BTF is forecasted. With today's methods of treatment, these drums are stacked in compartments with concrete moulds as partition walls. As the forecast now indicates, however, no additional moulds will be deposited in BTF. The future solution may be to pack S.13 waste in moulds, instead of steel drums. This would imply a simpler handling in the repository and a reduction in disposal volume.

A5.1.4 S.14

Containers in 1BLA containing drums of waste type S.14 (a total of 75 containers) may be retrieved from the repository since their material and radionuclide content may be incorrectly estimated. This is described more fully in Section D5.8 in Appendix D.

A5.2 Decommissioning waste

Since, so far, there is limited experience of decommissioning of nuclear facilities in Sweden, great uncertainty remains in the estimated quantity of generated waste. For SNAB and Svafo there are currently no detailed decommissioning studies, so the estimated waste quantities are based on predictions from the waste suppliers.

A5.2.1 Inventory

The quantity of materials that will be generated during decommissioning are presented in decommissioning studies for the different nuclear power plants. The inventory is based on e.g. design data, and the waste quantities for SFR have been estimated by means of radioactivity calculations. The contamination of the material has been calculated based on measurements of the radioactivity on site. The further away from the reactor (core), the more difficult it is to assess the radioactivity content and thereby also whether the material can be radiologically cleared or if it needs to be disposed of in SFR.

According to the decommissioning studies for FKA and OKG, the uncertainty of the inventoried quantities is estimated to be about ± 5–20% (Anunti et al. 2013, Larsson et al. 2013). For Ågesta, an uncertainty of + 0–20% (Lindow 2012) is indicated, and for Clink an the uncertainty interval is ± 20% (Edelborg et al. 2014). SNAB estimates that their forecasted decommissioning quantities may vary by ± 35%. Svafo estimates the uncertainty to be at least + 50%. No uncertainty range is indicated in the decommissioning studies for BKAB and RAB.

As a prerequisite for the decommissioning study for BKAB, all potentially contaminated concrete and soil has been included (the consequences of spill, leakage and events during the operation period). In contrast to other decommissioning studies, which have a waste category with materials that is assumed to getcleared for free release, the data from BKAB includes waste with specific radioactivity <500 Bq/kg. That kind of low very level waste from BKAB is estimated to a total of about 240 packages, for BLA, which corresponds to a volume of nearly 5,000 m^3.

The quantity of inventoried decommissioning waste might be overestimated as it is based on the whole system. After some treatment of materials (such as segmentation and separation of non-contaminated parts from contaminated parts within the same system) the forecasted waste volumes might be reduced, which then implies a reduced number of waste packages.

A5.2.2 Packing degree

In the decommissioning studies, the process waste (mainly scrap metal) is assumed to be packed with a packing degree of about 1.1 tonnes/m^3 (Jönsson 2013, Hansson et al. 2013, Anunti et al. 2013, Larsson et al. 2013, Londow 2012). If the maximum weight capacity for the packaging were used, instead of the packing degree, the number of packages would be reduced significantly.

The maximum allowed waste weight for tetramoulds and ISO-containers is 18 tonnes and for steel moulds the maximum weight is assumed to be about 4 tonnes (depending on wall thickness). This means that at a maximum weight capacity, the packing degree would be 2.8 tonnes/m^3 for tetramoulds, 1.2 tonnes/m^3 for ISO containers and 2.4 tonnes/m^3 for steel moulds. If the maximum weight was used for all process waste, the number of tetramoulds would decrease by about 450, the number of steel moulds by about 340 and the number of ISO-containers by about 120, which corresponds to a reduction of volume by about 3,700 m^3 for BMA and 2,400 m^3 for BLA. In BMA, however, it is required that the waste be embedded in concrete, which has not been taken into account in these calculations.

The above outcome is judged unlikely, but nevertheless, it indicates the potential for reducing the number of waste packages through the development of waste packaging and waste management equipment. In contrast to the operational waste, the decommissioning waste management will be a major industrialised process where large quantities of waste are generated and must be disposed of in a short time. For the decommissioning waste, environmental and optimisation demands, and dose to personnel vs cost and time, will determine what packing degree is achieved. The packing degree 1.1 tonnes/m^3 in the decommissioning studies is chosen based on experience from other decommissioning projects and considered to represent the most likely outcome.

A5.2.3 Segmented BWR reactor pressure vessels

As an alternative to the disposal of entire BWR reactor pressure vessels, they could be segmented prior to final deposition in SFR. Support data on this are available for the reactors B1 and B2 in (1403739 ver 1.0). For F1–F3, O1–O3, and R1, data from A, B and C in Table A5-2 have been used as working material. Segmented reactor pressure vessels would be placed in tetramoulds (excluding B1 and B2) and deposited in BMA. The reactor pressure vessels from BKAB would instead be placed in moulds and ISO-containers, because other conditions prevailed when decommissioning studies for these reactors were conducted. If segmentation of reactor pressure vessels were considered, the number of tetramoulds and moulds to BMA would increase by about 650 (4,500 m^3) and 80 (140 m^3) and the number of ISO-containers to BLA would increase by about 50 (1,000 m^3). At the same time, BRT would be available for deposition of other waste.

A5.2.4 Melting

Certain low-level scrap metal might be reduced with regard to volume by melting. If melting takes place according to today's assumptions (requirements from the melting facility at SNAB), the surface dose rate of the material must not exceed 0.2 mSv/h and the specific activity must not exceed 5×10^5 Bq/kg.

In the decommissioning studies for BKAB and RAB, the number of packages for SFR is calculated based on a situation where melting will take place (Jönsson 2013, Hansson et al. 2013).

The decommissioning studies for FKA and OKG include a total of about 700 ISO containers which, according to the above criteria, might be candidates for melting (Anunti et al. 2013, Larsson et al. 2013). When melting is carried out, a reduction of volume by 75% is assumed, which would reduce the number of containers to 162, a decrease of more than 10,000 m^3 for BLA. For an optimal handling of the ingot, a new waste packaging must be developed so that volume content can be maximised in relation to the allowed weight.

A5.2.5 Biological shield

By disposing of the biological shields in the form of major blocks without packaging, the volume can be reduced. This is due to the fact that the void increases when the material is broken down in order to be accommodated in existing waste packaging. In the envisaged waste packaging, the concrete is assumed to be packed with a packing degree of 1.1–1.5 tonnes/m^3 (Jönsson 2013, Hansson et al. 2013, Anunti et al. 2013, Larsson et al. 2013, Londow 2012). In order to calculate the volume for the waste without packaging, a density of 2.46 tonnes/m^3 has been assumed for the concrete. The actual disposal volume in SFR for the major blocks has not been taken into account. An important prerequisite for this, however, is that transport and handling at SFR is possible.

If the deposition of major blocks is assumed for all nuclear power plants, the total volume would decrease by about 800 m^3 in BMA and 5,000 m^3 in BLA.

Table A5-2. Support data for the segmentation of reactor pressure vessels.

Serial number	Support data
A	Westinghouse, Decommissioning study of Forsmark 1-3, rev 0, 2011
B	Westinghouse, Decommissioning study of Oskarshamn 1-3, rev 0, 2011
C	TLG Services extension, Decommissioning Waste package Analysis for the Ringhals Unit 1, rev 2, 2012

A5.2.6 System decontamination

System decontamination of the primary systems will be performed to reduce the activity levels in the facility and thereby facilitate handling during dismantling and demolition. An effect of this is that the distribution of waste between the waste vaults in SFR may be affected. It will be possible to deposit the waste in the different waste vaults depending on how well the decontamination is carried out. The decontamination factor is set to 10, i.e. the radioactivity in the components will be reduced by 90% after system decontamination and the radioactivity is relocated to ion-exchange resins. Most often, however, the decontamination is more efficient than the assigned factor and an underestimated decontamination factor may result in certain forecasted waste to BMA potentially being deposited in BLA instead. The released radioactivity from the decontamination will, however, be deposited in the Silo irrespective of the decontamination factor achieved.

A5.2.7 Secondary waste

The decommissioning studies and forecast data mainly include inventoried material such as scrap metal and concrete. During demolition, however, secondary waste will also arise, such as clothing, cloths, and tools. The decommissioning studies for BKAB, RAB and Ågesta present quantities of secondary waste assumed to occur during decommissioning but for other facilities, estimates of quantities of secondary waste are missing. The quantities of secondary waste are, of course, subordinate to the inventoried waste, but are estimated to be about 5–20% of the total volume of scrap metal, concrete and sand waste in the decommissioning studies for BKAB, RAB and Ågesta. In order to avoid underestimating the total quantity of waste that may need to be disposed in SFR, the other waste suppliers are assumed to deliver secondary waste comparable to BKAB, RAB and Ågesta, see Section A3.3.

In the present report, a total of about 200 ISO containers with waste for BLA (corresponding to 4,000 m^3) have been added to compensate for any amounts of secondary waste from FKA, OKG, Clink, SNAB, and Svafo.

It should be considered that there are great uncertainties in the estimates of secondary waste which could imply either more or less waste than the estimated quantities. There is a possibility that much of the secondary waste has such a low activity content that it does not need to be disposed of in SFR. According to the decommissioning studies for RAB, about 35 wt% of the secondary waste is assumed to be “soft waste” and 65 wt% “hard waste”. A part of the soft waste may possibly be burnt and the hard waste melted for maximum volume reduction.

According to the experience of incineration of operational waste, about 4% of the original raw volume still remains and about 10% of the original weight after incineration. The resulting ashes are chemically stable and can be assumed to correspond to waste type S.13.

A5.3 Free release and recycling

The forecasts for how large quantities of operational and decommissioning waste will be deposited in SFR are based on previous limits for free release.

Since January 1st, 2012, new nuclide-specific limit values apply, with a general permissible limit of 100 Bq/kg per nuclide, compared to 500 Bq/kg previously. For individual nuclides, the allowed limit has, at the same time, been raised to 1,000 Bq/kg.

Specialists at the nuclear power plants have been asked whether the quantity of waste for SFR will increase due to the new regulations for free release. Both FKA and OKG consider that the reduction of the limit values will not affect, to any significant extent, their possibility for clearing. Economic and environmental requirements will rather control to what extent clearance is carried out. The clearance work will entail increased working efforts to separate contaminated material from non-contaminated material, decontaminate material and perform more measurements. For decommissioning waste, it is also extra important that the clearance work can be carried out in an effective manner, as the decommissioning of the nuclear power plants will generate large quantities of waste during a relatively short period of time. Currently, however, there is an on-going adaptation to the new regulations at the nuclear power plants, such as the acquisition of more advanced measuring equipment, training of personnel and setting up suitable premises.

A5.4 Near-surface repositories

Deposition of very low-level operational waste in near-surface repositories is assumed to be possible during the reactors' remaining operating times. Thus, the quantity of very low-level operational waste is not included in the waste inventory of the present report. The nuclear power plant waste plans, included as a support document for the different reactors' SAR, demonstrate how large a volume of very low-level operational waste is annually deposited in near-surface repositories. For FKA and OKG a normal average production is around 300 m^3, of which at least two thirds consists of compactable/combustible material. For RAB, the average annual production is between 260 and 450 m^3. Furthermore, RAB has about 3,000 m^3 of very low-level operational waste in interim-store.

The precondition in this report concerning very low-level decommissioning waste is that near-surface repositories are not available for final disposal. Instead, this waste is assumed to be deposited in BLA. The nuclear power plants, however, demand the possibility of disposal of very low-level decommissioning waste in near-surface repositories. The quantity of waste is estimated based on the nuclide inventory that is calculated in the decommissioning studies (Anunti et al. 2013, Larsson et al. 2013, Hansson et al. 2013). SSM's regulations and general advice concerning free release of materials, premises, buildings and soil around facilities with ionizing radiation, and the current licenses of the power plants regarding near-surface repositories, have served as a basis for the underlying limit values to estimate the amount of decommissioning waste for possible disposal in near-surface repositories. If this outcome were applied to the nuclear power plants that have existing near-surface repositories (FKA, OKG and RAB), the total quantity of decommissioning waste for BLA from these waste suppliers would decrease by about half.

A5.5 Other waste

The planned research facility ESS (European Spallation Source) in Lund will generate radioactive waste that could be a case for final disposal in SFR. This waste is not considered in the calculated inventory in the present report.

Appendix B

Deposition volumes of waste

B1 Background and purpose

The purpose of this appendix is to present a compilation of the disposal volumes that will be occupied by waste for SFR, and to describe the main uncertainties.

The reported volume is based on the external volume of the different packaging, alternatively a cuboid that includes the packaging, the so-called disposal volume. Table B1-1 provides information on disposal volumes of the different packaging for waste to the existing and extended SFR, and the volume of BWR reactor pressure vessels.

B2 Operational waste

B2.1 Available information sources

The disposal volume in SFR is calculated based on the number of deposited and forecasted waste packages, which are presented in Section A2 in Appendix A, and the disposal volumes given in Table B1-1.

B2.2 Assumptions

Table B2-1 presents the disposal capacity of existing waste vaults in SFR. Otherwise, the assumptions given in Appendix A apply.

B2.3 Calculation methods

The total disposal volume is calculated in the report and forecasting tool Triumf NG v1.0.1.3. The volume has been calculated by multiplying the number of waste packages presented in Appendix A with the corresponding disposal volume for the different waste packaging, see Table B1-1.

Table B1-1. Disposal volume for different types of waste packaging and BWR reactor pressure vessels.

Packaging	Disposal volume [m^3]
Tetramould	6.912
Concrete/steel mould	1.728
Concrete tank	10
BWR reactor pressure vessel B1	860
BWR reactor pressure vessel B2	860
BWR reactor pressure vessel O1	645
BWR reactor pressure vessel O2	790
BWR reactor pressure vessel O3	1,190
BWR reactor pressure vessel F1	1,190
BWR reactor pressure vessel F2	1,190
BWR reactor pressure vessel F3	1,190
BWR reactor pressure vessel R1	850
Container, 10-foot half height	10
Container, 10-foot whole height/20-foot half height	20
Container, 20-foot whole height	40
Cortén box	3.375
Steel drums (on drum tray)	0.324
Drum box	1.728
Steel box	10
Reactor pressure vessel lid	100

Table B2-1. Disposal capacity of the waste vaults in the existing SFR.

Waste vault	Disposal volume [m^3]
Silo	17,740
1BMA	13,090
1BTF	7,655
2BTF	7,655
1BLA	14,280
In total	**60,420**

B2.4 Results

Table B2-2 presents disposal volumes of operational waste per waste vault at the closure of SFR. The table presents the volumes of waste deposited in SFR today and the volumes of waste interim-stored at the waste suppliers or forecasted to arise.

Table B2-2. Disposal volume of operational waste.

Waste vault	Deposited waste [m^3] 2012-12-31	Interim-stored waste [m^3] 2012-12-31	Forecasted waste [m^3] 2013–2075	Total disposal volume of operational waste [m^3] 2075-12-31
Silo	5,621	2,744	7,924	16,289
BMA	9,343	832	4,517	14,692
BTF	9,614	839	4,060	14,513
BLA	10,200	3,990	8,320	22,510
In total	**34,779**	**8,406**	**24,821**	**68,005**

B3 Decommissioning waste

B3.1 Available information sources

The disposal volume in SFR is calculated based on the number of waste packages presented in Section A3 in Appendix A, and the disposal volumes specified for waste packaging and BWR reactor pressure vessels in Table B1-1.

B3.2 Assumptions

In addition to the packaging used today for deposition of operational waste, decommissioning waste is assumed to be packed in the new tetramould with a disposal volume corresponding to four moulds. The disposal volumes of BWR reactor pressure vessels have been estimated based on a cuboid that includes the reactor pressure vessel and any flanges or fittings that stick out from the reactor pressure vessel. Otherwise, the assumptions given in Appendix A apply.

B3.3 Calculation methods

Calculations have been done in Excel by multiplying the number of waste packages presented in Appendix A by the corresponding disposal volume for the waste packaging, see Table B1-1.

B3.4 Results

Table B3-1 presents the disposal volume of decommissioning waste per waste vault at the closure of SFR.

Table B3-1. Disposal volume of decommissioning waste.

Waste vault	Disposal volume of decommissioning waste [m^3] 2075-12-31
Silo	169
BRT	8,765
BMA	10,098
BLA	67,720
In total	**86,753**

B4 Total disposal volume

Table B4-1 presents the disposal volume of operational and decommissioning waste per waste vault at the closure of SFR.

Table B4-1. Total disposal volume of operational and decommissioning waste.

Waste vault	Total disposal volume of operational and decommissioning waste [m^3] 2075-12-31
Silo	16,459
BRT	8,765
BMA	24,791
BTF	14,513
BLA	90,230
In total	**154,758**

B5 Uncertainties

The volume calculations are based on the forecasted number of waste packages presented in Appendix A and the specified disposal volumes for each kind of packaging in Table B1-1. The uncertainties discussed in Appendix A thus concern the volumes. The uncertainty of the disposal volume for the packaging, is, on the other hand, limited.

B5.1 Operational waste

Section A5.1.1 in Appendix A presents an uncertainty interval in the forecasted number of packages for SFR, based on estimates made by each waste supplier. Based on the variations in the number of packages presented there, a range of minimum and maximum disposal volumes for SFR has been calculated. In addition, the expected volume given in Section B2.4, is also presented.

In addition to the range in Table B5-1, the operational waste is also affected by the uncertainties presented for Clab canisters, waste types S.13 and S.14 and near-surface repositories described in Appendix A.

Table B5-1. Disposal volume of operational waste according to the minimum/maximum range of forecasted number of packages.

Waste vault	Minimum volume [m^3] 2075-12-31	Expected volume [m^3] 2075-12-31	Largest volume [m^3] 2075-12-31
Silo	14,899	16,289	17,721
BMA	13,896	14,692	15,373
BTF	14,000	14,513	15,026
BLA	20,050	22,510	27,850
In total	**62,845**	**68,005**	**75,971**

B5.2 Decommissioning waste

Estimated uncertainty intervals for the number of packages given in decommissioning studies and support data are discussed in Section A5.2.1 in Appendix A. Based on the uncertainty ranges, a minimum and a maximum disposal volume for SFR have been calculated, see Table B5-2. The range is based on an uncertainty of ± 20% for waste from FKA, OKG, RAB and Clink and an uncertainty of ± 35% for waste from SNAB. For Svafo, an uncertainty allowance of + 20% and for Ågesta + 50% have been calculated. For BKAB, waste with an activity of < 500 Bq/kg has been excluded in the calculations of the minimum volume (see Appendix A).

In addition to the range in Table B5-2, the decommissioning waste is also affected by the uncertainties that are presented for system decontamination, clearance for free release, near-surface repositories, secondary waste, packing degrees and volume reducing methods in Appendix A.

Table B5-2. Disposal volume of decommissioning waste according to minimum/maximum range of the forecasted number of packages.

Waste vault	Minimum volume [m^3] 2075-12-31	Expected volume [m^3] 2075-12-31	Largest volume [m^3] 2075-12-31
Silo	135	169	203
BRT	8,765	8,765	8,765
BMA	8,200	10,098	12,066
BLA	54,629	67,720	81,081
In total	**71,729**	**86,753**	**102,116**

Appendix C

Materials in the waste

C1 Background and purpose

Waste packages for SFR include waste types with different materials from waste, conditioning material and packaging. The material may in the longer term affect the chemical and physical conditions prevailing in the repository. Changed conditions may result in degraded barriers, which may accelerate the transport of radionuclides to the surrounding environment. Material properties that are important to consider are those that affect complexing ability, gas formation and swelling, which includes materials such as organic substances, metals and bitumen-embedded waste.

The purpose of this appendix is to provide a forecast of the materials and their quantities to be deposited in SFR and to describe the main uncertainties in the forecast. The account of material quantities only includes materials for waste packages that are deposited and not materials in the engineered barriers or other structures in the repository.

C2 Operational waste

C2.1 Available information sources

The total amount of material quantities in SFR is calculated based on the number of deposited and forecasted packages that are presented in Section A2 in Appendix A, and material quantities, void and corrosion surfaces presented per waste type given in Appendix E.

C2.2 Premises

The material content per waste type given in Appendix E is an estimation of what an average package consists of and is based on operating experience and waste type descriptions. The material quantities include both waste materials, conditioning material and packaging material.

Among the waste materials are ion exchange resins, iron/steel, aluminium/zinc, ashes, concrete, evaporator concentrates, cellulose, sludge, filter aids, other organic materials and other inorganic materials. Other organic materials comprise for example air filters, oil and combustible or non-combustible trash. Other inorganic materials may be for example blasting sand, cables, brass, or water and oil filters.

For packaging and conditioning materials there is concrete, cement, iron/steel, bitumen, other organic and other inorganic materials. There is also estimates on empty spaces, so called void, which occur in the packages, in particular when hard material is packed.

For the materials aluminium/zinc and iron/steel, besides mass, the surface area that may be exposed to corrosion is also calculated. Surfaces for packaging are calculated from the dimensions. Surfaces for waste are calculated from an assumption that the material is a 5 mm thick plate. Metal surfaces in contact with bitumen are assumed not to corrode and are thus not included in the calculations of corrosion surfaces.

For the number of packages per waste type, the assumptions given in Appendix A apply.

C2.3 Calculation methods

A summary of material quantities of operational waste for the different waste vaults is made using the report and forecast tool Triumf NG v1.0.1.3 with the number of deposited and forecasted packages per waste type multiplied per waste type.

C2.4 Results

Table C2-1 shows the deposited material from operational waste, including conditioning material and packaging material, at the closure of SFR in 2075.

The material quantities are presented in the unit kg. The table also presents corrosion surface and void with the units m^2 and m^3.

Table C2-1. Quantity of material of operational waste, including conditioning material and packaging material.

Material	Deposited waste					Forecasted waste				Total
	Silo	1BMA	1BTF	2BTF	1BLA	Silo	BMA	BTF	BLA	
Aluminium/Zinc [kg]	0.00E+00	3.22E+03	3.12E+04	0.00E+00	4.68E+04	8.26E+03	8.54E+03	2.16E+04	5.23E+04	1.72E+05
Aluminium/Zinc [m^2]	0.00E+00	4.46E+02	4.62E+03	0.00E+00	6.88E+03	1.22E+03	1.26E+03	3.19E+03	7.82E+03	2.54E+04
Ash [kg]	0.00E+00	0.00E+00	3.05E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.11E+05	0.00E+00	5.15E+05
Concrete [kg]	3.11E+06	7.28E+06	1.84E+06	7.72E+06	2.42E+05	8.61E+06	4.20E+06	4.84E+06	3.99E+04	3.79E+07
Bitumen [kg]	5.20E+05	1.16E+06	0.00E+00	0.00E+00	1.18E+05	5.18E+05	7.67E+05	0.00E+00	0.00E+00	3.08E+06
Cellulose [kg]	2.88E+02	6.31E+04	8.04E+02	0.00E+00	2.11E+05	1.77E+04	8.26E+04	2.46E+02	2.87E+05	6.63E+05
Cement [kg]	4.71E+06	3.86E+06	1.75E+05	0.00E+00	0.00E+00	7.31E+06	9.81E+05	6.16E+04	7.50E+04	1.72E+07
Filter aids [kg]	2.84E+03	2.72E+04	7.04E+03	1.29E+05	0.00E+00	7.23E+03	5.66E+04	6.79E+04	0.00E+00	2.98E+05
Evaporator concentrate [kg]	0.00E+00	5.54E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.78E+05	0.00E+00	2.70E+02	4.34E+05
Ion exchange resins [kg]	1.37E+06	1.43E+06	8.10E+04	7.90E+05	8.80E+04	1.86E+06	7.00E+05	3.81E+05	9.43E+03	6.71E+06
Iron/steel [kg]	1.17E+06	1.78E+06	3.44E+05	1.75E+06	2.70E+06	3.72E+06	1.48E+06	1.01E+06	3.33E+06	1.73E+07
Iron/steel [m^2]	5.22E+04	8.56E+04	3.62E+04	3.86E+04	1.82E+05	1.67E+05	5.49E+04	4.20E+04	1.81E+05	8.39E+05
Sludge [kg]	3.20E+04	6.46E+04	4.10E+03	4.28E+04	0.00E+00	3.33E+03	3.85E+04	2.21E+04	7.25E+02	2.08E+05
Other inorganic [kg]	8.88E+02	1.15E+04	0.00E+00	0.00E+00	1.29E+05	1.07E+06	1.05E+05	0.00E+00	1.71E+05	1.49E+06
Other organic [kg]	1.37E+04	1.55E+05	6.40E+03	8.27E+04	9.89E+05	3.94E+04	1.88E+05	4.32E+04	1.50E+06	3.02E+06
Void [m^3]	6.70E+02	1.46E+03	1.27E+02	6.23E+02	3.21E+03	1.45E+03	6.57E+02	4.05E+02	3.92E+03	1.25E+04

C3 Decommissioning waste

C3.1 Available information sources

The amount of scrap metal, concrete, sand and decontamination waste assumed to occur during demolition are presented in the decommissioning studies for FKA, OKG, Ågesta and Clink (Anunti et al. 2013, Larsson et al. 2013, Lindow 2012, Edelborg et al. 2014). No further specification of material is given.

In the decommissioning studies for BKAB and RAB, the number of packages of scrap metal, concrete, sand and decontamination waste is given, as well as the packing degree of the packages (Jönsson 2013, Hansson et al. 2013). For BKAB, RAB and Ågesta, the number of packages with secondary waste is also presented. No further specification of the material is given.

The number of packages and disposal volume for the waste assumed to be generated by SNAB and Svafo is presented in forecast data from the waste suppliers. The information is given per building/ facility and for the reactor S-R2. SNAB indicates planned waste types and waste categories as for the operational waste but otherwise there is no detailed specification of the material content.

C3.2 Assumptions

The material specification in decommissioning studies and forecast data is less detailed than in the operational waste. In the present report, therefore, a number of assumptions for the calculation of material quantities for SFR are made.

Forecasted waste from SNAB and Svafo is categorised in scrap metal and concrete as in the decommissioning studies. The categorisation is based on assumptions from the specification presented in forecast data. In addition to these two categories, ashes and contaminated soil are reported and handled as additional material categories.

Scrap metal waste, both from decommissioning studies and forecast data, and the BWR reactor pressure vessels are assumed to consist entirely of iron/steel. For BKAB, RAB, SNAB and Svafo, the weight of the iron/steel waste for SFR is calculated based on the number of packages and the packing degree 1.1 tonnes/m^3 (Jönsson 2013, Hansson et al. 2013).

The mass of concrete and sand is calculated for BKAB and RAB based on the number of packages and the packing degree 1.1 tonnes/m^3 (Jönsson 2013, Hansson et al. 2013). For SNAB and Svafo, the mass of concrete is calculated from the packing degree 1.5 tonnes/m^3 based on information from the decommissioning studies for FKA and OKG (Anunti et al. 2013, Larsson et al. 2013). The volume of waste presented for Svafo as contaminated ground consisting of asphalt, gravel and soil is calculated with a packing degree of 1.5 tonnes/m^3.

Intermediate-level waste is packed in tetramoulds, steel moulds or in concrete moulds and is expected to be embedded in concrete. The amount of concrete for grouting is calculated from the inner volume of the packaging, assumed void, i.e. void in the package, and the density of the waste and conditioning material. The inner volume is assumed to be 6.5 m^3 for tetramould, 1.7 m^3 for steel mould and 0.92 m^3 for concrete mould (a mould with thick bottom and lid). The density is assumed to 2.4 tonnes/m^3 for concrete, 7.8 tonnes/m^3 for iron/steel and 1.54 tonnes/m^3 for sand. Void is assumed according to operating experience to be about 25% but also to be adapted in order for the package not to exceed the maximum weight capacities of 5 tonnes and 20 tonnes. Packaging material for the steel and concrete moulds is assumed to be the same as for the operational waste and material quantities for tetramoulds are based on a preliminary waste description.

Low-level waste is packed in ISO containers of iron/steel with the dimension of 20-foot half height. No grouting is done of the waste in BLA. Void is calculated based on the inner volume of the containers, 15 m^3, and the density of the waste materials, i.e. concrete, sand and iron/steel.

Decontamination waste consists of ion exchange resins. The weight of ion exchange resins is calculated based on the number of packages and an assumption that the waste is embedded by equal part of cement or bitumen. The density for ion exchange resins is assumed to be 1.05 tonnes/m^3, cement 2.4 tonnes/m^3 and bitumen 1.1 tonnes/m^3. The void is assumed to be 10% according to experience from the handling of operational waste.

The reactor pressure vessels are assumed to consist of iron/steel and the mass is indicated in the different decommissioning studies. Void is calculated based on the specified dimensions of the reactor pressure vessels.

The ashes which will be delivered by SNAB (S.25:D) are assumed to be similar to the waste type S.13 available for operational waste, see Appendix E.

Quantities of secondary waste are only given in the decommissioning studies for BKAB, RAB and Ågesta. For other facilities, the amounts of secondary waste have been estimated based on the decommissioning study for RAB, see Appendix A for an estimation of the number of waste packages. In the decommissioning study for RAB, it is stated that secondary waste consists of dry active waste such as paper, plastics and similar materials, decommissioning tools and similar non-permanent building tools and scaffolding. For the unit R1, 25 containers with secondary waste containing 143 tonnes of tools and scaffoldings and 78 tonnes of dry active waste (Hansson et al. 2013) are presented. The corresponding number of containers of the waste type R.12 with trash and scrap metal from operational waste contains 125 tonnes of scrap metal and other inorganic materials and 88 tonnes of cellulose and other material. When the amounts of "hard" and "soft" waste coincide relatively well in the decommissioning study compared to the operational waste, low-level secondary waste is assumed to be similar to the waste type, R.12, with regard to materials. The material composition for all low-level secondary waste, regardless of facility, is assumed to be similar to the waste type R.12. In the decommissioning study for BKAB, intermediate-level secondary waste is also presented. The intermediate-level secondary waste is assumed to be similar to the waste type B.23 of operational waste. B.23 is trash and scrap metal deposited in BMA.

For the materials aluminium/zinc and iron/steel, besides mass, the surface area that may be exposed to corrosion is also calculated. Surfaces for packaging materials are calculated from the dimensions of the packaging. Surfaces for wastes are calculated according to an assumption that the material is a 5mm thick plate. Metal surfaces in contact with bitumen are assumed not to corrode and thus are not included in the calculations of corrosion surfaces.

For the number of packages per waste type, the assumptions given in Appendix A apply.

C3.3 Calculation methods

Waste weights are calculated, based on decommissioning studies and forecast data, per package and then estimates of conditioning material and packaging material are made, see Section C3.2. In addition to the waste that is described in the background reports it is assumed that additional secondary waste will be generated that will be calculated based on the number of packages and assumptions on materials according to similar types of operational waste, see Section C3.2. The total quantity of materials is summarised for all waste types within each waste vault. The calculations have been done in Excel.

C3.4 Results

Table C3-1 shows the deposited material from decommissioning waste, including conditioning material and packaging material, at the closure of SFR in 2075.

The quantities of materials are presented in the unit kg. The table presents also corrosion surface and void with the units m^2 and m^3.

The calculated material content for the different waste types is presented in Appendix E.

Table C3-1. The quantity of materials of decommissioning waste, including conditioning material and packaging material.

Material	Silo	BRT	BMA	BLA	Total
Aluminium/Zinc [kg]	0.00E+00	0.00E+00	1.60E+04	3.36E+04	4.96E+04
Aluminium/Zinc [m^2]	0.00E+00	0.00E+00	2.37E+03	5.04E+03	7.41E+03
Asphalt, gravel, soil [kg]	0.00E+00	0.00E+00	0.00E+00	3.60E+06	3.60E+06
Ashes [kg]	0.00E+00	0.00E+00	1.51E+05	0.00E+00	1.51E+05
Concrete [kg]	0.00E+00	0.00E+00	1.43E+07	1.79E+07	3.22E+07
Bitumen [kg]	1.77E+04	0.00E+00	0.00E+00	0.00E+00	1.77E+04
Cellulose [kg]	0.00E+00	0.00E+00	5.37E+03	1.68E+05	1.73E+05
Cement [kg]	1.41E+05	0.00E+00	0.00E+00	0.00E+00	1.41E+05
Ion exchange resins [kg]	7.87E+04	0.00E+00	0.00E+00	0.00E+00	7.87E+04
Iron/steel [kg]	4.43E+04	5.55E+06	8.86E+06	3.29E+07	4.74E+07
Iron/steel [m^2]	1.76E+03	7.24E+03	4.10E+05	1.71E+06	2.13E+06
Sand [kg]	0.00E+00	0.00E+00	1.06E+05	5.26E+06	5.37E+06
Other inorganic [kg]	0.00E+00	0.00E+00	0.00E+00	1.34E+05	1.34E+05
Other organic [kg]	0.00E+00	0.00E+00	1.22E+04	1.01E+06	1.02E+06
Void [m^3]	1.67E+01	4.67E+03	2.22E+03	3.19E+04	3.88E+04

C4 Total quantity of materials

Table C4-1 shows the total quantity of material from operational and decommissioning waste, including conditioning material and packaging material, per waste vault at the closure of SFR in 2075.

The material quantities are presented in the unit kg. The table also presents corrosion surface and void with units m^2 and m^3.

Table C4-1. Quantity of materials of operational and decommissioning wastes, including conditioning material and packaging material.

Material	Silo	BRT	BMA	BTF	BLA	Total
Aluminium/Zinc [kg]	8.26E+03	0.00E+00	2.77E+04	5.28E+04	1.33E+05	2.21E+05
Aluminium/Zinc [m^2]	1.22E+03	0.00E+00	4.07E+03	7.81E+03	1.97E+04	3.28E+04
Asphalt, gravel, soil [kg]	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.60E+06	3.60E+06
Ashes [kg]	0.00E+00	0.00E+00	1.51E+05	5.15E+05	0.00E+00	6.67E+05
Concrete [kg]	1.17E+07	0.00E+00	2.58E+07	1.44E+07	1.81E+07	7.00E+07
Bitumen [kg]	1.06E+06	0.00E+00	1.93E+06	0.00E+00	1.18E+05	3.10E+06
Cellulose [kg]	1.80E+04	0.00E+00	1.51E+05	1.05E+03	6.66E+05	8.36E+05
Cement [kg]	1.22E+07	0.00E+00	4.84E+06	2.37E+05	7.50E+04	1.73E+07
Filter aids [kg]	1.01E+04	0.00E+00	8.37E+04	2.04E+05	0.00E+00	2.98E+05
Evaporator concentrate [kg]	0.00E+00	0.00E+00	4.34E+05	0.00E+00	2.70E+02	4.34E+05
Ion exchange resins [kg]	3.31E+06	0.00E+00	2.13E+06	1.25E+06	9.74E+04	6.78E+06
Iron/steel [kg]	4.93E+06	5.55E+06	1.21E+07	3.11E+06	3.89E+07	6.47E+07
Iron/steel [m^2]	2.21E+05	7.24E+03	5.51E+05	1.17E+05	2.08E+06	2.97E+06
Sand [kg]	0.00E+00	0.00E+00	1.06E+05	0.00E+00	5.26E+06	5.37E+06
Sludge [kg]	3.53E+04	0.00E+00	1.03E+05	6.90E+04	7.25E+02	2.08E+05
Other inorganic [kg]	1.07E+06	0.00E+00	1.16E+05	0.00E+00	4.35E+05	1.62E+06
Other organic [kg]	5.31E+04	0.00E+00	3.55E+05	1.32E+05	3.50E+06	4.04E+06
Void [m^3]	2.14E+03	4.67E+03	4.33E+03	1.15E+03	3.90E+04	5.13E+04

C5 Uncertainties

The estimates of materials in the waste are calculated based on the forecasted number of waste packages that are presented in Section A2 and A3 in Appendix A, and the estimated mean values for the different materials for each waste type, see Appendix E. The uncertainties that are discussed in Appendix A concern also the estimates of the materials. In addition, the uncertainties in the estimated content of the materials in each waste type are added.

C5.1 Operational waste

The material content in the different waste types for operational waste is based primarily on the values of the corresponding waste type description. The waste type descriptions state a mean value for all waste packages within the waste type. These values are set to the best of their ability by specialists at each facility with long experience in waste management. In spite of this, the material content must be considered a rough estimate and may vary widely between individual packages of the same waste type. One component often consists of a number of different materials, difficult to discern and to estimate with regard to weight. At the weighing in of different components and materials also the moisture content may have a large impact on the weight.

Material estimates for waste types containing ion exchange resins are based on a recipe for waste conditioning. The embedment is a monitored process that requires a given mass of materials for good results. This means that major variations from the estimated quantities in the recipes are not reasonable and that the uncertainties for these waste types are thereby fairly limited. In general, a void of 5–15% in these packages, depending on conditioning method and waste, is left. This, however, may be somewhat modified from package to package since the maximum allowed surface dose rate needs to be considered when a waste package is created.

Waste packages containing trash and scrap metal, in general, have a more roughly estimated composition of materials than waste types containing ion exchange resins, since there is often a wider variation in such packages. The waste package consists in any case largely of a waste packaging that is well-defined with regard to materials, that often forms the main part of the estimated amount of iron/steel for the entire package. The quantity of aluminium/zinc stated for the waste types may, however, be assumed to be overestimated. Zinc occurs mainly in the form of an electroplating layer on steel components but the weight given is normally based on the weight of the entire component. The amount of aluminium and zinc specified in the waste type descriptions do not represent a mean value for the different waste types, but rather are chosen to accommodate single variations. For Silo, there is also a benchmark for the amount of aluminium that must not be exceeded. The void in waste packages with trash and scrap metal is generally of at least 20% and may vary to a higher degree than for packages with ion exchange resins since scrap parts cannot be compacted in the same way. Grouting of trash and scrap metal in moulds is done to achieve stability during transport, but also to reduce the available void and thereby the flow paths through the waste, and to increase the sorption capacity.

Steel drums containing ashes have a high uncertainty in the void, since the ashes bind air during refill and then collapse to a smaller volume.

C5.1.1 S.14

The half-height containers in 1BLA containing drums of waste type S.14 (a total of 75 containers) may be retrieved from the repository since their material and radionuclide content data may be incorrectly estimated. This is described more fully in Section D5.8 in Appendix D.

C5.2 Decommissioning waste

The material estimates for the decommissioning waste are based on inventoried material quantities from decommissioning studies and forecast data but also include a number of assumptions according to Section C3.2. Since there is limited experience of how much waste and which type of waste are generated in conjunction with demolition, the material estimates for the decommissioning waste suffer from large uncertainties.

The total amounts of decommissioning waste are affected mainly by the uncertainties presented in Section A5 in Appendix A. The quantities of waste materials are affected by uncertainties in the inventory work, uncertainties in the planned system decontamination, and uncertainties concerning which waste it will be possible to deposit in near-surface repositories, or to get cleared for free release. Packing degrees or volume reducing methods affect conditioning material and packaging material. It should also be noted that the quantity of contaminated soil is currently only reported for Svafo, but, contaminated soil possibly needs to be taken into account for the decommissioning of other plants as well. That the handling of soil is not yet decided upon implies additional uncertainties.

For the category scrap metal, it is assumed that the entire material quantity consists of iron/steel. The scrap metal waste may include other materials. The same applies to the decommissioning material from SNAB and Svafo where waste from different buildings has been assumed to consist entirely of iron/steel or concrete in spite of the fact that additional materials occur. For example, there is aluminium from the reactor pressure vessel S-R2.

The material content in secondary waste from the decommissioning has been assumed to be similar to waste types with trash and scrap metal from the operational waste, which entails large uncertainties. In addition to the packages with secondary waste presented in the decommissioning studies for BKAB, RAB and Ågesta, in this report additional secondary waste has been added for FKA, OKG, Clink, SNAB, and Svafo, see Appendix A.

Appendix D

Radionuclide inventory

D1 Background and purpose

The purpose of this appendix is to present calculation methods and results for the prognosis of the radionuclide inventory in SFR at closure of the repository in 2075, and to describe the main uncertainties.

D2 Operational waste

D2.1 Available sources of information

Calculations are carried out with the report and forecasting tool Triumf NG. The data for Triumf NG is based on the following sources:

- Number of packages per waste type forecasted to be deposited in SFR at closure in 2075, see Section A2 in Appendix A.
- Data for deposited packages is annually migrated from the TRIUMF database at SFR to Triumf NG. In TRIUMF, measured activity data is stored directly from the waste suppliers' waste data files before the deposition.
- Non-package-specific data for determination of the content of transuranics, Sr-90, nickel, C-14, Cl-36, Mo-93, Tc-99, I-129 and Cs-135 in the waste is obtained annually from the waste suppliers and from Studsvik ALARA Engineering. Older data for transuranics and Sr-90 has been compiled by SKB. Data for C-14 has been calculated based on the reports "C-14 accumulated in ion exchange resins in Swedish nuclear power plants" (SKBdoc 1339709), "Measurement of ^{14}C in process water from CLAB and estimation of the accumulated amount in spent ion exchange resins" (SKBdoc 1393796) and "Uppskattning av aktivitet C-14 och Cl-36 i driftavfall från forskningsreaktorn R2 i Studsvik" (SKBdoc 1393446). Data for Cl-36 comes from the "Uppskattning av aktivitet C-14 och Cl-36 i driftavfall från forskningsreaktorn R2 i Studsvik" (SKBdoc 1393446) and "Klor-36 – Uppskattning av aktivitet i driftavfall från svenska LWR" (SKBdoc 1393449). Data for Mo-93, Tc-99, I-129 and Cs-135 stems from "Uppskattning av Mo-93, Tc-99, I-129 och Cs-135 i driftavfall – Uppdatering till och med 2011" (SKBdoc 1341356).
- General correlation factors used are taken from Lindgren et al. (2007), with the exception of the correlation factor for Tc-99, which is based on "Mo-93, Tc-99 och Cs-135: Uppskattning av aktivitet i driftavfall från svenska LWR, Clab och Studsvik" (SKBdoc 1393496) and the correlation factor for Pu-241, which is obtained from Thierfeldt and Deckert (1995).
- Estimated activity data, used for forecasted waste packages that have never previously been deposited, is obtained from waste type descriptions. Forecasted values for waste types from SNAB stem from measurements made on interim-stored waste at SNAB.
- Half-lives are obtained from (Firestone et al. 1998), with the exception of Se-79 (Jörg et al. 2010) and Ag-108m (Schrader 2004).

D2.2 Assumptions

The database and calculation tool Triumf NG v1.0.1.3 is used for activity calculations in SFR.

D2.3 Calculation methods

In Triumf NG there are calculation methods to handle different types of input data, since certain nuclides can be determined by direct measurements while most nuclides are determined by indirect measurements and calculation models.

Table D2-1 shows an outline of the methods used to determine the radioactivity of the already deposited operational waste, i.e. waste deposited up to 2012-12-31. Further explanation of this is given in Sections D2.3.1–D2.3.3. The radioactivity content in the forecasted waste packages, and the waste packages that are interim-stored on site, is calculated according to the methodology described in Section D2.3.4. Decay and chain decay are described in Section D2.3.5.

Table D2-1. Method for activity determination in deposited operational waste.

Nuclide	Origin of the waste	Waste class		
		Ion exchange resin	**Trash and scrap metal**	**Other**
Co-60, Cs-137	NPP, Clab, SNAB, Svafo	Gamma measurement of waste package	Gamma measurement of waste package	Gamma measurement of waste package
TRU, Sr-90	NPP, Clab	α/β-sampling in water systems at the waste supplier. Distribution nuclide Co-60	Specific correlation at waste supplier based on measurements in water systems. Distribution nuclide Co-60.	General correlation factors with Pu-239/Pu-240
	SNAB, Svafo	General correlation factors with Cs-137 or Pu-239/Pu-240	General correlation factors with Cs-137 or Pu-239/Pu-240	General correlation factors with Cs-137 or Pu-239/Pu-240
Ni-63	FKA, OKG, RAB	α/β-sampling in water systems at the waste supplier. Distribution nuclide Co-60	Specific correlation at waste supplier based on measurements in water systems. Distribution nuclide Co-60.	General correlation factors with Co-60
	BKAB, Clab, SNAB, Svafo	General correlation factors with Co-60	General correlation factors with Co-60	General correlation factors with Co-60
Ni-59	FKA, OKG, RAB	Specific correlation at waste supplier on the basis of measurements of Ni-63 in water systems. Distribution nuclide Co-60	Specific correlation at waste supplier on the basis of measurements of Ni-63 in water systems. Distribution nuclide Co-60	General correlation factors with Co-60
	BAKB, Clab, SNAB, Svafo	General correlation factors with Co-60	General correlation factors with Co-60	General correlation factors with Co-60
C-14	NPP, SNAB	Calculation based on reactor-specific energy production. Distribution kg ion exchange resins.	No contribution	No contribution
	Clab	Calculation based on annual uptake in ion exchange resins. Distribution kg ion exchange resins.	No contribution	No contribution
	Svafo	No contribution	No contribution	No contribution
Cl-36	NPP, SNAB	Calculation based on reactor-specific energy production. Distribution kg ion exchange resins.	General correlation factors with Co-60	General correlation factors with Co-60
	Clab, Svafo	General correlation factors with Co-60	General correlation factors with Co-60	General correlation factors with Co-60
Mo-93, Tc-99	NPP, Clab, SNAB	Calculation by Studsvik ALARA Engineering. Distribution nuclide Co-60	Calculation by Studsvik ALARA Engineering. Distribution nuclide Co-60	Calculation by Studsvik ALARA Engineering. Distribution nuclide Co-60
	Svafo	General correlation factors with Co-60 and Cs-137	General correlation factors with Co-60 and Cs-137	General correlation factors with Co-60 and Cs-137
I-129, Cs-135	NPP, Clab, SNAB	Calculation by Studsvik ALARA Engineering. Distribution nuclide Cs-137	Calculation by Studsvik ALARA Engineering. Distribution nuclide Cs-137	Calculation by Studsvik ALARA Engineering. Distribution nuclide Cs-137
	Svafo	General correlation factors with Cs-137	General correlation factors with Cs-137	General correlation factors with Cs-137
Other nuclides	NPP, Clab, SNAB, Svafo	General correlation factors with Co-60, Cs-137 and Pu-239/Pu-240	General correlation factors with Co-60, Cs-137 and Pu-239/Pu-240	General correlation factors with Co-60, Cs-137 and Pu-239/Pu-240

D2.3.1 Nuclide specific measurement of waste package

SKB requires each waste package deposited in SFR to be measured by the waste supplier with gamma spectrometry. The activity for a number of gamma-emitting nuclides, for example Co-60 and

Cs-137, can be detected by this measurement. These values are registered in Triumf NG for each package, including information about manufacturing date and measurement date.

The activation product Co-60 and the fission product Cs-137 are relatively simple to measure and are used in most of the calculation methods to determine the activity or distribute other difficult-to-measure nuclides in the different waste vaults in SFR. Due to the important role that Co-60 and Cs-137 play, they are named as "key nuclides".

The activity of interest when using the key nuclides is the original activity at the waste origin. The date of waste origin can, however, be difficult to clearly determine for a package, which is why the activity at the package's manufacturing date is set as the original activity in Triumf NG. The measured activity is re-calculated in Triumf NG, with the aid of half-lives, to the activity the package had at the manufacturing date, see equation D2-1 in Section D2.3.5. The activity of the key nuclides at the manufacturing date is then used in most of the calculation methods in Triumf NG.

D2.3.2 Non-package-specific data

It is not possible to measure all nuclides in each waste package. Nuclides that cannot be detected by gamma spectrometry are generally determined for the package by correlation, see Section D2.3.3. For activity determination of most transuranics, Sr-90, Ni-59, Ni-63, C-14, Cl-36, Mo-93, Tc-99, I-129 and Cs-135, however, specific measurement and calculation methods have been designed. Common to these methods is that measurements and calculations of the amount of radioactivity are made on active systems in the different facilities instead of on individual packages. These data are therefore called "non-package-specific data".

The measured or calculated non-package-specific activity is registered per waste supplier in Triumf NG. The activity is thereafter distributed in the deposited packages according to a distribution methodology based on the origin of the waste, its nature, and each package's content of ion exchange resins or the key nuclides Co-60 or Cs-137 at the manufacturing date.

In order to follow how the activity of non-package-specific nuclides is distributed, it is important to understand how the different waste types are divided into waste classes and linked to so-called waste streams, see Table D2-2. The waste is divided into the waste classes "Ion exchange resins", "Trash and scrap metal" and "Other waste", based on the nature of the waste. The waste class "Ion exchange resins" comprises the waste streams "CCU" (waste from condensate clean-up), "Rest" (waste from other clean-up systems), "Total Silo" (waste for Silo) and "Total Other" (other waste). These waste streams are adapted for distributing the activity of C-14 and Cl-36. A more detailed description of how the distribution is made is available in the following sections.

Transuranics, strontium and nickel

Every year, SKB receives information from the nuclear power plants and Clab on the amount of radioactivity generated during the previous year regarding the alpha-emitting transuranics Pu-238, Pu-239, Pu-240, Am-241, Am-243, Cm-242, Cm-243, Cm-244 and the beta-emitting nuclides Sr-90, Ni-59 and Ni-63 in waste from the waste classes "Ion exchange resins" and "Trash and scrap metal".

In the different clean-up systems, water samples are taken to quantify the absorbed amount of activity in ion exchange resins in the current year. For nickel nuclides, measurements are made on Ni-63, while the difficult-to-measure Ni-59 is calculated, by the waste supplier, with specific correlation factors based on the results of measurements on Ni-63. To determine the activity of "Trash and scrap metal", the waste suppliers use specific correlation factors with Co-60. These correlation factors are developed based on the nuclide ratio in the water samples, i.e. the same activity ratio is assumed to be valid for "Trash and scrap metal" as for ion-exchange resins.

Pu-239 and Pu-240, as well as Pu-238 and Am-241, have congruent alpha energies, which render them difficult to separate from each other. The acitivity amount of the nuclides is therefore usually reported as one. In Triumf NG, Pu-239 and Pu-240 are separated with the ratio 1:1.4, according to the relationship between these nuclides in a calculated equilibrium core. In the cases where Pu-238 and Am-241 are reported as one, it is conservatively assumed that all activity is Pu-238 and Am-241 is calculated instead by general correlation factors, see Section D2.3.3.

Table D2-2. Waste type links to waste classes and waste streams.

Waste class	Waste stream	Silo	BMA	BTF	BLA
Ion exchange resins	CCU		F.17	B.07	
				O.07	
	Rest	B.04	B.05	O.01	B.20
		B.06	F.05	O.99:1	F.20
		F.18	F.15		O99:3
		O.02	O.01		
	Total Silo	C.02	C.01		
		R.02			
		R.16			
		S.04			
		S.11			
	Total Other		R.01	R.01	
			R.15		
Trash and scrap metal	–	B.24	B.23	R.23	B.12
		C.24	C.23		F.12
		F.24	F.23		O.12
		O.24	O.23		R.12
		R.24	R.23		S.12
		S.24	S.23		S.14
Other	–		F.99:1	F.99:2	
			R.10	R.99:1	
			R.29	R.10	
				S.13	
			S.21		

In Triumf NG, the annually reported amounts of activity are registered per nuclide, system, waste class and waste supplier. Samples have been registered for the transuranics and Sr-90 since 1988. The amount that can be found in operational waste up to 1987 has been compiled by facility, based on knowledge and estimates by the waste suppliers from the start of operation. From FKA and RAB, values for nickel are reported since 2010 and from OKG there are values since 2007. For other waste suppliers, and for FKA and RAB prior to 2010 and OKG prior to 2007, the amount of nickel is determined by correlation with the key nuclide Co-60, see Section D2.3.3.

The distribution of the reported yearly productions in the deposited packages in SFR is carried out in the following way. All reported annual productions are summarised per nuclide and per waste supplier and waste class, i.e. "Ion exchange resins" or "Trash and scrap metal". Thereafter, the summed annual productions are distributed amongst the waste packages that match the right waste supplier and waste class. The distribution is proportional to the measured activity of Co-60, at the manufacturing date, in each individual package.

Activity determination of transuranics, Sr-90 and nickel in waste from the waste class "Other" and in waste from SNAB and Svafo is done by correlation factors according to Section D2.3.3. This is because it is miscellaneous waste, to which the above-mentioned methodology is not applied. In SNAB and Svafo, a project is under way to develop specific nuclide vectors to correlate transuranics in the packages that are allocated for deposition in SFR.

C-14 and Cl-36

The nuclides C-14 and Cl-36 are mainly found in clean-up resins and are therefore determined particularly for waste in the waste class "Ion exchange resins". C-14 is formed during neutron irradiation of carbon, nitrogen and oxygen that are present in control rods, reactor water and other internal parts. Cl-36 occur by activation of chloride ions in the reactor water. The calculation methods for activity determination of these nuclides assume a production of C-14 and Cl-36 proportional to the thermal energy production in a reactor. Additional variable parameters for Cl-36 are chloride concentration in the reactor water and moisture content in the formed vapour.

The two calculation models for C-14 and Cl-36 were first introduced in 2008 and are described in "C-14 accumulated in ion exchange resins in Swedish nuclear power plants" (SKBdoc 1339709) and "Klor-36 – Uppskattning av aktivitet i driftavfall från svenska LWR" (SKBdoc 1393449). Today FKA, OKG and RAB report the variable parameters annually. Previous activity amounts are based on historic data from "C-14 accumulated in ion exchange resins in Swedish nuclear power plants" (SKBdoc 1339709) and "Klor-36 – Uppskattning av aktivitet i driftavfall från svenska LWR " (SKBdoc 1393449). For the closed reactors at SNAB and BKAB, the calculated sum of the total produced activity of C-14 and Cl-36 during each reactor's operating time is used. For C-14, data is taken from "Uppskattning av aktivitet C-14 och Cl-36 i driftavfall från forsmkningsreaktorn R2 i Studsvik" (SKBdoc 1393446) and "C-14 accumulated in ion exchange resins in Swedish Nuclear Power plants" (SKBdoc 1339709) and for Cl-36 data is taken from "Uppskattning av aktivitet C-14 och Cl-36 i driftavfall från forskningsreaktorn R2 i Studsvik" (SKBdoc 1393446) and "Klor-36 – Uppskattning av aktivitet i driftavfall från svenska LWR" (SKBdoc 1393449). C-14 is also assumed to be present in the fuel storage pools in Clab. Since no energy is produced in Clab, an estimate of the annual uptake in the ion exchange resins has been made, based on measurements according to "Measurement of ^{14}C in process power from CLAB and estimation of the accumulated amount in spent ion exchange resins" (SKBdoc 1393796).

C-14 can form organic or inorganic compounds. Organic and inorganic C-14 have different properties, which need to be taken into account in the assessment of the long-term safety. For this reason, C-14 is divided into organic and inorganic activity. The proportion of organic activity for the Swedish nuclear power reactors has been taken from "C-14 accumulated in ion exchange resins in Swedish nuclear power plants" (SKBdoc 1339709) and varies between 1–28%, see Table D2-3. For waste from Clab and SNAB, the proportion of organic activity is conservatively assumed to be 30%.

The distribution of the calculated annual productions for the deposited packages in SFR is carried out in the following way. All annual productions are summarised per nuclide and per waste supplier. Thereafter the summed annual productions are distributed amongst the waste packages from the corresponding waste supplier, belonging to the waste class "Ion exchange resins". Furthermore, the activity is directed to specific waste types, according to the so-called waste streams, and within each waste stream the distribution of activity is proportional to the amount of ion exchange resins in each package.

For waste that only stems from BWR reactors, i.e. from BKAB, FKA and OKG, activity is registered on the waste streams "CCU" and "Rest", where "CCU" is activity absorbed in ion exchange resins from condensate clean-up, and "Rest" is activity absorbed in ion exchange resins from other clean-up systems, e.g. reactor water clean-up and system drainage. Activity from RAB, with mixed ion exchange resins from BWRs and PWRs, and activity in waste from Clab and SNAB, will be divided into waste streams named after the determined final waste vault, "Total Silo" or "Total Other". The properties waste class and waste stream are specified for each waste type and serve as a way to link the non-package-specific activity to the right waste types and the right final waste vault.

Cl-36 can also be found in waste belonging to the waste classes "Trash and scrap metal" and "Other", and in waste from Clab and Svafo. The amount of Cl-36 in such waste is determined by correlation factors; see Table D2-4 in Section D2.3.3. C-14 is assumed to be negligible in other waste than ion exchange resins and is determined with the above-mentioned methodology only.

Mo-93, Tc-99, I-129 and Cs-135

Mo-93 is an activation product that mainly occurs during neutron irradiation of fuel crud. Tc-99 is both an activation product from fuel crud and a fission product. I-129 and Cs-135 are both fission products. The activity from Mo-93, Tc-99, I-129 and Cs-135 disposed of in SFR is determined on the basis of calculations performed by Studsvik ALARA Engineering for waste from the nuclear power plants, Clab and SNAB. Since 2008, the annual production of these nuclides is reported. For previously generated activity, a compilation has been made up to 2007 in "Uppskattning av Mo-93, Tc-99, I-129 och Cs-135 i driftavfall – Uppdatering till och med 2011" (SKBdoc 1341356).

Table D2-3. Fraction of organic activity for C-14 per reactor.

Reactor	Fraction of organic C-14 [%]
B1, B2	3.7
C*	30
F1, F2	1.1
F3	2.6
O1, O2	3.7
O3	1.2
R1	1.5
R2, R3, R4	28
Studsvik R2	30

* The letter C is assumed for waste from Clab, no reactor.

Table D2-4. Nuclides correlated with Co-60.

Nuclide	General correlation factors for waste from BWRs, Clab, SNAB and Svafo	General correlation factors for waste from PWRs
H-3	1×10^{-4}	
Be-10	6×10^{-10}	
Cl-36	6×10^{-7}	
Fe-55	1	
Ni-59	1×10^{-3}	3×10^{-2}
Ni-63	8×10^{-2}	4
Mo-93	1×10^{-6}	
Zr-93	1×10^{-6}	
Nb-93m	1×10^{-3}	
Nb-94	1×10^{-5}	
Tc-99	3×10^{-6}	
Ag-108m	6×10^{-5}	
Sb-125	1×10^{-1}	
Ba-133	1×10^{-5}	
Ho-166m	4×10^{-6}	

The distribution of the reported annual productions amongst the deposited packages in SFR, is calculated in the following way. All annual productions are summarised per nuclide and per waste supplier. Thereafter the summed annual productions are distributed amongst the waste packages that match the right waste supplier. The distribution is proportional to the measured activity of Co-60 or Cs-137, at the manufacturing date, in each package. Mo-93 and Tc-99 follows the content of Co-60 while I-129 and Cs-135 follows the content of Cs-137.

The activity of Mo-93, Tc-99, I-129 and Cs-135 in waste from Svafo is correlated according to Section D2.3.3.

D2.3.3 Correlation factors

The nuclides that cannot be measured or calculated with the aforementioned methods are determined by correlation factors where nuclides are correlated based on the activity of the key nuclides Co-60, Cs-137 or the sum of Pu-239 and Pu-240. The current correlation factors used in the preparation of the nuclide inventory are presented in Table D2-4 to Table D2-6. The general correlation factors used are taken from Lindgren et al. (2007), with the exception of Tc-99 which is based on "Mo-93, Tc-99 och Cs-135: Uppskattning av aktivitet i driftavfall från svenska LWR, Clab och Studsvik" (SKBdoc 1393496) and Pu-241, which stems from Thierfeldt and Deckert (1995).

Table D2-5. Nuclides correlated with Cs-137.

Nuclide	General correlation factors	Specific correlation factors for certain waste from SNAB and Svafo
Se-79	4×10^{-6}	
Sr-90	1×10^{-1}	
Tc-99	1×10^{-4}	
Pd-107	1×10^{-6}	
Cd-113m	6×10^{-4}	
Sn-126	5×10^{-7}	
I-129	3×10^{-6}	
Cs-134	1	
Cs-135	1×10^{-5}	
Pm-147	9×10^{-1}	
Sm-151	3×10^{-3}	
Eu-152	7×10^{-5}	
Eu-154	1×10^{-1}	
Eu-155	7×10^{-2}	
Pu-238	0	1×10^{-2}
Pu-239	0	5×10^{-3}
Pu-240	0	5×10^{-3}
Am-241	0	4×10^{-2}
Cm-242[1)]	0	1×10^{-2}
Cm-244	0	1×10^{-2}

[1)] The nuclide is of importance for decay chain calculations but is not accounted for in the results due to its short half-life.

Table D2-6. Nuclides correlated with Pu-239 and Pu-240.

Nuclide	General correlation factors
U-232	3×10^{-5}
U-234	1×10^{-3}
U-235	2×10^{-5}
U-236	3×10^{-4}
U-238	4×10^{-4}
Np-237	4×10^{-4}
Pu-238	4
Pu-241	1.2×10^{2}
Pu-242	3×10^{-3}
Am-241	1
Am-242m	1×10^{-2}
Am-243	3×10^{-2}
Cm-243	2×10^{-2}
Cm-244	3
Cm-245	3×10^{-4}
Cm-246	8×10^{-5}

Generally, activation products are correlated with Co-60, fission products with Cs-137 and transuranics with the sum of Pu-239 and Pu-240. Correlation is calculated with activity data for the key nuclides at the package's manufacturing date.

Among the nuclides with correlation factors are also nuclides that can be determined with specific measurement or calculation methods, see Sections D2.3.1–D2.3.2. The correlation factors for these nuclides are used only in cases where the waste packages lack measured or otherwise calculated values.

For the nuclides Ni-59 and Ni-63, different correlation factors have been developed for waste that stems from BWR and PWR, as the nickel production is considerably higher in PWR reactors than BWR reactors. Waste from RAB, which can be produced by both types of reactors, is therefore correlated with both correlation factors. According to measurements performed on ion exchange resins from RAB between the years 1988–2005, on average 65% of the activity of Co-60 in the waste stems from the BWR reactor and 35% from PWR reactors. 65% of the waste from RAB is therefore assigned to the BWR factor and 35% to the PWR factor. Nickel that is correlated with waste from other waste suppliers, including Clab and Svafo, is assigned 100% correlation with the BWR factor.

Tc-99 is correlated with both Co-60 and Cs-137, since the nuclide can arise in conjunction with activation of materials as well as during fission, and the contribution from the two correlations are added up, in accordance with "Mo-93, Tc-99 och Cs-135: Uppskattning av aktivitet i driftavfall från svenska LWR, Clab och Studsvik" (SKBdoc 1393496).

A number of transuranics for waste from SNAB and Svafo are also correlated with Cs-137.

D2.3.4 Forecasts

Estimates of expected deposition of packages from the waste suppliers according to Appendix A are used for the forecasts. When the activity is calculated, all waste packages that are interim-stored in plants are assumed to be deposited during the first forecasted year, i.e. 2013. Deposition of these waste packages and future waste packages is assumed to take place on the 1st of July every year. All future waste packages are also given the same manufacturing date as deposition date.

For each waste type that is deposited, package-specific activity has been measured by gamma spectrometry, see Section D2.3.1. From these data nuclide-specific mean values are calculated, at each package's manufacturing date, per waste type. Forecasted waste packages with the same waste type are assigned these mean values. This includes mean values for the key nuclides Co-60 and Cs-137. These mean values constitute the basis for the forecast of all other nuclides besides C-14.

There are a number of waste types in the forecasts that have never previously been deposited. The activity for these waste packages is determined on the basis of assumed mean values for Co-60 and Cs-137, manually registered per waste type. Manually registered mean values have also been used for forecasted waste delivered by SNAB, although deposited packages exist. This is due to the fact that SNAB has measured higher activity levels on waste packages that are interim-stored at present than the average activity on already deposited packages. SNAB believes that the measured activity in the interim-stored waste packages better represents future waste, which is why these values are used in the calculations. SNAB has mentioned nuclides other than Co-60 and Cs-137 and all nuclides mentioned by SNAB are used in the calculations.

For waste in the waste classes "Ion exchange resins" and "Trash and scrap metal" from the nuclear power plants and Clab/Clink, the forecasted activity of transuranics, Sr-90 and nickel is calculated based on reported annual productions of these nuclides, in accordance with the methodology described for the deposited waste in Section D2.3.2. The ratio prevailing between the aforementioned nuclides and Co-60 in disposed of waste is assumed to apply for forecasted waste as well. The ratio is calculated before decay, i.e. with activity data at the manufacturing date. The forecasted waste is assigned transuranics, Sr-90 and nickel according to the same ratio, with knowledge of the forecasted amount of Co-60 in future waste. The ratio is calculated separately within each waste class and for each waste supplier. For waste in the waste class "Other" or waste from SNAB and Svafo, values for transuranics are assigned by manually registered values or correlation, see Section D2.3.3

Forecasted activity of C-14 is calculated in the same way as for deposited packages, see Section D2.2.2, with forecasted values for annual energy production. The forecasted energy production is conservatively based on maximum availability for each reactor. The forecasted uptake of C-14 in ion exchange resins from Clab/Clink is assumed to be the same as today's annual productions. The distribution of activity on forecasted waste packages is done in the same way as for deposited packages, with an estimate of the amount of ion exchange resins in each waste type. Waste types that have never previously been deposited are, however, not assigned any activity of C-14.

The remaining nuclides are determined by correlation in the same manner as described in Section D2.3.3, with calculated values for the key nuclides according to the above.

D2.3.5 Radioactive decay

To calculate the radionuclide content in SFR at the time of closure of the repository, the decay of nuclides must be taken into account. Activity at a given point in time, t, is calculated according to equation D2-1, where t_0 represents the package's manufacturing date and $T_{½}$ the nuclide's half-life (Firestone et al. 1998, Jörg et al. 2010, Schrader 2004), see Table D2-7.

a) $A(t) = A(t_0)e^{-\lambda\tau}$ D2-1

b) $\lambda = \frac{ln2}{T_{½}}$

c) $\tau = t - t_0$

$A = activity$ [Bq]

$t = time$ [years]

$T_{½} = half\text{-}life$ [years]

Table D2-7. Half-lives for calculated radionuclides.

Nuclide	Half-life [years]	Nuclide	Half-life [years]	Nuclide	Half-life [years]
H-3	1.23×10^{1}	Sb-125	2.76×10^{0}	Pu-241	1.44×10^{1}
Be-10	1.51×10^{6}	I-129	1.57×10^{7}	Pu-242	3.73×10^{5}
C-14	5.73×10^{3}	Cs-134	2.06×10^{0}	Am-241	4.32×10^{2}
Cl-36	3.01×10^{5}	Cs-135	2.30×10^{6}	Am-242m	1.41×10^{2}
Ca-41[1)]	1.03×10^{5}	Cs-137	3.01×10^{1}	Am-243	7.37×10^{3}
Fe-55	2.73×10^{0}	Ba-133	1.05×10^{1}	Cm-242[2)]	0.45×10^{0}
Co-60	5.27×10^{0}	Pm-147	2.62×10^{0}	Cm-243	2.91×10^{1}
Ni-59	7.60×10^{4}	Sm-151	9.00×10^{1}	Cm-244	1.81×10^{1}
Ni-63	1.00×10^{2}	Eu-152	1.35×10^{1}	Cm-245	8.50×10^{3}
Se-79	3.27×10^{5}	Eu-154	8.59×10^{0}	Cm-246	4.73×10^{3}
Sr-90	2.88×10^{1}	Eu-155	4.76×10^{0}		
Zr-93	1.53×10^{6}	Ho-166m	1.20×10^{3}		
Nb-93m	1.61×10^{1}	U-232	6.89×10^{1}		
Nb-94	2.03×10^{4}	U-234	2.46×10^{5}		
Mo-93	4.00×10^{3}	U-235	7.04×10^{8}		
Tc-99	2.11×10^{5}	U-236	2.34×10^{7}		
Pd-107	6.50×10^{6}	U-238	4.47×10^{9}		
Ag-108m	4.38×10^{2}	Np-237	2.14×10^{6}		
Cd-113m	1.41×10^{1}	Pu-238	8.77×10^{1}		
In-115[1)]	4.41×10^{14}	Pu-239	2.41×10^{4}		
Sn-126	1.00×10^{5}	Pu-240	6.56×10^{3}		

[1)] Nuclides only occur in decommissioning waste.

[2)] Nuclide is of importance for decay chain calculations but is not accounted for in the results due to its short half-life.

Each decay gives rise to a new nuclide, a daughter nuclide. Decay giving rise to a new radioactive nuclide, with a considerable half-life for the time span up to closure of the repository, is handled by an addition of activity to already registered activity. Chain decay is calculated in Triumf NG according to the theory presented in equation D2-2. Index *m* stands for parent nuclide, *d* for daughter nuclide and *dd* for granddaughter nuclide. t_0 represents the package's manufacturing date and $T_{½}$ is the nuclide's half-life according to Table D2-7. The decay chains included are shown in Table D2-8.

a) $A_d(t) = A_d(t_0)e^{-\lambda_d\tau} + \frac{\lambda_d}{\lambda_d-\lambda_m}A_m(t_0)\left(e^{-\lambda_m\tau} - e^{-\lambda_d\tau}\right)$ D2-2

b) $$A_{dd}(t) = A_{dd}(t_0)e^{-\lambda_{dd}\tau} + \frac{\lambda_{dd}}{\lambda_{dd}-\lambda_d}A_d(t_0)\left(e^{-\lambda_d\tau} - e^{-\lambda_{dd}\tau}\right) +$$
$$+\frac{\lambda_{dd}\lambda_d}{\lambda_d - \lambda_m}A_m(t_0)\left(\left(\frac{e^{-\lambda_m\tau} - e^{-\lambda_{dd}\tau}}{\lambda_{dd} - \lambda_m}\right) - \left(\frac{e^{-\lambda_d\tau} - e^{-\lambda_{dd}\tau}}{\lambda_{dd} - \lambda_d}\right)\right)$$

c) $\boldsymbol{\lambda = \frac{ln2}{T_{½}}}$

d) $\boldsymbol{\tau = t - t_0}$

A = activity [*Bq*]

t = time [years]

$T_{½}$ = *half-life* [years]

m = parent nuclide

d = daughter nuclide

dd = granddaughter nuclide

Table D2-8. Decay chains in Triumf NG.

Parent nuclide	Daughter nuclide	Granddaughter nuclide
Pu-239	U-235	
Pu-240	U-236	
Pu-241	Am-241	Np-237
Am-241	Np-237	
Am-243	Pu-239	U-235
Cm-242	Pu-238	
Cm-243	Pu-239	U-235
Cm-244	Pu-240	U-236

D2.4 Results

Table D2-9 presents the calculated nuclide inventory for operational waste at the closure of SFR. Activity is presented at 2075-12-31, for waste deposited in SFR up to the year 2012 and for forecasted waste that is expected to be deposited during the years 2013–2075.

Table D2-9. Radioactivity in operational waste per nuclide and waste vault in 2075.

Nuclide	Deposited waste [Bq]					Forecasted waste [Bq]				Total
	Silo	1BMA	1BTF	2BTF	1BLA	Silo	BMA	BTF	BLA	[Bq]
H-3	7.03E+08	2.05E+08	1.77E+06	4.62E+07	6.69E+05	8.27E+09	8.47E+08	1.27E+08	2.34E+08	1.04E+10
Be-10	3.81E+05	1.73E+05	1.32E+03	2.47E+04	3.60E+02	6.08E+05	6.65E+04	1.25E+04	7.12E+02	1.27E+06
C-14 org	2.92E+11	1.33E+11	6.85E+09	5.39E+09	7.92E+07	4.62E+11	1.68E+10	3.67E+09	0.00E+00	9.20E+11
C-14 inorg	1.12E+12	1.15E+12	3.47E+10	2.14E+11	4.02E+09	1.60E+12	7.57E+11	2.10E+11	0.00E+00	5.09E+12
Cl-36	2.52E+08	2.86E+08	8.13E+06	1.03E+07	2.14E+07	6.42E+08	6.65E+07	1.25E+07	7.11E+05	1.30E+09
Fe-55	3.39E+06	2.99E+05	2.02E+03	2.22E+05	2.21E+03	2.68E+12	1.34E+11	1.97E+08	1.89E+07	2.81E+12
Co-60	2.23E+10	3.57E+09	3.44E+07	1.56E+09	1.96E+07	1.14E+13	7.54E+11	3.88E+10	2.13E+09	1.22E+13
Ni-59	2.77E+12	1.92E+12	1.46E+10	3.80E+10	2.79E+09	3.72E+12	3.06E+11	1.87E+10	3.18E+09	8.80E+12
Ni-63	1.95E+14	1.32E+14	9.96E+11	2.12E+12	2.01E+11	3.16E+14	2.50E+13	1.19E+12	2.57E+11	6.73E+14
Se-79	5.16E+08	1.75E+08	1.36E+06	1.96E+07	3.10E+05	5.25E+08	4.17E+07	1.01E+07	3.03E+05	1.29E+09
Sr-90	8.26E+11	3.24E+11	1.59E+09	4.39E+10	1.61E+08	2.66E+12	2.68E+11	4.69E+10	2.07E+09	4.17E+12
Zr-93	6.34E+08	2.88E+08	2.30E+06	4.11E+07	6.01E+05	1.13E+09	1.11E+08	2.09E+07	1.19E+06	2.23E+09
Nb-93m	2.00E+10	6.49E+09	5.46E+07	1.30E+09	1.91E+07	1.39E+11	1.48E+10	2.42E+09	1.23E+08	1.84E+11
Nb-94	6.33E+09	2.88E+09	2.29E+07	4.10E+08	5.99E+06	1.01E+10	1.11E+09	2.33E+08	1.04E+08	2.12E+10
Mo-93	1.98E+09	7.60E+08	4.00E+07	1.84E+08	4.35E+07	1.65E+09	1.10E+08	2.07E+07	1.17E+06	4.79E+09
Tc-99	1.05E+10	5.24E+09	1.85E+09	6.87E+08	2.15E+09	3.66E+10	1.37E+09	3.16E+08	1.11E+07	5.87E+10
Pd-107	1.29E+08	4.38E+07	3.41E+05	4.90E+06	7.75E+04	1.31E+08	1.04E+07	2.54E+06	7.58E+04	3.22E+08
Ag-108m	3.35E+10	1.50E+10	1.22E+08	2.17E+09	1.97E+08	5.62E+10	6.15E+09	1.43E+09	4.66E+07	1.15E+11
Cd-113m	1.04E+09	3.01E+08	2.22E+06	3.33E+07	7.79E+05	8.53E+09	5.74E+08	1.05E+08	3.92E+06	1.06E+10
Sn-126	6.45E+07	2.19E+07	1.70E+05	2.45E+06	3.87E+04	6.56E+07	5.21E+06	1.27E+06	3.79E+04	1.61E+08
Sb-125	4.04E+05	3.62E+04	2.44E+02	3.01E+04	1.45E+02	1.32E+11	1.18E+08	1.78E+07	1.11E+06	1.32E+11
I-129	5.97E+08	1.21E+08	1.22E+07	1.31E+07	3.87E+05	3.82E+08	3.13E+07	7.61E+06	2.27E+05	1.17E+09
Cs-134	4.86E+02	1.69E+01	8.09E-02	1.06E+01	3.23E-01	2.20E+11	3.66E+08	1.59E+05	6.02E+04	2.20E+11
Cs-135	3.11E+09	7.66E+08	7.86E+07	1.75E+07	2.99E+06	1.29E+09	1.04E+08	2.54E+07	7.58E+05	5.40E+09
Cs-137	1.66E+13	5.34E+12	4.05E+10	5.88E+11	1.13E+10	4.22E+13	3.30E+12	7.08E+11	2.33E+10	6.89E+13
Ba-133	3.30E+07	8.83E+06	7.76E+04	2.17E+06	3.11E+04	5.83E+08	5.75E+07	7.99E+06	3.98E+05	6.93E+08
Pm-147	1.12E+05	6.86E+03	3.72E+01	1.71E+03	7.11E+01	3.58E+11	7.59E+08	8.41E+06	1.06E+06	3.59E+11
Sm-151	1.94E+11	6.49E+10	5.01E+08	7.22E+09	1.22E+08	2.67E+11	2.12E+10	4.95E+09	1.53E+08	5.60E+11
Eu-152	1.12E+08	2.93E+07	1.51E+07	3.24E+06	9.09E+07	7.49E+08	7.37E+07	5.02E+07	5.53E+07	1.18E+09
Eu-154	1.23E+10	3.01E+09	2.20E+07	3.46E+08	1.29E+07	5.11E+11	2.36E+10	3.41E+09	7.02E+07	5.54E+11
Eu-155	5.04E+07	7.50E+06	5.01E+04	1.06E+06	4.38E+04	9.95E+10	1.26E+09	1.07E+08	4.66E+06	1.01E+11
Ho-166m	2.42E+09	1.10E+09	8.74E+06	1.57E+08	2.29E+06	4.40E+09	4.31E+08	8.10E+07	4.60E+06	8.60E+09
U-232	1.80E+05	6.62E+04	7.41E+03	6.04E+03	8.91E+01	3.92E+05	7.85E+04	9.50E+03	1.01E+04	7.50E+05
U-234	1.32E+07	5.47E+06	4.42E+05	4.57E+05	6.97E+03	2.25E+07	4.24E+06	5.42E+05	5.64E+05	4.75E+07
U-235	5.87E+06	1.68E+06	1.09E+07	1.04E+05	2.15E+08	8.30E+06	1.40E+06	7.54E+06	4.06E+08	6.57E+08
U-236	5.98E+06	2.28E+06	1.43E+05	3.51E+05	2.05E+03	7.46E+06	1.28E+06	2.63E+05	1.70E+05	1.79E+07
U-238	1.27E+07	4.18E+06	1.90E+05	8.75E+05	7.15E+08	2.01E+07	3.01E+06	6.65E+05	1.95E+08	9.52E+08
Np-237	5.23E+07	1.51E+07	2.87E+05	1.96E+06	7.99E+03	4.81E+08	1.36E+07	8.02E+05	2.58E+05	5.65E+08
Pu-238	1.24E+10	5.02E+09	1.09E+09	4.17E+08	1.20E+07	4.47E+10	1.05E+10	1.04E+09	1.50E+09	7.67E+10
Pu-239	5.51E+09	2.27E+09	2.18E+08	1.90E+08	3.19E+06	9.37E+09	1.77E+09	2.49E+08	2.81E+08	1.99E+10
Pu-240	7.68E+09	3.17E+09	2.25E+08	2.65E+08	3.83E+06	1.32E+10	2.48E+09	2.93E+08	2.85E+08	2.76E+10
Pu-241	3.79E+10	8.70E+09	3.41E+09	1.17E+09	1.48E+07	2.25E+11	5.55E+10	5.14E+09	5.69E+09	3.42E+11
Pu-242	3.97E+07	1.64E+07	1.33E+06	1.37E+06	2.09E+04	6.75E+07	1.27E+07	1.62E+06	1.69E+06	1.42E+08
Am-241	5.45E+10	2.31E+10	3.08E+09	1.85E+09	4.41E+07	2.31E+13	2.00E+10	3.04E+09	2.15E+09	2.32E+13
Am-242m	9.00E+07	3.51E+07	3.32E+06	3.06E+06	4.59E+04	1.72E+08	3.34E+07	4.16E+06	4.38E+06	3.46E+08
Am-243	4.21E+08	1.73E+08	1.32E+07	1.79E+07	2.14E+05	9.52E+08	1.24E+08	1.92E+07	1.69E+07	1.74E+09
Cm-243	3.62E+07	1.18E+07	2.24E+06	3.83E+05	1.82E+04	1.20E+08	2.64E+07	1.61E+06	3.31E+06	2.02E+08
Cm-244	4.53E+08	1.80E+08	1.47E+08	1.52E+07	1.40E+06	5.30E+09	2.02E+09	1.34E+08	2.35E+08	8.49E+09
Cm-245	3.95E+06	1.63E+06	1.32E+05	1.36E+05	2.08E+03	6.72E+06	1.27E+06	1.62E+05	1.68E+05	1.42E+07
Cm-246	1.05E+06	4.32E+05	3.51E+04	3.62E+04	5.51E+02	1.79E+06	3.37E+05	4.30E+04	4.48E+04	3.76E+06
Total	**2.17E+14**	**1.41E+14**	**1.11E+12**	**3.03E+12**	**2.23E+11**	**4.06E+14**	**3.07E+13**	**2.24E+12**	**3.00E+11**	**8.02E+14**

D3 Decommissioning waste

D3.1 Available sources of information

For FKA, OKG, RAB and Clink, activity data can be found in the decommissioning studies (Anunti et al. 2013, Larsson et al. 2013, Hansson et al. 2013, Edelborg et al. 2014). The activity data for BKAB and Ågesta are based on Jönsson (2013) and Lindow (2012). For the reactors F1, F2, F3, R3 and R4, the activity calculations for the present report have been updated with regard to prolonged operating time (60 years). The calculations have been carried out according to the methodology described in Lundgren (2012c), but with the assumption that the activity in exhaust systems increases proportionally to the amount of dissolved uranium.

There are no activity data available for facilities operated by SNAB and Svafo.

D3.2 Assumptions

Nuclide-specific activity is presented per system for each unit in the decommissioning studies. The activity is presented at an assumed reference date. Table D3-1 shows the current closure date and reference date for each unit.

For Clink, the information on closure date differs between operational and decommissioning waste, where the forecast for operational waste extends up to 2070. This is due to the fact that data are produced at different times with different assumptions. This is not judged to have a significant impact, which is why the data have been used in their existing form.

At present, there exist no decommissioning waste and therefore no acceptance criteria in terms of radiological properties have been defined for this kind of waste. On the other hand, the decommissioning waste is assumed to fulfil the acceptance criteria that apply today for operational waste, such as maximum surface dose rate and surface contamination and the opportunity to determine the content of radionuclides. Calculations show that the relationship between the surface dose rate on an ISO container and the specific activity is linear and that a surface dose rate of 2 mSv/h corresponds to about 7×10^6 Bq/kg (Carlsson 2011). The postulated value for delimitation between low- and intermediate-level wastes is conservatively assumed to be 10^6 Bq/kg. For individual systems this may change, but without exceeding the level of 7×10^6 Bq/kg for waste that is classified as low-level.

The activity in the decommissioning waste is calculated with the assumption that BWR reactor pressure vessels are deposited while still intact. The PWR reactor pressure vessels from RAB and Ågesta will, due to their high activity content, be deposited in SFL and are, according to type, meant to be interim-stored at RAB and the Studsvik facility. Systems with a total activity of C-14 > 10^{10} Bq, which previously were planned to be deposited in BMA, have been excluded in this report and are instead assumed to be managed in the planned final repository for long-lived waste, SFL.

The waste that will be deposited in the Silo comes from system decontamination at the nuclear power plants and consists of ion exchange resins. For this waste, the same distribution between organic and inorganic C-14 as for ion-exchange resins in the operational waste is assumed, see Table D2-3. The same distribution of C-14 applies to decommissioning waste containing concrete where the contamination primarily consists of leakage from cooling water. For isolated process systems concerned and for the RPVs, the activity for C-14 is induced in steel and thereby no division between organic and inorganic C-14 is made. Decommissioning waste with sand is assumed to contain no C-14 activity.

In addition to the nuclides presented for operational waste, Ca-41 and In-115 are included for the decommissioning waste. The reason for these only existing in the decommissioning waste is that Ca-41 occurs in the large amounts of concrete in the biological shield and In-115 comes from alloys in the control rods from the PWR reactor in Ågesta. The activity of the natural uranium isotopes U-234, U-235 and U-238 is not considered in the decommissioning studies, as the natural content of these nuclides in the decommissioning materials exceeds the concentration that occurs due to nuclear activities. The activity U-235 that generates from chain decay is, however, presented even for the decommissioning waste.

The total amount of calculated radioactivity for the decommissioning waste includes activity contributions from neutron-induced activity, contamination on system surfaces and contaminated concrete. The contributions from these three categories are calculated with the methods described below.

Activation calculations for the neutron-induced activity are made with three different energy ranges; thermal, epithermal and fast neutrons. Neutron flux densities are determined generally by three-dimensional calculation methods of the type MCNP (Monte Carlo N-Particle Transport Code). The neutron fluxes are then used together with typical material compositions and activation cross sections according to the three-range model for determining the neutron-induced activity in different components.

The radioactivity from contamination on system surfaces includes contribution from the following sources:

- Activated corrosion products.
  The build-up of corrosion products on the fuel, activation and release of these from the fuel and the resulting system contamination is calculated with the calculation model CrudAct. There are separate models for BWR and PWR.
- Remnants from fuel leakage.
  The system oxides, besides activated corrosion products, also contain small amounts of actinides and insoluble fission products. The amounts are dependent on the history of fuel damage, especially periods with secondary damage where dissolution of the fuel has occurred. In order to estimate the activity, a leakage model for uranium, the estimated amount of dissolved uranium that adheres to system surfaces and the nuclide vector for the distribution of activity in the fuel are used.
- Remaining ion exchange resins.
  Remaining amounts of ion exchange resins are expected to give rise to the main contamination in the systems that handle ion exchange resins. The annually accumulated activity in ion exchange resins, as well as the annual consumption of ion exchange resins in the different systems, serves as a basis for the calculations. It is also assumed that 0.1 kg/m^2 ion exchange resins remain in the systems, based on the total surface area for systems that handle ion exchange resins.
- Exhaust systems.
  In the reactor exhaust systems, volatile long-lived nuclides and certain noble gas daughters accumulate as a function of the fuel damage statistics and the amount of dissolved uranium leading to free uranium on the core.

The activity in the amount of contaminated concrete is determined according to the following model:

- Surface contamination in Bq/m^2 is obtained for Co-60 and Cs-137 from data based on experience from the reactors B1 and B2.
- Other nuclides are determined on the basis of a nuclide vector for ion exchange resins from system 342 (system drainage) for each reactor, where activated corrosion products are correlated with Co-60 and fission products and actinides are correlated with Cs-137.

The emission rates obtained from the above-mentioned activity sources are compiled by Studsvik ALARA Engineering to determine the total decommissioning inventory. The model also covers which emission rates each system is linked to and the scaling factor for this link. The scaling factors are based on approximate original radiation levels in the different systems, which are taken from databases for dose rate measurements. In addition to this an estimated uncertainty factor is included for each system.

Based on this the activity for each system at the reference date is obtained and a summation of all systems gives the total activity inventory. A more detailed description of the theory as well as the calculations behind the different emission rates can be found in a separate methodological report (Lundgren 2012c).

D3.3 Calculation methods

The reported activity at the closure of the repository, in 2075, has been calculated in Excel by starting from the specified reference date for each unit. To calculate the radionuclide content in SFR in 2075, the decay of nuclides must be taken into account. Activity at the time of closure is calculated according to equation D2-1, where t_0 represents the reference date presented in Table D3-1 and $T_{½}$ is the nuclide's half-life according to Table D2-7.

Each decay gives rise to a new nuclide, a daughter nuclide. Decay giving rise to a new radioactive nuclide, with a considerable half-life for the time span up to closure of the repository, is handled by an addition of activity for the nuclide in question. Decay chain calculations for decommissioning waste are made according to equation D2-2. The reference date, presented in Table D3-1, is represented by t_0, while $T_{½}$ is the nuclide's half-life according to Table D2-7. The decay chains included are shown in Table D2-8.

In order to calculate the total radioactivity in SFR at closure, all nuclide activities are summarised per waste vault. For calculation of the activity content in an average package for each waste type, presented in Appendix E, the systems for the waste type are summarised, after which the total activity of the waste type is divided by the number of packages of the waste type.

Table D3-1. Closure and reference date.

Unit	Closure date	Reference date
B1, B2	1999, 2005	2015
F1, F2	2040, 2041	2042
F3	2045	2046
O1	2032	2036
O2	2035	2036
O3	2045	2046
R1	2025	2034
R2	2025	2034
R3	2041	2043
R4	2043	2044
Clink	2075	2075
Ågesta	1974	2020

D3.4 Results

Table D3-2 presents the calculated nuclide inventory for decommissioning waste at the closure of SFR on 2075-12-31.

D4 Total radioactivity content

Table D4-1 presents the total nuclide inventory calculated for operational and decommissioning waste at the closure of SFR on 2075-12-31.

Table D3-2. Radioctivity in decommissioning waste per nuclide and waste vault in 2075.

Nuclide	Silo [Bq]	BRT [Bq]	BMA [Bq]	BLA [Bq]	Total [Bq]
H-3	0.00E+00	0.00E+00	3.31E+12	1.94E+11	3.51E+12
Be-10	0.00E+00	0.00E+00	3.33E+03	8.41E+02	4.17E+03
C-14 org	1.03E+09	0.00E+00	9.91E+08	2.25E+08	2.25E+09
C-14 inorg	2.82E+09	0.00E+00	3.17E+09	9.27E+08	6.92E+09
C-14 ind	0.00E+00	1.02E+10	5.09E+09	1.19E+09	1.65E+10
Cl-36	1.98E+05	7.20E+06	1.83E+08	4.56E+07	2.36E+08
Ca-41	0.00E+00	0.00E+00	1.56E+10	3.91E+09	1.95E+10
Fe-55	5.14E+10	1.49E+10	2.38E+10	4.35E+08	9.06E+10
Co-60	1.49E+12	1.93E+11	1.64E+12	2.48E+10	3.34E+12
Ni-59	3.68E+11	1.60E+11	8.19E+11	9.56E+09	1.36E+12
Ni-63	3.61E+13	1.44E+13	8.18E+13	9.67E+11	1.33E+14
Se-79	1.11E+07	0.00E+00	3.69E+04	5.73E+06	1.68E+07
Sr-90	1.20E+11	2.32E+10	3.18E+11	2.25E+10	4.84E+11
Zr-93	2.71E+09	1.84E+08	1.03E+09	2.88E+07	3.96E+09
Nb-93m	9.16E+12	1.06E+12	1.31E+13	1.34E+11	2.34E+13
Nb-94	7.00E+10	7.94E+09	9.09E+10	9.03E+08	1.70E+11
Mo-93	5.83E+09	2.99E+09	4.15E+09	4.72E+07	1.30E+10
Tc-99	2.88E+09	4.50E+08	1.03E+09	1.95E+08	4.55E+09
Pd-107	1.49E+07	0.00E+00	2.55E+09	1.67E+06	2.57E+09
Ag-108m	1.40E+11	1.62E+09	3.89E+10	1.48E+09	1.82E+11
Cd-113m	1.45E+07	0.00E+00	1.58E+07	3.39E+06	3.37E+07
In-115	0.00E+00	0.00E+00	3.13E+05	0.00E+00	3.13E+05
Sn-126	7.49E+07	7.52E+05	1.66E+07	7.90E+06	1.00E+08
Sb-125	4.74E+08	1.34E+07	1.88E+08	3.83E+06	6.79E+08
I-129	4.61E+06	0.00E+00	1.13E+06	1.75E+06	7.49E+06
Cs-134	3.26E+07	0.00E+00	4.15E+06	1.35E+06	3.81E+07
Cs-135	6.99E+07	0.00E+00	2.47E+07	1.75E+08	2.69E+08
Cs-137	8.43E+11	0.00E+00	4.20E+11	4.79E+11	1.74E+12
Ba-133	9.62E+04	0.00E+00	1.25E+08	1.24E+07	1.38E+08
Pm-147	7.23E+06	1.37E+06	1.88E+07	4.29E+05	2.78E+07
Sm-151	1.91E+09	3.42E+08	3.21E+10	5.79E+09	4.02E+10
Eu-152	2.67E+06	5.41E+05	1.33E+11	1.72E+10	1.51E+11
Eu-154	5.01E+08	9.26E+07	3.53E+09	2.24E+08	4.35E+09
Eu-155	1.19E+07	2.40E+06	1.25E+08	8.46E+06	1.48E+08
Ho-166m	3.39E+04	8.00E+03	4.02E+08	8.76E+07	4.89E+08
U-232	4.74E+04	6.86E+03	8.99E+04	1.52E+03	1.46E+05
U-235	8.00E+01	1.49E+01	1.99E+02	2.09E+00	2.96E+02
U-236	2.41E+06	3.92E+05	5.09E+06	7.46E+04	7.96E+06
Np-237	2.72E+06	4.70E+05	6.20E+06	6.18E+04	9.45E+06
Pu-238	1.59E+10	2.71E+09	3.61E+10	3.55E+08	5.51E+10
Pu-239	2.11E+09	4.16E+08	5.51E+09	5.80E+07	8.09E+09
Pu-240	3.04E+09	5.93E+08	7.42E+09	7.41E+07	1.11E+10
Pu-241	4.41E+10	9.06E+09	1.26E+11	1.32E+09	1.80E+11
Pu-242	1.57E+07	3.11E+06	4.11E+07	3.97E+05	6.03E+07
Am-241	1.16E+10	1.99E+09	2.73E+10	2.65E+08	4.12E+10
Am-242m	5.92E+07	1.32E+07	1.59E+08	1.44E+06	2.33E+08
Am-243	2.22E+08	4.14E+07	5.68E+08	5.48E+06	8.37E+08
Cm-243	3.27E+07	6.38E+06	8.30E+07	8.28E+05	1.23E+08
Cm-244	3.51E+09	6.74E+08	9.18E+09	9.68E+07	1.35E+10
Cm-245	4.21E+06	6.82E+05	9.22E+06	8.70E+04	1.42E+07
Cm-246	1.46E+06	2.24E+05	3.10E+06	3.13E+04	4.82E+06
Total	**4.85E+13**	**1.59E+13**	**1.02E+14**	**1.87E+12**	**1.68E+14**

Table D4-1. Radioactivity in operational and decommissioning waste per nuclide and waste vault in 2075.

Nuclide	Silo [Bq]	BRT [Bq]	BMA [Bq]	BTF [Bq]	BLA [Bq]	Total [Bq]
H-3	8.97E+09	0.00E+00	3.31E+12	1.75E+08	1.95E+11	3.52E+12
Be-10	9.89E+05	0.00E+00	2.43E+05	3.85E+04	1.91E+03	1.27E+06
C-14 org	7.55E+11	0.00E+00	1.51E+11	1.59E+10	3.04E+08	9.22E+11
C-14 inorg	2.72E+12	0.00E+00	1.91E+12	4.58E+11	4.95E+09	5.09E+12
C-14 ind	0.00E+00	1.02E+10	5.09E+09	0.00E+00	1.19E+09	1.65E+10
Cl-36	8.94E+08	7.20E+06	5.36E+08	3.10E+07	6.77E+07	1.54E+09
Ca-41	0.00E+00	0.00E+00	1.56E+10	0.00E+00	3.91E+09	1.95E+10
Fe-55	2.73E+12	1.49E+10	1.58E+11	1.98E+08	4.54E+08	2.91E+12
Co-60	1.29E+13	1.93E+11	2.39E+12	4.04E+10	2.70E+10	1.55E+13
Ni-59	6.86E+12	1.60E+11	3.05E+12	7.13E+10	1.55E+10	1.02E+13
Ni-63	5.47E+14	1.44E+13	2.39E+14	4.30E+12	1.43E+12	8.06E+14
Se-79	1.05E+09	0.00E+00	2.17E+08	3.11E+07	6.34E+06	1.31E+09
Sr-90	3.61E+12	2.32E+10	9.09E+11	9.24E+10	2.47E+10	4.66E+12
Zr-93	4.48E+09	1.84E+08	1.43E+09	6.43E+07	3.06E+07	6.19E+09
Nb-93m	9.31E+12	1.06E+12	1.31E+13	3.78E+09	1.34E+11	2.36E+13
Nb-94	8.65E+10	7.94E+09	9.49E+10	6.66E+08	1.01E+09	1.91E+11
Mo-93	9.46E+09	2.99E+09	5.02E+09	2.45E+08	9.18E+07	1.78E+10
Tc-99	5.00E+10	4.50E+08	7.64E+09	2.85E+09	2.35E+09	6.33E+10
Pd-107	2.75E+08	0.00E+00	2.60E+09	7.78E+06	1.82E+06	2.89E+09
Ag-108m	2.30E+11	1.62E+09	6.01E+10	3.72E+09	1.72E+09	2.97E+11
Cd-113m	9.58E+09	0.00E+00	8.91E+08	1.40E+08	8.09E+06	1.06E+10
In-115	0.00E+00	0.00E+00	3.13E+05	0.00E+00	0.00E+00	3.13E+05
Sn-126	2.05E+08	7.52E+05	4.37E+07	3.89E+06	7.98E+06	2.61E+08
Sb-125	1.32E+11	1.34E+07	3.06E+08	1.78E+07	4.94E+06	1.32E+11
I-129	9.84E+08	0.00E+00	1.54E+08	3.29E+07	2.37E+06	1.17E+09
Cs-134	2.20E+11	0.00E+00	3.70E+08	1.59E+05	1.41E+06	2.20E+11
Cs-135	4.47E+09	0.00E+00	8.95E+08	1.21E+08	1.79E+08	5.67E+09
Cs-137	5.97E+13	0.00E+00	9.05E+12	1.34E+12	5.14E+11	7.06E+13
Ba-133	6.16E+08	0.00E+00	1.92E+08	1.02E+07	1.28E+07	8.31E+08
Pm-147	3.58E+11	1.37E+06	7.78E+08	8.41E+06	1.49E+06	3.59E+11
Sm-151	4.63E+11	3.42E+08	1.18E+11	1.27E+10	6.06E+09	6.00E+11
Eu-152	8.64E+08	5.41E+05	1.33E+11	6.85E+07	1.74E+10	1.52E+11
Eu-154	5.24E+11	9.26E+07	3.01E+10	3.78E+09	3.07E+08	5.58E+11
Eu-155	9.96E+10	2.40E+06	1.40E+09	1.08E+08	1.32E+07	1.01E+11
Ho-166m	6.82E+09	8.00E+03	1.93E+09	2.47E+08	9.45E+07	9.09E+09
U-232	6.20E+05	6.86E+03	2.35E+05	2.29E+04	1.17E+04	8.96E+05
U-234	3.58E+07	0.00E+00	9.71E+06	1.44E+06	5.71E+05	4.75E+07
U-235	1.42E+07	1.49E+01	3.08E+06	1.85E+07	6.21E+08	6.57E+08
U-236	1.58E+07	3.92E+05	8.65E+06	7.57E+05	2.46E+05	2.59E+07
U-238	3.28E+07	0.00E+00	7.18E+06	1.73E+06	9.10E+08	9.52E+08
Np-237	5.36E+08	4.70E+05	3.49E+07	3.04E+06	3.28E+05	5.75E+08
Pu-238	7.29E+10	2.71E+09	5.17E+10	2.55E+09	1.87E+09	1.32E+11
Pu-239	1.70E+10	4.16E+08	9.55E+09	6.58E+08	3.43E+08	2.80E+10
Pu-240	2.39E+10	5.93E+08	1.31E+10	7.84E+08	3.63E+08	3.87E+10
Pu-241	3.07E+11	9.06E+09	1.90E+11	9.72E+09	7.03E+09	5.23E+11
Pu-242	1.23E+08	3.11E+06	7.02E+07	4.32E+06	2.11E+06	2.03E+08
Am-241	2.32E+13	1.99E+09	7.04E+10	7.97E+09	2.46E+09	2.33E+13
Am-242m	3.22E+08	1.32E+07	2.28E+08	1.05E+07	5.86E+06	5.79E+08
Am-243	1.59E+09	4.14E+07	8.64E+08	5.03E+07	2.26E+07	2.57E+09
Cm-243	1.89E+08	6.38E+06	1.21E+08	4.24E+06	4.16E+06	3.25E+08
Cm-244	9.26E+09	6.74E+08	1.14E+10	2.96E+08	3.34E+08	2.19E+10
Cm-245	1.49E+07	6.82E+05	1.21E+07	4.30E+05	2.58E+05	2.84E+07
Cm-246	4.29E+06	2.24E+05	3.87E+06	1.14E+05	7.66E+04	8.58E+06
Total	6.71E+14	1.59E+13	2.74E+14	6.37E+12	2.39E+12	9.70E+14

D5 Uncertainties

The calculated activities from operational waste are primarily based on the activity of the already deposited waste packages, the forecasted number of future waste packages and the average activity of Co-60 and Cs-137 per waste type at the manufacturing date. The uncertainty in the number of forecasted waste packages, discussed in Appendix A, will therefore also influence the uncertainties in the activity. The activity for future waste packages may change due to power increases or changes in the allowed surface dose rates, after which the mean value of the deposited packages would no longer be representative for newly produced packages.

The estimated amounts of activity in the decommissioning waste from the Swedish nuclear power plants are presented in the decommissioning studies (Anunti et al. 2013, Larsson et al. 2013, Hansson et al. 2013, Edelborg et al. 2014), after which SKB has updated the calculations F1, F2, F3, R3 and R4 with respect to the extended operating time of 60 years, see Section D3.1. For BKAB and Ågesta, the calculations are presented in Jönsson (2013) and Lindow (2012). The uncertainties have been estimated for each system. Activities in the decommissioning waste from the facilities SNAB and Svafo are not considered at present, since there are no activity calculations available for these, which means an underestimation.

D5.1 Gamma measurements

Before each fabricated waste package is sent to SFR, it is gamma-measured to determine the activity of the key nuclides Co-60 and Cs-137. The activity of difficult-to-measure nuclides that are not determined by sample analyses or specific calculations are correlated based on the activity for these key nuclides. The uncertainty in gamma measurements is thus mirrored in the activity of the nuclides that are calculated by correlation, since the correlations are based on the activity of Co-60 and Cs-137. Forecasted activities are also correlated with the activity of the key nuclides, which means that the uncertainty in gamma measurements constitutes an error source for predictions as well.

Measurement errors can be divided into the three categories according to “Kontroll av SFR-avfall. Delprojekt mätteknik” (SKBdoc 1400742): temporary errors, systematic errors and gross errors. Temporary errors can in turn be broken down into sampling, calibration, measurement time, inhomogeneity in samples and the intensity and geometry of the radiation source. Systematic errors include e.g. the determined uncertainty of the calibration radiation sources, errors in the measurement of distance between the calibration radiation source and the detector and errors in the applied calculation model. Gross errors are most often associated with manual input of measured data. Manual input is, however, no longer as common, but is still used by BKAB and FKA.

The uncertainty in the measurement of Co-60 and Cs-137 in the Swedish nuclear facilities has been determined from an extensive completed study, “Kontroll av SFR-avfall. Delprojekt mätteknik” (SKBdoc 1400742), and is of the order of ± 10–20% (1σ) for homogeneous waste. For inhomogeneous packages, the uncertainty is considerably larger. For inhomogeneous packages with many point sources or widespread distribution of activity where gamma radiation is measured from all sides, on the other hand, the total error is not much greater than for homogeneous packages. For waste packages with a small number of concentrated activity sources such as drums or moulds containing scrap metal, the error is of greater importance. From estimations made at SNAB and FKA, the error for these facilities may be of the same size as the measured value. As a standard, +100% and –50% of the value in “Kontroll av SFR-avfall. Delprojekt mätteknik” (SKBdoc 1400742) is assigned. A typical systematic error is that the amount of Cs-137 can escape detection, due to the detector’s low sensitivity when measuring the activity of nuclides with low gamma energy, such as Cs-137, in the presence of nuclides with high gamma energy, such as Co-60. MDA (the minimum detectable activity) for Cs-137 can vary between 1 and 6% of the activity for Co-60, depending on which detector is used, according to “Kontroll av SFR-avfall. Delprojekt mätteknik” (SKBdoc 1400742).

D5.2 Actinides

The so-called transuranics include all nuclides with mass numbers higher than uranium and belong to the element series actinides. Water sampling with subsequent alpha measurements is carried out to determine the activity for these in the waste class “Ion exchange resins” in operational waste. The nuclides that are usually measured by the nuclear power plants are Pu-238/Am-241, Pu-239/Pu-240, Am-243, Cm-242 and Cm-243/Cm-244, where some have congruent alpha energies and therefore in some cases are reported as one. The activity for the remaining actinides is determined by correlation with Pu-239/Pu-240, for which the uncertainties are presented in Section D5.5.

The uncertainty in the actinide inventory in SFR has been calculated at ±10% or ±14% (1σ) depending on which sampling method is used (continuous sampling or intermittent random sampling) (Ingemansson 2000a). Earlier assessments made by the chemists at the nuclear power plants resulted in an uncertainty of about ±25%, whereof half stems from the sampling and the other half from the measurement analysis of the samples. In the nuclear power plants and Clab, the amount of actinides in the waste class "Trash and scrap metal" is correlated with the quantity of Co-60 and the uncertainty for this has been calculated at ±20% (1σ) (Ingemansson 2000a).

The forecasted amount of actinides in the operational waste is estimated conservatively, as the annually generated activity of transuranics is distributed between the packages already deposited in SFR and not in the waste that has actually been produced and may still be interim-stored on site. This double calculation of activity contributes to a large extent to the total uncertainty for the actinide inventory when the proportion of waste packages that are presently interim-stored at the nuclear power plants is about 25% of the waste already deposited in SFR.

D5.2.1 Fuel damage

Fuel damage leads to actinides leaking out into the reactor water, which exposes the ion exchange resins and thereby the operational waste to activity from these nuclides. Furthermore, activity adheres to the reactor system surfaces, which also affects the activity in the decommissioning waste. Development and optimisation of the fuel is done continuously, such as its size and form, but also the position of the fuel assemblies in the core with respect to age and burnup. Historically, the amount of fuel damage has decreased with time. For example, the nuclear power plants in the beginning used fuel rods in matrices of 8×8 in a fuel assembly for BWR reactors. These fuel types generated a lot of damage in BWR reactors and have progressively been phased out and superseded by fuel with a larger number of rods in a fuel assembly (e.g. 96 rods in Svea 96). As the number of fuel rods in an assembly increased, the diameter of each fuel rod reduced. As a result of this, a more even radial temperature is achieved for each individual pellet and thus less thermal stress in the pellets. This has led to greater sustainability for the fuel and less fuel damage. At present, most fuel damage is due to debris in the core, which can give rise to damage, rather than the fuel having poor sustainability. The trend of a decreased amount of fuel damage is expected to continue even in the future.

The forecasted amount of activity for actinides in operational waste that is measured directly and will be deposited in SFR in the future is based on the ratio between existing data reported for the actinides and the Co-60 activity at the manufacturing date of existing waste. The forecasts thereby take into account the history of fuel damage, so that the amount of future actinides can be regarded as conservatively estimated for operational waste also from this aspect, since the amount of fuel damage is assumed to continue to decrease in the future. The activity of actinides in decommissioning waste is calculated from a model based on reactor-specific amounts of I-131 and Cs-137 in the reactor water. Also for decommissioning waste, the history of fuel damage has served as a basis for the calculated future operation and the formation of actinides, which can be considered a conservative estimate.

D5.3 Ni-59/63 and Sr-90

At the nuclear power plants in operation, water samples are taken to determine the activity of Ni-63 and Sr-90 in operational waste, while BKAB and Clab only take water samples to determine the activity of Sr-90. The activity of Ni-59 is calculated from the amount of Ni-63. With these nuclide-specific measurements the uncertainty for the activity estimations will be relatively low. The nickel measurements have, however, only been carried out for a few years, which means that the main uncertainty for the activity of Ni-63 and Ni-59 stems from the correlation with Co-60 which was done previously and is still used in the forecast, see Section D5.5.

At OKG, the method for measurement of Ni-63 has been evaluated. The interfering amount of Co-60 is gamma-measured and subtracted from the measured value for Ni-63. The evaluation showed that Ni-63 nevertheless was overestimated by almost a factor of 4. This was due to the samples not being allowed to decay some months prior to measurement, with interference from Co-58 as a result. The result of the evaluation was that overestimation may be reduced by double separation of the samples prior to measurement, which reduces the amount of cobalt. In combination with letting the samples decay some months prior to measurement, the amount of Ni-63 is judged to be overestimated by at most 20%.

Sr-90 activity is determined in a similar manner to Ni-63, i.e. by liquid scintillation. But the nuclide actually detected is the daughter nuclide Y-90, from which the amount of Sr-90 can be determined (due to secular equilibrium). The inventory for Sr-90 has been evaluated together with actinides and includes uncertainties of the same order of magnitude as these, i.e. ±10–14% (1σ) for ion exchange resins and ±20% (1σ) for trash and scrap metal (Ingemansson 2000a). The activity for Sr-90 in trash and scrap metal is, by the nuclear power plants and Clab, correlated with the activity for Co-60 with specific correlate factors.

D5.4 Calculated activities

For some nuclides, the activity in operational waste is calculated with specifically developed calculation models. This applies to nuclides C-14, Cl-36, Mo-93, Tc-99, I-129 and Cs-135.

On behalf of SKB, the annual contribution of Mo-93, Tc-99, I-129 and Cs-135 is calculated by Studsvik ALARA Engineering, previously with Excel-based calculation programs, now with SciLab/FISPACT, which is described in "Uppskattning av Mo-93, Tc-99, I-129 och Cs-135 i driftavfall – Uppdatering till och med 2011" (SKBdoc 1341356).

The calculation model for I-129 is based on values for the measurable nuclides I-131 and Cs-137 in combination with leakage models for BWR and PWR fuel. The amount of I-129 that stems from Clab is also estimated based on the amount of Cs-137 and the ratio I-129/Cs-137 in the gap inventory. The amount of I-129 from SNAB's R2 reactor (operational waste from R2 belongs to SNAB while the decommissioning waste belongs to Svafo) is calculated in a similar manner, but both the amount of Cs-137 and the ratio for I-129/Cs-137 are estimated more roughly.

The calculation model for Mo-93, Tc-99 and Cs-135 is based on the model for I-129 and uses the nuclides Mo-99, Tc-99m and Cs-137 as indicators.

Apart from the R2 reactor at SNAB, the calculations only apply to waste produced at the nuclear power plants and Clab and not the waste delivered from SNAB and Svafo.

The uncertainty in the calculation models has been analysed for the different input parameters and a general standard deviation of 20% has been set for these according to "Uppskattning av Mo-93, Tc-99, I-129 och Cs-135 i driftavfall – Metodik" (SKBdoc 1393365). This results in an uncertainty for calculated activities of ± 33% on the level 2σ according to "Uppskattning av Mo-93, Tc-99, I-129 och Cs-135 i driftavfall – Uppdatering till och med 2011" (SKBdoc 1341356).

To determine the activity of C-14, calculated production rates in Bq/MWh_t are used for BWR and PWR, which together with the expected reactor-specific energy production give an estimated amount of C-14. The uptake in ion exchange resins is determined by sampling and quantification, where samples are taken on storage tanks for ion exchange resins after air bubbling and mixing. Thereafter, the sample is filtered and mixed with water and acid so that carbon dioxide forms from the carbonates in the sample. Only organic C-14 compounds remain thereafter in the sample, which oxidises catalytically in the formation of carbon dioxide. Quantification of organic compounds is done by liquid scintillation. The carbon dioxide that has formed from inorganic C-14 is quantified in a similar manner. The uncertainty in the sampling for C-14 has been estimated at ±20% (1σ), based on sampling made at the nuclear power plants during the past years, and reported in "C-14 accumulated in ion exchange resins in Swedish Nuclear Power plants" (SKBdoc 1339709). In Clab, no activity for C-14 has been detected and therefore a fraction of the detection limit is assigned conservatively. For waste deposited in SNAB, no C-14 activity is assumed to occur except from the activity estimation made for the R2 reactor. Most of the C-14 that nevertheless occurs in SNAB's waste can be assumed to disappear in the form of gas during incineration. The amount of C-14 that has been deposited in SFR in older waste from SNAB and Svafo is not currently included in the radionuclide inventory, leading to an uncertainty, particularly regarding the early non-nuclear waste.

The amount of Cl-36 in operational waste is also determined by indirect calculation methods. The method is based on reactor effect and the amount of chloride in the reactor water according to "Klor-36 – Uppskattning av aktivitet i driftavfall från svenska LWR" (SKBdoc 1393449). The reactor-specific history of chloride content since the start of operations has been calculated and the amount of Cl-36 produced so far as well as the forecasted total future activity is conservatively calculated and considered to not exceed 1 GBq. At SKB, the annual activity for Cl-36 is registered based on this method and the parameters chloride content in water, thermal energy production and moisture content in vapour. Predictions are made by correlation with Co-60, which results in a total Cl-36 activity in operational waste of 1.6 GBq at closure and can be considered a conservative estimate.

An uncertainty factor for the calculated activities in decommissioning waste has been estimated per system and is valid for all reactors (Lundgren 2012a, b, d, Jonasson 2012a, b, c, d, e, f, g, h, i, j). The uncertainty factor is presented per system together with each identified critical factor in Table D5-1. For the calculations for Ågesta, the uncertainty is judged to be a factor of 3 (Jiselmark and Viertel 2011).

The system-specific uncertainty factor has thereafter been multiplied (maximum case) or divided (minimum case) by the calculated activity for each system. Based on the planned deposition location for each system in SFR, a summation has been made per waste vault over all input activities. The uncertainty factors for decommissioning waste are obtained by dividing the activity including uncertainty by the activity excluding uncertainty and are presented per nuclide and waste vault in Table D5-2 and Table D5-3.

Table D5-1. Uncertainty factor for decommissioning waste per system.

System	Uncertainty factor [±]	Critical factor
Reactor internals	0.5	Material composition (e.g. concentration of Co)
Reactor pressure vessels	2	Reactor core operation (high- and low-leakage operation)
Biological shield and isolation	3	Reactor core operation and the composition of concrete
Near-core process systems	2	Future operational conditions as regards chemistry and corrosion
Turbine systems	10	Future moisture content in the vapour and fuel damage
Waste systems	10	Future fuel damage and how well ion exchange resins can be removed
Contaminated concrete	> 10	Leakage (e.g. behind pool plates), fuel damage situation, surface treatment of concrete and the possibility for cleaning

D5.5 Correlation factors

Correlations are made with measured activities for Co-60, Cs-137 and Pu-239/Pu-240 and are associated with relatively large uncertainties. The activity for nuclides that according to previous safety assessments provide a high dose contribution is now determined with alternative methods yielding lower uncertainty values (see Sections D2.3.2 and D5.2–D5.4). On the other hand, the correlation for all nuclides besides C-14, transuranics and Sr-90 is used to forecast activity in future waste. When forecasting, the correlation is made with the average activity of Co-60, Cs-137 and Pu-239/Pu-240 for deposited packages of each waste type at the manufacturing date. The correlation factors and their uncertainty are presented in Table D5-4.

The correlation of fission products with the activity of Cs-137 includes additional uncertainties, as cesium is soluble in water and has a considerably lower boiling point than most of the other fission products whose activity is determined in SFR (Haynes 2010, 4-121, 4-122). An exception is, however, I-129 in reactor water which has an even lower boiling point and thereby presumably can be correlated in a representative way with cesium in reactor water. Because of their low boiling points, these two fission products occur in their gas phases inside the fuel cladding. For fuel damage that does not result in the dissolution of uranium, gases escape immediately and dissolve in the reactor water while the solid substances largely remain inside the cladding. The reactor water concentration is thus higher in relation to other fission products for Cs-137 and I-129 and thereby also the concentration in the ion exchange resins for SFR from this system. The original ratio between each fission product and Cs-137 in the fuel (based on which the correlation is determined) does not represent the ratio in the reactor water, which means that the activity is overestimated and the correlated values for fission products can be considered the theoretical maximum. Since iodine occurs to a greater extent in the form of volatile compounds, however, a larger fraction of I-129 passes into condensate clean-up than Cs-137. Activity determination of I-129 by correlation with Cs-137 for waste from this system thereby probably leads to underestimation. The correlation of Cs-135 with Cs-137 is also a poor representation of the ratio in condensate clean-up, which has to do with the half-life of the parent nuclides and whether they have time to enter the secondary side before decay. However, it should be pointed out that the activity for both I-129 and Cs-135 in existing operational waste is determined by calculations (see Section D2.3.2 and D5.4) and correlation with Cs-137 is only made for forecasted waste.

Table D5-2. Maximum factor for decommissioning waste per nuclide and waste vault.

Nuclide	Maximum factor Silo	Maximum factor BRT	Maximum factor BMA	Maximum factor BLA
H-3	0.00	0.00	7.19	3.17
Be-10	0.00	0.00	4.71	3.00
C-14	9.73	2.00	3.99	6.31
Cl-36	9.39	2.00	4.27	2.72
Ca-41	0.00	0.00	4.52	2.91
Fe-55	5.01	2.00	2.53	6.99
Co-60	5.16	2.00	2.20	7.81
Ni-59	4.18	2.00	2.32	7.37
Ni-63	4.20	2.00	2.31	7.42
Se-79	9.63	0.00	9.25	9.69
Sr-90	4.42	2.00	2.52	9.05
Zr-93	4.87	2.00	2.45	7.04
Nb-93m	4.65	2.00	2.10	7.65
Nb-94	4.59	2.00	2.17	7.38
Mo-93	4.68	2.00	2.17	6.23
Tc-99	8.07	2.00	3.21	9.17
Ag-108m	4.73	2.00	2.39	4.60
Pd-107	9.79	0.00	2.00	8.16
Cd-113m	9.78	0.00	3.30	9.06
In-115	0.00	0.00	3.00	0.00
Sn-126	9.50	2.00	2.48	8.19
Sb-125	5.02	2.00	2.42	7.00
I-129	9.68	0.00	10.00	9.05
Cs-134	9.88	0.00	9.29	8.00
Cs-135	9.69	0.00	11.00	8.20
Cs-137	9.71	0.00	10.68	8.44
Ba-133	9.82	0.00	4.51	3.00
Pm-147	5.61	2.00	2.03	8.42
Sm-151	4.42	2.00	4.63	3.04
Eu-152	4.76	2.00	4.14	3.00
Eu-154	4.71	2.00	3.41	3.41
Eu-155	5.03	2.00	3.12	3.34
Ho-166m	4.85	2.00	5.27	3.00
U-232	5.19	2.00	2.01	8.47
U-235	4.33	2.00	2.35	7.68
U-236	4.98	2.00	1.98	8.36
Np-237	4.38	2.00	1.98	7.94
Pu-238	4.38	2.00	1.98	7.93
Pu-239	4.31	2.00	2.20	7.80
Pu-240	4.31	2.00	2.10	7.99
Pu-241	4.36	2.00	1.92	8.10
Pu-242	4.33	2.00	1.97	7.99
Am-241	4.37	2.00	2.03	7.96
Am-242m	4.21	2.00	1.96	7.84
Am-243	4.38	2.00	1.98	8.01
Cm-243	4.35	2.00	1.97	7.96
Cm-244	4.46	2.00	1.98	8.08
Cm-245	4.47	2.00	2.04	7.87
Cm-246	4.49	2.00	2.03	8.03

Table D5-3. Minimum factor for decommissioning waste per nuclide and waste vault.

Nuclide	Minimum factor Silo	Minimum factor BRT	Minimum factor BMA	Minimum factor BLA
H-3	0.00	0.00	0.19	0.33
Be-10	0.00	0.00	0.28	0.33
C-14	0.11	0.50	0.31	0.22
Cl-36	0.11	0.50	0.30	0.30
Ca-41	0.00	0.00	0.29	0.32
Fe-55	0.22	0.50	0.47	0.18
Co-60	0.23	0.50	0.53	0.16
Ni-59	0.28	0.50	0.48	0.18
Ni-63	0.28	0.50	0.48	0.18
Se-79	0.11	0.00	0.12	0.11
Sr-90	0.27	0.50	0.55	0.12
Zr-93	0.22	0.50	0.46	0.17
Nb-93m	0.25	0.50	0.53	0.17
Nb-94	0.24	0.50	0.51	0.17
Mo-93	0.22	0.50	0.51	0.20
Tc-99	0.15	0.50	0.46	0.12
Ag-108m	0.22	0.50	0.48	0.25
Pd-107	0.11	0.00	0.50	0.16
Cd-113m	0.11	0.00	0.32	0.13
In-115	0.00	0.00	0.33	0.00
Sn-126	0.11	0.50	0.46	0.15
Sb-125	0.22	0.50	0.49	0.18
I-129	0.11	0.00	0.10	0.12
Cs-134	0.10	0.00	0.12	0.15
Cs-135	0.11	0.00	0.10	0.14
Cs-137	0.11	0.00	0.10	0.13
Ba-133	0.10	0.00	0.28	0.33
Pm-147	0.24	0.50	0.56	0.15
Sm-151	0.26	0.50	0.31	0.33
Eu-152	0.27	0.50	0.30	0.33
Eu-154	0.26	0.50	0.39	0.32
Eu-155	0.26	0.50	0.38	0.32
Ho-166m	0.26	0.50	0.26	0.33
U-232	0.24	0.50	0.55	0.14
U-235	0.26	0.50	0.53	0.17
U-236	0.25	0.50	0.56	0.15
Np-237	0.26	0.50	0.55	0.16
Pu-238	0.26	0.50	0.55	0.16
Pu-239	0.27	0.50	0.54	0.17
Pu-240	0.27	0.50	0.55	0.16
Pu-241	0.27	0.50	0.57	0.16
Pu-242	0.27	0.50	0.56	0.16
Am-241	0.26	0.50	0.55	0.16
Am-242m	0.28	0.50	0.56	0.16
Am-243	0.26	0.50	0.55	0.16
Cm-243	0.27	0.50	0.56	0.16
Cm-244	0.27	0.50	0.55	0.16
Cm-245	0.25	0.50	0.53	0.16
Cm-246	0.25	0.50	0.54	0.16

Table D5-4. Uncertainty of correlation with key nuclides.

Nuclide	Uncertainty factor	Nuclide	Uncertainty factor
H-3[1]	50	Pm-147[3]	2
Be-10[1]	50	Sm-151[3]	2
Cl-36[2]	6	Eu-152[3]	2
Fe-55[1]	5	Eu-154[3]	2
Ni-59[2]	3	Eu-155[3]	2
Ni-63[2]	3	Ho-166m[3]	2
Se-79[1]	50	U-232[3]	2
Sr-90[1]	5	U-234[3]	2
Zr-93[1]	50	U-235[3]	2
Nb-93m[1]	20	U-236[3]	2
Mo-93[2]	20	Np-237[3]	2
Nb-94[1]	5	Pu-238[3]	2
Tc-99[2]	5	U-238[3]	2
Pd-107[1]	40	Pu-241[3]	2
Ag-108m[1]	50	Am-241[1]	10
Cd-113m[1]	50	Am-242m[3]	2
Sb-125[1]	10	Pu-242[3]	2
Sn-126[1]	40	Am-243[3]	2
I-129[2]	5	Cm-243[3]	2
Ba-133[3]	2	Cm-244[3]	2
Cs-134[3]	1,2	Cm-245[3]	2
Cs-135[2]	3	Cm-246[3]	2

[1] Cronstrand 2005.

[2] "Assessing uncertainty to correlation factors for ^{14}C, ^{36}Cl, ^{59}Ni, ^{63}Ni, ^{93}Mo, ^{99}Tc, ^{129}I, and ^{135}Cs in operational waste for SFR 1" (SKBdoc 1393443).

[3] (Forsyth 1997).

Waste from SNAB and Svafo is associated with larger uncertainties as its content varies more than the waste from the nuclear power plants. This is due to the fact that SNAB, in addition to waste from the nuclear power industry, manages radioactive waste that is generated in medical care and research, while Svafo manages legacy nuclear waste. Since sampling cannot be made in the same way to determine the activity of certain nuclides, hard-to-measure nuclides are to a greater extent correlated with the activity of Co-60 and Cs-137 from the gamma measurements. Furthermore, the general correlation factors used are less adapted to their waste as they primarily have been developed to represent the waste from nuclear power plants. Although SNAB manages nuclear power plant waste, after-treatment in the form of melting and incineration can change the nuclide composition in the waste.

D5.6 System decontamination

System decontamination carried out during operation of the nuclear power reactors leads to the radioactivity on system surfaces being released and accumulated in ion exchange resins deposited in SFR. Water sampling during ongoing system decontamination is, as a rule, not included in the regular sampling that serves as a basis for determination of non-package-specific activity data. The radionuclides whose activity is determined on the basis of this water sampling include Ni-59, Ni-63, Sr-90, Mo-93, Tc-99, I-129, Cs-135 and actinides, which thereby are probably underestimated. Of these nuclides, it is mainly the actinides that provide a significant activity contribution in relation to what has been measured during operation, which is based on experience from final decontamination at the reactors at BKAB. In the activity inventory for decommissioning waste, the activity of actinides and insoluble fission products is estimated based on the accumulated amount of dissolved uranium for each reactor up to final shutdown, as well as the experience from system decontamination at BKAB that about 10% of the activity that occurs due to dissolution of uranium remains on system surfaces at shutdown. In this estimate, no adjustment is made for the activity released at system decontaminations that have been carried out during operation, which means that the total activity exists in the decommissioning inventory.

D5.7 Production date vs measurement date

The measured activity for Co-60 should be calculated back to the production date of the waste, since the correlations made refer to the ratio between each nuclide and Co-60 at the time of origin of the activity. If the activity for Co-60 has too much time to decay before gamma measurements are conducted (Co-60 has a half-life of 5.3 years), the correlated activities are underestimated. If the production date (when the activity arises) differs from the manufacturing date (when the package is manufactured), this can also constitute an uncertainty, since only manufacturing date and measurement date are stated. In the cases where the manufacturing date differs from the measurement date, this is taken into account for activity calculations.

The nuclear power plants and Clab were asked how much operational waste each facility manufactures that is older than five years at the measurement date, which would lead to an error in the determination of activity by a factor of 2. Apart from some exceptional cases, this does not seem to be frequent. At Clab, granular ion exchange resins have been stored in tanks since 1986 in the waste type C.02. All data on backwash and activity uptake are, however, registered for each filled tank. Waste types C.23 and C.24, which consist of trash and scrap metal, are fabricated for a longer time and are not always gamma-measured immediately. For this waste, however, the date of grouting and the measurement date are registered when filling is completed. The time between the first and last filling does not normally exceed five years. RAB also fabricates the waste types R.23 and R.24 with a periodicity of about four years. It should be noted that this waste, containing trash and scrap metal, does not normally contribute with particularly high activities. At FKA, evaporator concentrates can have cycles exceeding five years. Single large miscellaneous components can also be kept longer than five years, like after treatment and return by SNAB that are deposited in SFR as F.12 or F.24. At OKG, granular ion exchange resins have been interim-stored in tanks at O3 since the start of operation in 1984, which entails a problem with regard to the date for activity origin. The amount of waste is calculated to include about 300 concrete moulds O.02. On BKAB, waste containing ion exchange resins may be older than five years before gamma measurements are made.

D5.8 Distribution models

Radioactivity that is not associated with specific packages, but obtained as a lump sum from each waste supplier and production year, is distributed by SKB according to different models to the packages that have been deposited in SFR. Nuclide activities in operational waste that are obtained in this way include C-14, Cl-36, Ni-59, Ni-63, Sr-90, Mo-93, Tc-99, I-129, Cs-135 and actinides. In addition to the uncertainties previously mentioned for these nuclides, they are also affected by the uncertainty in the distribution models used to distribute the calculated activity to the different waste vaults in a representative way. This may imply that the activity of a certain nuclide is underestimated in one waste vault and overestimated in another.

In forecasts for future waste, activities that stem from non-package-specific data are overestimated, except for C-14. This is due to the fact that the measured activities for these nuclides are distributed to the packages that are already deposited in SFR, even though some packages are still interim-stored at the nuclear power plants. The activity for interim-stored waste will, in addition, be determined once more by correlation, which leads to a double calculation for these waste packages. The proportion of waste packages that are interim-stored at the nuclear power plants at present is about 25% of all waste disposed of in SFR.

The activity of actinides Sr-90 and Ni-59/Ni-63 is distributed according to the proportion of Co-60 in each deposited package containing ion exchange resins. A study of how the actinide activity up to year 1998 has been distributed between the waste vaults in SFR has been carried out based on the origin of the activity with respect to waste stream and waste classes (Ingemansson 2000b). The amounts estimated in the study are in good agreement with the reported amounts within a margin of error on ±10% (1σ). The same reasoning can be assumed to be valid for Sr-90, Ni-59/Ni-63, Mo-93 and Tc-99, which are distributed in the same way as the actinides, as these nuclides follow the activity of Co-60 chemically. The activity for I-129 and Cs-135 is distributed according to the proportion of Cs-137 instead. For the same reasons as described in Section D5.5, this is a poor representation of the ratio between these nuclides in the different systems and the uncertainty should thus be higher than for the other distributed activities.

For C-14, which chemically does not follow the activity of Co-60, it is better to distribute the calculated activity according to the amount of ion exchange resins in the waste. For BKAB, FKA and OKG, the activity of C-14 can mainly be found in condensate clean-up and to some extent in other systems such as reactor clean-up, fuel pool clean-up and system drainage. Ion exchange resins from condensate clean-up at FKA are only deposited in BMA, while ion exchange resins from condensate clean-up at BKAB and OKG are exclusively deposited in BTF. Thereby, no additional uncertainty is added as a result of the distribution in these cases. Ion exchange resins from other systems, however, may be mixed and end up in different waste types that in turn are deposited in different waste vaults, which means that the estimated activity of C-14 for these waste packages can deviate from the actual activity. However, it should be pointed out that 78–97% of the activity of C-14 can be found in condensate clean-up. For RAB the activity of C-14 will mainly be deposited in the Silo, whereas the activity for Clab/Clink will exclusively end up there. Therefore, almost no additional uncertainty is added as a result of the distribution in these cases. The uncertainty in the distribution of C-14 activity should thus be limited. How the activity is distributed between the different clean-up systems is reactor-specific and has been determined on the basis of measured amounts of C-14 in ion exchange resins from each system in "C-14 accumulated in ion exchange resins in Swedish nuclear power plants" (SKBdoc 1339709), "Measurement of ^{14}C in process water from CLAB and estimation of the accumulated amount in spent ion exchange resins" (SKBdoc 1393796) and "Uppskatting av aktivitet C-14 och Cl-36 i driftavfall från forskningsreaktorn R2 i Studsvik" (SKBdoc 1393446).

Cl-36 also does not follow the activity for Co-60 chemically, and is distributed after the amount of ion exchange resins in the waste instead. In contrast to C-14, however, the distribution of the activity of Cl-36 is made to a greater extent on different waste types and to different repository parts, which means that the uncertainty as a result of the distribution is considerably larger. This only includes waste from BKAB, FKA and OKG. Waste of the type ion exchange resins from other waste suppliers containing activity for Cl-36 is without exception placed in the Silo and thus does not contribute any additional uncertainty due to the distribution model.

D5.9 S.14

At the end of 2012, Svafo completed a project with the purpose of clarifying the content of about 7, 500 historical waste drums that are interim-stored on site. These drums will be deposited in SFL and are thus not included in the present inventory report. In the project, all drums were X-rayed and gamma-measured, on the basis of which it was noted that their content did not completely correspond to what was claimed in the documentation. Among other things, drums with liquid, suspected mercury and suspected safeguard-material were identified.

During the years 1995–2001, drums of a similar type from Svafo were deposited in BLA as waste type S.14, so SKB has reason to suspect that there may also be errors in the drums deposited in BLA. It concerns a total of 75 deposited half-height containers, containing a total of 2,844 drums.

SKB has concluded that more information is needed to make a correct assessment of the long-term safety. As it is not currently clear if S.14 can completely or partially be disposed of in BLA, the waste is included in the present inventory. Both activity and material content is, however, associated with large uncertainties.

Prior to the beginning of a possible retrieval, SKB has to present how this work will be carried out to SSM, taking into account the protection of personnel as well as further handling of the waste and reasons for the chosen strategy.

Appendix E

Description of the waste types

Contents

E1 Introduction

Each package deposited in SFR belongs to a waste type, which follows a waste type description that describes the handling process from manufacturing to disposal of the waste. This appendix presents information on the waste types assumed to be deposited in SFR, and the overall acceptance criteria that apply for disposal of waste in SFR.

E1.1 Waste in Silo

The waste vault Silo is located in the existing facility at SFR.

Mainly ion exchange resins but also some trash and scrap metal and a small amount of sludge are deposited in the Silo.

E1.1.1 Acceptance criteria

Waste packages deposited in SFR must conform to the general, radiological, chemical, physical and mechanical requirements, which are defined in the acceptance criteria found in the Waste handling manual for low and intermediate-level waste. The following is a general description of the acceptance criteria for the Silo.

- The waste packages shall have packaging of the type steel or concrete moulds with geometry 1.2×1.2×1.2 m or drums with a diameter of 0.59 m and a height of 0.88 m, placed on drum trays with base area 1.2×1.2 m or in drum boxes with the geometry 1.2×1.2×1.2 m.
- A mould or a drum box including waste may weigh a maximum of 5 tonnes. Four drums including the drum tray may weigh a maximum of 2 tonnes, and a maximum of 500 kg per drum.
- Each waste package shall have an individual identity in the form of a unique and clear label.
- The radionuclide content of gamma-emitting nuclides in the waste package must be known.
- The maximum permissible surface dose rate on waste packages is 500 mSv/h.
- Surface contamination on the waste package must be less than 40 kBq/m^2 for gamma and beta-emitting nuclides and 4 kBq/m^2 for alpha-emitting nuclides.
- For waste packages of the type cement or bitumen-solidified ion exchange resins, the integrated radiation dose must not exceed 10^6 Gy.
- Waste content shall be distributed so that the radiological properties established from a radiation protection perspective are not jeopardized.
- The chemical composition and structure of the waste form and the packaging shall comply with given specifications.
- Waste packages containing bitumen or cement-solidified waste should be homogeneous enough that the physical and chemical properties established according to radiation safety and long-term safety are not jeopardized.
- The waste should be solidified in cement, bitumen or concrete inside the packaging.
- The waste package must withstand storage at temperatures between 0 and 30°C and storage down to –20°C during short periods.
- The waste package's contents must not be liquid. The waste must not contain free or confined liquid.
- The waste package and its contents must not give rise to gas evolution at such a rate or amount that safety during operation or long-term safety be jeopardized.
- Combustible waste must not give rise to spontaneous combustion. Waste packages must withstand a short-lived fire without unacceptable dispersion of radionuclides. Explosive substances must not occur in the waste.
- The waste package's content of chemical species that can form mobile complexes shall be known both in kind and amount and as far as possible be avoided.

- Small amounts of lead, asbestos and epoxy may be deposited after assessment. Disposal of other environmentally hazardous substances are in principle prohibited.
- Moulds must withstand being stacked, 42 packages on top of each other, with grouting.
- The waste packages shall withstand a fall of 9 m without unacceptable dispersion of radioactivity.
- Swelling shall be reviewed per waste type.
- Waste packaging must be sufficiently corrosion resistant to endure all processing steps up to deposition.

E1.2 Waste in BRT

BRT is a waste vault for intact reactor pressure vessels planned to be built in the extension of SFR. The waste vault will accommodate the BWR tanks from BKAB, FKA, OKG and RAB.

All RPVs in Sweden of BWR type are delivered by ASEA-ATOM. The RPVs resemble each other according to the following:

- F1 and F2.
- B1, B2 and O2.
- O1.
- R1.
- O3 and F3.

The waste consists of iron/steel. No packaging will be used and no treatment of the waste is intended. The exception is possibly some kind of surface treatment to prevent spreading of surface contamination.

E1.2.1 Acceptance criteria

Waste packages deposited in SFR must meet certain requirements, which are defined in the acceptance criteria. Acceptance criteria for BRT waste are under development.

E1.3 Waste in BMA

BMA consists of waste vaults for intermediate-level waste. 1BMA is the existing waste vault in SFR. XBMA is additional waste vault(s) that will be built in the extension of SFR.

Mainly ion exchange resins and waste in the form of trash and scrap metal and concrete, as well as small amounts of sludge and evaporator concentrates are deposited in BMA.

E1.3.1 Acceptance criteria

Waste packages deposited in SFR must conform to the general, radiological, chemical, physical and mechanical requirements, which are defined in the acceptance criteria found in the Waste handling manual for low and intermediate-level waste. The following is a general description of the acceptance criteria for 1BMA. Acceptance criteria for XBMA waste are under development.

- The waste packages shall have packaging of the type steel or concrete moulds with geometry 1.2×1.2×1.2 m or drums with a diameter of 0.59 m and height 0.88 m placed on drum trays with base area 1.2×1.2 m or in drum boxes with geometry 1.2×1.2×1.2 m.
- A mould or a drum box including waste may weigh a maximum of 5 tonnes. Four drums including drum trays may weigh a maximum of 2 tonnes, and a maximum of 500 kg per drum.
- Each waste package shall have an individual identity in the form of a unique and clear label.
- The radionuclide content of gamma-emitting nuclides in the waste package must be known.

- The maximum permissible surface dose rate on the waste package is 100 mSv/h (at most 20% of all packages > 30 mSv/h).
- Surface contamination on the waste package must be less than 40 kBq/m^2 for gamma and beta-emitting nuclides and 4 kBq/m^2 for alpha-emitting nuclides.
- For waste packages of the type cement or bitumen-solidified ion exchange resins, the integrated radiation dose must not exceed 10^6 Gy.
- Waste contents shall be distributed so that the radiological properties established from a radiation protection perspective are not jeopardized.
- The chemical composition and structure of the waste form and the waste packaging shall comply with given specifications.
- Waste packages containing bitumen or cement-solidified waste should be homogeneous enough that the physical and chemical properties established according to radiation safety and long-term safety are not jeopardized.
- The waste should be solidified in cement, bitumen or concrete inside the packaging.
- The waste package must withstand storage at temperatures between 0 and 30°C and storage down to –20°C during short periods.
- The waste package's contents must not be liquid. The waste must not contain free or confined liquid.
- The waste package and its contents must not give rise to gas evolution in such a rate or amount that safety during operation or long-term safety be jeopardized.
- Combustible waste must not give rise to spontaneous combustion. Waste packages must withstand a short-lived fire without unacceptable dispersion of radionuclides. Explosive substances must not occur in the waste.
- The waste package's content of chemical species that can form mobile complexes shall be known both in kind and amount and be avoided as far as possible.
- Small amounts of lead, asbestos and epoxy may be deposited after assessment. Other environmentally hazardous substances are in principle prohibited.
- Packages must withstand being stacked, 6 moulds or 8 drums on top of each other.
- The waste packages shall withstand a fall of 9 m without unacceptable dispersion of radioactivity.
- Swelling shall be reviewed per waste type.
- Waste packaging must be sufficiently resistant to corrosion to endure all processing steps up to deposition.

E1.4 Waste in BTF

BTF consists of 1BTF and 2BTF which are waste vaults for concrete tanks in the existing SFR.

Ash drums, low-level ion exchange resins and filter aids as well as a small amount of trash and scrap metal are deposited in 1BTF. Low-level ion exchange resins and filter aids are deposited in 2BTF. In both waste vaults, however, larger metal components such as steam separators or reactor pressure vessel lids have been permitted for deposition.

The waste in 1BTF is normally packaged in steel drums and concrete tanks, but concrete moulds also occur. A few waste packages of waste types that are normally deposited in 1BMA (O.01, R.01, R.10 and R.23) are deposited in 1BTF. These packages consist of concrete moulds with low activity content, and have been deposited in 1BTF to build support walls for ash drums. Concrete tanks are normally deposited in 2BTF.

E1.4.1 Acceptance criteria

Waste packages deposited in SFR must conform to the general, radiological, chemical, physical and mechanical requirements, which are defined in the acceptance criteria found in the Waste handling manual for low and intermediate-level waste. The following is a general description of the acceptance criteria for BTF. Miscellaneous waste can deviate from the acceptance criteria and is managed from case to case.

- The waste packages shall have packaging of the type concrete tanks with the outer dimensions LxBxH = 3.3×1.3×2.3 m, or drums with a diameter of 0.59 m and the height 0.88 m.
- Concrete tanks may weigh a maximum of 20 tonnes. Drums may weigh a maximum of 400 kg.
- Each waste package shall have an individual identity in the form of a unique and clear label.
- The radionuclide content of gamma-emitting nuclides in the waste package must be known.
- Maximum permissible surface dose rate on the waste package is 8 mSv/h.
- Surface contamination on the waste package must be less than 40 kBq/m^2 for gamma and beta-emitting nuclides and 4 kBq/m^2 for alpha-emitting nuclides.
- Waste contents shall be distributed so that the radiological properties established from a radiation protection perspective are not jeopardized.
- The chemical composition and structure of the waste form and the waste packaging shall comply with given specifications.
- The waste shall be cement- solidified inside the packaging or the packaging shall consist of a concrete tank.
- The waste package must withstand storage at temperatures between 0 and 30°C and storage down to –20°C during short periods.
- The waste package's content must not be liquid. The waste must not contain free or confined liquid.
- The waste package and its content must not give rise to gas evolution in such a rate or amount that safety during operation or long-term safety be jeopardized.
- Combustible waste must not give rise to spontaneous combustion. Waste packages must withstand a short-lived fire without unacceptable dispersion of radionuclides occur. Explosive substances must not occur in the waste.
- The waste package's content of chemical species that can form mobile complexes shall be known both in kind and amount and as far as possible be avoided.
- Small amounts of lead, asbestos and epoxy may be deposited after assessment. Other environmentally hazardous substances are in principle prohibited.
- Concrete tanks must withstand being stacked, 2 tanks on top of each other, with an overload of 30 kN and drums must withstand being stacked lying, 10 drums on top of each other. Waste packages shall withstand normal handling with fork-lift truck.
- Concrete tanks shall withstand tipping over, as well as a fall of 2.5 m without unacceptable dispersion of radioactivity. Drums shall withstand a fall of 5 m.
- Swelling shall be reviewed per waste type.
- Waste packaging must be sufficiently resistant to corrosion to endure all processing steps up to deposition.

E1.5 Waste in BLA

BLA consists of waste vaults for low-level waste. 1BLA is the existing waste vault in SFR. XBLA consists of additional waste vaults that will be built in the extension of SFR.

Mainly, waste in the form of trash and scrap metal, is deposited in BLA. In addition, a small number of ion exchange resins are deposited.

E1.5.1 Acceptance criteria

Waste packages deposited in SFR must conform to the general, radiological, chemical, physical and mechanical requirements, which are defined in the acceptance criteria found in the Waste handling manual for low and intermediate-level waste. The following is a general description of the acceptance criteria for 1BLA. Acceptance criteria for XBLA waste are under development.

- The waste package should have packaging of the type ISO container with dimensions 10 or 20 foot, full or half height.
- 10-foot containers may weigh a maximum of 10 tonnes. 20-foot containers may weigh a maximum of 20 tonnes. This applies regardless of height.
- Each waste package shall have an individual identity in the form of a unique and clear label.
- The radionuclide content of gamma-emitting nuclides in the waste package must be known.
- The maximum permissible surface dose rate on the waste package is 2 mSv/h, and 0.1 mSv/h at a distance of 2 metres.
- Surface contamination on the waste package must be less than 40 kBq/m^2 for gamma and beta-emitting nuclides and 4 kBq/m^2 for alpha-emitting nuclides.
- For waste packages of the type cement or bitumen-solidified ion exchange resins, the integrated radiation dose must not exceed 10^6 Gy.
- Waste contents shall be distributed so that the radiological properties established from a radiation protection perspective are not jeopardized.
- The chemical composition and structure of the waste form and the waste packaging shall comply with given specifications.
- Waste packages of the type container should show impermeability to washing and precipitation.
- The waste package must withstand storage at temperatures between 0 and 30°C and storage down to –20°C during short periods.
- The waste package's contents must not be liquid. The waste must not contain free or confined liquid.
- The waste package and its contents must not give rise to gas evolution in such a rate or amount that safety during operation or long-term safety be jeopardized.
- Combustible waste must not give rise to spontaneous combustion. Waste packages must withstand a short-lived fire without unacceptable dispersion of radionuclides. Explosive substances must not occur in the waste.
- The waste package's content of chemical species that can form mobile complexes shall be known both in kind and amount and be avoided as far as possible.
- Small amounts of lead, asbestos and epoxy may be deposited after assessment. Other environmentally hazardous substances are in principle prohibited.
- The containers must withstand being stacked, 3 full height or 6 half-height containers, on top of each other.
- Waste packages shall withstand a fall of 6.5 m without unacceptable dispersion of radioactivity.
- Swelling shall be reviewed per waste type.
- Waste packaging must be sufficiently resistant to corrosion to endure all processing steps up to deposition.

E1.6 Definitions

Waste types for decommissioning waste have been adopted for this report and are denoted by a D (Decommissioning). For further explanation of these waste types, see Appendix A.

Corrosion surface is defined as the area of metal that may be exposed to corrosion. Metal surfaces in contact with bitumen are not included in the definition for corrosion surface.

Void is defined as real voids, i.e. for example pores with air that are formed in embedment/grouting are not included.

Other organic material comprises e.g. air filters, oil, combustible and non-combustible trash.

Other inorganic material includes e.g. blasting sand, fuel samples, electrical cables, brass, and water and oil filters.

E2 B.BWR:D

E2.1 Description of the waste type

The waste type B.BWR:D is a waste type adopted for reactor pressure vessels without internals from BKAB (RPV B1 and B2).

There is no approved waste type description for deposition of this waste type. Data are based on Jönsson (2013) and Farías et al. (2008).

Acceptance criteria for BRT are under development, see Section E1.2.1.

E2.1.1 Waste

The waste consists of surface contaminated and induced steel or steel alloys (C1070/SIS2333).

E2.1.2 Packaging

No waste packaging is used. The RPV is transported and stored intact, without packaging. The RPVs B1 and B2 have a height of 20 m (without lid) and an outside diameter of 5.9 m.

The disposal volume for a reactor pressure vessel is about 860 m^3 based on a cuboid with sides of 6.1 m and a length of 23.2 m, where the dimensions refer to RPV measurements including lid and 0.1 m air around it.

E2.1.3 Treatment

Connections are sealed and radiation shielding is mounted as needed. No other treatment is planned, with the exception of covering with tarpaulin, painting or other surface treatment that can be done to avoid any surface contamination from spreading.

Specified void is based on the inside volume of the reactor pressure vessels.

E2.1.4 Activity determination of radionuclides

The fully treated reactor pressure vessel is measured with respect to surface dose rate. The dominant gamma-emitting nuclide is Co-60.

The surface dose rate may not exceed 2 mSv/h. The reactor pressure vessels are assumed to be free of surface contamination on the outside.

E2.1.5 Production of the waste type

Table E2-1 lists the number of packages for SFR.

The reactor pressure vessels B1 and B2 will be deposited in 2023. The waste vault given is according to the acceptance criteria for the waste type.

Table E2-1. Number of packages of the waste type.

Number of packages	Waste vault	B1	B2
Forecasted	(BRT)	1	1

E2.2 Average package for the waste type

E2.2.1 Material – waste, packaging and matrix

Table E2-2 gives values for an estimated average of the material content in waste type B.BWR:D. The material data refer to one reactor pressure vessel. Besides weights, corrosion surface and thickness for metals and void in the waste package are given. The RPVs are internally plated with a stainless layer of at least 3 mm.

Table E2-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	B1	B2
Iron/steel [kg]	Waste	540,000	540,000
Iron/steel surface [m^2]	Waste	828	828
Iron/steel thickness [mm]	Waste	126	126
Void [m^3]	Matrix	522	522

E2.2.2 Radionuclide content

Table E2-3 provides values for a calculated average of the nuclide content in waste type B.BWR:D at the closure of SFR on 2075-12-31. Activity data refer to one reactor pressure vessel.

Table E2-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	B1		B2	
	Induced activity [Bq]	Surface activity [Bq]	Induced activity [Bq]	Surface activity [Bq]
H-3	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Be-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00
C-14 org	0.00E+00	0.00E+00	0.00E+00	0.00E+00
C-14 inorg	0.00E+00	0.00E+00	0.00E+00	0.00E+00
C-14 ind	8.05E+08	0.00E+00	8.31E+08	0.00E+00
Cl-36	6.16E+05	0.00E+00	6.35E+05	0.00E+00
Ca-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Fe-55	1.22E+05	2.74E+04	4.34E+05	7.25E+04
Co-60	8.83E+07	1.94E+08	1.63E+08	3.60E+08
Ni-59	3.96E+09	8.31E+09	4.09E+09	1.73E+10
Ni-63	2.63E+11	6.58E+11	2.79E+11	1.42E+12
Se-79	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-90	0.00E+00	2.83E+07	0.00E+00	1.12E+08
Zr-93	0.00E+00	6.46E+07	0.00E+00	7.05E+07
Nb-93m	1.83E+09	1.26E+10	2.23E+09	3.37E+10
Nb-94	2.63E+07	3.06E+08	2.71E+07	6.65E+08
Mo-93	1.10E+08	6.10E+06	1.13E+08	6.98E+06
Tc-99	1.47E+07	1.01E+06	1.52E+07	1.25E+06
Pd-107	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ag-108m	0.00E+00	3.64E+08	0.00E+00	4.13E+08
Cd-113m	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sn-126	0.00E+00	1.56E+03	0.00E+00	5.61E+03
Sb-125	3.55E+00	1.01E+03	1.24E+01	4.03E+03
I-129	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-134	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-133	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pm-147	0.00E+00	1.33E–01	0.00E+00	1.75E+00
Sm-151	0.00E+00	5.18E+05	0.00E+00	1.92E+06
Eu-152	0.00E+00	2.65E+02	0.00E+00	1.17E+03
Eu-154	0.00E+00	1.49E+04	0.00E+00	7.50E+04
Eu-155	0.00E+00	3.33E+01	0.00E+00	2.33E+02
Ho-166m	0.00E+00	9.64E+00	0.00E+00	3.48E+01
U-232	0.00E+00	1.10E+01	0.00E+00	4.10E+01
U-235	0.00E+00	5.04E–02	0.00E+00	1.82E–01
U-236	0.00E+00	8.44E+02	0.00E+00	3.04E+03
Np-237	0.00E+00	8.96E+02	0.00E+00	3.19E+03
Pu-238	0.00E+00	3.68E+06	0.00E+00	1.36E+07
Pu-239	0.00E+00	8.39E+05	0.00E+00	3.02E+06
Pu-240	0.00E+00	1.45E+06	0.00E+00	5.24E+06
Pu-241	0.00E+00	4.33E+06	0.00E+00	1.89E+07
Pu-242	0.00E+00	5.95E+03	0.00E+00	2.14E+04
Am-241	0.00E+00	5.54E+06	0.00E+00	1.88E+07
Am-242m	0.00E+00	1.53E+04	0.00E+00	5.61E+04
Am-243	0.00E+00	6.55E+04	0.00E+00	2.36E+05
Cm-243	0.00E+00	5.68E+03	0.00E+00	2.23E+04
Cm-244	0.00E+00	3.37E+05	0.00E+00	1.41E+06
Cm-245	0.00E+00	7.21E+02	0.00E+00	2.60E+03
Cm-246	0.00E+00	2.21E+02	0.00E+00	7.95E+02

E3 B.04

E3.1 Description of the waste type

The waste type B.04 consists of steel drums containing cement-solidified low and intermediate-level ion exchange resins from BKAB.

There is an approved waste type description for deposition of this waste type. Data are based on information in the waste type description and Triumf NG v1.0.1.3.

The acceptance criteria for the Silo, described in Section E1.1.1, apply for this waste type.

E3.1.1 Waste

The waste is well defined and consists of both bead and powder resins from reactor water clean-up (system 331), clean-up of reactor and turbine drainage in the waste facility (systems 352 and 482), clean-up system for fuel storage and handling pools (system 324) and system decontamination. The waste type will also contain waste in the form of ion exchange resins from the system decontamination carried out prior to the decommissioning of the nuclear power plant.

E3.1.2 Packaging

The waste is packed in standard 200-litre steel drums. The drum has a diameter of 0.59 m, a height of 0.88 m and a thickness of 1.2 mm. The drum has an empty weight of about 21 kg and is also equipped with a stirrer and a bottom cross, which together weigh 10 kg.

The drums are positioned four by four on a drum tray of carbon steel. The drum tray has the outer dimensions 1.2×1.2 m and weighs about 70 kg.

Maximum permissible weight for drums including waste is 450 kg. The disposal volume for a drum on a drum tray is 0.324 m^3.

E3.1.3 Treatment

The ion exchange resins are homogenised and dewatered before they are mixed with cement in the intended waste packaging. About 197 kg cement and 109 kg water, added and bound, are used per package. The method with a so-called lost stirrer is used, which means that the stirrer remains in the waste package after mixing is completed and thereby acts as reinforcement. The total fill volume of a package is about 90%. After filling, the drum is sealed with a steel lid.

E3.1.4 Activity determination of radionuclides

Before the waste is transported from BKAB to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are determined with measurement and calculation methods according to Appendix D.

The activity content must not exceed 500 GBq. The usually measured surface dose rate is less than 75 mSv/h. The highest permissible surface dose rate is 500 mSv/h. The waste packages are usually free from surface contamination.

E3.1.5 Production of the waste type

Table E3-1 lists the number of packages for SFR.

Deposited packages refer to packages in the existing SFR as of 2012-12-31. The waste type started being produced in 2004 and started being deposited in 2012.

Forecasted packages refer to waste that is interim-stored at the waste supplier. No future production of the waste type is planned. The waste vault indicated for forecasted waste is according to the acceptance criteria for the waste type.

Table E3-1. Number of packages of the waste type.

Number of packages	Waste vault	B.04
Deposited	Silo	96
Forecasted	(Silo)	672

E3.2 Average package for the waste type

E3.2.1 Material – waste, packaging and matrix

Table E3-2 gives values for an estimated average of the material content in waste type B.04. The material data refer to a steel drum including a ¼ drum tray. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E3-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	B.04
Ion exchange resins [kg]	Waste	33
Iron/steel [kg]	Packaging	39
Iron/steel surface [m^2]	Packaging	5.1
Iron/steel thickness [mm]	Packaging (steel drums)	1.2
Iron/steel thickness [mm]	Packaging (drum tray)	5.0
Cement [kg]	Matrix	197
Iron/steel [kg]	Matrix (stirrer)	10
Iron/steel surface [m^2]	Matrix (stirrer)	0.50
Iron/steel thickness [mm]	Matrix (stirrer)	5.0
Void [m^3]	Matrix	0.024

E3.2.2 Radionuclide content

Table E3-3 provides values for a calculated average of the nuclide content in waste type B.04 at the closure of SFR on 2075-12-31. Activity data refer to a steel drum.

Table E3-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	B.04 [Bq]	Nuclide	B.04 [Bq]	Nuclide	B.04 [Bq]
H-3	5.54E+04	Ag-108m	1.04E+06	U-235	3.03E+01
Be-10	1.15E+01	Cd-113m	1.26E+04	U-236	3.02E+01
C-14 org	1.28E+04	Sn-126	1.86E+03	U-238	2.79E+01
C-14 inorg	3.33E+05	Sb-125	8.97E+00	Np-237	1.60E+02
Cl-36	3.93E+04	I-129	1.11E+03	Pu-238	4.17E+04
Fe-55	2.26E+03	Cs-134	6.09E–02	Pu-239	1.24E+04
Co-60	3.86E+06	Cs-135	1.80E+04	Pu-240	1.74E+04
Ni-59	1.91E+07	Cs-137	2.35E+08	Pu-241	7.57E+04
Ni-63	2.33E+09	Ba-133	3.01E+03	Pu-242	4.63E+01
Se-79	1.49E+04	Pm-147	2.18E+02	Am-241	6.02E+04
Sr-90	8.00E+07	Sm-151	6.93E+06	Am-242m	1.13E+02
Zr-93	1.74E+05	Eu-152	1.05E+04	Am-243	4.61E+02
Nb-93m	1.28E+06	Eu-154	2.37E+06	Cm-243	6.91E+01
Nb-94	1.91E+05	Eu-155	2.84E+04	Cm-244	4.02E+03
Mo-93	8.56E+05	Ho-166m	6.72E+05	Cm-245	4.61E+00
Tc-99	1.80E+07	U-232	2.46E–01	Cm-246	1.22E+00
Pd-107	3.73E+03	U-234	1.54E+01		

E4 B.05/B:05:2/B.05:9

E4.1 Description of the waste type

The waste type B.05 consists of steel drums containing bitumen-solidified low and intermediate-level ion exchange resins and evaporator concentrates from BKAB.

In addition to the waste type B.05, there are two variants, B.05:2 and B.05:9. Both the variants refer to steel drums of carbon steel. These were found to be susceptible to corrosion and therefore the production was changed in May 1985 to use steel drums of stainless steel. Waste type B.05:2 consists of steel drums of carbon steel deposited in drum boxes instead of on drum trays. This means that a larger amount of iron/steel is deposited per steel drum in B.05:2 than in B.05 and B.05:9. The only difference between B.05 and B.05:9 is the material of the steel drum.

There are approved waste type descriptions for deposition of this waste type and the two variants. Data are based on information in the waste type descriptions and Triumf NG v1.0.1.3.

The acceptance criteria for BMA, described in Section E1.3.1, apply for this waste type.

E4.1.1 Waste

The waste consists of powder resins and ground bead resins from reactor water clean-up (system 331), the waste facility's clean-up system (system 342) and the clean-up system for fuel storage and handling pools (system 324). The waste also contains concentrates from evaporation of backwash water from the filters in the clean-up system for the fuel storage pools.

E4.1.2 Packaging

The waste is packed in standard 200-litre steel drums. The drum has a diameter of 0.59 m, a height of 0.88 m and a thickness of about 1.2 mm. The drum has an empty weight of about 23 kg. The drums for waste type B.05 are made of stainless steel while the drums for waste type B.05:2 and B.05:9 are made of carbon steel.

The drums are positioned four by four on a drum tray or in a drum box of carbon steel. The drum tray has the outer dimensions 1.2×1.2 m and weighs 65 kg. The drum box has the dimensions 1.2×1.2×1.2 m and weighs 246 kg.

The maximum permissible weight for a drum including waste is 500 kg. The disposal volume for a drum on a drum tray is 0.324 m^3 and for a drum in a drum box 0.432 m^3.

E4.1.3 Treatment

The bead resins are ground in a mill before the solidification process. Evaporator concentrates and powder resins are mixed and reprocessed together with the ground bead resins. The waste material is dried before it is mixed with bitumen in the intended waste packaging. In conjunction with the bituminisation process, an emulsifier is added to ensure that the mixture of bitumen and waste is homogeneous. About 150 kg bitumen is used per package, but the contents can vary between 123–175 kg. The total fill volume is about 86% of the inner volume of a packaging. After completed filling, the drums are allowed to cool down for about 15 h and are then provided with a steel lid. To avoid contamination as far as possible, an outer lid consisting of 0.3 mm steel plate is also rolled on.

E4.1.4 Activity determination of radionuclides

Before the waste is transported from BKAB to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are determined with measurement and calculation methods according to Appendix D.

The activity content must not exceed 0.2 TBq. The highest permissible surface dose rate is 100 mSv/h. The waste packages are usually free from surface contamination.

E4.1.5 Production of the waste type

Table E4-1 lists the number of packages for SFR.

Deposited packages refer to packages in the existing SFR as of 2012-12-31. The waste type B.05 started being produced in 1978 and has been deposited since 1994. The variant B.05:2 started being produced in 1975 and has been deposited since 1990. The variant B.05:9 started being produced in 1976 and has been deposited since 1991.

No future production of the waste type is planned.

Table E4-1. Number of packages of the waste type.

Number of packages	Waste vault	B.05 steel drum on drum tray	B.05:2 steel drum in drum box	B.05:9 steel drum on drum tray
Deposited	1BMA	304	892 (224 boxes)	3,056
Forecasted	–	0	0	0

E4.2 Average package for the waste type

E4.2.1 Material – waste, packaging and matrix

Table E4-2 gives values for an estimated average of the material content in waste type B.05. The material data refer to one steel drum including a ¼ drum tray or ¼ drum box. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E4-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	B.05/B.05:9	B.05:2
Ion exchange resins [kg]	Waste	50	50
Iron/steel [kg]	Packaging	39	85
Iron/steel surface [m^2]	Packaging	3.7	6.6
Iron/steel thickness [mm]	Packaging (steel drum)	1.2	1.2
Iron/steel thickness [mm]	Packaging (drum tray or box)	1.0	1.0
Bitumen [kg]	Matrix	150	150
Other inorganic [kg]	Matrix	0.50	0.50
Void [m^3]	Matrix	0.04	0.04

E4.2.2 Radionuclide content

Table E4-3 provides values for a calculated average of the nuclide content in waste type B.05 at the closure of SFR on 2075-12-31. Activity data refer to one steel drum.

Table E4-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	B.05/B.05:2/B.05:9 [Bq]	Nuclide	B.05/B.05:2/B.05:9 [Bq]
H-3	5.01E+03	Pm-147	2.98E–02
Be-10	5.39E+00	Sm-151	1.07E+06
C-14 org	5.07E+04	Eu-152	4.55E+02
C-14 inorg	1.32E+06	Eu-154	4.35E+04
Cl-36	1.29E+04	Eu-155	8.18E+01
Fe-55	1.62E+00	Ho-166m	3.40E+04
Co-60	5.44E+04	U-232	5.60E–02
Ni-59	8.97E+06	U-234	4.74E+00
Ni-63	3.78E+08	U-235	1.41E+02
Se-79	2.89E+03	U-236	1.21E+02
Sr-90	2.19E+05	U-238	1.04E+02
Zr-93	8.98E+03	Np-237	7.21E+02
Nb-93m	1.69E+05	Pu-238	2.64E+04
Nb-94	8.95E+04	Pu-239	1.97E+03
Mo-93	1.46E+04	Pu-240	2.76E+03
Tc-99	3.33E+04	Pu-241	6.67E+03
Pd-107	7.23E+02	Pu-242	1.42E+01
Ag-108m	4.65E+05	Am-241	2.14E+04
Cd-113m	4.71E+03	Am-242m	3.00E+01
Sn-126	3.61E+02	Am-243	1.41E+02
Sb-125	1.95E-01	Cm-243	1.05E+01
I-129	1.24E+02	Cm-244	1.85E+02
Cs-134	6.88E–05	Cm-245	1.41E+00
Cs-135	1.40E+02	Cm-246	3.74E–01
Cs-137	8.65E+07		
Ba-133	2.05E+02		

E5 B.06

E5.1 Description of the waste type

The waste type B.06 consists of steel drums containing bitumen-solidified low and intermediate-level ion exchange resins and evaporator concentrates from BKAB.

There is an approved waste type description for deposition of this waste type. Data are based on the information in the waste type description and Triumf NG v1.0.1.3.

The acceptance criteria for Silo, described in Section E1.1.1, apply for this waste type.

E5.1.1 Waste

The waste is well defined and consists of both powder resins and ground bead resins from the systems reactor water clean-up (system 331), the waste facility's clean-up system (system 342) and the clean-up system for fuel storage and handling pools (system 324). The waste also contains evaporator concentrates from evaporation of backwash water from the filters in the clean-up system for the fuel storage pools.

E5.1.2 Packaging

The waste is packed in standard 200-litre steel drums. The drum has a diameter of 0.59 m, a height of 0.88 m and a thickness of about 1.2 mm. The drum is made of stainless steel and has an empty weight of about 23 kg.

The drums are positioned four by four on a drum tray of carbon steel. The drum tray has outer dimensions 1.2×1.2 m and weighs about 70 kg. Between the four drums a swelling body is placed to permit volume expansion if the bitumen matrix should start swelling. The swelling body, which consists of a hollow box, is made of steel and has a weight of about 20 kg.

The maximum permissible weight for a drum including waste is 500 kg. The disposal volume for a drum on a drum tray is 0.324 m^3.

E5.1.3 Treatment

The bead resins are ground in a mill before the solidification process. Evaporator concentrates and powder resins are mixed and reprocessed together with the ground bead resins. The waste material is dried before it is mixed with bitumen in the intended waste packaging. In conjunction with the bituminisation process, an emulsifier is added to ensure that the mixture of bitumen and waste is homogeneous. About 150 kg bitumen is used per package, but the contents can vary between 129–165 kg (123–175 kg before 1993). The total fill volume of a package is about 86%. After completed filling, the drums are allowed to cool down for about 15 h and are then provided with a steel lid. To avoid contamination as far as possible, an outer lid consisting of 0.3 mm steel plate is also rolled on.

E5.1.4 Activity determination of radionuclides

Before the waste is transported from BKAB to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The activity content must not exceed 1 TBq. The highest permissible surface dose rate is 500 mSv/h. The waste packages are usually free from surface contamination.

E5.1.5 Production of the waste type

Table E5-1 lists the number of packages for SFR.

Deposited packages refer to packages in the existing SFR as of 2012-12-31. The waste type started being produced in 1980 and has been deposited since 1994.

No future production of the waste type is planned.

Table E5-1. Number of packages of the waste type.

Number of packages	Waste vault	B.06
Deposited	Silo	1,776
Forecasted	–	0

E5.2 Average package for the waste type

E5.2.1 Material – waste, packaging and matrix

Table E5-2 gives values for an estimated average of the material content in waste type B.06. The material data refer to one steel drum including a ¼ drum tray and swelling body. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E5-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	B.06
Ion exchange resins [kg]	Waste	50
Iron/steel [kg]	Packaging	46
Iron/steel surface [m^2]	Packaging	3.5
Iron/steel thickness [mm]	Packaging (steel drum)	1.2
Iron/steel thickness [mm]	Packaging (drum tray)	5.0
Iron/steel thickness [mm]	Packaging (swelling body)	1.0
Bitumen [kg]	Matrix	150
Other inorganic [kg]	Matrix	0.50
Void [m^3]	Matrix	0.035

E5.2.2 Radionuclide content

Table E5-3 provides values for a calculated average of the nuclide content in waste type B.06 at the closure of SFR on 2075-12-31. Activity data refer to one steel drum.

Table E5-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	B.06 [Bq]	Nuclide	B.06 [Bq]	Nuclide	B.06 [Bq]
H-3	4.16E+04	I-129	8.20E+02	Pu-242	9.92E+01
Be-10	3.76E+01	Cs-134	4.06E-04	Am-241	1.50E+05
C-14 org	5.08E+04	Cs-135	9.24E+02	Am-242m	2.13E+02
C-14 inorg	1.32E+06	Cs-137	5.91E+08	Am-243	9.84E+02
Cl-36	1.29E+04	Ba-133	1.76E+03	Cm-243	7.86E+01
Fe-55	1.58E+01	Pm-147	3.14E–01	Cm-244	1.37E+03
Co-60	5.50E+05	Sm-151	7.13E+06	Cm-245	9.85E+00
Ni-59	6.26E+07	Eu-152	3.25E+03	Cm-246	2.61E+00
Ni-63	2.70E+09	Eu-154	3.29E+05		
Se-79	1.91E+04	Eu-155	7.11E+02		
Sr-90	1.59E+06	Ho-166m	2.38E+05		
Zr-93	6.27E+04	U-232	4.03E–01		
Nb-93m	1.35E+06	U-234	3.31E+01		
Nb-94	6.25E+05	U-235	9.84E+02		
Mo-93	1.02E+05	U-236	8.47E+02		
Tc-99	2.33E+05	U-238	7.23E+02		
Pd-107	4.78E+03	Np-237	5.04E+03		
Ag-108m	3.26E+06	Pu-238	1.87E+05		
Cd-113m	3.35E+04	Pu-239	1.38E+04		
Sn-126	2.39E+03	Pu-240	1.92E+04		
Sb-125	1.72E+00	Pu-241	5.41E+04		

E6 B.07/B.07:9

E6.1 Description of the waste type

The waste type B.07 consists of concrete tanks containing dewatered low-level ion exchange resins, filter aids and sludge from BKAB.

There is a variant of the waste type, B.07:9. B.07:9 consists of concrete tanks manufactured before 1983. The variants deviate somewhat in treatment and product composition compared with waste type B.07. The differences are considered to be so small that the same data are used for B.07:9 as for B.07.

The waste type will contain a small amount of decommissioning waste in the form of sludge from the system decontamination.

There are approved waste type descriptions for deposition of this waste type, and both variants. Data are based on information in the waste type descriptions and Triumf NG v1.0.1.3.

The acceptance criteria described for BTF, in Section E1.4.1, apply for this waste type.

E6.1.1 Waste

The waste is well defined and consists of powder resins and filter aids from the condensate clean-up system (system 332) and clean-up systems in the waste facility as well as sludge from decontamination.

E6.1.2 Packaging

The waste is packed in concrete tanks with the outer dimensions 3.3×1.3×2.3 m. The tank is made of reinforced concrete. The wall thickness is 15 cm and the reinforcing bars have a diameter of about 8 mm. The empty weight of the tank is about 11 tonnes, of which the concrete weighs about 10.3 tonnes and the reinforcing bars about 650 kg. The tank is internally lined with a 2 mm thick sack of butyl rubber that weighs about 50 kg. The tank is provided with a bolted reinforced lid with filling hole including a 50 mm steel plate that weighs about 1.7 tonnes.

The maximum permissible weight for a concrete tank including waste is 18 tonnes. The disposal volume is 10 m^3.

E6.1.3 Treatment

The various wastes are mixed and pumped into the concrete tank. The waste is dewatered through a suction pipe that is connected to the filter cartridges, which are located under a bed of sand in the conical bottom of the tank. The cycle of pumping and suction is normally repeated three times to fill the tank with dewatered waste. Some void will always occur at the top of the tank. The total fill volume of a package is about 5.5 m^3 compared with the inner volume of 6 m^3.

E6.1.4 Activity determination of radionuclides

Before the waste is transported from BKAB to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The activity content must not exceed 100 GBq. The highest permissible surface dose rate is 3 mSv/h. The limitation comes from manufacturing. The waste packages are usually free from surface contamination.

E6.1.5 Production of the waste type

Table E6-1 lists the number of packages for SFR.

Deposited packages refer to packages in the existing SFR as of 2012-12-31. The waste type B.07 started being produced in 1985 and has been deposited since 1988. B.07:9 started being produced in 1988 and has been deposited since 1989.

Forecasted packages refer to waste that has not yet been deposited and includes both waste that is interim-stored at the waste supplier and waste that has not yet been produced. Currently, there are 12 packages in interim storage and a production of 2 packages per year up to 2014 is planned. The waste vault indicated for forecasted waste is according to the acceptance criteria for the waste type.

Table E6-1. Number of packages of the waste type.

Number of packages	Waste vault	B.07	B.07:9
Deposited	1BTF	21	3
Deposited	2BTF	174	18
Forecasted	(BTF)	16	0

E6.2 Average package for the waste type

E6.2.1 Material – waste, packaging and matrix

Table E6-2 gives values for an estimated average of the material content in waste type B.07. The material data refer to one concrete tank. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E6-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	B.07/B.07:9
Filter aids [kg]	Waste	170
Ion exchange resins [kg]	Waste	1,400
Sludge [kg]	Waste	60
Other organic [kg]	Waste	66
Concrete [kg]	Packaging	10,350
Iron/steel [kg]	Packaging	2,333
Iron/steel surface [m^2]	Packaging	49
Iron/steel thickness [mm]	Packaging (transport lid)	50
Iron/steel thickness [mm]	Packaging (reinforcing bar)	8.0
Other organic [kg]	Packaging	50
Void [m^3]	Matrix	0.50

E6.2.2 Radionuclide content

Table E6-3 provides values for a calculated average of the nuclide content in waste type B.07 at the closure of SFR on 2075-12-31. Activity data refer to one concrete tank.

Table E6-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	B.07/B.07:9 [Bq]
H-3	3.99E+04
Be-10	1.91E+01
C-14 org	8.23E+06
C-14 inorg	2.14E+08
Cl-36	2.10E+04
Fe-55	3.95E+02
Co-60	1.56E+06
Ni-59	3.18E+07
Ni-63	1.47E+09
Se-79	7.27E+03
Sr-90	9.05E+05
Zr-93	3.18E+04
Nb-93m	1.10E+06
Nb-94	3.17E+05
Mo-93	5.04E+04
Tc-99	1.31E+05
Pd-107	1.82E+03
Ag-108m	1.68E+06
Cd-113m	2.17E+04
Sn-126	9.08E+02
Sb-125	1.01E+01
I-129	7.01E+02
Cs-134	2.09E–02

Nuclide	B.07/B.07:9 [Bq]
Cs-135	1.70E+03
Cs-137	2.84E+08
Ba-133	1.91E+03
Pm-147	9.86E+00
Sm-151	2.92E+06
Eu-152	2.16E+03
Eu-154	3.24E+05
Eu-155	2.04E+03
Ho-166m	1.22E+05
U-232	2.27E–01
U-234	1.68E+01
U-235	5.00E+02
U-236	4.30E+02
U-238	3.67E+02
Np-237	2.56E+03
Pu-238	9.56E+04
Pu-239	6.98E+03
Pu-240	9.75E+03
Pu-241	4.68E+04
Pu-242	5.04E+01
Am-241	7.66E+04
Am-242m	1.14E+02
Am-243	5.00E+02

Nuclide	B.07/B.07:9 [Bq]
Cm-243	5.13E+01
Cm-244	7.41E+02
Cm-245	5.01E+00
Cm-246	1.33E+00

E7 B.12/B.12:1

E7.1 Description of the waste type

The waste type B.12 consists of steel containers containing low-level trash and scrap metal from BKAB.

There is a variant of the waste type, B.12:1. The difference between B.12 and B.12:1 is that high-pressure-compacted drums are packed in the variant, providing a considerably larger amount of iron/steel than that included in B.12.

There are approved waste type descriptions for deposition of both the waste type and its variant. Data are based on information in the waste type descriptions and Triumf NG v1.0.1.3.

The acceptance criteria for BLA, described in Section E1.5.1, apply for this waste type.

E7.1.1 Waste

The waste consists of trash and scrap metal. The trash consists of compacted or non-compacted garbage bags containing e.g. textiles, paper, insulation, small pieces of aluminium, copper and plastics. The scrap metal consists of e.g. fittings, scrapped components, cables, suspensions and insulation. The mixture of waste materials has changed over time, depending on the maintenance work, revisions or other work carried out.

E7.1.2 Packaging

The waste is packed in ISO containers of carbon steel with dimensions 20-foot half height or 20-foot full height.

The half-height container has a length of 6.1 m, a width of 2.5 m and a height of 1.3 m. The thickness of the walls and roof is normally about 1.5 mm. An empty container weighs about 1,900 kg.

The full-height container has a height of 2.6 m but otherwise the same geometry and thickness as the half-height container. It has an empty weight of 2,200 kg. The floor of the full-height container can consist of about 15–30 mm of plywood with load-bearing steel construction. The plywood floor weighs about 310 kg.

Open containers are sealed with a lid.

The maximum permissible weight for a 20-foot container including waste is 20 tonnes. The disposal volume for a half-height container is 20 m^3 and for a full-height container 40 m^3.

E7.1.3 Treatment

Combustible waste that cannot be burned due to high activity content is mixed with compactible waste and compacted and wrapped in plastic. Non-combustible and non-compactible waste is placed in plastic bags, steel drums or drum boxes or is placed directly in the container, without treatment.

As high a fill volume as possible should always be targeted; it can, however, vary widely depending on the nature of the waste. The void in a waste package is assumed to be 7.5 m^3 for both types of packaging, even though there is likely to be more void in the larger packaging.

E7.1.4 Activity determination of radionuclides

Before the waste is transported from BKAB to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The usually measured activity content is about 1 GBq/container. The highest permissible surface dose rate is 2 mSv/h. The waste packages are usually free from surface contamination.

E7.1.5 Production of the waste type

Table E7-1 lists the number of packages for SFR.

Deposited packages refer to packages in the existing SFR as of 2012-12-31. The waste type B.12 started being produced in 1984 and has been deposited since 1990. The variant B.12:1 started being produced in 1984 and has been deposited since 1990.

Forecasted packages refer to waste that has not yet been deposited and includes both waste that is interim-stored at the waste supplier and waste that has not yet been produced. Currently, there are 12 packages in interim storage and a production of 2 packages per year up to 2020 is planned. The waste vault indicated for forecasted waste is according to the acceptance criteria for the waste type.

Table E7-1. Number of packages of the waste type.

Number of packages	Waste vault	B.12 half height	B.12 full height	B.12:1 half height
Deposited	1BLA	171	33	22
Forecasted	(BLA)	0	28	0

E7.2 Average package for the waste type

E7.2.1 Material – waste, packaging and matrix

Table E7-2 gives values for estimated average of the material content in waste type B.12. The material data refer to one container. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E7-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	B.12 half height	B.12 full height	B.12:1 half height
Aluminium/zinc [kg]	Waste	100	100	100
Aluminium/zinc surface [m^2]	Waste	15	15	15
Aluminium/zinc thickness [mm]	Waste	5.0	5.0	5.0
Cellulose [kg]	Waste	500	500	500
Iron/steel [kg]	Waste	4,500	4,500	7,720
Iron/steel surface [m^2]	Waste	229	229	862
Iron/steel thickness [mm]	Waste	5.0	5.0	5.0
Other inorganic [kg]	Waste	400	400	400
Other organic [kg]	Waste	3,000	3,000	3,000
Cellulose [kg]	Packaging	–	310	–
Iron/steel [kg]	Packaging	1,900	2,200	1,900
Iron/steel surface [m^2]	Packaging	105	150	105
Iron/steel thickness [mm]	Packaging	1.5	1.5	1.5
Void [m^3]	Matrix	7.5	7.5	7.5

E7.2.2 Radionuclide content

Table E7-3 provides values for a calculated average of the nuclide content in waste type B.12 at the closure of SFR on 2075-12-31. Activity data refer to one container.

Table E7-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	B.12/B.12:1 half height [Bq]	B.12 full height [Bq]
H-3	6.39E+02	2.62E+03
Be-10	3.30E–01	6.41E–01
C-14 org	0.00E+00	0.00E+00
C-14 inorg	0.00E+00	0.00E+00
Cl-36	3.30E+02	6.41E+02
Fe-55	1.55E+00	1.36E+02
Co-60	1.80E+04	2.25E+05
Ni-59	5.50E+05	1.07E+06
Ni-63	2.53E+07	5.37E+07
Se-79	4.73E+01	1.18E+02
Sr-90	5.56E+04	1.47E+05
Zr-93	5.50E+02	1.07E+03
Nb-93m	1.82E+04	6.20E+04
Nb-94	5.48E+03	1.07E+04
Mo-93	8.97E+02	1.43E+03
Tc-99	2.04E+03	4.97E+03
Pd-107	1.18E+01	2.95E+01
Ag-108m	2.91E+04	5.77E+04
Cd-113m	1.28E+02	6.67E+02
Sn-126	5.91E+00	1.47E+01
Sb-125	1.08E–01	4.41E+00
I-129	2.03E+00	4.34E+01
Cs-134	2.72E–05	5.66E–04
Cs-135	2.29E+00	1.39E+02
Cs-137	1.79E+06	6.24E+06
Ba-133	2.97E+01	1.40E+02
Pm-147	7.65E–03	1.80E+00

Nuclide	B.12/B.12:1 half height [Bq]	B.12 full height [Bq]
Sm-151	1.89E+04	5.25E+04
Eu-152	1.26E+01	6.81E+01
Eu-154	1.66E+03	1.45E+04
Eu-155	6.39E+00	1.75E+02
Ho-166m	2.10E+03	4.11E+03
U-232	9.22E–03	2.04E–02
U-234	6.85E–01	1.33E+00
U-235	1.37E–02	2.67E–02
U-236	4.83E–03	9.09E–03
U-238	2.74E–01	5.33E–01
Np-237	3.41E–01	6.39E–01
Pu-238	1.08E+03	2.33E+03
Pu-239	2.85E+02	5.54E+02
Pu-240	4.01E+02	7.81E+02
Pu-241	1.81E+03	6.63E+03
Pu-242	2.06E+00	4.00E+00
Am-241	3.04E+03	5.93E+03
Am-242m	4.63E+00	9.58E+00
Am-243	2.04E+01	3.97E+01
Cm-243	2.06E+00	5.44E+00
Cm-244	8.84E+01	2.82E+02
Cm-245	2.04E–01	3.97E–01
Cm-246	5.42E–02	1.06E–01

E8 B.12:D/B.12C:D/B.12S:D

E8.1 Description of the waste type

Waste types B.12:D, B.12C:D and B.12S:D are waste types adopted for low-level decommissioning waste in steel containers from BKAB. B.12:D contains scrap metal or secondary waste. B.12C:D contains concrete and B.12S:D contains sand.

There is no approved waste type description for deposition of these waste types. Material quantities and activity have been calculated based on Jönsson (2013), supplemented with assumptions about secondary waste, and material composition for the decommissioning waste, packaging and solidification material.

The acceptance criteria for BLA, described in Section E1.5.1, are assumed to be valid for the waste types.

E8.1.1 Waste

The scrap metal in B.12:D consists mainly of fittings and scrapped components. The secondary waste in B.12:D is assumed to consist of trash and scrap metal similar to waste type R.12 from operational waste. The waste in B.12C:D consists of concrete from the biological shield and contaminated concrete from the controlled area in the facility. The waste in B.12S:D consists of sand from the sand beds in system 341.

E8.1.2 Packaging

The waste is packed in ISO containers of carbon steel with the dimension 20-foot half height. The container has a length of 6.1 m, a width of 2.5 m and a height of 1.3 m. The thickness of the walls and roof is normally about 1.5 mm. An empty container weighs about 1,900 kg. Open containers are sealed with a lid.

The maximum permissible weight for a container including waste is 20 tonnes. The disposal volume is 20 m^3.

E8.1.3 Treatment

The scrap metal, concrete waste and sand are assumed to be packed with a packing degree of 1.1 tonnes/m^3.

The secondary waste is assumed to be treated like waste type R.12 from operational waste.

E8.1.4 Activity determination of radionuclides

Before the waste is transported from BKAB to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The highest permissible surface dose rate is 2 mSv/h. The waste packages are assumed to be free from surface contamination.

E8.1.5 Production of the waste type

Table E8-1 lists the number of packages for SFR.

The waste will be deposited during the years 2023–2029. The waste vault given is according to the acceptance criteria for the waste types.

Table E8-1. Number of packages of the waste type.

Number of packages	Waste vault	B.12:D scrap metal	B.12:D secondary waste	B.12C:D	B.12S:D
Forecasted	(BLA)	267	30	389	190

E8.2 Average package for the waste type

E8.2.1 Material – waste, packaging and matrix

Table E8-2 gives values for an estimated average of the material content in waste types B.12:D, B.12C:D and B.12S:D. The material data refer to one container. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

E8.2.2 Radionuclide content

Table E8-3 provides values for a calculated average of the nuclide content in waste types B.12:D, B.12C:D and B.12S:D at the closure of SFR on 2075-12-31. Activity data refer to one container.

Table E8-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	B.12:D scrap metal	B.12:D secondary waste	B.12C:D	B.12S:D
Aluminium/zinc [kg]	Waste	–	100	–	–
Aluminium/zinc surface [m^2]	Waste	–	15	–	–
Aluminium/zinc thickness [mm]	Waste	–	5.0	–	–
Concrete [kg]	Waste	–	–	16,500	–
Cellulose [kg]	Waste	–	500	–	–
Iron/steel [kg]	Waste	16,500	4,500	–	–
Iron/steel surface [m^2]	Waste	846	229	–	–
Iron/steel thickness [mm]	Waste	5.0	5.0	–	–
Sand [kg]	Waste	–	–	–	16,500
Other inorganic [kg]	Waste	–	400	–	–
Other organic [kg]	Waste	–	3,000	–	–
Iron/steel [kg]	Packaging	1,900	1,900	1,900	1,900
Iron/steel surface [m^2]	Packaging	105	105	105	105
Iron/steel thickness [mm]	Packaging	1.5	1.5	1.5	1.5
Void [m^3]	Matrix	13	7.5	8.1	4.3

Table E8-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	B.12:D [Bq]	B.12C:D [Bq]	B.12S:D [Bq]
H-3	2.49E+01	2.31E+08	0.00E+00
Be-10	0.00E+00	1.74E+00	0.00E+00
C-14 org	8.47E+03	4.85E+03	0.00E+00
C-14 inorg	2.20E+05	1.26E+05	0.00E+00
C-14 ind	2.25E+05	2.27E+06	0.00E+00
Cl-36	3.04E+04	7.55E+04	0.00E+00
Ca-41	8.10E+05	7.47E+06	0.00E+00
Fe-55	6.10E+01	2.44E+01	0.00E+00
Co-60	7.65E+04	4.86E+04	0.00E+00
Ni-59	2.53E+06	4.47E+05	0.00E+00
Ni-63	1.99E+08	3.36E+07	0.00E+00
Se-79	5.28E–02	2.65E+00	0.00E+00
Sr-90	2.20E+04	2.90E+04	3.59E+05
Zr-93	1.34E+04	5.87E+03	0.00E+00
Nb-93m	4.44E+06	1.60E+06	0.00E+00
Nb-94	1.10E+05	6.47E+04	0.00E+00
Mo-93	1.80E+04	5.98E+02	0.00E+00
Tc-99	4.14E+03	2.39E+03	0.00E+00
Pd-107	7.72E–02	2.29E–01	0.00E+00
Ag-108m	6.46E+04	1.36E+06	0.00E+00
Cd-113m	4.98E–03	3.07E+02	0.00E+00
Sn-126	9.13E–01	1.10E+00	0.00E+00
Sb-125	3.90E–01	1.35E–02	0.00E+00
I-129	2.97E+00	6.46E+00	4.22E+00
Cs-134	3.77E–05	2.05E–03	0.00E+00
Cs-135	3.26E+01	7.64E+01	7.20E+02
Cs-137	5.81E+05	1.76E+06	2.96E+06
Ba-133	1.42E–04	1.19E+04	0.00E+00
Pm-147	2.57E–04	1.56E–02	0.00E+00
Sm-151	3.11E+02	1.13E+07	0.00E+00
Eu-152	1.85E–01	2.26E+07	0.00E+00
Eu-154	1.17E+01	1.41E+05	0.00E+00

Nuclide	B.12:D [Bq]	B.12C:D [Bq]	B.12S:D [Bq]
Eu-155	3.53E–02	6.60E+02	0.00E+00
Ho-166m	5.66E–03	1.78E+05	0.00E+00
U-232	4.75E–03	2.21E–03	0.00E+00
U-235	2.39E–05	3.21E–04	0.00E+00
U-236	3.54E–01	1.65E–01	0.00E+00
Np-237	3.66E–01	1.72E–01	0.00E+00
Pu-238	1.72E+03	1.11E+03	0.00E+00
Pu-239	4.34E+02	5.82E+03	0.00E+00
Pu-240	7.11E+02	5.59E+02	0.00E+00
Pu-241	2.13E+03	1.00E+03	0.00E+00
Pu-242	2.50E+00	1.16E+00	0.00E+00
Am-241	1.99E+03	1.01E+03	0.00E+00
Am-242m	6.52E+00	3.03E+00	0.00E+00
Am-243	2.75E+01	1.27E+01	0.00E+00
Cm-243	2.57E+00	1.20E+00	0.00E+00
Cm-244	1.60E+02	7.50E+01	0.00E+00
Cm-245	3.02E–01	1.40E–01	0.00E+00
Cm-246	9.26E–02	4.29E–02	0.00E+00

E9 B.20

E9.1 Description of the waste type

The waste type, B.20 consists of steel containers containing steel drums with low-level bitumen-solidified ion exchange resins from BKAB. The waste type is almost identical to waste type B.05 and B.06, the difference being the amount of radionuclides and dose rate.

There is an approved waste type description for deposition of this waste type. Data are based on information in the waste type description and Triumf NG v1.0.1.3.

The acceptance criteria for BLA, described in Section E1.5.1, apply for this waste type.

E9.1.1 Waste

The waste is well defined and consists of both powder resins and ground bead resins from the systems reactor water clean-up (system 331), the waste facility's clean-up system (system 342), the clean-up system for fuel storage and handling pools (system 324) and the condensate clean-up system (332).

E9.1.2 Packaging

The waste is packed in standard 200-litre steel drums placed in an ISO container with the dimension 20-foot half height. On average, 39.3 drums fit in a container.

The steel drum has a diameter of 0.59 m, a height of 0.88 m and a thickness of about 1.2 mm. The drum is made of carbon steel and has an empty weight of about 23 kg.

The container is made of carbon steel and has a length of 6.1 m, a width of 2.5 m and a height of 1.3 m. The thickness of the walls and roof is normally about 1.5 mm. An empty container weighs about 1,900 kg.

The maximum permissible weight for a steel drum including waste is 500 kg. The maximum permissible weight for a container including waste is 20 tonnes. The disposal volume for a container is 20 m^3.

E9.1.3 Treatment

The waste material is dried, heat treated and mixed with bitumen before filling the steel drums. About 150 kg bitumen is used per container, but the contents can vary between 123–175 kg. The total fill volume in a steel drum is about 86%. After filling, the drums are provided with a lid.

E9.1.4 Activity determination of radionuclides

Before the waste is transported from BKAB to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The activity content must not exceed 5 GBq/drum, i.e. less than 200 GBq/container. The highest permissible surface dose rate is 2 mSv/h. The waste packages are usually free from surface contamination.

E9.1.5 Production of the waste type

Table E9-1 lists the number of packages for SFR.

Deposited packages refer to packages in the existing SFR as of 2012-12-31. The waste type B.20 started being produced in 1974 and has been deposited since 1989.

No future production of the waste type is planned.

Table E9-1. Number of packages of the waste type.

Number of packages	Waste vault	B.20
Deposited	1BLA	12
Forecasted	–	0

E9.2 Average package for the waste type

E9.2.1 Material – waste, packaging and matrix

Table E9-2 gives values for an estimated average of the material content in waste type B.20. The material data refer to one container including drums. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E9-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	B.20
Ion exchange resins [kg]	Waste	1,967
Iron/steel [kg]	Packaging (drums)	905
Iron/steel surface [m^2]	Packaging (drums)	98
Iron/steel thickness [mm]	Packaging (drums)	1.2
Iron/steel [kg]	Packaging (container)	1,900
Iron/steel surface [m^2]	Packaging (container)	105
Iron/steel thickness [mm]	Packaging (container)	1.5
Bitumen [kg]	Matrix	5,900
Other inorganic [kg]	Matrix	20
Void [m^3]	Matrix	7.5

E9.2.2 Radionuclide content

Table E9-3 provides values for a calculated average of the nuclide content in waste type B.20 at the closure of SFR on 2075-12-31. Activity data refer to one container including drums.

Table E9-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	B.20 [Bq]
H-3	4.49E+00
Be-10	6.27E–03
C-14 org	1.99E+06
C-14 inorg	5.19E+07
Cl-36	5.08E+05
Fe-55	2.20E–04
Co-60	3.05E+01
Ni-59	1.04E+04
Ni-63	4.27E+05
Se-79	2.44E–02
Sr-90	2.27E+02
Zr-93	1.05E+01
Nb-93m	1.62E+02
Nb-94	1.04E+02
Mo-93	1.70E+01
Tc-99	3.88E+01
Pd-107	6.10E–03

Nuclide	B.20 [Bq]
Ag-108m	5.38E+02
Cd-113m	2.96E–02
Sn-126	3.05E–03
Sb-125	1.83E–05
I-129	1.05E–03
Cs-134	3.14E–11
Cs-135	1.18E–03
Cs-137	6.37E+02
Ba-133	1.75E–01
Pm-147	3.12E–08
Sm-151	8.60E+00
Eu-152	2.82E–03
Eu-154	2.25E–01
Eu-155	2.72E–04
Ho-166m	3.95E+01
U-232	6.24E–05
U-234	5.52E–03

Nuclide	B.20 [Bq]
U-235	1.64E–01
U-236	1.41E–01
U-238	1.21E–01
Np-237	8.40E–01
Pu-238	2.96E+01
Pu-239	2.29E+00
Pu-240	3.21E+00
Pu-241	6.22E+00
Pu-242	1.66E–02
Am-241	2.49E+01
Am-242m	3.43E–02
Am-243	1.64E–01
Cm-243	1.10E–02
Cm-244	1.78E–01
Cm-245	1.64E–03
Cm-246	4.35E–04

E10 B.23

E10.1 Description of the waste type

The waste type B.23 consists of steel moulds containing concrete-solidified intermediate-level trash and scrap metal from BKAB. The waste type will also contain a small amount of decommissioning waste in the form of filters from system decontamination.

There is no approved waste type description for deposition of this waste type. Data are based on information from BKAB's waste plan and Triumf NG v1.0.1.3.

The acceptance criteria for BMA, described in Section E1.3.1, apply for this waste type.

E10.1.1 Waste

The waste consists of trash and scrap metal. The trash consists of combustible and non-combustible material such as textiles, paper and plastics. The scrap metal consists of e.g. fittings, scrapped components, cables, suspensions and valves. The mixture of waste materials has varied over time due to maintenance work, revisions or other work carried out.

E10.1.2 Packaging

The waste is packed in steel moulds. The mould is a cubic box with dimensions 1.2×1.2×1.2 m. It is made of carbon steel and has a wall thickness of 5 mm and a bottom thickness of 6 mm. The mould also contains press plates. The steel mould including press plates weighs about 660 kg.

The maximum permissible weight for a mould including waste is 5,000 kg. The disposal volume is 1.728 m^3.

E10.1.3 Treatment

The waste is placed directly in the mould. Compactible waste is compacted. Springback and floating to the surface at subsequent solidification is prevented by special press plates or reinforcing bars which are attached to the wall of the mould. When the mould is filled to the maximum, the waste is solidified with concrete. The void is assumed to be about 25% of the inner volume of a packaging. When the concrete has been hardened for two days, a lid with a thickness of at least 10 cm is cast in place.

E10.1.4 Activity determination of radionuclides

Before the waste is transported from BKAB to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The activity content must not exceed 1.0 TBq. The highest permissible surface dose rate is 100 mSv/h. The waste packages are usually free from surface contamination.

E10.1.5 Production of the waste type

Table E10-1 lists the number of packages for SFR.

There are no deposited packages of waste type B.23 in SFR as of 2012-12-31.

Forecasted packages refer to waste that has not yet been deposited and includes both waste that is interim-stored at the waste supplier and waste that has not yet been produced. Currently, there are 25 packages in interim storage and a production of 2 packages per year up to 2016 is planned. The waste vault indicated for forecasted waste is according to the acceptance criteria for the waste type.

Table E10-1. Number of packages of the waste type.

Number of packages	Waste vault	B.23
Deposited	–	0
Forecasted	(BMA)	33

E10.2 Average package for the waste type

E10.2.1 Material – waste, packaging and matrix

Table E10-2 gives values for an estimated average of the material content in waste type B.23. The material data refer to one mould. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E10-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	B.23
Aluminium/zinc [kg]	Waste	4.0
Aluminium/zinc surface [m^2]	Waste	0.60
Aluminium/zinc thickness [mm]	Waste	5.0
Cellulose [kg]	Waste	44
Iron/steel [kg]	Waste	100
Iron/steel surface [m^2]	Waste	5.1
Iron/steel thickness [mm]	Waste	5.0
Other organic [kg]	Waste	100
Concrete [kg]	Packaging (lid)	500
Iron/steel [kg]	Packaging	661
Iron/steel surface [m^2]	Packaging	23
Iron/steel thickness [mm]	Packaging	5.0–6.0
Concrete [kg]	Matrix	1,356
Void [m^3]	Matrix	0.43

E10.2.2 Radionuclide content

Table E10-3 provides values for a calculated average of the nuclide content in waste type B.23 at the closure of SFR on 2075-12-31. Activity data refer to one mould.

Table E10-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	B.23 [Bq]	Nuclide	B.23 [Bq]	Nuclide	B.23 [Bq]	Nuclide	B.23 [Bq]
H-3	7.90E+04	Nb-94	2.59E+05	Pm-147	4.15E+01	Pu-239	1.35E+04
Be-10	1.56E+01	Mo-93	2.57E+04	Sm-151	1.13E+06	Pu-240	1.90E+04
C-14 org	0.00E+00	Tc-99	1.39E+05	Eu-152	1.77E+03	Pu-241	1.95E+05
C-14 inorg	0.00E+00	Pd-107	6.10E+02	Eu-154	4.06E+05	Pu-242	9.72E+01
Cl-36	1.56E+04	Ag-108m	1.41E+06	Eu-155	5.04E+03	Am-241	1.44E+05
Fe-55	3.72E+03	Cd-113m	1.72E+04	Ho-166m	1.00E+05	Am-242m	2.39E+02
Co-60	7.36E+06	Sn-126	3.05E+02	U-232	5.20E–01	Am-243	9.67E+02
Ni-59	2.60E+07	Sb-125	4.37E+02	U-234	3.24E+01	Cm-243	1.47E+02
Ni-63	1.35E+09	I-129	1.83E+03	U-235	6.49E–01	Cm-244	8.01E+03
Se-79	2.44E+03	Cs-134	5.52E–01	U-236	2.18E–01	Cm-245	9.67E+00
Sr-90	3.97E+06	Cs-135	6.10E+03	U-238	1.30E+01	Cm-246	2.57E+00
Zr-93	2.60E+04	Cs-137	1.46E+08	Np-237	1.53E+01		
Nb-93m	1.80E+06	Ba-133	4.31E+03	Pu-238	5.88E+04		

E11 B.23:D

E11.1 Description of the waste type

The waste type B.23:D is a waste type adopted for decommissioning waste from BKAB. It consists of steel moulds containing cement-solidified intermediate-level scrap metal or secondary waste.

There is no approved waste type description for the waste. Waste materials and activity have been calculated based on Jönsson (2013), supplemented with assumptions about secondary waste, and material composition for the decommissioning waste, packaging and solidification material.

The acceptance criteria for BMA, described in Section E1.3.1, are assumed to be valid for the waste type.

E11.1.1 Waste

The scrap metal consists mainly of fittings and scrapped components. The secondary waste is assumed to consist of trash and scrap metal similar to waste type B.23 from operational waste.

E11.1.2 Packaging

The waste is packed in steel moulds. The mould is a cubic box with dimensions 1.2×1.2×1.2 m. It is made of carbon steel and has a wall thickness of 5 mm and a bottom thickness of 6 mm. The mould also contains press plates. The steel mould including press plates weighs about 660 kg.

The maximum permissible weight for a mould including waste is 5,000 kg. The disposal volume is 1.728 m^3.

E11.1.3 Treatment

The scrap metal is assumed to be packed with a packing degree of 1.1 tonnes/m^3. The waste is solidified with concrete. The void is calculated to be about 26% of the inner volume of the packaging, in order to not exceed the maximum weight for a package.

The secondary waste is assumed to be treated like waste type B.23 from operational waste.

E11.1.4 Activity determination of radionuclides

Before the waste is transported from BKAB to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The highest permissible surface dose rate is 100 mSv/h. The waste packages are assumed to be free from surface contamination.

E11.1.5 Production of the waste type

Table E11-1 lists the number of packages for SFR.

The waste will be deposited during the years 2023–2029. The waste vault given is according to the acceptance criteria for the waste type.

Table E11-1. Number of packages of the waste type.

Number of packages	Waste vault	B.23:D scrap metal	B.23:D secondary waste
Forecasted	(BMA)	486	122

E11.2 Average package for the waste type

E11.2.1 Material – waste, packaging and matrix

Table E11-2 gives values for an estimated average of the material content in waste type B.23:D. The material data refer to one mould. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E11-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	B.23:D scrap metal	B.23:D secondary waste
Aluminium/zinc [kg]	Waste	–	4.0
Aluminium/zinc surface [m^2]	Waste	–	0.60
Aluminium/zinc thickness [mm]	Waste	–	5.0
Cellulose [kg]	Waste	–	44
Iron/steel [kg]	Waste	1,870	100
Iron/steel surface [m^2]	Waste	96	5.1
Iron/steel thickness [mm]	Waste	5.0	5.0
Other organic [kg]	Waste	–	100
Concrete [kg]	Packaging (lid)	500	500
Iron/steel [kg]	Packaging	661	661
Iron/steel surface [m^2]	Packaging	23	23
Iron/steel thickness [mm]	Packaging	5.0–6.0	5.0–6.0
Concrete [kg]	Matrix	1,944	1,356
Void [m^3]	Matrix	0.44	0.43

E11.2.2 Radionuclide content

Table E11-3 provides values for a calculated average of the nuclide content in waste type B.23:D at the closure of SFR on 2075-12-31. Activity data refer to one mould.

Table E11-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	B.23:D [Bq]
H-3	3.88E+03
Be-10	4.17E-03
C-14 org	4.12E+04
C-14 inorg	1.07E+06
C-14 ind	9.35E+04
Cl-36	3.64E+02
Ca-41	0.00E+00
Fe-55	7.23E+02
Co-60	3.26E+06
Ni-59	1.53E+08
Ni-63	1.23E+10
Se-79	5.52E-02
Sr-90	5.33E+05
Zr-93	4.73E+05
Nb-93m	2.68E+08
Nb-94	5.63E+06
Mo-93	9.71E+04
Tc-99	1.56E+04
Pd-107	4.19E+06
Ag-108m	6.37E+05
Cd-113m	6.98E-02

Nuclide	B.23:D [Bq]
Sn-126	6.25E+00
Sb-125	2.86E+01
I-129	6.78E+00
Cs-134	2.96E-05
Cs-135	4.17E+01
Cs-137	1.23E+06
Ba-133	1.46E-04
Pm-147	1.66E-03
Sm-151	2.12E+03
Eu-152	1.25E+00
Eu-154	7.85E+01
Eu-155	2.33E-01
Ho-166m	3.87E-02
U-232	2.41E-01
U-235	8.09E-04
U-236	1.47E+01
Np-237	1.46E+01
Pu-238	6.54E+04
Pu-239	1.47E+04
Pu-240	2.53E+04
Pu-241	8.68E+04

Nuclide	B.23:D [Bq]
Pu-242	1.04E+02
Am-241	5.47E+04
Am-242m	2.70E+02
Am-243	1.14E+03
Cm-243	1.06E+02
Cm-244	6.54E+03
Cm-245	1.25E+01
Cm-246	9.25E+00

E12 C.01:9

E12.1 Description of the waste type

The waste type consists of concrete moulds containing cement-solidified intermediate-level ion exchange resins and filter aids from Clab. The waste type has been manufactured by OKG but the waste packages are owned by Clab, since the waste in them comes from Clab. In order to distinguish the waste owned by Clab from waste that is owned by OKG with the same waste type, the waste type is denoted with the letter C in the present report. In reality the waste is deposited with the letter O. See further description of the waste type in Section E33, O.01:9.

E12.2 Average package for the waste type

Information on the average package for the waste type can be found in Section E33, O.01:9.

E13 C.02

E13.1 Description of the waste type

The waste type C.02 consists of concrete moulds containing cement-solidified intermediate-level ion exchange resins and filter aids from Clab/Clink.

The waste type has previously been deposited as waste type O.02. There is an approved waste type description for deposition of this waste type. Data are based on information in the waste type description and Triumf NG v1.0.1.3.

The acceptance criteria for Silo, described in Section E1.1.1, apply for this waste type.

E13.1.1 Waste

The waste is well defined and consists of both powder and bead resins and inert filter aids from the systems cooling and clean-up system for receiving pools (system 313), cooling and clean-up system for storage pools (system 324), system for treatment of process water (system 371) and system for treatment of floor drainage (system 372).

E13.1.2 Packaging

The waste is packed in concrete moulds. The mould is a cubic box made of reinforced concrete with dimensions 1.2×1.2×1.2 m and a wall thickness of 10 cm. The reinforcement consists of 12 mm steel bars with a total weight of 274 kg. The mould is provided with internal lining in the form of an expansion canister, a disposable stirrer and splash plate. The expansion canister consists of a 20 mm thick pressure-absorbing layer of plastic which weighs about 10 kg. The stirrer is made of carbon steel and weighs about 16 kg. The mould has an empty weight of about 1,600 kg, including the weight for the expansion canister, the stirrer and the splash plate.

The maximum permissible weight for a waste package including waste is 5,000 kg.

The disposal volume for a mould is 1.728 m^3.

E13.1.3 Treatment

The waste is placed in tanks in the waste treatment facility before it is pumped into the concrete mould, where cement is added during mixing. After completed cement dosage, mixing proceeds until a homogeneous waste and cement matrix is obtained. The total fill volume of a package with expansion canister is about 67%. The matrix is allowed to harden for at least two days before a concrete lid is cast on the package. The thickness of the lid is at least 10 cm.

E13.1.4 Activity determination of radionuclides

Before the waste is transported from Clab/Clink to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The activity content must not exceed 0.55 TBq for Co-60 and 0.02 TBq for Cs-137. The highest permissible surface dose rate is 300 mSv/h. This limitation comes from the manufacturing of the waste package. The waste packages are usually free from surface contamination.

E13.1.5 Production of the waste type

Table E13-1 lists the number of packages for SFR.

Deposited packages refer to packages in the existing SFR as of 2012-12-31. The waste type C.02 started being produced in 1986 and has been deposited since 1989.

Forecasted packages refer to waste that has not yet been deposited and includes both waste that is interim-stored at the waste supplier and waste that has not yet been produced. Currently, there are 251 packages in interim storage and a production of 15 packages per year is planned up to 2025 and then 17 packages per year up to 2070. The waste vault that is indicated for forecasted waste is according to the acceptance criteria for the waste type.

Table E13-1. Number of packages of the waste type.

Number of packages	Waste vault	C.02
Deposited	Silo	150
Forecasted	(Silo)	1,211

E13.2 Average package for the waste type

E13.2.1 Material – waste, packaging and matrix

Table E13-2 gives values for an estimated average of the material content in waste type C.02. The material data refer to one mould. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E13-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	C.02
Filter aids [kg]	Waste	3.7
Ion exchange resins [kg]	Waste	130
Concrete [kg]	Packaging (including lid)	1,840
Iron/steel [kg]	Packaging	274
Iron/steel surface [m^2]	Packaging	12
Iron/steel thickness [mm]	Packaging	12
Other organic [kg]	Packaging	10
Cement [kg]	Matrix	1,540
Iron/steel [kg]	Matrix (stirrer)	16
Iron/steel surface [m^2]	Matrix (stirrer)	1.0
Iron/steel thickness [mm]	Matrix (stirrer)	5.0
Void [m^3]	Matrix	0.33

E13.2.2 Radionuclide content

Table E13-3 provides values for a calculated average of the nuclide content in waste type C.02 at the closure of SFR on 2075-12-31. Activity data refer to one mould.

Table E13-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	C.02 [Bq]	Nuclide	C.02 [Bq]	Nuclide	C.02 [Bq]
H-3	1.49E+06	Ag-108m	5.00E+06	U-235	1.78E+01
Be-10	5.35E+01	Cd-113m	3.19E+04	U-236	2.67E+02
C-14 org	1.42E+07	Sn-126	1.35E+02	U-238	3.56E+02
C-14 inorg	3.31E+07	Sb-125	2.21E+07	Np-237	3.95E+02
Cl-36	5.35E+04	I-129	4.54E+03	Pu-238	6.13E+06
Fe-55	1.23E+09	Cs-134	3.12E+06	Pu-239	3.70E+05
Co-60	4.39E+09	Cs-135	8.77E+03	Pu-240	5.21E+05
Ni-59	8.92E+07	Cs-137	1.11E+08	Pu-241	2.15E+07
Ni-63	5.32E+09	Ba-133	1.21E+05	Pu-242	2.67E+03
Se-79	1.08E+03	Pm-147	3.06E+06	Am-241	3.28E+06
Sr-90	2.07E+07	Sm-151	5.86E+05	Am-242m	7.19E+03
Zr-93	8.92E+04	Eu-152	3.55E+03	Am-243	2.46E+05
Nb-93m	2.04E+07	Eu-154	2.80E+06	Cm-243	1.45E+04
Nb-94	8.91E+05	Eu-155	7.85E+05	Cm-244	6.22E+05
Mo-93	3.58E+05	Ho-166m	3.48E+05	Cm-245	2.66E+02
Tc-99	8.07E+05	U-232	1.75E+01	Cm-246	7.07E+01
Pd-107	2.71E+02	U-234	8.89E+02		

E14 C.12:D/C.12C:D

E14.1 Description of the waste type

Waste types C.12:D and C.12C:D are waste types adopted for low-level decommissioning waste in steel containers from Clink. C.12:D contains scrap metal or secondary waste and C.12C:D contains concrete.

There is no approved waste type description for deposition of these waste types. Material quantities and activity have been calculated based on Edelborg et al. (2014), supplemented with assumptions about secondary waste, and material composition for the decommissioning waste, packaging and solidification material.

The acceptance criteria for BLA, described in Section E1.5.1, are assumed to be valid for the waste types.

E14.1.1 Waste

The scrap metal in C.12:D consists mainly of fittings and scrapped components. The secondary waste in C.12:D is assumed to consist of trash and scrap metal like waste type R.12 from operational waste. The waste in C.12C:D comes mainly from contaminated concrete behind the pool plate in the storage pools.

E14.1.2 Packaging

The waste is packed in ISO containers of carbon steel with the dimension 20-foot half height. The container has a length of 6.06 m, a width of 2.5 m and a height of 1.3 m. The thickness of the walls and roof is normally 1.5 mm. An empty container weighs about 1,900 kg. Open containers are sealed with a lid.

The maximum permissible weight for a container including waste is 20 tonnes. The disposal volume is 20 m^3.

E14.1.3 Treatment

A C.12:D container is assumed to be filled with about 16,200 kg of scrap metal. The secondary waste is assumed to be treated like waste type R.12 from operational waste. A C.12C:D container is assumed to be filled with about 15,700 kg of concrete waste.

E14.1.4 Activity determination of radionuclides

Before the waste is transported from Clink to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The highest permissible surface dose rate is 2 mSv/h. The waste packages are assumed to be free from surface contamination.

E14.1.5 Production of the waste type

Table E14-1 lists the number of packages for SFR.

The waste will be deposited during the years 2073–2075. The waste vault given is according to the acceptance criteria for the waste types.

Table E14-1. Number of packages of the waste type.

Number of packages	Waste vault	C.12:D scrap metal	C.12:D secondary waste	C.12C:D
Forecasted	(BLA)	9	2	7

E14.2 Average package for the waste type

E14.2.1 Material – waste, packaging and matrix

Table E14-2 gives values for an estimated average of the material content in waste type C.12:D and C.12C:D. The material data refer to one container. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E14-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	C.12:D scrap metal	C.12:D secondary waste	C.12C:D
Aluminium/zinc [kg]	Waste	–	100	–
Aluminium/zinc surface [m^2]	Waste	–	15	–
Aluminium/zinc thickness [mm]	Waste	–	5.0	–
Concrete [kg]	Waste	–	–	15,714
Cellulose [kg]	Waste	–	500	–
Iron/steel [kg]	Waste	16,189	4,500	–
Iron/steel surface [m^2]	Waste	830	229	–
Iron/steel thickness [mm]	Waste	5.0	5.0	–
Other inorganic [kg]	Waste	–	400	–
Other organic [kg]	Waste	–	3,000	–
Iron/steel [kg]	Packaging	1,900	1,900	1,900
Iron/steel surface [m^2]	Packaging	105	105	105
Iron/steel thickness [mm]	Packaging	1.5	1.5	1.5
Void [m^3]	Matrix	13	7.5	8.5

E14.2.2 Radionuclide content

Table E14-3 provides values for a calculated average of the nuclide content in waste type C.12:D and C.12C:D at the closure of SFR on 2075-12-31. Activity data refer to one container.

Table E14-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	C.12:D [Bq]	C.12C:D [Bq]	Nuclide	C.12:D [Bq]	C.12C:D [Bq]	Nuclide	C.12:D [Bq]	C.12C:D [Bq]
H-3	0.00E+00	0.00E+00	Sb-125	8.61E+03	9.43E+02	Am-242m	0.00E+00	0.00E+00
Be-10	0.00E+00	0.00E+00	I-129	0.00E+00	1.26E+01	Am-243	1.33E+01	1.32E+00
C-14 org	0.00E+00	0.00E+00	Cs-134	0.00E+00	2.20E+02	Cm-243	0.00E+00	0.00E+00
C-14 inorg	0.00E+00	0.00E+00	Cs-135	0.00E+00	0.00E+00	Cm-244	2.94E+02	2.95E+01
C-14 ind	0.00E+00	0.00E+00	Cs-137	0.00E+00	1.63E+07	Cm-245	0.00E+00	0.00E+00
Cl-36	0.00E+00	0.00E+00	Ba-133	0.00E+00	0.00E+00	Cm-246	0.00E+00	0.00E+00
Ca-41	0.00E+00	0.00E+00	Pm-147	0.00E+00	0.00E+00			
Fe-55	1.54E+06	1.57E+05	Sm-151	0.00E+00	0.00E+00			
Co-60	6.16E+07	6.29E+06	Eu-152	0.00E+00	0.00E+00			
Ni-59	1.08E+06	1.07E+05	Eu-154	0.00E+00	0.00E+00			
Ni-63	1.33E+08	1.32E+07	Eu-155	0.00E+00	0.00E+00			
Se-79	0.00E+00	0.00E+00	Ho-166m	0.00E+00	0.00E+00			
Sr-90	0.00E+00	1.07E+05	U-232	0.00E+00	0.00E+00			
Zr-93	4.52E+03	4.46E+02	U-235	0.00E+00	0.00E+00			
Nb-93m	1.50E+07	1.51E+06	U-236	0.00E+00	0.00E+00			
Nb-94	5.15E+04	5.09E+03	Np-237	0.00E+00	0.00E+00			
Mo-93	6.36E+03	6.29E+02	Pu-238	9.53E+02	9.43E+01			
Tc-99	9.55E+02	9.43E+01	Pu-239	1.65E+02	1.63E+01			
Pd-107	0.00E+00	0.00E+00	Pu-240	2.29E+02	2.83E+01			
Ag-108m	1.40E+05	1.51E+04	Pu-241	5.54E+03	5.66E+02			
Cd-113m	0.00E+00	0.00E+00	Pu-242	0.00E+00	0.00E+00			
Sn-126	0.00E+00	0.00E+00	Am-241	1.65E+02	1.63E+01			

E15 C.16:D

E15.1 Description of the waste type

The waste type C.16:D is a waste type adopted for decommissioning waste from Clink. It consists of steel moulds containing cement-solidified intermediate-level ion exchange resins from system decontamination.

There is no approved waste type description for deposition of this waste type. Material quantities and activity have been calculated based on Edelborg et al. (2014), supplemented with assumptions about packaging and solidification material.

The acceptance criteria for Silo, described in Section E1.1.1, are assumed to be valid for the waste type.

E15.1.1 Waste

The waste consists of ion exchange resins produced at system decontamination prior to decommissioning.

E15.1.2 Packaging

The waste is packed in steel moulds. The mould is a cubic box made of carbon steel with dimensions 1.2×1.2×1.2 m. The wall is 5 mm thick and the bottom is 6 mm thick. The mould weighs about 400 kg. The mould contains a stirrer of carbon steel that weighs about 25 kg. The waste packaging is also provided with a splash plate.

The maximum permissible weight for a waste package including waste is 5,000 kg.

The disposal volume for a mould is 1.728 m^3.

E15.1.3 Treatment

The mould is assumed to be filled with equal parts ion exchange resin and cement. The waste matrix is homogenised with the aid of the stirrer. The void is assumed to be about 10% of the inner volume of the packaging. The matrix is allowed to harden before a steel lid is placed on the mould.

E15.1.4 Activity determination of radionuclides

Before the waste is transported from OKG to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclide is Co-60. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The highest permissible surface dose rate is 500 mSv/h. The waste packages are assumed to be free from surface contamination.

E15.1.5 Production of the waste type

Table E15-1 lists the number of packages for SFR.

The waste will be deposited during 2074. The waste vault given is according to the acceptance criteria for the waste type.

Table E15-1. Number of packages of the waste type.

Number of packages	Waste vault	C.16:D
Forecasted	(Silo)	7

E15.2 Average package for the waste type

E15.2.1 Material – waste, packaging and matrix

Table E15-2 gives values for an estimated average of the material content in waste type C.16:D. The material data refer to one mould. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E15-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	C.16:D
Ion exchange resins [kg]	Waste	803
Iron/steel [kg]	Packaging	400
Iron/steel surface [m^2]	Packaging	17
Iron/steel thickness [mm]	Packaging	5.0–6.0
Cement [kg]	Matrix	1,836
Iron/steel [kg]	Matrix	25
Iron/steel surface [m^2]	Matrix	3.0
Iron/steel thickness [mm]	Matrix	5.0
Void [m^3]	Matrix	0.17

E15.2.2 Radionuclide content

Table E15-3 provides values for a calculated average of the nuclide content in waste type C.16:D at the closure of SFR on 2075-12-31. Activity data refer to one mould.

Table E15-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	C.16:D [Bq]	Nuclide	C.16:D [Bq]	Nuclide	C.16:D [Bq]
H-3	0.00E+00	Pd-107	0.00E+00	U-236	0.00E+00
Be-10	0.00E+00	Ag-108m	7.07E+07	Np-237	0.00E+00
C-14 org	0.00E+00	Cd-113m	0.00E+00	Pu-238	4.82E+05
C-14 inorg	0.00E+00	Sn-126	0.00E+00	Pu-239	8.35E+04
C-14 ind	0.00E+00	Sb-125	4.49E+06	Pu-240	1.16E+05
Cl-36	0.00E+00	I-129	0.00E+00	Pu-241	2.86E+06
Ca-41	0.00E+00	Cs-134	0.00E+00	Pu-242	0.00E+00
Fe-55	8.02E+08	Cs-135	0.00E+00	Am-241	8.35E+04
Co-60	3.21E+10	Cs-137	0.00E+00	Am-242m	0.00E+00
Ni-59	5.46E+08	Ba-133	0.00E+00	Am-243	6.75E+03
Ni-63	6.75E+10	Pm-147	0.00E+00	Cm-243	0.00E+00
Se-79	0.00E+00	Sm-151	0.00E+00	Cm-244	1.51E+05
Sr-90	0.00E+00	Eu-152	0.00E+00	Cm-245	0.00E+00
Zr-93	2.28E+06	Eu-154	0.00E+00	Cm-246	0.00E+00
Nb-93m	7.71E+09	Eu-155	0.00E+00		
Nb-94	2.60E+07	Ho-166m	0.00E+00		
Mo-93	3.21E+06	U-232	0.00E+00		
Tc-99	4.82E+05	U-235	0.00E+00		

E16 C.23

E16.1 Description of the waste type

The waste type C.23 consists of concrete moulds containing concrete-solidified intermediate-level trash and scrap metal from Clab/Clink. C.23 was previously deposited as the waste type O.23.

There is an approved waste type description for deposition of this waste type. Data are based on information in the waste type description and Triumf NG v1.0.1.3.

The acceptance criteria for BMA, described in Section E1.3.1, apply for this waste type.

E16.1.1 Waste

The waste consists of trash and scrap metal. Scrap metal in the form of active components/parts such as valves, fittings, gaskets, filters etc., as well as trash in the form of e.g. plastics, rags, packaging and concrete.

E16.1.2 Packaging

The waste is packed in concrete moulds. The mould is a cubic box made of reinforced concrete with dimensions 1.2×1.2×1.2 m and a wall thickness of 10 cm. The reinforcement consists of 12 mm steel bars with a total weight of 274 kg. The mould has an empty weight of about 1,600 kg.

The maximum permissible weight for a waste package including waste is 5,000 kg.

The disposal volume for a mould is 1.728 m^3.

E16.1.3 Treatment

The waste is placed directly in moulds and anchored with special restraints to prevent it from floating to the surface when the concrete is added. The void is assumed to be about 25% of the inner volume of the packaging. After solidification, the matrix is hardened for two days before a concrete lid is cast on site.

E16.1.4 Activity determination of the radionuclides

Before the waste is transported from Clab/Clink to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

Normal activity content per mould is 5–500 GBq. The activity content must not exceed 0.96 TBq for Co-60 and 0.014 TBq for Cs-137. The usually measured value of surface dose rate is 1–15 mSv/h. The highest permissible surface dose rate is 100 mSv/h. The waste packages are usually free from surface contamination.

E16.1.5 Production of the waste type

Table E16-1 lists the number of packages for SFR.

Deposited packages refer to packages in the existing SFR as of 2012-12-31. The waste type C.23 started being produced in 1986 and has been deposited since 1993.

Forecasted packages refer to waste that has not yet been deposited and includes both waste that is interim-stored at the waste supplier and waste that has not yet been produced. Currently, there are 15 packages in interim storage and a production is planned of one package per year up to 2025 and then two packages per year up to 2070. The waste vault that is indicated for forecasted waste is according to the acceptance criteria for the waste type.

Table E16-1. Number of packages of the waste type.

Number of packages	Waste vault	C.23
Deposited	1BMA	43
Forecasted	(BMA)	118

E16.2 Average package for the waste type

E16.2.1 Material – waste, packaging and matrix

Table E16-2 gives values for an estimated average of the material content in waste type C.23. The material data refer to one mould. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E16-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	C.23
Aluminium/zinc [kg]	Waste	3.5
Aluminium/zinc surface [m^2]	Waste	0.50
Aluminium/zinc thickness [mm]	Waste	5.0
Cellulose [kg]	Waste	30
Iron/steel [kg]	Waste	105
Iron/steel surface [m^2]	Waste	5.3
Iron/steel thickness [mm]	Waste	5.0
Sludge [kg]	Waste	53
Other inorganic [kg]	Waste	18
Other organic [kg]	Waste	67
Concrete [kg]	Packaging (including lid)	1,840
Iron/steel [kg]	Packaging	274
Iron/steel surface [m^2]	Packaging	12
Iron/steel thickness [mm]	Packaging	12
Concrete [kg]	Matrix	565
Void [m^3]	Matrix	0.25

E16.2.2 Radionuclide content

Table E16-3 provides values for a calculated average of the nuclide content in waste type C.23 at the closure of SFR on 2075-12-31. Activity data refer to one mould.

Table E16-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	C.23 [Bq]	Nuclide	C.23 [Bq]	Nuclide	C.23 [Bq]
H-3	9.81E+05	Cd-113m	3.80E+04	U-238	2.35E+01
Be-10	3.58E+01	Sn-126	1.63E+02	Np-237	2.58E+01
C-14 org	0.00E+00	Sb-125	6.74E+04	Pu-238	5.96E+05
C-14 inorg	0.00E+00	I-129	1.18E+04	Pu-239	2.45E+04
Cl-36	3.58E+04	Cs-134	2.24E+06	Pu-240	3.46E+04
Fe-55	8.16E+08	Cs-135	2.09E+04	Pu-241	1.40E+06
Co-60	2.91E+09	Cs-137	1.33E+08	Pu-242	1.76E+02
Ni-59	5.96E+07	Ba-133	8.01E+04	Am-241	1.93E+05
Ni-63	3.54E+09	Pm-147	3.66E+06	Am-242m	4.75E+02
Se-79	1.30E+03	Sm-151	7.05E+05	Am-243	3.85E+04
Sr-90	1.31E+07	Eu-152	4.22E+03	Cm-243	4.66E+02
Zr-93	5.96E+04	Eu-154	3.34E+06	Cm-244	5.87E+04
Nb-93m	1.34E+07	Eu-155	9.38E+05	Cm-245	1.76E+01
Nb-94	5.95E+05	Ho-166m	2.32E+05	Cm-246	4.68E+00
Mo-93	4.96E+05	U-232	1.15E+00		
Tc-99	1.04E+06	U-234	5.88E+01		
Pd-107	3.26E+02	U-235	1.18E+00		
Ag-108m	3.33E+06	U-236	1.77E+01		

E17 C.4K23:D

E17.1 Description of the waste type

The waste type, C.4K23:D is a waste type adopted for decommissioning waste from Clink. It consists of tetramoulds containing cement-solidified intermediate-level scrap metal.

There is no approved waste type description for deposition of this waste type. Material quantities and activity have been calculated based on Edelborg et al. (2014), supplemented with assumptions about material composition and packaging and solidification material.

The acceptance criteria for BMA, described in Section E1.3.1, are assumed to be valid for the waste type.

E17.1.1 Waste

The waste consists mainly of scrap metal in the form of fittings and scrapped components.

E17.1.2 Packaging

The waste is packed in tetramoulds. The tetramould is a mould of steel plate with outer dimensions 2.4×2.4×1.2 m. The thickness of the walls is 4 mm, the floor 8 mm and the lid 15 mm. The packaging weighs about 1,700 kg.

The maximum permissible weight for a tetramould including waste is 20 tonnes. The disposal volume is 6,912 m^3.

E17.1.3 Treatment

The tetramould is assumed to be filled with about 6,700 kg of waste. The waste is solidified with concrete. The void is estimated to be 25% of the inner volume of the packaging.

E17.1.4 Activity determination of radionuclides

A fully treated package is measured with respect to gamma-emitting nuclides. The dominant gamma-emitting nuclide is Co-60. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The highest permissible surface dose rate is 100 mSv/h. Surface contamination should not exceed 40 kBq/m^2 for loose surface contamination.

E17.1.5 Production of the waste type

Table E17-1 lists the number of packages for SFR.

The waste will be deposited during the years 2073–2075. The waste vault given is according to the acceptance criteria for the waste type.

Table E17-1. Number of packages of the waste type.

Number of packages	Waste vault	C.4K23:D
Forecasted	(BMA)	3

E17.2 Average package for the waste type

E17.2.1 Material – waste, packaging and matrix

Table E17-2 gives values for an estimated average of the material content in waste type C.4K23:D. The material data refer to one tetramould. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E17-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	C.4K23:D
Iron/steel [kg]	Waste	6,667
Iron/steel surface [m^2]	Waste	342
Iron/steel thickness [mm]	Waste	5.0
Iron/steel [kg]	Packaging	1,722
Iron/steel surface [m^2]	Packaging	46
Iron/steel thickness [mm]	Packaging	4.0–15
Concrete [kg]	Matrix	9,649
Void [m^3]	Matrix	1.6

E17.2.2 Radionuclide content

Table E17-3 provides values for a calculated average of the nuclide content in waste type C.4K23:D at the closure of SFR on 2075-12-31. Activity data refer to one tetramould.

Table E17-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	C.4K23:D [Bq]	Nuclide	C.4K23:D [Bq]	Nuclide	C.4K23:D [Bq]
H-3	0.00E+00	Sn-126	0.00E+00	Pu-242	0.00E+00
Be-10	0.00E+00	Sb-125	1.12E+06	Am-241	2.08E+04
C-14 org	0.00E+00	I-129	0.00E+00	Am-242m	0.00E+00
C-14 inorg	0.00E+00	Cs-134	0.00E+00	Am-243	1.68E+03
C-14 ind	0.00E+00	Cs-135	0.00E+00	Cm-243	0.00E+00
Cl-36	0.00E+00	Cs-137	0.00E+00	Cm-244	3.76E+04
Ca-41	0.00E+00	Ba-133	0.00E+00	Cm-245	0.00E+00
Fe-55	2.00E+08	Pm-147	0.00E+00	Cm-246	0.00E+00
Co-60	8.00E+09	Sm-151	0.00E+00		
Ni-59	1.36E+08	Eu-152	0.00E+00		
Ni-63	1.68E+10	Eu-154	0.00E+00		
Se-79	0.00E+00	Eu-155	0.00E+00		
Sr-90	0.00E+00	Ho-166m	0.00E+00		
Zr-93	5.68E+05	U-232	0.00E+00		
Nb-93m	1.92E+09	U-235	0.00E+00		
Nb-94	6.48E+06	U-236	0.00E+00		
Mo-93	8.00E+05	Np-237	0.00E+00		
Tc-99	1.20E+05	Pu-238	1.20E+05		
Pd-107	0.00E+00	Pu-239	2.08E+04		
Ag-108m	1.76E+07	Pu-240	2.88E+04		
Cd-113m	0.00E+00	Pu-241	7.12E+05		

E18 C.24

E18.1 Description of the waste type

The waste type C.24 consists of concrete moulds containing cement-solidified intermediate-level solid waste in the form of components and scrap metal of steel from Clab/Clink.

There is no approved waste type description for deposition of this waste type. Data are based on information in Triumf NG v1.0.1.3.

The acceptance criteria for Silo, described in Section E1.1.1, apply for this waste type.

E18.1.1 Waste

The waste consists of components and scrap metal of steel, steel alloys or other materials containing activity, e.g. valves or filters for clean-up of water and air.

E18.1.2 Packaging

The waste is packed in concrete moulds. The mould is a cubic box made of reinforced concrete with dimensions 1.2×1.2×1.2 m. The walls are 10 cm thick. The reinforcement consists of 12 mm steel bars with a weight of 274 kg. The empty weight of the mould is about 1,600 kg. The mould normally contains a smaller steel container that weighs about 420 kg. Other inner packaging may also be used, e.g. different types of steel drums and drum baskets.

The maximum permissible weight for a mould including waste is 5,000 kg. The disposal volume is 1.728 m^3.

E18.1.3 Treatment

When the mould has been filled with waste including any inner packaging, it is solidified with concrete. The void is assumed to be about 25% of the inner volume of a packaging. After the grout has hardened, the mould is provided with a lid which is cast on site.

E18.1.4 Activity determination of radionuclides

Before the waste is transported from Clab/Clink to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The activity content must not exceed 500 GBq. The highest permissible surface dose rate is 300 mSv/h. The limitation comes from the manufacturing of the waste package. The waste packages are usually free from surface contamination.

E18.1.5 Production of the waste type

Table E18-1 lists the number of packages for SFR.

No packages are deposited in the current SFR as of 2012-12-31.

Forecasted packages refer to waste that has not yet been deposited and includes both waste that is interim-stored at the waste supplier and waste that has not yet been produced. Currently, there are 54 packages in interim storage and a production is planned of 2 packages per year up to 2025 and then 6 packages per year up to 2070. The waste vault given is according to the acceptance criteria for the waste type.

Table E18-1. Number of packages of the waste type.

Number of packages	Waste vault	C.24
Deposited	–	0
Forecasted	(Silo)	350

E18.2 Average package for the waste type

E18.2.1 Material – waste, packaging and matrix

Table E18-2 gives values for an estimated average of the material content in waste type C.24. The material data refer to one mould. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E18-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	C.24
Cellulose [kg]	Waste	35
Iron/steel [kg]	Waste	17
Iron/steel surface [m^2]	Waste	0.9
Iron/steel thickness [mm]	Waste	5.0
Other inorganic [kg]	Waste	24
Concrete [kg]	Packaging (including lid)	1,840
Iron/steel [kg]	Packaging	274
Iron/steel surface [m^2]	Packaging	12
Iron/steel thickness [mm]	Packaging	12
Concrete [kg]	Matrix	1,000
Iron/steel [kg]	Matrix (steel containers)	420
Iron/steel surface [m^2]	Matrix (steel containers)	21
Iron/steel thickness [mm]	Matrix (steel containers)	5.0
Void [m^3]	Matrix	0.25

E18.2.2 Radionuclide content

Table E18-3 provides values for a calculated average of the nuclide content in waste type C.24 at the closure of SFR on 2075-12-31. Activity data refer to one mould.

Table E18-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	C.24 [Bq]	Nuclide	C.24 [Bq]	Nuclide	C.24 [Bq]
H-3	3.29E+06	Ag-108m	8.52E+06	U-235	2.96E+00
Be-10	9.00E+01	Cd-113m	9.93E+06	U-236	4.45E+01
C-14 org	0.00E+00	Sn-126	3.25E+04	U-238	5.92E+01
C-14 inorg	0.00E+00	Sb-125	2.90E+08	Np-237	6.33E+01
Cl-36	9.00E+04	I-129	1.95E+05	Pu-238	1.61E+06
Fe-55	2.83E+09	Cs-134	6.16E+08	Pu-239	6.17E+04
Co-60	1.01E+10	Cs-135	6.50E+05	Pu-240	8.71E+04
Ni-59	1.50E+08	Cs-137	3.15E+10	Pu-241	4.63E+06
Ni-63	9.49E+09	Ba-133	2.72E+05	Pu-242	4.44E+02
Se-79	2.60E+05	Pm-147	1.01E+09	Am-241	4.57E+05
Sr-90	3.95E+07	Sm-151	1.50E+08	Am-242m	1.25E+03
Zr-93	1.50E+05	Eu-152	1.11E+06	Am-243	9.69E+04
Nb-93m	4.38E+07	Eu-154	9.09E+08	Cm-243	1.40E+03
Nb-94	1.50E+06	Eu-155	2.58E+08	Cm-244	1.88E+05
Mo-93	1.49E+05	Ho-166m	5.88E+05	Cm-245	4.43E+01
Tc-99	6.95E+06	U-232	3.17E+00	Cm-246	1.18E+01
Pd-107	6.50E+04	U-234	1.48E+02		

E19 F.BWR:D

E19.1 Description of the waste type

The waste type F.BWR:D is a waste type adopted for reactor pressure vessels without internals from FKA (RPV F1, F2 and F3).

There is no approved waste type description for deposition of this waste type. Data are based on Anunti et al. (2013).

Acceptance criteria for BRT are under development, see Section E1.2.1.

E19.1.1 Waste

The waste consists of surface contaminated and induced steel or steel alloys (C1070/SIS2333).

E19.1.2 Packaging

No waste packaging is used. The reactor pressure vessel is transported and stored intact, without packaging. Reactor pressure vessels F1 and F2 have a height of 21.5 m and an outside diameter of 7.2 m. Reactor pressure vessel F3 has a height of 21.4 m and an outside diameter of 6.75 m.

The disposal volume for a reactor pressure vessel is about 1,190 m^3 based on a cuboid with sides of 7.4 m and a length of 21.7 m, where the dimensions refer to reactor pressure vessel measurements including 0.1 m surrounding air.

E19.1.3 Treatment

Connections are sealed and radiation shielding is mounted as needed. No other treatment is planned, with the exception of covering with tarpaulin, painting or other surface treatment that can be carried out to avoid any surface contamination spreading.

The specified void is based on the inner volume of the reactor pressure vessel.

E19.1.4 Activity determination of radionuclides

The fully treated reactor pressure vessel is measured with respect to surface dose rate. The dominant gamma-emitting nuclide is Co-60.

The highest permissible surface dose rate is 2 mSv/h. The reactor pressure vessels are assumed to be free from surface contamination on the outside.

E19.1.5 Production of the waste type

Table E19-1 lists the number of packages for SFR.

The reactor pressure vessels F1, F2 and F3 are assumed to be deposited in years 2043, 2044 and 2048 respectively. The waste vault given is according to the acceptance criteria for the waste type.

Table E19-1. Number of packages of the waste type.

Number of packages	Waste vault	F1	F2	F3
Forecasted	(BRT)	1	1	1

E19.2 Average package for the waste type

E19.2.1 Material – waste, packaging and matrix

Table E19-2 gives values for an estimated average of the material content in waste type F.BWR:D. The material data refer to one reactor pressure vessel. Besides weights, corrosion surface and thickness for metals and void in the waste package are given. The reactor pressure vessels are internally plated with a stainless layer of at least 3 mm.

Table E19-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	F1	F2	F3
Iron/steel [kg]	Waste	705,000	705,000	760,000
Iron/steel surface [m^2]	Waste	903	903	880
Iron/steel thickness [mm]	Waste	159	159	156
Void [m^3]	Matrix	597	597	615

E19.2.2 Radionuclide content

Table E19-3 provides values for a calculated average of the nuclide content in waste type F.BWR:D at the closure of SFR on 2075-12-31. Activity data refer to one reactor pressure vessel.

Table E19-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	F1 Induced activity [Bq]	Surface activity [Bq]	F2 Induced activity [Bq]	Surface activity [Bq]	F3 Induced activity [Bq]	Surface activity [Bq]
H-3	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Be-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
C-14 org	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
C-14 inorg	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
C-14 ind	7.74E+08	0.00E+00	7.74E+08	0.00E+00	8.85E+08	0.00E+00
Cl-36	5.81E+05	0.00E+00	5.81E+05	0.00E+00	6.66E+05	0.00E+00
Ca-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Fe-55	1.67E+09	4.85E+07	1.93E+09	5.61E+07	5.87E+09	8.30E+08
Co-60	1.04E+10	1.09E+10	1.12E+10	1.17E+10	1.68E+10	4.84E+10
Ni-59	2.82E+09	1.99E+09	2.82E+09	1.99E+09	4.24E+09	1.85E+10
Ni-63	2.18E+11	2.09E+11	2.19E+11	2.10E+11	3.43E+11	1.99E+12
Se-79	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-90	0.00E+00	2.33E+09	0.00E+00	3.98E+09	0.00E+00	3.15E+09
Zr-93	0.00E+00	1.99E+06	0.00E+00	2.10E+06	0.00E+00	8.25E+06
Nb-93m	1.79E+10	8.58E+10	1.84E+10	8.79E+10	1.06E+10	1.38E+11
Nb-94	5.45E+07	4.54E+08	5.45E+07	4.54E+08	3.11E+07	6.65E+08
Mo-93	3.15E+08	2.08E+07	3.15E+08	2.08E+07	1.45E+08	4.79E+06
Tc-99	4.23E+07	4.98E+06	4.23E+07	6.19E+06	1.85E+07	3.08E+06
Pd-107	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ag-108m	0.00E+00	2.12E+08	0.00E+00	2.12E+08	0.00E+00	6.24E+07
Cd-113m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sn-126	0.00E+00	7.04E+04	0.00E+00	1.18E+05	0.00E+00	8.55E+04
Sb-125	1.19E+05	5.95E+05	1.37E+05	6.94E+05	1.72E+05	6.07E+06
I-129	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-134	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-133	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pm-147	0.00E+00	1.19E+05	0.00E+00	2.34E+05	0.00E+00	4.84E+05
Sm-151	0.00E+00	3.39E+07	0.00E+00	5.73E+07	0.00E+00	4.07E+07
Eu-152	0.00E+00	6.24E+04	0.00E+00	1.08E+05	0.00E+00	8.78E+04
Eu-154	0.00E+00	9.79E+06	0.00E+00	1.72E+07	0.00E+00	1.79E+07
Eu-155	0.00E+00	2.43E+05	0.00E+00	4.44E+05	0.00E+00	5.92E+05
Ho-166m	0.00E+00	5.57E+02	0.00E+00	9.37E+02	0.00E+00	1.37E+03
U-232	0.00E+00	5.74E+02	0.00E+00	9.72E+02	0.00E+00	8.12E+02
U-235	0.00E+00	1.30E+00	0.00E+00	2.19E+00	0.00E+00	1.36E+00
U-236	0.00E+00	3.67E+04	0.00E+00	6.18E+04	0.00E+00	4.48E+04
Np-237	0.00E+00	4.35E+04	0.00E+00	7.32E+04	0.00E+00	5.39E+04
Pu-238	0.00E+00	2.44E+08	0.00E+00	4.12E+08	0.00E+00	3.39E+08
Pu-239	0.00E+00	3.94E+07	0.00E+00	6.63E+07	0.00E+00	4.69E+07
Pu-240	0.00E+00	5.23E+07	0.00E+00	8.79E+07	0.00E+00	6.19E+07
Pu-241	0.00E+00	1.01E+09	0.00E+00	1.74E+09	0.00E+00	1.49E+09
Pu-242	0.00E+00	2.93E+05	0.00E+00	4.93E+05	0.00E+00	3.67E+05
Am-241	0.00E+00	1.89E+08	0.00E+00	3.17E+08	0.00E+00	2.14E+08
Am-242m	0.00E+00	1.12E+06	0.00E+00	1.89E+06	0.00E+00	1.33E+06
Am-243	0.00E+00	3.84E+06	0.00E+00	6.46E+06	0.00E+00	5.26E+06
Cm-243	0.00E+00	5.71E+05	0.00E+00	9.73E+05	0.00E+00	9.12E+05
Cm-244	0.00E+00	6.08E+07	0.00E+00	1.05E+08	0.00E+00	1.13E+08
Cm-245	0.00E+00	4.75E+04	0.00E+00	7.99E+04	0.00E+00	9.54E+04
Cm-246	0.00E+00	1.67E+04	0.00E+00	2.81E+04	0.00E+00	2.53E+04

E20 F.05:1/F.05:2

E20.1 Description of the waste type

The waste type F.05 consists of standard 200-litre steel drums containing bitumen-solidified low and intermediate-level ion exchange resins from FKA.

The waste type only occurs as its two variants, F.05:1 and F.05:2. That is, there is no waste type F.05 for deposition. The two variants are so similar that they in this chapter will be reported as one when it comes to activity, however, not when it comes to material. F.05:1 contains ion exchange resins from Forsmark 1 while F.05:2 contains ion exchange resins from Forsmark 3.

There are approved waste type descriptions for the deposition of the two variants. Data are based on information in the waste type descriptions and Triumf NG v1.0.1.3.

The acceptance criteria for BMA, described in Section E1.3.1, apply for this waste type.

E20.1.1 Waste

The waste is well defined and consists mostly of powder resins from condensate clean-up (system 332) and water clean-up (system 342/1). Waste type F.05:1 also contains powder resins from system 324 and waste type F.05:2 contains powder resins from treatment of floor drainage/chemical water (system 342/2). Ground bead resins from the systems reactor clean-up (system 331) and system drainage clean-up (system 342) and inert filter aids also occur in both waste types.

E20.1.2 Packaging

The waste is packed in steel drums. The drum has a diameter of 0.59 m, a height of 0.88 m and a thickness of about 1.2 mm. The drum is made of carbon steel and has an empty weight of about 25 kg.

The drums are positioned four by four on a drum tray of carbon steel. The drum tray has the outer dimensions 1.2×1.2 m and weighs about 66 kg.

The maximum permissible weight for a drum including waste is 500 kg. The disposal volume for a drum on a drum tray is 0.324 m^3.

E20.1.3 Treatment

The waste material is gathered in tanks in the waste facility. From here, the waste material is pumped to a drying system where it is dried. The waste is mixed to a homogeneous mass together with bitumen before filling the drums. For F.05:1 about 95 kg of bitumen per package is used and for F.05:2 about 75 kg of bitumen per package is used. The total fill volume of a package is about 85–90% in cold state. A steel lid with a thickness of 0.5 mm seals the drum.

E20.1.4 Activity determination of radionuclides

Before the waste was transported from FKA to SFR, a measurement of gamma-emitting nuclides was made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The measured activity content for F.05:1 is about 0.01–0.1 GBq/kg of dry waste and for F.05:2 about 0.1 GBq/kg of dry waste. Surface dose rate for F.05:1 is normally about 1–10 mSv/h and for F.05:2 typically less than 5 mSv/h. The highest permissible surface dose rate is 30 mSv/h, which is based on the manufacturing of the waste package. The waste packages are normally free from surface contamination.

E20.1.5 Production of the waste type

Table E20-1 lists the number of packages for SFR.

Deposited packages refer to packages in the existing SFR as of 2012-12-31. The variant F.05:1 started being produced in 1981 and has been deposited since 1988. The variant F.05:2 started being produced in 1986 and has been deposited since 1990.

No future production of the waste type is planned.

Table E20-1. Number of packages of the waste type.

Number of packages	Waste vault	F.05:1	F.05:2
Deposited	1BMA	1,454	258
Forecasted	–	0	0

E20.2 Average package for the waste type

E20.2.1 Material – waste, packaging and matrix

Table E20-2 gives values for an estimated average of the material content in waste type F.05. The material data refer to one steel drum including a ¼ drum tray. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E20-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	F.05:1	F.05:2
Ion exchange resins [kg]	Waste	130	140
Iron/steel [kg]	Packaging	42	42
Iron/steel surface [m^2]	Packaging	4.2	4.2
Iron/steel thickness [mm]	Packaging (steel drum)	1.2	1.2
Iron/steel thickness [mm]	Packaging (drum tray)	5.0	5.0
Bitumen [kg]	Matrix	95	75
Void [m^3]	Matrix	0.035	0.035

E20.2.2 Radionuclide content

Table E20-3 provides values for a calculated average of the nuclide content in waste type F.05 at the closure of SFR on 2075-12-31. Activity data refer to one steel drum.

Table E20-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	F.05:1/F.05:2 [Bq]	Nuclide	F.05:1/F.05:2 [Bq]	Nuclide	F.05:1/F.05:2 [Bq]
H-3	3.25E+03	I-129	1.22E+02	Pu-242	8.26E+01
Be-10	3.06E+00	Cs-134	2.59E–06	Am-241	1.10E+05
C-14 org	1.44E+05	Cs-135	1.11E+02	Am-242m	1.77E+02
C-14 inorg	7.66E+06	Cs-137	7.11E+06	Am-243	2.19E+02
Cl-36	3.37E+04	Ba-133	1.36E+02	Cm-243	3.20E+01
Fe-55	6.44E–01	Pm-147	2.71E–03	Cm-244	1.76E+03
Co-60	3.77E+04	Sm-151	8.45E+04	Cm-245	8.20E+00
Ni-59	4.92E+06	Eu-152	3.98E+01	Cm-246	2.18E+00
Ni-63	2.22E+08	Eu-154	4.06E+03		
Se-79	2.25E+02	Eu-155	8.49E+00		
Sr-90	5.67E+06	Ho-166m	1.94E+04		
Zr-93	5.11E+03	U-232	3.34E–01		
Nb-93m	1.07E+05	U-234	2.75E+01		
Nb-94	5.09E+04	U-235	1.04E+02		
Mo-93	1.49E+04	U-236	8.31E+00		
Tc-99	2.54E+04	U-238	1.14E+02		
Pd-107	5.63E+01	Np-237	9.36E+02		
Ag-108m	2.66E+05	Pu-238	2.82E+04		
Cd-113m	4.10E+02	Pu-239	1.14E+04		
Sn-126	2.81E+01	Pu-240	1.61E+04		
Sb-125	8.13E–02	Pu-241	4.36E+04		

E21 F.12

E21.1 Description of the waste type

The waste type F.12 consists of steel containers containing low-level trash and scrap metal from FKA.

There is an approved waste type description for deposition of this waste type. Data are based on information in the waste type description and Triumf NG v1.0.1.3.

The acceptance criteria for BLA, described in Section E1.5.1, apply for this waste type.

E21.1.1 Waste

The waste consists of trash and scrap metal. The trash consists of compacted or non-compacted garbage bags containing e.g. textiles, paper, insulation, small pieces of aluminium, copper and plastics. The scrap metal consists of e.g. fittings, scrapped components, cables, suspensions and insulation. The mixture of waste materials has changed over time depending on the maintenance work, revisions or other work carried out.

E21.1.2 Packaging

The waste is packed in ISO containers of carbon steel with the dimension 20-foot half height or 10-foot half height.

The 20-foot container has a length of 6.1 m, a width of 2.5 m and a height of 1.3 m. The thickness of the walls and roof is normally about 1.5 mm. An empty container weighs about 1,900 kg.

The 10-foot container has a length of 3 m, a width of 2.4 m and a height of 1.3 m. Otherwise it has the same specification as the 20-foot container.

Open containers are sealed with a lid.

The maximum permissible weight for a 20-foot container including waste is 20 tonnes and for a 10-foot container 10 tonnes. The disposal volumes are 20 m^3 and 10 m^3 respectively.

E21.1.3 Treatment

Combustible waste that cannot be burned due to high activity content is mixed with compactible waste, compacted and wrapped in plastic. Non-combustible and non-compactible waste is placed in steel drums, steel boxes or garbage bags or is placed directly in the container, without treatment.

As high a fill volume as possible should always be targeted; it can, however, vary widely depending on the nature of the waste. The void in a waste package is assumed to be 7.5 m^3 for both types of packaging, even though there is likely to be more void in the larger packaging.

E21.1.4 Activity determination of radionuclides

Before the waste is transported from FKA to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The usually measured activity content is about 10 GBq/container. The highest permissible surface dose rate is 2 mSv/h. The waste packages are usually free from surface contamination.

E21.1.5 Production of the waste type

Table E21-1 lists the number of packages for SFR.

Deposited packages refer to packages in the existing SFR as of 2012-12-31. The waste type F.12 started being produced in 1988 and has been deposited since 1991.

Forecasted packages refer to waste that has not yet been deposited and includes both waste that is interim-stored at the waste supplier and waste that has not yet been produced. Currently, there are 25 packages in interim storage, of which 4 are 20-foot containers and 21 are 10-foot containers. A production is planned of 0.5 20-foot containers and 0.2 10-foot containers per year up to 2042. The waste vault that is indicated for forecasted waste is according to the acceptance criteria for the waste type.

Table E21-1. Number of packages of the waste type.

Number of packages	Waste vault	F.12 20-foot half height	F.12 10-foot half height
Deposited	1BLA	24	0
Forecasted	(BLA)	19	27

E21.2 Average package for the waste type

E21.2.1 Material – waste, packaging and matrix

Table E21-2 gives values for an estimated average of the material content in waste type F.12. The material data refer to one container. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E21-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	F.12
Aluminium/zinc [kg]	Waste	100
Aluminium/zinc surface [m^2]	Waste	15
Aluminium/zinc thickness [mm]	Waste	5.0
Cellulose [kg]	Waste	500
Iron/steel [kg]	Waste	4,500
Iron/steel surface [m^2]	Waste	229
Iron/steel thickness [mm]	Waste	5.0
Other inorganic [kg]	Waste	400
Other organic [kg]	Waste	3,000
Iron/steel [kg]	Packaging	1,900
Iron/steel surface [m^2]	Packaging	105
Iron/steel thickness [mm]	Packaging	1.5
Void [m^3]	Matrix	7.5

E21.2.2 Radionuclide content

Table E21-3 provides values for a calculated average of the nuclide content in waste type F.12 at the closure of SFR on 2075-12-31. Activity data refer to one container.

Table E21-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	F.12 20-foot half height [Bq]	F.12 10-foot half height [Bq]
H-3	2.83E+04	3.34E+04
Be-10	4.79E+00	4.79E+00
C-14 org	0.00E+00	0.00E+00
C-14 inorg	0.00E+00	0.00E+00
Cl-36	4.79E+03	4.79E+03
Fe-55	9.37E+04	8.34E+04
Co-60	9.89E+06	8.99E+06
Ni-59	6.34E+06	6.35E+06
Ni-63	3.37E+08	3.57E+08
Se-79	7.32E+02	7.32E+02
Sr-90	3.34E+06	3.88E+06
Zr-93	7.98E+03	7.98E+03
Nb-93m	5.69E+05	6.84E+05
Nb-94	7.96E+04	7.96E+04
Mo-93	1.65E+04	7.90E+03
Tc-99	4.08E+04	4.22E+04
Pd-107	1.83E+02	1.83E+02
Ag-108m	4.30E+05	4.36E+05
Cd-113m	5.87E+03	6.71E+03
Sn-126	9.15E+01	9.15E+01
Sb-125	1.56E+02	9.15E+03
I-129	4.65E+02	5.49E+02
Cs-134	8.46E+00	1.08E+02
Cs-135	1.01E+03	1.83E+03
Cs-137	4.27E+07	4.80E+07
Ba-133	1.74E+03	2.00E+03
Pm-147	1.33E+03	1.19E+03
Sm-151	3.33E+05	3.50E+05
Eu-152	6.14E+02	7.01E+02
Eu-154	2.05E+05	2.16E+05
Eu-155	9.08E+03	7.98E+03
Ho-166m	3.07E+04	3.09E+04
U-232	9.32E–02	1.01E–01
U-234	6.08E+00	6.08E+00
U-235	3.43E+01	3.43E+01
U-236	1.83E+00	1.83E+00
U-238	2.62E+01	2.62E+01
Np-237	1.67E+02	1.67E+02
Pu-238	7.08E+03	7.55E+03
Pu-239	2.53E+03	2.53E+03
Pu-240	3.54E+03	3.54E+03
Pu-241	3.94E+04	4.72E+04
Pu-242	1.82E+01	1.82E+01
Am-241	2.34E+04	2.35E+04
Am-242m	4.36E+01	4.56E+01
Am-243	3.97E+02	3.98E+02
Cm-243	1.11E+00	1.29E+00
Cm-244	4.83E+02	5.76E+02
Cm-245	1.81E+00	1.82E+00
Cm-246	4.82E–01	4.83E–01

E22 F.12:D/F.12C:D/F.12S:D

E22.1 Description of the waste type

Waste types F.12:D, F.12C:D and F.12S:D are waste types adopted for low-level decommissioning waste in steel containers from FKA. F.12:D contains scrap metal or secondary waste. F.12C:D contains concrete and F.12S:D contains sand.

There is no approved waste type description for deposition of these waste types. Material quantities and activity have been calculated based on Anunti et al. (2013), supplemented with assumptions about secondary waste, and material composition for the decommissioning waste, packaging and solidification material.

The acceptance criteria for BLA, described in Section E1.5.1, are assumed to be valid for the waste types.

E22.1.1 Waste

The scrap metal in F.12:D consists mainly of fittings and scrapped components. The secondary waste in F.12:D is assumed to consist of trash and scrap metal similar to waste type R.12 from operational waste. The waste in F.12C:D consists of concrete from the outer parts of the biological shield and contaminated concrete from the controlled area in the facility, and the waste in F.12S:D consists of sand from the sand beds in system 341.

E22.1.2 Packaging

The waste is packed in ISO containers of carbon steel with dimensions 20-foot half height. The container has a length of 6.06 m, a width of 2.5 m and a height of 1.3 m. The thickness of the walls and roof is normally 1.5 mm. An empty container weighs about 1,900 kg. Open containers are sealed with a lid.

The maximum permissible weight for a container including waste is 20 tonnes. The disposal volume is 20 m^3.

E22.1.3 Treatment

A container F.12:D is assumed to be filled with about 16,500 kg of scrap metal. The secondary waste is assumed to be treated like waste type R.12 from operational waste. A container F.12C:D is assumed to be filled with about 17,800 kg of concrete waste and F.12S:D is assumed to be filled with about 17,900 kg of sand.

E22.1.4 Activity determination of radionuclides

Before the waste is transported from FKA to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. The radioactivity in F.12S:D system 341 is dominated during manufacturing of the waste package by the long-lived noble gas daughters Sr-90, Cs-135 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The highest permissible surface dose rate is 2 mSv/h. The waste packages are assumed to be free from surface contamination.

E22.1.5 Production of the waste type

Table E22-1 lists the number of packages for SFR.

The waste will be deposited during the years 2040–2052. The waste vault given is according to the acceptance criteria for the waste types.

Table E22-1. Number of packages of the waste type.

Number of packages	Waste vault	F.12:D scrap metal	F.12:D secondary waste	F.12C:D	F.12S:D
Forecasted	(BLA)	454	75	152	53

E22.2 Average package for the waste type

E22.2.1 Material – waste, packaging and matrix

Table E22-2 gives values for an estimated average of the material content in waste types F.12:D, F.12C:D and F.12S:D. The material data refer to one container. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E22-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	F.12:D scrap metal	F.12:D secondary waste	F.12C:D	F.12S:D
Aluminium/zinc [kg]	Waste	–	100	–	–
Aluminium/zinc surface [m^2]	Waste	–	15	–	–
Aluminium/zinc thickness [mm]	Waste	–	5.0	–	–
Concrete [kg]	Waste	–	–	17,827	–
Cellulose [kg]	Waste	–	500	–	–
Iron/steel [kg]	Waste	16,474	4,500	–	–
Iron/steel surface [m^2]	Waste	845	229	–	–
Iron/steel thickness [mm]	Waste	5.0	5.0	–	–
Sand [kg]	Waste	–	–	–	17,866
Other inorganic [kg]	Waste	–	400	–	–
Other organic [kg]	Waste	–	3,000	–	–
Iron/steel [kg]	Packaging	1,900	1,900	1,900	1,900
Iron/steel surface [m^2]	Packaging	105	105	105	105
Iron/steel thickness [mm]	Packaging	1.5	1.5	1.5	1.5
Void [m^3]	Matrix	13	7.5	7.6	3.4

E22.2.2 Radionuclide content

Table E22-3 provides values for a calculated average of the nuclide content in waste types F.12:D, F.12C:D and F.12S:D at the closure of SFR on 2075-12-31. Activity data refer to one container.

Table E22-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	F.12:D [Bq]	F.12C:D [Bq]	F.12S:D [Bq]
H-3	0.00E+00	3.71E+08	0.00E+00
Be-10	0.00E+00	5.22E–01	0.00E+00
C-14 org	1.36E+03	3.63E+03	0.00E+00
C-14 inorg	7.38E+04	1.86E+05	0.00E+00
C-14 ind	0.00E+00	6.80E+05	0.00E+00
Cl-36	1.68E+00	2.26E+04	0.00E+00
Ca-41	0.00E+00	2.24E+06	0.00E+00
Fe-55	1.82E+05	1.58E+05	0.00E+00
Co-60	1.56E+07	7.69E+06	0.00E+00
Ni-59	4.47E+06	3.47E+05	0.00E+00
Ni-63	4.79E+08	3.85E+07	0.00E+00
Se-79	2.13E–01	1.54E+00	0.00E+00
Sr-90	2.72E+06	2.79E+05	1.49E+08
Zr-93	2.74E+03	1.16E+03	0.00E+00
Nb-93m	7.77E+07	2.56E+07	0.00E+00
Nb-94	3.96E+05	1.35E+05	0.00E+00
Mo-93	1.34E+04	2.90E+03	0.00E+00
Tc-99	1.02E+04	1.53E+04	0.00E+00
Pd-107	3.48E–01	8.83E–01	0.00E+00
Ag-108m	1.37E+05	4.65E+05	0.00E+00
Cd-113m	1.11E+01	4.46E+02	0.00E+00
Sn-126	7.09E+01	4.08E+00	0.00E+00
Sb-125	1.46E+03	9.98E+01	0.00E+00
I-129	4.60E+01	6.28E+01	1.18E+04
Cs-134	1.96E+02	1.26E+03	0.00E+00
Cs-135	2.07E+03	5.56E+02	1.89E+06
Cs-137	1.54E+07	1.96E+07	4.34E+09
Ba-133	6.83E–03	3.16E+04	0.00E+00
Pm-147	2.54E+02	3.52E+02	0.00E+00
Sm-151	3.43E+04	4.57E+06	0.00E+00
Eu-152	6.92E+01	3.03E+07	0.00E+00
Eu-154	1.21E+04	6.23E+05	0.00E+00
Eu-155	3.62E+02	3.40E+04	0.00E+00
Ho-166m	6.89E–01	6.27E+04	0.00E+00
U-232	5.94E–01	2.16E–02	0.00E+00
U-235	1.26E–03	1.45E–04	0.00E+00
U-236	3.65E+01	9.22E–01	0.00E+00
Np-237	4.34E+01	1.15E+00	0.00E+00
Pu-238	2.52E+05	1.22E+04	0.00E+00
Pu-239	3.93E+04	4.66E+03	0.00E+00
Pu-240	5.20E+04	2.25E+03	0.00E+00
Pu-241	1.08E+06	8.96E+04	0.00E+00
Pu-242	2.93E+02	7.52E+00	0.00E+00
Am-241	1.87E+05	1.11E+04	0.00E+00
Am-242m	1.11E+03	3.16E+01	0.00E+00
Am-243	3.92E+03	1.07E+02	0.00E+00
Cm-243	6.14E+02	3.30E+01	0.00E+00
Cm-244	6.89E+04	5.52E+03	0.00E+00
Cm-245	5.40E+01	1.84E+00	0.00E+00
Cm-246	1.75E+01	5.05E–01	0.00E+00

E23 F.15

E23.1 Description of the waste type

The waste type F.15 consists of steel moulds containing cement-solidified intermediate-level ion exchange resins, filter aids and evaporator concentrates from FKA.

There is an approved waste type description for deposition of this waste type. Data are based on information in the waste type description and Triumf NG v1.0.1.3.

The acceptance criteria for BMA, described in Section E1.3.1, apply for this waste type.

E23.1.1 Waste

The waste is well defined and consists of powder resins, including possible filter aids, from the systems condensate clean-up (system 332), system drainage (system 342/1), floor drainage/chemical water clean-up (system 342/2) and evaporator concentrates from evaporation of water (system 342/5).

E23.1.2 Packaging

The waste is packed in steel moulds. The mould is a cubic box made of carbon steel with dimensions 1.2×1.2×1.2 m. The wall is 5 mm thick, the bottom is 6 mm thick. The mould weighs about 400 kg. The mould contains a stirrer of carbon steel. This weighs about 25 kg. The waste packaging is also provided with a splash plate.

The maximum permissible weight for a mould including waste is 5,000 kg. The disposal volume is 1.728 m^3.

E23.1.3 Treatment

The waste material is dried and heat treated and then mixed with water and cement additive. The slurry is pumped into the steel mould and cement is added. The waste matrix is homogenised with the aid of the stirrer. The total fill volume of a package is about 90%. The matrix is allowed to harden before a steel lid is placed on the mould.

E23.1.4 Activity determination of radionuclides

Before the waste was transported from FKA to SFR, a measurement of gamma-emitting nuclides was made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The measured activity content at manufacturing is less than 0.08 GBq/kg. The usually measured surface dose rate is less than 1 mSv/h. The highest permissible surface dose rate is 100 mSv/h. The waste packages are usually free from surface contamination.

E23.1.5 Production of the waste type

Table E23-1 lists the number of packages for SFR.

Deposited packages refer to packages in the existing SFR as of 2012-12-31. The waste type F.15 started being produced in 1988 and has been deposited since 1993.

No future production of the waste type is planned.

Table E23-1. Number of packages of the waste type.

Number of packages	Waste vault	F.15
Deposited	1BMA	11
Forecasted	–	0

E23.2 Average package for the waste type

E23.2.1 Material – waste, packaging and matrix

Table E23-2 gives values for an estimated average of the material content in waste type F.15. The material data refer to one mould. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E23-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	F.15
Evaporator concentrates [kg]	Waste	161
Ion exchange resins [kg]	Waste	375
Iron/steel [kg]	Packaging	400
Iron/steel surface [m^2]	Packaging	17
Iron/steel thickness [mm]	Packaging	5.0–6.0
Cement [kg]	Matrix	805
Iron/steel [kg]	Matrix (stirrer)	25
Iron/steel surface [m^2]	Matrix (stirrer)	3.0
Iron/steel thickness [mm]	Matrix (stirrer)	5.0
Void [m^3]	Matrix	0.17

E23.2.2 Radionuclide content

Table E23-3 provides values for a calculated average of the nuclide content in waste type F.15 at the closure of SFR on 2075-12-31. Activity data refer to one mould.

Table E23-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	F.15 [Bq]
H-3	2.48E+03
Be-10	2.06E+00
C-14 org	1.09E+06
C-14 inorg	5.83E+07
Cl-36	2.56E+05
Fe-55	7.27E–01
Co-60	3.36E+04
Ni-59	3.31E+06
Ni-63	1.52E+08
Se-79	8.94E+03
Sr-90	4.04E+06
Zr-93	3.44E+03
Nb-93m	7.93E+04
Nb-94	3.43E+04
Mo-93	1.01E+04
Tc-99	1.71E+04
Pd-107	2.23E+03
Ag-108m	1.80E+05
Cd-113m	1.80E+04
Sn-126	1.12E+03
Sb-125	9.13E–02
I-129	4.85E+03
Cs-134	9.01E–05

Nuclide	F.15 [Bq]
Cs-135	4.40E+03
Cs-137	2.96E+08
Ba-133	1.06E+02
Pm-147	1.73E–01
Sm-151	3.41E+06
Eu-152	1.75E+03
Eu-154	1.89E+05
Eu-155	4.45E+02
Ho-166m	1.31E+04
U-232	2.30E–01
U-234	1.86E+01
U-235	7.00E+01
U-236	5.60E+00
U-238	7.67E+01
Np-237	6.31E+02
Pu-238	1.94E+04
Pu-239	7.71E+03
Pu-240	1.08E+04
Pu-241	3.26E+04
Pu-242	5.57E+01
Am-241	7.41E+04
Am-242m	1.21E+02
Am-243	1.47E+02

Nuclide	F.15 [Bq]
Cm-243	2.28E+01
Cm-244	1.29E+03
Cm-245	5.53E+00
Cm-246	1.47E+00

E24 F.17/F.17:1

E24.1 Description of the waste type

The waste type F.17 consists of steel moulds containing bitumen-solidified intermediate-level ion exchange resins and evaporator concentrates from FKA.

There is a variant of the waste type, F.17:1. The difference for the variant compared with F.17 is that the ion exchange resins and evaporator concentrates come from different systems. The differences are considered to be so small that the same data are used for F.17:1 as for F.17.

There are approved waste type descriptions for deposition of this waste type and the variant. Data are based on information in the waste type descriptions and Triumf NG v1.0.1.3.

The acceptance criteria for BMA, described in Section E1.3.1, apply for this waste type.

E24.1.1 Waste

The waste is well defined and consists of powder resins and possibly filter aids, bead resins and evaporator concentrates, from the systems condensate clean-up (system 332), waste facility water clean-up systems and evaporator circuit (systems 342/1, 342/2 and 342/5) and condensation pool clean-up via 324 (systems 324/316).

E24.1.2 Packaging

The waste is packed in steel moulds. The mould is a cubic box made of carbon steel with dimensions 1.2×1.2×1.2 m. The wall is 5 mm thick, the bottom is 6 mm thick. The mould weighs about 550 kg.

The maximum permissible weight for a mould including waste is 5,000 kg. The disposal volume is 1.728 m^3.

E24.1.3 Treatment

The waste material is dried and heat treated before it is homogenised with bitumen. The mixed mass is poured directly into a steel mould. About 820 kg of bitumen is used per mould. The total fill volume of a package is about 90%. The waste package is provided with a steel lid before the cooling process.

E24.1.4 Activity determination of radionuclides

Before the waste is transported from FKA to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

Usually measured activity content is about 0.2 GBq/kg of dry waste and the maximum measured activity content is 400 GBq per package. Usually measured surface dose rate is about 20 mSv/h. The highest permissible surface dose rate is 100 mSv/h. The waste packages are normally free from surface contamination.

E24.1.5 Production of the waste type

Table E24-1 lists the number of packages for SFR.

Deposited packages refer to packages in the existing SFR as of 2012-12-31. The waste type F.17 started being produced in 1988 and has been deposited since 1991. The variant F.17:1 started being produced in 1991 and has been deposited since 1992.

Forecasted packages refer to waste that has not yet been deposited and includes both waste that is interim-stored at the waste supplier and waste that has not yet been produced. Currently, there are 185 packages in interim storage and a production is planned of 25 packages per year up to 2042. The waste vault that is indicated for forecasted waste is according to the acceptance criteria for the waste type.

Table E24-1. Number of packages of the waste type.

Number of packages	Waste vault	F.17	F.17:1
Deposited	1BMA	415	32
Forecasted	(BMA)	935	0

E24.2 Average package for the waste type

E24.2.1 Material – waste, packaging and matrix

Table E24-2 gives values for an estimated average of the material content in waste type F.17. The material data refer to one mould. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E24-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	F.17/F.17:1
Cellulose [kg]	Waste	4.3*
Filter aids [kg]	Waste	60
Evaporator concentrates [kg]	Waste	120
Ion exchange resins [kg]	Waste	650
Iron/steel [kg]	Packaging	550
Iron/steel surface [m^2]	Packaging	10
Iron/steel thickness [mm]	Packaging	5.0–6.0
Bitumen [kg]	Matrix	820
Void [m^3]	Matrix	0.17

*Until the end of the 1980s filter aids with cellulose content were used. This refers to 195 packages of waste type F.17, deposited in SFR during the early 90s. Other packages contain no cellulose.

E24.2.2 Radionuclide content

Table E24-3 provides values for a calculated average of the nuclide content in waste type F.17 at the closure of SFR on 2075-12-31. Activity data refer to one mould.

Table E24-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	F.17/F.17:1 [Bq]	Nuclide	F.17/F.17:1 [Bq]	Nuclide	F.17/F.17:1 [Bq]
H-3	3.37E+05	Ag-108m	4.03E+06	U-235	1.51E+03
Be-10	4.44E+01	Cd-113m	3.45E+05	U-236	1.20E+02
C-14 org	1.92E+07	Sn-126	5.01E+03	U-238	1.65E+03
C-14 inorg	1.03E+09	Sb-125	2.79E+02	Np-237	1.36E+04
Cl-36	3.18E+04	I-129	2.62E+04	Pu-238	5.14E+05
Fe-55	1.11E+06	Cs-134	2.42E+03	Pu-239	1.66E+05
Co-60	1.23E+08	Cs-135	6.27E+04	Pu-240	2.33E+05
Ni-59	7.13E+07	Cs-137	2.40E+09	Pu-241	3.25E+06
Ni-63	3.93E+09	Ba-133	2.11E+04	Pu-242	1.20E+03
Se-79	4.01E+04	Pm-147	7.90E+04	Am-241	1.58E+06
Sr-90	1.74E+08	Sm-151	1.83E+07	Am-242m	2.95E+03
Zr-93	7.41E+04	Eu-152	3.62E+04	Am-243	3.18E+03
Nb-93m	6.52E+06	Eu-154	1.26E+07	Cm-243	9.74E+02
Nb-94	7.39E+05	Eu-155	5.65E+05	Cm-244	9.02E+04
Mo-93	1.27E+05	Ho-166m	2.86E+05	Cm-245	1.19E+02
Tc-99	8.13E+05	U-232	6.49E+00	Cm-246	3.17E+01
Pd-107	1.00E+04	U-234	3.99E+02		

E25 F.18

E25.1 Description of the waste type

The waste type F.18 consists of steel moulds containing bitumen-solidified intermediate-level ion exchange resins and evaporator concentrates from FKA.

There is an approved waste type description for deposition of this waste type. Data are based on information in the waste type description and Triumf NG v1.0.1.3.

The acceptance criteria for Silo, described in Section E1.1.1, apply for this waste type.

E25.1.1 Waste

The waste is well defined and consists of bead and powder resins, in some cases mixed with filter aids, from the systems reactor water clean-up (system 331), drainage clean-up (system 342/1), pool water clean-up (system 324) and condensate clean-up (system 332).

E25.1.2 Packaging

The waste is packed in steel moulds. The mould is a cubic box made of carbon steel with dimensions 1.2×1.2×1.2 m. The wall is 5 mm thick and the bottom is 6 mm thick. The mould weighs about 550 kg.

The maximum permissible weight for a mould including waste is 5,000 kg. The disposal volume is 1.728 m^3.

E25.1.3 Treatment

The waste material is dried and heat treated before it is homogenised with bitumen. About 960 kg of bitumen is used per package. The mixed mass is poured directly into a steel mould. The total fill volume of a package is about 90%. The waste package is provided with a steel lid before the cooling process.

E25.1.4 Activity determination of radionuclides

Before the waste is transported from FKA to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The usually measured activity content is about 3 GBq/kg dry waste. The highest permissible surface dose rate is 500 mSv/h. The waste packages are usually free from surface contamination.

E24.1.5 Production of the waste type

Table E25-1 lists the number of packages for SFR.

Deposited packages refer to packages in the existing SFR as of 2012-12-31. The waste type F.18 started being produced in 1992 and has been deposited since 1994.

Forecasted packages refer to waste that has not yet been deposited and includes both waste that is interim-stored at the waste supplier and waste that has not yet been produced. Currently, there are 90 packages in interim storage and a production is planned of 15 packages per year up to 2042. The waste vault that is indicated for forecasted waste is according to the acceptance criteria for the waste type.

Table E25-1. Number of packages of the waste type.

Number of packages	Waste vault	F.18
Deposited	Silo	264
Forecasted	(Silo)	540

E25.2 Average package for the waste type

E25.2.1 Material – waste, packaging and matrix

Table E25-2 gives values for an estimated average of the material content in waste type F.18. The material data refer to one mould. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E25-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	F.18
Ion exchange resins [kg]	Waste	600
Iron/steel [kg]	Packaging	550
Iron/steel surface [m^2]	Packaging	10
Iron/steel thickness [mm]	Packaging	5.0–6.0
Bitumen [kg]	Matrix	960
Void [m^3]	Matrix	0.17

E25.2.2 Radionuclide content

Table E25-3 provides values for a calculated average of the nuclide content in waste type F.18 at the closure of SFR on 2075-12-31. Activity data refer to one mould.

Table E25-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	F.18 [Bq]
H-3	3.60E+06
Be-10	4.31E+02
C-14 org	3.46E+06
C-14 inorg	1.45E+08
Cl-36	3.64E+05
Fe-55	1.20E+07
Co-60	1.33E+09
Ni-59	6.92E+08
Ni-63	3.90E+10
Se-79	2.58E+05
Sr-90	1.79E+09
Zr-93	7.19E+05
Nb-93m	6.90E+07
Nb-94	7.17E+06
Mo-93	1.17E+06
Tc-99	6.95E+06
Pd-107	6.45E+04
Ag-108m	3.93E+07
Cd-113m	2.72E+06
Sn-126	3.22E+04
Sb-125	3.53E+03
I-129	1.76E+05
Cs-134	2.13E+04
Cs-135	4.75E+05
Cs-137	1.74E+10
Ba-133	2.27E+05
Pm-147	6.64E+05
Sm-151	1.23E+08
Eu-152	2.87E+05

Nuclide	F.18 [Bq]
Eu-154	1.04E+08
Eu-155	4.74E+06
Ho-166m	2.78E+06
U-232	6.48E+01
U-234	3.88E+03
U-235	1.46E+04
U-236	1.17E+03
U-238	1.60E+04
Np-237	1.32E+05
Pu-238	5.11E+06
Pu-239	1.61E+06
Pu-240	2.27E+06
Pu-241	3.45E+07
Pu-242	1.16E+04
Am-241	1.53E+07
Am-242m	2.91E+04
Am-243	3.09E+04
Cm-243	1.00E+04
Cm-244	9.48E+05
Cm-245	1.16E+03
Cm-246	3.07E+02

E26 F.18:D

E26.1 Description of the waste type

The waste type F.18:D is a waste type adopted for decommissioning waste from FKA. It consists of steel moulds containing bitumen-solidified intermediate-level ion exchange resins from system decontamination.

There is no approved waste type description for deposition of this waste type. Material quantities and activity have been calculated based on Anunti et al. (2013), supplemented with assumptions about packaging and solidification material.

The acceptance criteria for Silo, described in Section E1.1.1, are assumed to be valid for this waste type.

E26.1.1 Waste

The waste consists of ion exchange resins produced at system decontamination prior to decommissioning.

E26.1.2 Packaging

The waste is packed in steel moulds. The mould is a cubic box made of carbon steel with dimensions 1.2×1.2×1.2 m. The wall is 5 mm thick and the bottom is 6 mm thick. The mould weighs about 550 kg.

The maximum permissible weight for a waste package including waste is 5,000 kg. The disposal volume for a mould is 1.728 m^3.

E26.1.3 Treatment

The waste material is dried and heat treated before it is homogenised with bitumen. The mould is assumed to be filled with equal parts ion exchange resin and bitumen. The mixed mass is poured directly into a steel mould. The void of a package is assumed to be about 10%. The waste package is provided with a steel lid.

E26.1.4 Activity determination of radionuclides

Before the waste is transported from FKA to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The highest permissible surface dose rate is 500 mSv/h. The waste packages are assumed to be free from surface contamination.

E26.1.5 Production of the waste type

Table E26-1 lists the number of packages for SFR.

The waste will be deposited during the years 2041-2042 and 2046. The waste vault given is according to the acceptance criteria for the waste type.

Table E26-1. Number of packages of the waste type.

Number of packages	Waste vault	F.18:D
Forecasted	(Silo)	21

E26.2 Average package for the waste type

E26.2.1 Material – waste, packaging and matrix

Table E26-2 gives values for an estimated average of the material content in waste type F.18:D. The material data refer to one mould. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E26-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	F.18:D
Ion exchange resins [kg]	Waste	803
Iron/steel [kg]	Packaging	550
Iron/steel surface [m^2]	Packaging	10
Iron/steel thickness [mm]	Packaging	5.0–6.0
Bitumen [kg]	Matrix	842
Void [m^3]	Matrix	0.17

E26.2.2 Radionuclide content

Table E26-3 provides values for a calculated average of the nuclide content in waste type F.18:D at the closure of SFR on 2075-12-31. Activity data refer to one mould.

Table E26-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	F.18:D [Bq]
H-3	0.00E+00
Be-10	0.00E+00
C-14 org	6.52E+04
C-14 inorg	4.78E+06
C-14 ind	0.00E+00
Cl-36	5.99E+02
Ca-41	0.00E+00
Fe-55	4.27E+07
Co-60	3.50E+09
Ni-59	1.03E+09
Ni-63	1.11E+11
Se-79	7.48E+01
Sr-90	5.38E+08
Zr-93	6.10E+05
Nb-93m	1.66E+10
Nb-94	8.41E+07
Mo-93	2.71E+06
Tc-99	3.13E+06
Pd-107	1.24E+02
Ag-108m	2.73E+07
Cd-113m	4.70E+03
Sn-126	1.52E+04
Sb-125	3.38E+05
I-129	9.08E+03
Cs-134	5.44E+04
Cs-135	8.00E+04
Cs-137	2.81E+09
Ba-133	1.30E+00
Pm-147	6.12E+04

Nuclide	F.18:D [Bq]
Sm-151	7.38E+06
Eu-152	1.53E+04
Eu-154	2.75E+06
Eu-155	8.49E+04
Ho-166m	1.51E+02
U-232	1.27E+02
U-235	2.68E-01
U-236	7.74E+03
Np-237	9.21E+03
Pu-238	5.40E+07
Pu-239	8.38E+06
Pu-240	1.11E+07
Pu-241	2.33E+08
Pu-242	6.22E+04
Am-241	3.99E+07
Am-242m	2.35E+05
Am-243	8.34E+05
Cm-243	1.32E+05
Cm-244	1.49E+07
Cm-245	1.16E+04
Cm-246	3.73E+03

E27 F.20

E27.1 Description of the waste type

The waste type F.20 consists of steel containers containing steel drums with low-level bitumen-solidified ion exchange resins from FKA.

There is an approved waste type description for deposition of this waste type. Data are based on information in the waste type description and Triumf NG v1.0.1.3.

The acceptance criteria for BLA, described in Section E1.5.1, apply for this waste type.

E27.1.1 Waste

The waste is well defined and consists of powder resins from the waste facility's clean-up system (system 342) and the condensate clean-up system (332).

E27.1.2 Packaging

The waste is packed in standard 200-litre steel drums placed in ISO containers with the dimension 20-foot half height. On average, 33 drums fit in a container.

The steel drum has a diameter of 0.59 m, a height of 0.88 m and a thickness of about 1.2 mm. The drum is made of carbon steel and has an empty weight of about 25 kg.

The container is made of carbon steel and has a length of 6.1 m, a width of 2.5 m and a height of 1.3 m. An empty container weighs about 1,900 kg and has a thickness of about 1.5 mm.

The maximum permissible weight for a steel drum including waste is 500 kg. The maximum permissible weight for a container including waste is 20 tonnes. The disposal volume for a container is 20 m^3.

E27.1.3 Treatment

The waste material is dried, heat treated and mixed with bitumen before filling the steel drums. About 95 kg of bitumen is used per container. The total fill volume of a steel drum is about 85–90% in cold state. The drum is provided with a 0.5 mm thick steel lid before the cooling process.

E27.1.4 Activity determination of radionuclides

Before the waste is transported from FKA to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The measured activity content is less than 5 GBq/drum. The highest permissible surface dose rate is 2 mSv/h. The waste packages are usually free from surface contamination.

E27.1.5 Production of the waste type

Table E27-1 lists the number of packages for SFR.

Deposited packages refer to packages in the existing SFR as of 2012-12-31. The waste type F.20 started being produced in 1981 and has been deposited since 1989.

No future production of the waste type is planned.

Table E27-1. Number of packages of the waste type.

Number of packages	Waste vault	F.20
Deposited	1BLA	15
Forecasted	–	0

E27.2 Average package for the waste type

E27.2.1 Material – waste, packaging and matrix

Table E27-2 gives values for an estimated average of the material content in waste type F.20. The material data refer to one container including drums. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E27-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	F.20
Bitumen [kg]	Waste	33*95
Ion exchange resins [kg]	Waste	33*130
Iron/steel [kg]	Packaging (steel drum)	33*25
Iron/steel surface [m^2]	Packaging (steel drum)	33*2.5
Iron/steel thickness [mm]	Packaging (steel drum)	1.2
Iron/steel [kg]	Packaging (container)	1,900
Iron/steel surface [m^2]	Packaging (container)	105
Iron/steel thickness [mm]	Packaging (container)	1.5
Void [m^3]	Matrix	7.5

E27.2.2 Radionuclide content

Table E27-3 provides values for a calculated average of the nuclide content in waste type F.20 at the closure of SFR on 2075-12-31. Activity data refer to one container including drums.

Table E27-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	F.20 [Bq]
H-3	2.47E+01
Be-10	2.85E-02
C-14 org	3.68E+06
C-14 inorg	2.27E+08
Cl-36	9.95E+05
Fe-55	2.31E-03
Co-60	2.16E+02
Ni-59	4.57E+04
Ni-63	2.01E+06
Se-79	4.96E+00
Sr-90	4.83E+04
Zr-93	4.74E+01
Nb-93m	8.52E+02

Nuclide	F.20 [Bq]
Nb-94	4.73E+02
Mo-93	1.39E+02
Tc-99	2.36E+02
Pd-107	1.24E+00
Ag-108m	2.45E+03
Cd-113m	7.51E+00
Sn-126	6.19E-01
Sb-125	3.21E-04
I-129	2.69E+00
Cs-134	1.77E-08
Cs-135	2.44E+00
Cs-137	1.44E+05
Ba-133	9.94E-01

Nuclide	F.20 [Bq]
Pm-147	2.10E-05
Sm-151	1.81E+03
Eu-152	7.24E-01
Eu-154	6.59E+01
Eu-155	1.07E-01
Ho-166m	1.80E+02
U-232	3.00E-03
U-234	2.56E-01
U-235	9.65E-01
U-236	7.72E-02
U-238	1.06E+00
Np-237	8.69E+00
Pu-238	2.55E+02

Nuclide	F.20 [Bq]
Pu-239	1.06E+02
Pu-240	1.49E+02
Pu-241	3.40E+02
Pu-242	7.67E-01
Am-241	1.02E+03
Am-242m	1.62E+00
Am-243	2.03E+00
Cm-243	2.73E-01
Cm-244	1.43E+01
Cm-245	7.62E-02
Cm-246	2.02E-02

E28 F.23

E28.1 Description of the waste type

The waste type F.23 consists of steel or concrete moulds containing concrete-solidified intermediate-level trash, scrap metal and sludge from FKA.

There is an approved waste type description for deposition of this waste type. Data are based on information in the waste type description and Triumf NG v1.0.1.3.

The acceptance criteria for BMA, described in Section E1.3.1, apply for this waste type.

E28.1.1 Waste

The waste consists of trash, scrap metal and sludge. The trash consists of compacted or non-compacted garbage bags containing e.g. textiles, paper, insulation, small pieces of aluminium, copper and plastics. The scrap metal consists of e.g. fittings, scrapped components, cables, suspensions and insulation. The mixture of waste materials has changed over time depending on the maintenance work, revisions or other work carried out.

E28.1.2 Packaging

The waste is packed in steel or concrete moulds. The moulds are cubic boxes with dimensions 1.2×1.2×1.2 m. The steel mould is made of carbon steel and has a wall thickness of 5 mm and a bottom thickness of 6 mm. The steel mould weighs about 460 kg. The concrete moulds are made of ready mixed concrete with reinforcement and are available in two designs with wall thicknesses of 10 cm and 25 cm. The reinforcement consists of 12 mm steel bars with a total weight of 274 kg.

The maximum permissible weight for a mould including waste is 5,000 kg. The disposal volume is 1.728 m^3.

E28.1.3 Treatment

Non-compactible waste and dewatered dried sludge are placed in a mould and a lid is cast on. Compactible waste is placed in a mould and compacted, after which the lid is cast on. Filling with concrete is done by mixing concrete paste and pumping it into the mould with waste material in conjunction with casting on the lid. The void in a package is assumed to be about 25%.

E28.1.4 Activity determination of radionuclides

Before the waste is transported from FKA to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The usually measured activity content is about 0.1 GBq/kg. The highest permissible surface dose rate is 100 mSv/h. The waste packages are usually free from surface contamination.

E28.1.5 Production of the waste type

Table E28-1 lists the number of packages for SFR.

Deposited packages refer to packages in the existing SFR as of 2012-12-31. The waste type F.23 started being produced in 1986 and has been deposited since 1993.

Forecasted packages refer to waste that has not yet been deposited and includes both waste that is interim-stored at the waste supplier and waste that has not yet been produced. Currently, there are 19 packages in interim storage and a production is planned of 10 packages per year up to 2042. The waste vault that is indicated for forecasted waste is according to the acceptance criteria for the waste type.

Table E28-1. Number of packages of the waste type.

Number of packages	Waste vault	F.23 concrete mould	F.23 steel mould
Deposited	1BMA	57	151
Forecasted	(BMA)	0	319

E28.2 Average package for the waste type

E28.2.1 Material – waste, packaging and matrix

Table E28-2 gives values for an estimated average of the material content in waste type F.23. The material data refer to one mould. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E28-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	F.23 concrete mould	F.23 steel mould
Aluminium/zinc [kg]	Waste	–	5.0
Aluminium/zinc surface [m^2]	Waste	–	0.70
Aluminium/zinc thickness [mm]	Waste	–	5.0
Cellulose [kg]	Waste	29	150
Iron/steel [kg]	Waste	30	150
Iron/steel surface [m^2]	Waste	1.5	7.6
Iron/steel thickness [mm]	Waste	5.0	5.0
Sludge [kg]	Waste	–	18
Other organic [kg]	Waste	186	450
Concrete [kg]	Packaging (including lid)	1,840	500
Iron/steel [kg]	Packaging	274	462
Iron/steel surface [m^2]	Packaging	12	21
Iron/steel thickness [mm]	Packaging	12	5.0–6.0
Concrete [kg]	Matrix	565	1,356
Void [m^3]	Matrix	0.25	0.43

E28.2.2 Radionuclide content

Table E28-3 provides values for a calculated average of the nuclide content in waste type F.23 at the closure of SFR on 2075-12-31. Activity data refer to one mould.

Table E28-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	F.23 steel mould [Bq]	F.23 concrete mould [Bq]
H-3	8.78E+04	1.21E+04
Be-10	9.59E+00	7.66E+00
C-14 org	0.00E+00	0.00E+00
C-14 inorg	0.00E+00	0.00E+00
Cl-36	9.59E+03	7.66E+03
Fe-55	3.05E+05	2.17E+01
Co-60	3.36E+07	2.74E+05
Ni-59	1.27E+07	1.02E+07
Ni-63	7.28E+08	4.82E+08
Se-79	8.63E+02	8.91E+02
Sr-90	8.48E+06	3.40E+06
Zr-93	1.60E+04	1.28E+04
Nb-93m	1.66E+06	3.61E+05
Nb-94	1.60E+05	1.27E+05
Mo-93	2.58E+04	3.73E+04
Tc-99	7.27E+04	6.35E+04
Pd-107	2.16E+02	2.23E+02
Ag-108m	8.77E+05	6.71E+05
Cd-113m	9.92E+03	4.82E+03
Sn-126	1.08E+02	1.11E+02
Sb-125	8.99E+02	2.63E+00
I-129	5.90E+02	4.84E+02
Cs-134	6.51E+01	3.12E–02
Cs-135	1.60E+03	4.39E+02
Cs-137	6.13E+07	4.69E+07
Ba-133	5.58E+03	5.47E+02
Pm-147	2.53E+03	3.52E+00
Sm-151	4.20E+05	3.97E+05
Eu-152	1.05E+03	4.90E+02
Eu-154	3.87E+05	9.56E+04
Eu-155	1.80E+04	8.31E+02
Ho-166m	6.19E+04	4.87E+04
U-232	2.09E–01	1.26E–01
U-234	1.22E+01	9.73E+00
U-235	6.88E+01	5.49E+01
U-236	3.67E+00	2.93E+00
U-238	5.24E+01	4.19E+01
Np-237	4.01E+02	2.67E+02
Pu-238	1.55E+04	9.95E+03
Pu-239	5.07E+03	4.05E+03
Pu-240	7.09E+03	5.65E+03
Pu-241	1.18E+05	2.16E+04
Pu-242	3.65E+01	2.92E+01
Am-241	3.57E+06	3.80E+04
Am-242m	9.24E+01	6.46E+01
Am-243	7.97E+02	6.35E+02
Cm-243	2.81E+00	1.13E+00
Cm-244	1.36E+03	3.48E+02
Cm-245	3.64E+00	2.90E+00
Cm-246	9.67E–01	7.69E–01

E29 F.4K23:D/F.4K23C:D

E29.1 Description of the waste type

Waste types F.4K23:D and F.4K23C:D are waste types adopted for low-level decommissioning waste from FKA. F.4K23:D consists of tetramoulds containing cement-solidified intermediate-level scrap metal and F.4K23C:D consists of tetramoulds containing concrete-solidified intermediate-level concrete.

There is no approved waste type description for deposition of these waste types. Material quantities and activity have been calculated based on Anunti et al. (2013), supplemented with assumptions about material composition and packaging and solidification material.

The acceptance criteria for BMA, described in Section E1.3.1, are assumed to be valid for these waste types.

E29.1.1 Waste

The scrap metal in F.4K23:D consists mainly of fittings and scrapped components. The waste in F.4K23C:D consists of parts from the biological shield that have been close to the core.

E29.1.2 Packaging

The waste is packed in tetramoulds. The tetramould is a mould of steel plate with outer dimensions 2.4×2.4×1.2 m. The thickness of the walls is 4 mm, the floor 8 mm and the lid 15 mm. The packaging weighs about 1,700 kg.

The maximum permissible weight for a tetramould including waste is 20 tonnes. The disposal volume is 6.912 m^3.

E29.1.3 Treatment

A tetramould F.4K23:D is assumed to be filled with about 5,100 kg of waste and a tetramould F.4K23C:D is assumed to be filled with about 10,000 kg of waste. The waste is solidified with concrete. The void is estimated to be about 25% of the inner volume of the packaging.

E29.1.4 Activity determination of radionuclides

Before the waste is transported from FKA to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137.

The highest permissible surface dose rate is 100 mSv/h. The waste packages are assumed to be free from surface contamination.

E29.1.5 Production of the waste type

Table E29-1 lists the number of packages for SFR.

The waste will be deposited during the years 2040–2052. The waste vault given is according to the acceptance criteria for the waste types.

Table E29-1. Number of packages of the waste type.

Number of packages	Waste vault	F.4K23:D	F.4K23C:D
Forecasted	(BMA)	237	70

E29.2 Average package for the waste type

E29.2.1 Material – waste, packaging and matrix

Table E29-2 gives values for an estimated average of the material content in waste type F.4K23:D and F.4K23C:D. The material data refer to one tetramould. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E29-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	F.4K23:D	F.4K23C:D
Concrete [kg]	Waste	–	10,003
Iron/steel [kg]	Waste	5,106	–
Iron/steel surface [m^2]	Waste	262	–
Iron/steel thickness [mm]	Waste	5.0	–
Iron/steel [kg]	Packaging	1,722	1,722
Iron/steel surface [m^2]	Packaging	46	46
Iron/steel thickness [mm]	Packaging	4.0–15	4.0–15
Concrete [kg]	Matrix	10,129	1,697
Void [m^3]	Matrix	1.6	1.6

E29.2.2 Radionuclide content

Table E29-3 provides values for a calculated average of the nuclide content in waste type F.4K23:D and F.4K23C:D at the closure of SFR on 2075-12-31. Activity data refer to one tetramould.

Table E29-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	F.4K23:D [Bq]	F.4K23C:D [Bq]
H-3	1.25E+02	7.97E+09
Be-10	0.00E+00	1.09E+01
C-14 org	2.56E+02	0.00E+00
C-14 inorg	2.30E+04	0.00E+00
C-14 ind	9.40E+05	1.42E+07
Cl-36	3.24E+04	4.72E+05
Ca-41	1.50E+06	4.69E+07
Fe-55	4.78E+07	1.83E+06
Co-60	4.33E+09	2.52E+07
Ni-59	1.13E+09	1.11E+05
Ni-63	1.21E+11	8.87E+06
Se-79	1.11E+00	2.01E+01
Sr-90	8.19E+08	9.00E+04
Zr-93	7.60E+05	3.53E+02
Nb-93m	2.44E+10	2.80E+06
Nb-94	1.26E+08	1.12E+05
Mo-93	4.91E+06	2.54E+02
Tc-99	1.44E+06	5.15E+01
Pd-107	1.84E+00	0.00E+00
Ag-108m	4.94E+07	9.58E+06
Cd-113m	6.56E+01	9.21E+03
Sn-126	2.41E+04	0.00E+00
Sb-125	3.68E+05	2.68E+01
I-129	5.37E+02	0.00E+00
Cs-134	7.37E+03	4.56E+03
Cs-135	2.01E+04	0.00E+00

Nuclide	F.4K23:D [Bq]	F.4K23C:D [Bq]
Cs-137	2.18E+08	9.80E+04
Ba-133	2.53E-02	6.71E+05
Pm-147	5.75E+04	1.81E+03
Sm-151	1.16E+07	9.53E+07
Eu-152	2.22E+04	6.51E+08
Eu-154	3.67E+06	1.32E+07
Eu-155	9.84E+04	7.40E+05
Ho-166m	2.17E+02	1.30E+06
U-232	2.02E+02	0.00E+00
U-235	4.37E-01	1.78E-03
U-236	1.26E+04	0.00E+00
Np-237	1.50E+04	0.00E+00
Pu-238	8.54E+07	0.00E+00
Pu-239	1.35E+07	5.55E+04
Pu-240	1.79E+07	0.00E+00
Pu-241	3.61E+08	0.00E+00
Pu-242	1.01E+05	0.00E+00
Am-241	6.41E+07	0.00E+00
Am-242m	3.83E+05	0.00E+00
Am-243	1.34E+06	0.00E+00
Cm-243	2.05E+05	0.00E+00
Cm-244	2.25E+07	0.00E+00
Cm-245	1.77E+04	0.00E+00
Cm-246	5.90E+03	0.00E+00

E30 F.99:1

E30.1 Description of the waste type

The waste type F.99 exists only in two variants, F.99:1 which is presented here, and F.99:2, which is presented in the subsequent section. That is, there is no waste type F.99 for deposition.

The variant F.99:1 consists of miscellaneous waste in the form of steel drums in steel moulds from FKA.

There is an approved waste type description for deposition of F.99:1. Data are based on information in the waste type description and Triumf NG v1.0.1.3.

The acceptance criteria for BMA, described in Section E1.3.1, apply for this waste type.

E30.1.1 Waste

The waste consists of residual slurry from wet combustion experiments.

E30.1.2 Packaging

The waste is packed in steel drums inside steel moulds. A total of six steel drums are placed in two steel moulds.

The steel drum has a diameter of 0.59 m, a height of 0.88 m and a thickness of about 1.2 mm. The drum is made of carbon steel and has an empty weight of about 25 kg.

The mould is a cubic box made of carbon steel with dimensions 1.2×1.2×1.2 m and a wall thickness of 5 mm. The mould weighs about 400 kg.

The maximum permissible weight for a waste package including waste is 5,000 kg. The disposal volume for a mould is 1.728 m^3.

E30.1.3 Treatment

The waste is solidified in cement in the steel drums. The package has a relatively large void of about 58%.

E30.1.4 Activity determination of radionuclides

Before the waste is transported from FKA to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The highest permissible surface dose rate is 100 mSv/h. The waste packages are usually free from surface contamination.

E30.1.5 Production of the waste type

Table E30-1 lists the number of packages for SFR.

Deposited packages refer to packages in the existing SFR as of 2012-12-31. The variant F.99:1 started being produced in 1994 and has been deposited since 1995.

No future production of the waste type is planned.

Table E30-1. Number of packages of the waste type.

Number of packages	Waste vault	F.99:1
Deposited	1BMA	2
Forecasted	–	0

E30.2 Average package for the waste type

E30.2.1 Material – waste, packaging and matrix

Table E30-2 gives values for an estimated average of the material content in waste type F.99:1. The material data refer to one mould including 3 drums. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E30-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	F.99:1
Other inorganic [kg]	Waste	3*105
Iron/steel [kg]	Packaging (mould)	400
Iron/steel surface [m^2]	Packaging (mould)	17
Iron/steel thickness [mm]	Packaging (mould)	5.0
Iron/steel [kg]	Packaging (steel drum)	3*25
Iron/steel surface [m^2]	Packaging (steel drum)	3*4.4
Iron/steel thickness [mm]	Packaging (steel drum)	1.5
Cement [kg]	Matrix (steel drum)	3*200
Void [m^3]	Matrix	0.99

E30.2.2 Radionuclide content

Table E30-3 provides values for a calculated average of the nuclide content in waste type F.99:1 at the closure of SFR on 2075-12-31. Activity data refer to one mould including drums.

Table E30-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	F.99:1 [Bq]
H-3	7.32E+02
Be-10	4.20E–01
C-14 org	0.00E+00
C-14 inorg	0.00E+00
Cl-36	4.20E+02
Fe-55	7.89E–01
Co-60	1.63E+04
Ni-59	7.00E+05
Ni-63	3.20E+07
Se-79	6.58E+02
Sr-90	2.33E+06
Zr-93	7.01E+02
Nb-93m	2.14E+04
Nb-94	6.99E+03
Mo-93	2.05E+03
Tc-99	3.49E+03
Pd-107	1.64E+02
Ag-108m	3.70E+04
Cd-113m	1.83E+03
Sn-126	8.22E+01
Sb-125	9.74E–02
I-129	3.57E+02
Cs-134	1.54E–05
Cs-135	3.24E+02
Cs-137	2.53E+07
Ba-133	3.32E+01

Nuclide	F.99:1 [Bq]
Pm-147	7.21E–02
Sm-151	2.64E+05
Eu-152	1.80E+02
Eu-154	2.36E+04
Eu-155	8.51E+01
Ho-166m	2.67E+03
U-232	0.00E+00
U-234	0.00E+00
U-235	0.00E+00
U-236	0.00E+00
U-238	0.00E+00
Np-237	0.00E+00
Pu-238	0.00E+00
Pu-239	0.00E+00
Pu-240	0.00E+00
Pu-241	0.00E+00
Pu-242	0.00E+00
Am-241	0.00E+00
Am-242m	0.00E+00
Am-243	0.00E+00
Cm-243	0.00E+00
Cm-244	0.00E+00
Cm-245	0.00E+00
Cm-246	0.00E+00

E31 F.99:2

E31.1 Description of the waste type

The waste type F.99 only exists in two variants, F.99:1 which is presented in the previous section, and F.99:2 which is presented here. That is, there is no waste type F.99 for deposition.

The variant F.99:2 consists of miscellaneous waste in the form of steam separators in steel boxes from FKA.

There is an approved waste type description for deposition of F.99:2. Data are based on information in the waste type description and Triumf NG v1.0.1.3.

The acceptance criteria for BTF, described in Section E1.4.1, apply for this waste type.

E31.1.1 Waste

The waste is well defined and consists of steam separators with auxiliary equipment manufactured in stainless steel. Three steam separators fit in each packaging.

E31.1.2 Packaging

The waste is packed in steel boxes. The steel box is made of 3 mm bent plate and 4 mm outer plate of stainless steel. It has outer dimensions 3.3×1.3×2.3 m and inner dimensions 3.1×2×1.1 m. The weight of an empty box is about 1,900 kg.

The maximum permissible weight for a waste package including waste is 24 tonnes. The disposal volume for a steel box is 10 m^3.

E31.1.3 Treatment

The waste is solidified with concrete until the package is completely filled. The void in a package is estimated to be 5% of the inner volume of the packaging. A tight lid is fitted after filling.

E31.1.4 Activity determination of radionuclides

Before the waste is transported from FKA to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The usually measured activity content is about 70 GBq/steam separator. The highest permissible surface dose rate is 10 mSv/h. The waste packages are usually free from surface contamination.

E31.1.5 Production of the waste type

Table E31-1 lists the number of packages for SFR.

Deposited packages refer to packages in the existing SFR as of 2012-12-31. The variant F.99:2 started being produced in 1999 and has been deposited since 2001.

No future production of the waste type is planned.

Table E31-1. Number of packages of the waste type.

Number of packages	Waste vault	F.99:2
Deposited	2BTF	18
Forecasted	–	0

E31.2 Average package for the waste type

E31.2.1 Material – waste, packaging and matrix

Table E31-2 gives values for an estimated average of the material content in waste type F.99:2. The material data refer to one steel box. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E31-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	F.99:2
Iron/steel [kg]	Waste	3,000
Iron/steel surface [m^2]	Waste	153
Iron/steel thickness [mm]	Waste	5.0
Iron/steel [kg]	Packaging	1,900
Iron/steel surface [m^2]	Packaging	53
Iron/steel thickness [mm]	Packaging	3.0–4.0
Concrete [kg]	Matrix	19,000
Void [m^3]	Matrix	0.34

E31.2.2 Radionuclide content

Table E31-3 provides values for a calculated average of the nuclide content in waste type F.99:2 at the closure of SFR on 2075-12-31. Activity data refer to one steel box.

Table E31-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	F.99:2 [Bq]
H-3	4.99E+05
Be-10	2.18E+02
C-14 org	0.00E+00
C-14 inorg	0.00E+00
Cl-36	2.18E+05
Fe-55	1.09E+02
Co-60	1.60E+07
Ni-59	3.63E+08
Ni-63	3.02E+10
Se-79	0.00E+00
Sr-90	7.83E+08
Zr-93	3.64E+05
Nb-93m	1.37E+07
Nb-94	3.63E+06
Mo-93	1.07E+06
Tc-99	1.81E+06
Pd-107	0.00E+00
Ag-108m	1.93E+07
Cd-113m	0.00E+00
Sn-126	0.00E+00
Sb-125	1.71E+02
I-129	0.00E+00
Cs-134	0.00E+00
Cs-135	0.00E+00
Cs-137	0.00E+00
Ba-133	2.37E+04
Pm-147	0.00E+00
Sm-151	0.00E+00

Nuclide	F.99:2 [Bq]
Eu-152	0.00E+00
Eu-154	0.00E+00
Eu-155	0.00E+00
Ho-166m	1.39E+06
U-232	7.29E+01
U-234	5.24E+03
U-235	1.05E+02
U-236	1.58E+03
U-238	2.10E+03
Np-237	3.26E+04
Pu-238	3.63E+06
Pu-239	2.18E+06
Pu-240	3.04E+06
Pu-241	1.60E+07
Pu-242	1.57E+04
Am-241	2.33E+07
Am-242m	3.60E+04
Am-243	1.56E+05
Cm-243	1.70E+04
Cm-244	6.25E+04
Cm-245	1.56E+03
Cm-246	4.15E+02

E32 O.BWR:D

E32.1 Description of the waste type

The waste type O.BWR:D is a waste type adopted for reactor pressure vessels without internals from OKG (RPV O1, O2 and O3).

There is no approved waste type description for deposition of this waste type. Data are based on Larsson et al. (2013).

Acceptance criteria for BRT are under development, see Section E1.2.1.

E32.1.1 Waste

The waste consists of surface contaminated and induced steel or steel alloys, O1 (A302/CN24/13), O2 and O3 (C1070/SIS2333).

E32.1.2 Packaging

No waste packaging is used. The reactor pressure vessel is transported and stored intact, without packaging. The reactor pressure vessels O1, O2 and O3 have heights of 18 m, 20.2 m, 21.4 m and outer diameters of 5.3 m, 5.5 m, 6.75 m, respectively.

The disposal volume for reactor pressure vessel O1 is about 645 m^3 based on a cuboid with sides of 5.95 m and a length of 18.3 m, where the dimensions refer to RPV measurements including flanges and fittings and 0.1 m surrounding air.

The disposal volume for reactor pressure vessel O2 is about 790 m^3 based on a cuboid with sides of 6.2 m and a length of 20.45 m, where the dimensions refer to RPV measurements including flanges and fittings and 0.1 m surrounding air.

The disposal volume for reactor pressure vessel O3 is about 1,190 m^3 based on a cuboid with sides of 7.4 m and a length of 21.7 m. The disposal volume is assumed to be the same as for reactor pressure vessel F3, as O3 and F3 are sister reactors.

E32.1.3 Treatment

Connections are sealed and radiation shielding is mounted as needed. No other treatment is planned, with the exception of covering with tarpaulin, painting or other surface treatment that can be carried out to avoid any surface contamination spreading.

The specified void is based on the inner volume of the reactor pressure vessels.

E32.1.4 Activity determination of radionuclides

The fully treated reactor pressure vessel is measured with respect to surface dose rate. The dominant gamma-emitting nuclide is Co-60. The surface dose rate may not exceed 2 mSv/h.

The reactor pressure vessels are assumed to be free from surface contamination on the outside.

E32.1.5 Production of the waste type

Table E32-1 lists the number of packages for SFR.

The reactor pressure vessels O1, O2 and O3 are assumed to be deposited in the years 2034, 2037 and 2048. The waste vault given is according to the acceptance criteria for the waste type.

Table E32-1. Number of packages of the waste type.

Number of packages	Waste vault	O1	O2	O3
Forecasted	(BRT)	1	1	1

E32.2 Average package for the waste type

E32.2.1 Material – waste, packaging and matrix

Table E32-2 gives values for an estimated average of the material content in waste type O.BWR:D. The material data refer to one reactor pressure vessel. Besides weights, corrosion surface and thickness for metals and void in the waste package are given. The reactor pressure vessels are internally plated with a stainless layer of at least 3 mm.

Table E32-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	O1	O2	O3
Iron/steel [kg]	Waste	414,000	530,000	760,000
Iron/steel surface [m^2]	Waste	578	674	880
Iron/steel thickness [mm]	Waste	125	134	156
Void [m^3]	Matrix	315	386	615

E32.2.2 Radionuclide content

Table E32-3 provides values for a calculated average of the nuclide content in waste type O.BWR:D at the closure of SFR on 2075-12-31. Activity data refer to one reactor pressure vessel.

Table E32-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	O1 Induced activity [Bq]	O1 Surface activity [Bq]	O2 Induced activity [Bq]	O2 Surface activity [Bq]	O3 Induced activity [Bq]	O3 Surface activity [Bq]
H-3	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Be-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
C-14 org	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
C-14 inorg	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
C-14 ind	3.49E+08	0.00E+00	2.02E+09	0.00E+00	4.58E+08	0.00E+00
Cl-36	2.22E+05	0.00E+00	1.54E+06	0.00E+00	3.44E+05	0.00E+00
Ca-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Fe-55	2.54E+08	1.45E+07	9.04E+08	4.63E+07	2.58E+09	5.67E+08
Co-60	4.91E+09	4.21E+09	9.31E+09	9.30E+09	8.00E+09	3.61E+10
Ni-59	4.42E+09	1.54E+10	9.90E+09	2.29E+10	2.19E+09	1.12E+10
Ni-63	3.78E+11	1.49E+12	7.45E+11	2.26E+12	1.76E+11	1.20E+12
Se-79	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-90	0.00E+00	2.02E+09	0.00E+00	5.05E+09	0.00E+00	3.41E+09
Zr-93	0.00E+00	1.04E+07	0.00E+00	1.56E+07	0.00E+00	5.08E+06
Nb-93m	4.22E+09	6.27E+10	1.19E+10	1.06E+11	5.53E+09	1.03E+11
Nb-94	2.47E+08	5.61E+08	6.57E+07	8.34E+08	1.63E+07	4.18E+08
Mo-93	3.13E+08	3.41E+06	2.75E+08	5.07E+06	7.74E+07	3.17E+06
Tc-99	4.94E+07	2.78E+06	3.67E+07	5.70E+06	9.81E+06	2.92E+06
Pd-107	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ag-108m	0.00E+00	3.55E+07	0.00E+00	5.30E+07	0.00E+00	4.87E+07
Cd-113m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sn-126	0.00E+00	7.58E+04	0.00E+00	1.72E+05	0.00E+00	9.29E+04
Sb-125	0.00E+00	2.43E+05	2.39E+04	7.66E+05	7.76E+04	4.47E+06
I-129	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-134	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-133	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pm-147	0.00E+00	9.48E+03	0.00E+00	6.26E+04	0.00E+00	4.55E+05
Sm-151	0.00E+00	4.54E+07	0.00E+00	6.55E+07	0.00E+00	4.67E+07
Eu-152	0.00E+00	3.59E+04	0.00E+00	9.47E+04	0.00E+00	8.47E+04
Eu-154	0.00E+00	9.32E+06	0.00E+00	1.28E+07	0.00E+00	1.87E+07
Eu-155	0.00E+00	2.18E+05	0.00E+00	2.50E+05	0.00E+00	5.89E+05
Ho-166m	0.00E+00	4.72E+02	0.00E+00	1.07E+03	0.00E+00	1.68E+03
U-232	0.00E+00	5.93E+02	0.00E+00	1.45E+03	0.00E+00	8.78E+02
U-235	0.00E+00	2.22E+00	0.00E+00	3.64E+00	0.00E+00	1.46E+00
U-236	0.00E+00	3.90E+04	0.00E+00	9.27E+04	0.00E+00	4.85E+04
Np-237	0.00E+00	5.94E+04	0.00E+00	9.09E+04	0.00E+00	5.87E+04
Pu-238	0.00E+00	3.18E+08	0.00E+00	4.67E+08	0.00E+00	3.58E+08
Pu-239	0.00E+00	5.63E+07	0.00E+00	9.23E+07	0.00E+00	4.95E+07
Pu-240	0.00E+00	7.34E+07	0.00E+00	1.59E+08	0.00E+00	6.43E+07
Pu-241	0.00E+00	9.04E+08	0.00E+00	1.42E+09	0.00E+00	1.55E+09
Pu-242	0.00E+00	3.04E+05	0.00E+00	6.54E+05	0.00E+00	3.91E+05
Am-241	0.00E+00	2.77E+08	0.00E+00	3.69E+08	0.00E+00	2.27E+08
Am-242m	0.00E+00	3.42E+06	0.00E+00	1.84E+06	0.00E+00	1.22E+06
Am-243	0.00E+00	3.08E+06	0.00E+00	7.21E+06	0.00E+00	5.67E+06
Cm-243	0.00E+00	8.69E+05	0.00E+00	1.00E+06	0.00E+00	9.18E+05
Cm-244	0.00E+00	5.70E+07	0.00E+00	8.50E+07	0.00E+00	1.22E+08
Cm-245	0.00E+00	5.98E+04	0.00E+00	7.94E+04	0.00E+00	8.54E+04
Cm-246	0.00E+00	1.83E+04	0.00E+00	2.43E+04	0.00E+00	3.59E+04

E33 O.01:9 (C.01:9)

E33.1 Description of the waste type

The waste type consists of concrete moulds containing cement-solidified intermediate-level ion exchange resins and filter aids from OKG and Clab. The waste type has been deposited both in BMA and in 1BTF, where the packages in 1BTF act as supportive walls for the steel drums deposited there.

The waste type has been manufactured by OKG, but some of the waste packages are owned by Clab, since the waste in them comes from Clab. In order to distinguish the waste owned by Clab, it is denoted with the letter C in the present report. In reality the waste is deposited with the letter O.

There is a variant of the waste type, O.01:9. Even though O.01 is approved for deposition, no O.01 packages have been manufactured and they will not be manufactured in the future either. This chapter will therefore only present the variant O.01:9.

There are approved waste type descriptions for deposition of the waste type and the variant. Data are based on information in the waste type descriptions and Triumf NG v1.0.1.3.

The acceptance criteria for BMA, described in Section E1.3.1, apply for this waste type. For the packages deposited in BTF, the acceptance criteria in Section E1.4.1 apply.

E33.1.1 Waste

The waste is well defined and consists of powder and bead resins and inert filter aids from the systems for reactor water clean-up (system 331), treatment of liquid waste (system 342), system decontamination (system 347) and fuel storage pool clean-up (system 324). Bead resins come from systems 331, 342 and 347 and powder resins and filter aids from system 324.

E33.1.2 Packaging

The waste is packed in concrete moulds. The mould is a cubic box made of reinforced concrete with dimensions 1.2×1.2× 1.2 m. The walls are normally 10 cm thick, but can in some exceptional cases be 25 cm thick. The reinforcement consists of 12 mm steel bars with a total weight of 274 kg. The mould is provided with internal lining in the form of an expansion canister (only the 10-cm mould), a disposable stirrer and splash plate. The expansion canister consists of a 20 mm thick pressure-receiving layer of plastic and weighs about 10 kg. The stirrer is made of carbon steel and weighs about 16 kg. The 10-cm mould has an empty weight of about 1,600 kg, including the weight of the expansion canister, the stirrer and the splash plate. The 25-cm mould has an empty weight of about 3,100 kg.

The maximum permissible weight for a waste package including waste is 5,000 kg. The disposal volume for a mould is 1.728 m^3.

E33.1.3 Treatment

The waste is pumped into the concrete mould after which cement is metered during mixing. The mixing proceeds until a homogeneous waste matrix is obtained. The total fill volume of a package with expansion canister is about 67%. The matrix is allowed to harden for two days before a concrete lid with a thickness of at least 10 cm is cast on the mould. The lid concrete is allowed to harden for 24 hours.

E33.1.4 Activity determination of radionuclides

Before the waste is transported from OKG to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The usually measured activity content is about 200–500 GBq. The highest permissible surface dose rate is 30 mSv/h, which is based on the manufacturing of the waste package. The waste packages are usually free from surface contamination.

E33.1.5 Production of the waste type

Table E33-1 lists the number of packages for SFR.

Deposited packages refer to packages in the existing SFR as of 2012-12-31. The variant O.01:9 started being produced in 1971 and has been deposited since 1994. Of the deposited packages, 68 are owned by Clab and are here called C.01:9.

Forecasted packages refer to waste that has not yet been deposited and includes both waste that is interim-stored at the waste supplier and waste that has not yet been produced. Currently, there are 5 packages in interim storage at OKG. No future production of the waste type is planned. The waste vault that is indicated for forecasted waste is according to the acceptance criteria for the waste type.

Table E33-1. Number of packages of the waste type.

Number of packages	Waste vault	O.01:9	C.01:9
Deposited	1BMA	670	68
Deposited	1BTF	28	0
Forecasted	(BMA)	5	0

E33.2 Average package for the waste type

E33.2.1 Material – waste, packaging and matrix

Table E33-2 gives values for an estimated average of the material content in waste type O.01:9/C.01:9. The material data refer to one mould. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E33-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	O.01:9/C.01:9
Cellulose [kg]	Waste	20.5*
Ion exchange resins [kg]	Waste	130
Concrete [kg]	Packaging (including lid)	1,840
Iron/steel [kg]	Packaging	274
Iron/steel surface [m^2]	Packaging	12
Iron/steel thickness [mm]	Packaging	12
Other organic [kg]	Packaging	10
Cement [kg]	Matrix	1,540
Iron/steel [kg]	Matrix (stirrer)	16
Iron/steel surface [m^2]	Matrix (stirrer)	1.0
Iron/steel thickness [mm]	Matrix (stirrer)	5.0
Void [m^3]	Matrix	0.33

*Only included in waste with a certain type of packaging, code 030. This concerns 339 packages. Other packages contain no cellulose.

E33.2.2 Radionuclide content

Table E33-3 provides values for a calculated average of the nuclide content in waste type O.01:9/C.01:9 at the closure of SFR on 2075-12-31. Activity data refer to one mould.

Table E33-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	O.01:9 BMA [Bq]	C.01:9 BMA [Bq]	O.01:9 1BTF [Bq]
H-3	2.57E+04	3.86E+04	1.15E+02
Be-10	2.45E+01	3.30E+01	2.02E–01
C-14 org	3.73E+05	3.69E+07	2.37E+05
C-14 inorg	1.36E+07	8.60E+07	1.35E+07
Cl-36	1.02E+05	3.30E+04	1.02E+05
Fe-55	9.16E+01	1.04E+01	2.50E–03
Co-60	4.42E+05	5.05E+05	5.75E+02
Ni-59	3.64E+07	5.50E+07	3.01E+05
Ni-63	1.76E+09	2.39E+09	1.40E+07
Se-79	5.30E+04	1.50E+01	6.89E+02
Sr-90	3.49E+07	3.89E+06	2.51E+05
Zr-93	4.08E+04	5.51E+04	3.37E+02
Nb-93m	8.39E+05	1.24E+06	4.39E+03
Nb-94	4.07E+05	5.49E+05	3.36E+03
Mo-93	2.20E+05	1.56E+06	1.84E+03
Tc-99	9.10E+05	3.05E+06	7.50E+03
Pd-107	1.33E+04	3.74E+00	1.72E+02
Ag-108m	2.12E+06	2.87E+06	1.73E+04
Cd-113m	7.63E+04	1.47E+01	7.40E+02
Sn-126	6.62E+03	1.87E+00	8.61E+01
Sb-125	1.62E+00	1.31E+00	3.24E–04
I-129	3.80E+04	4.78E+02	4.93E+02
Cs-134	3.23E–02	5.28E–09	2.66E–07
Cs-135	5.16E+04	7.96E+02	6.58E+02
Cs-137	1.48E+09	3.54E+05	1.70E+07
Ba-133	1.09E+03	1.64E+03	4.30E+00
Pm-147	9.41E+00	6.51E–06	4.71E–04
Sm-151	1.91E+07	5.10E+03	2.38E+05
Eu-152	7.35E+03	1.39E+00	7.03E+01
Eu-154	7.00E+05	9.79E+01	5.21E+03
Eu-155	2.10E+03	9.04E–02	5.39E+00
Ho-166m	1.55E+05	2.09E+05	1.27E+03
U-232	6.21E+00	6.77E+00	4.64E–02
U-234	5.17E+02	5.48E+02	4.27E+00
U-235	8.15E+00	1.10E+01	6.74E–02
U-236	3.44E+02	1.65E+02	2.84E+00
U-238	1.10E+03	2.19E+02	9.10E+00
Np-237	1.26E+03	2.75E+02	1.05E+01
Pu-238	4.51E+05	2.64E+06	3.57E+03
Pu-239	2.15E+05	2.28E+05	1.78E+03
Pu-240	3.00E+05	3.21E+05	2.48E+03
Pu-241	8.03E+05	9.42E+05	3.96E+03
Pu-242	1.55E+03	1.65E+03	1.28E+01
Am-241	2.03E+06	2.28E+06	1.66E+04
Am-242m	3.31E+03	3.55E+03	2.60E+01
Am-243	2.15E+04	1.51E+05	1.78E+02
Cm-243	7.96E+01	2.77E+03	5.73E–01
Cm-244	1.43E+04	5.13E+04	9.39E+01
Cm-245	1.54E+02	1.63E+02	1.27E+00
Cm-246	4.08E+01	4.33E+01	3.37E–01

E34 O.02/O.02:9

E34.1 Description of the waste type

The waste type O.02 consists of concrete moulds containing cement-solidified intermediate-level ion exchange resins and filter aids from OKG. Previously, waste from Clab has also been included in this waste type. In order to distinguish the waste owned by Clab, these are denoted with the letter C in the present report and described in Chapter 13.

There is a variant of this waste type, O.02:9. The differences between O.02 and O.02:9 mainly concern the packaging and solidification recipes. The differences are considered to be so small that the same data are used for O.02:9 as for O.02.

There are approved waste type descriptions for deposition of the waste type and the variant. Data are based on information in the waste type descriptions and Triumf NG v1.0.1.3.

The acceptance criteria for Silo, described in Section E1.1.1, apply for this waste type.

E34.1.1 Waste

The waste is well defined and consists of powder and bead resins and inert filter aids from the systems for reactor water clean-up (system 331), treatment of liquid waste (system 342), system decontamination (system 347) and fuel storage pool clean-up (system 324). Bead resins come from systems 331, 342 and 347 and powder resins and filter aids from system 324.

E34.1.2 Packaging

The waste is packed in concrete moulds. The mould is a cubic box made of reinforced concrete with dimensions 1.2×1.2×1.2 m. The walls are normally 10 cm thick, but can in some exceptional cases be 25 cm thick. The reinforcement consists of 12 mm steel bars with a total weight of 274 kg. The mould is provided with internal lining in the form of an expansion canister (only the 10-cm mould), a disposable stirrer and splash plate. The expansion canister consists of a 20 mm thick pressure-receiving layer of plastic and weighs about 10 kg. The stirrer is made of carbon steel and weighs about 16 kg. The 10-cm mould has an empty weight of about 1,600 kg, including the weight of the expansion canister, the stirrer and the splash plate. The 25-cm mould has an empty weight of about 3,100 kg.

The maximum permissible weight for a waste package including waste is 5,000 kg. The disposal volume for a mould is 1.728 m^3.

E34.1.3 Treatment

The waste is pumped into the concrete mould after which cement is metered during mixing. The mixing proceeds until a homogeneous waste matrix is obtained. The total fill volume of a package with expansion canister is about 67%. The matrix is allowed to harden for two days before a concrete lid with a thickness of at least 10 cm is cast on the package.

E34.1.4 Activity determination of radionuclides

Before the waste is transported from OKG to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The usually measured activity content is about 200–500 GBq. The highest permissible surface dose rate is 30 mSv/h. This limitation comes from the manufacturing of the waste package. The waste packages are usually free from surface contamination.

E34.1.5 Production of the waste type

Table E34-1 lists the number of packages for SFR.

Deposited packages refer to packages in the existing SFR as of 2012-12-31. The waste type O.02 started being produced in 1978 and has been deposited since 1989. The variant O.02:9 started being produced in 1972 and has been deposited since 1995.

Forecasted packages refer to waste that has not yet been deposited and includes both waste that is interim-stored at the waste supplier and waste that has not yet been produced. Currently, there are 445 packages in interim storage and a production is planned of 25 packages per year up to 2037. The waste vault that is indicated for forecasted waste is according to the acceptance criteria for the waste type.

Table E34-1. Number of packages of the waste type.

Number of packages	Waste vault	O.02	O.02:9
Deposited	Silo	597	277
Forecasted	(Silo)	1,070	0

E34.2 Average package for the waste type

E34.2.1 Material – waste, packaging and matrix

Table E34-2 gives values for an estimated average of the material content in waste type O.02. The material data refer to one mould. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E34-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	O.02/O.02:9
Filter aids [kg]	Waste	2.6
Ion exchange resins [kg]	Waste	130
Concrete [kg]	Packaging (including lid)	1,840
Iron/steel [kg]	Packaging	274
Iron/steel surface [m^2]	Packaging	12
Iron/steel thickness [mm]	Packaging	12
Other organic [kg]	Packaging	10
Cement [kg]	Matrix	1,540
Iron/steel [kg]	Matrix (stirrer)	16
Iron/steel surface [m^2]	Matrix (stirrer)	1.0
Iron/steel thickness [mm]	Matrix (stirrer)	5.0
Void [m^3]	Matrix	0.33

E34.2.2 Radionuclide content

Table E34-3 provides values for a calculated average of the nuclide content in waste type O.02 at the closure of SFR on 2075-12-31. Activity data refer to one mould.

Table E34-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	O.02/O.02:9 [Bq]	Nuclide	O.02/O.02:9 [Bq]	Nuclide	O.02/O.02:9 [Bq]
H-3	4.48E+05	Ag-108m	7.07E+06	U-235	2.61E+01
Be-10	7.85E+01	Cd-113m	4.82E+05	U-236	1.10E+03
C-14 org	5.51E+05	Sn-126	1.87E+04	U-238	3.53E+03
C-14 inorg	2.76E+07	Sb-125	1.11E+03	Np-237	3.99E+03
Cl-36	7.50E+04	I-129	1.08E+05	Pu-238	1.77E+06
Fe-55	4.56E+05	Cs-134	7.75E+02	Pu-239	6.89E+05
Co-60	9.99E+07	Cs-135	1.88E+05	Pu-240	9.64E+05
Ni-59	1.17E+08	Cs-137	5.37E+09	Pu-241	1.07E+07
Ni-63	6.75E+09	Ba-133	2.64E+04	Pu-242	4.97E+03
Se-79	1.49E+05	Pm-147	2.54E+04	Am-241	6.48E+06
Sr-90	2.16E+08	Sm-151	5.73E+07	Am-242m	1.20E+04
Zr-93	1.31E+05	Eu-152	4.92E+04	Am-243	6.91E+04
Nb-93m	9.41E+06	Eu-154	1.22E+07	Cm-243	4.89E+02
Nb-94	1.31E+06	Eu-155	3.46E+05	Cm-244	1.37E+05
Mo-93	3.71E+05	Ho-166m	5.04E+05	Cm-245	4.94E+02
Tc-99	2.17E+06	U-232	2.59E+01	Cm-246	1.31E+02
Pd-107	3.74E+04	U-234	1.66E+03		

E35 O.07/O.07:9

E35.1 Description of the waste type

The waste type O.07 consists of concrete tanks containing dewatered low-level ion exchange resins and filter aids from OKG. Small amounts of sludge from Clab can also occur.

There is a variant of the waste type, O.07:9. The differences between O.07 and O.07:9 mainly concern the design of the packaging. The differences are considered to be so small that the same data are used for O.07:9 as for O.07.

There are approved waste type descriptions for deposition of the waste type and the variant. Data are based on information in the waste type descriptions and Triumf NG v1.0.1.3.

The acceptance criteria for BTF, described in Section E1.4.1, apply for this waste type.

E35.1.1 Waste

The waste is well defined and consists of powder resins and inert filter aids from the systems for treatment of liquid waste (system 342), condensate clean-up (system 332) and fuel storage pool clean-up (system 324). Small amounts of sludge may also occur.

E35.1.2 Packaging

The waste is packed in concrete tanks with outer dimensions 3.3×1.3×2.3 m. The tank is made of reinforced concrete. The wall thickness is 15 cm and the reinforcing bars have a diameter of about 8 mm. The empty weight of the tank is about 11 tonnes, of which the concrete weighs about 10.3 tonnes and the reinforcing bars about 650 kg. The tank is internally lined with a 2 mm thick sack of butyl rubber that weighs about 50 kg. The tank is provided with a bolted reinforced lid with filling hole including a 50 mm steel plate that weighs about 1.7 tonnes.

The maximum permissible weight for a waste package including waste is 5,000 kg.

The disposal volume for a concrete tank is 10 m^3.

E35.1.3 Treatment

The waste is pumped in the form of slurry to the concrete tank where it is dewatered through the filter arrangement that each concrete tank is equipped with. The filter arrangement consists of two rows of sand-covered filter rods, located in the conical bottom of the concrete tank. The filter rods are connected to suction pipes. A filled concrete tank is provided with a 50 mm thick transport lid of steel. Some void will always occur at the top of the tank. The total fill volume of a package is about 5 m^3 compared with the inner volume of 6 m^3.

E35.1.4 Activity determination of radionuclides

Before the waste is transported from OKG to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The maximum measured activity content per package is 1.0 TBq. The highest permissible surface dose rate is 8 mSv/h. This limitation comes from the transport. The waste packages are usually free from surface contamination.

E35.1.5 Production of the waste type

Table E35-1 lists the number of packages for SFR.

Deposited packages refer to packages in the existing SFR as of 2012-12-31. The waste type O.07 started being produced in 1985 and has been deposited since 1988. The variant O.07:9 started being produced in 1976 and has been deposited since 1991.

Forecasted packages refer to waste that has not yet been deposited and includes both waste that is interim-stored at the waste supplier and waste that has not yet been produced. Currently, there are 3 packages in interim storage and a production is planned of 14 packages per year up to 2037. The waste vault that is indicated for forecasted waste is according to the acceptance criteria for the waste type.

Table E35-1. Number of packages of the waste type.

Number of packages	Waste vault	O.07	O.07:9
Deposited	1BTF	4	12
Deposited	2BTF	343	178
Forecasted	(BTF)	353	0

E35.2 Average package for the waste type

E35.2.1 Material – waste, packaging and matrix

Table E35-2 gives values for an estimated average of the material content in waste type O.07. The material data refer to one concrete tank. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E35-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	O.07/O.07:9
Filter aids [kg]	Waste	185
Ion exchange resins [kg]	Waste	1,000
Sludge [kg]	Waste	60
Other organic [kg]	Waste	66
Concrete [kg]	Packaging	10,350
Iron/steel [kg]	Packaging	2,333
Iron/steel surface [m^2]	Packaging	49
Iron/steel thickness [mm]	Packaging (transport lid)	50
Iron/steel thickness [mm]	Packaging (reinforcing bar)	8.0
Other organic [kg]	Packaging	50
Void [m^3]	Matrix	1.0

E35.2.2 Radionuclide content

Table E35-3 provides values for a calculated average of the nuclide content in waste type O.07 at the closure of SFR on 2075-12-31. Activity data refer to one concrete tank.

Table E35-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	O.07/O.07:1 [Bq]	Nuclide	O.07/O.07:1 [Bq]	Nuclide	O.07/O.07:1 [Bq]
H-3	1.73E+05	Ag-108m	2.91E+06	U-235	1.08E+01
Be-10	3.25E+01	Cd-113m	1.13E+05	U-236	4.56E+02
C-14 org	8.37E+06	Sn-126	3.34E+03	U-238	1.46E+03
C-14 inorg	4.30E+08	Sb-125	1.99E+04	Np-237	1.66E+03
Cl-36	1.59E+04	I-129	1.93E+04	Pu-238	7.11E+05
Fe-55	2.19E+05	Cs-134	1.70E+02	Pu-239	2.85E+05
Co-60	4.42E+07	Cs-135	3.48E+04	Pu-240	3.99E+05
Ni-59	4.83E+07	Cs-137	1.08E+09	Pu-241	4.08E+06
Ni-63	2.72E+09	Ba-133	1.03E+04	Pu-242	2.06E+03
Se-79	2.67E+04	Pm-147	8.77E+03	Am-241	2.68E+06
Sr-90	8.32E+07	Sm-151	1.06E+07	Am-242m	4.88E+03
Zr-93	5.42E+04	Eu-152	1.17E+04	Am-243	2.86E+04
Nb-93m	3.59E+06	Eu-154	3.33E+06	Cm-243	1.88E+02
Nb-94	5.40E+05	Eu-155	1.11E+05	Cm-244	5.21E+04
Mo-93	1.98E+05	Ho-166m	2.08E+05	Cm-245	2.05E+02
Tc-99	9.32E+05	U-232	1.03E+01	Cm-246	5.43E+01
Pd-107	6.68E+03	U-234	6.86E+02		

E36 O.12

E36.1 Description of the waste type

The waste type O.12 consists of steel containers containing low-level solid waste in the form of trash and scrap metal from OKG.

There is a variant of the waste type, O.12:1. The difference between O.12 and O.12:1 is that in the variant, parts of moulds are packed with solidified evaporator concentrates, yielding a different material content than in O.12.

There are approved waste type descriptions for deposition of the waste type and the variant. Data are based on information in the waste type descriptions and Triumf NG v1.0.1.3.

The acceptance criteria for BLA, described in Section E1.5.1, apply for this waste type.

E36.1.1 Waste

The waste consists of trash and scrap metal. The trash consists of compacted or non-compacted garbage bags containing e.g. textiles, paper, insulation, small pieces of aluminium, copper and plastics. The scrap metal consists of e.g. fittings, scrapped components, cables, suspensions and insulation. The mixture of waste materials has changed over time depending on the maintenance work, revisions or other work carried out. In O.12:1, parts from three moulds with solidified evaporator concentrates are included.

E36.1.2 Packaging

The waste is packed in ISO containers of carbon steel with dimensions 20-foot half height, 20-foot full height or 10-foot full height.

The 20-foot half-height container has a length of 6.1 m, a width of 2.5 m and a height of 1.3 m. The thickness of the walls and roof is normally about 1.5 mm. An empty container weighs about 1,900 kg.

The 20-foot full-height container has a height of 2.6 m but otherwise the same geometry as the half-height container. It has an empty weight of 2,200 kg. The floor of the full-height container can consist of about 15–30 mm of plywood with load-bearing steel construction. The plywood floor weighs about 310 kg.

The 10-foot full-height container has a length of 3 m, a width of 2.5 m and a height of 2.6 m. The thickness of the walls and roof is normally about 1.5 mm.

Open containers are sealed with a lid.

The maximum permissible weight for a 20-foot container including waste is 20 tonnes and the maximum permissible weight for a 10-foot container is 10 tonnes. The disposal volume is 20 m^3 or 40 m^3.

E36.1.3 Treatment

Compactible waste is compacted into bales that are placed in the container. Non-combustible or non-compactible waste is placed in Berglöf boxes without further treatment or directly in the container. Locking ring drums, steel boxes, paper boxes, garbage bags and plastic also occur as inner packaging.

As high a fill volume as possible should always be targeted; it can, however, vary widely depending on the nature of the waste. The void in a waste package is assumed to be 7.5 m^3 for all types of packaging, even though there is likely to be more void in the larger packaging.

E36.1.4 Activity determination of radionuclides

Before the waste is transported from OKG to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The usually measured activity content is about 25 GBq/container. The highest permissible surface dose rate is 2 mSv/h. The waste packages are usually free from surface contamination.

E36.1.5 Production of the waste type

Table E36-1 lists the number of packages for SFR.

Deposited packages refer to packages in the existing SFR as of 2012-12-31. The waste type O.12 started being produced in 2006 and has been deposited since 2006. The variant O.12:1 has not yet been deposited.

Forecasted packages refer to waste that has not yet been deposited and includes both waste that is interim-stored at the waste supplier and waste that has not yet been produced. Currently, there are 35 packages in interim storage, of which 21 are O.12 20-foot half height, 8 are O.12 10-foot full height, 5 are O.12 20-foot full height and 1 is O.12:1. A production is planned of 2 packages per year up to 2037, of which 1 is O.12 20-foot half height and 1 is O.12 10-foot full height. The waste vault that is indicated for forecasted waste is according to the acceptance criteria for the waste type.

Table E36-1. Number of packages of the waste type.

Number of packages	Waste vault	O.12 20-foot half height	O.12 10-foot full height	O.12 20-foot full height	O.12:1 20-foot half height
Deposited	1BLA	0	1	5	0
Forecasted	(BLA)	46	33	5	1

E36.2 Average package for the waste type

E36.2.1 Material – waste, packaging and matrix

Table E36-2 gives values for an estimated average of the material content in waste type O.12. The material data refer to one container. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E36-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	O.12 20-foot half height/10-foot full height	O.12 20-foot full height	O.12 20-foot half height
Aluminium/zinc [kg]	Waste	100	100	100
Aluminium/zinc surface [m^2]	Waste	15	15	15
Aluminium/zinc thickness [mm]	Waste	5.0	5.0	5.0
Concrete [kg]	Waste	–	–	1,170
Cellulose [kg]	Waste	500	500	500
Evaporator concentrates [kg]	Waste	–	–	270
Iron/steel [kg]	Waste	4,500	4,500	4,754
Iron/steel surface [m^2]	Waste	229	229	242
Iron/steel thickness [mm]	Waste	5.0	5.0	5.0
Other inorganic [kg]	Waste	–	400	–
Other organic [kg]	Waste	3,000	3,000	3,000
Cellulose [kg]	Packaging	–	310	–
Iron/steel [kg]	Packaging	1,900	2,200	1,900
Iron/steel surface [m^2]	Packaging	105	150	105
Iron/steel thickness [mm]	Packaging	1.5	1.5	1.5
Void [m^3]	Matrix	7.5	7.5	7.5

E36.2.2 Radionuclide content

Table E36-3 provides values for a calculated average of the nuclide content in waste type O.12 at the closure of SFR on 2075-12-31. Activity data refer to one container.

Table E36-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	O.12/O.12:1 20-foot half height [Bq]	O.12 10-foot full height [Bq]	O.12 20-foot full height [Bq]
H-3	3.33E+04	6.49E+01	3.94E+03
Be-10	4.20E+00	7.20E–03	9.50E–01
C-14 org	0.00E+00	0.00E+00	0.00E+00
C-14 inorg	0.00E+00	0.00E+00	0.00E+00
Cl-36	4.20E+03	7.20E+00	9.50E+02
Fe-55	3.78E+04	8.90E+01	1.17E+02
Co-60	8.21E+06	1.82E+04	2.96E+05
Ni-59	5.13E+06	8.79E+03	1.16E+06
Ni-63	2.80E+08	4.88E+05	5.92E+07
Se-79	3.24E+02	1.80E+02	1.15E+03
Sr-90	4.53E+06	8.22E+03	7.95E+05
Zr-93	7.00E+03	1.20E+01	1.58E+03
Nb-93m	6.68E+05	1.26E+03	9.37E+04
Nb-94	6.99E+04	1.20E+02	1.58E+04
Mo-93	6.93E+03	1.35E+01	5.12E+03
Tc-99	2.91E+04	4.41E+03	3.44E+04
Pd-107	8.10E+01	4.50E+01	2.88E+02
Ag-108m	3.84E+05	6.61E+02	8.56E+04
Cd-113m	3.35E+03	2.08E+03	6.81E+03
Sn-126	4.05E+01	2.25E+01	1.44E+02
Sb-125	4.22E+03	9.93E+00	1.28E+00
I-129	2.43E+02	1.35E+02	8.44E+02
Cs-134	1.46E+01	1.12E+01	6.93E–04
Cs-135	8.10E+02	4.42E+02	1.99E+03
Cs-137	2.26E+07	1.32E+07	6.30E+07
Ba-133	2.01E+03	4.01E+00	2.08E+02

Nuclide	O.12/O.12:1 20-foot half height [Bq]	O.12 10-foot full height [Bq]	O.12 20-foot full height [Bq]
Pm-147	2.55E+02	1.95E+02	9.96E+00
Sm-151	1.58E+05	8.93E+04	5.19E+05
Eu-152	3.51E+02	2.19E+02	6.95E+02
Eu-154	1.11E+05	7.35E+04	1.45E+05
Eu-155	3.47E+03	2.53E+03	1.52E+03
Ho-166m	2.71E+04	4.65E+01	6.10E+03
U-232	6.91E–01	1.21E–03	1.41E–01
U-234	4.04E+01	6.92E–02	9.13E+00
U-235	8.08E–01	1.39E–03	1.83E–01
U-236	1.22E+01	2.08E–02	2.75E+00
U-238	1.62E+01	2.77E–02	3.65E+00
Np-237	1.84E+01	3.13E–02	4.27E+00
Pu-238	5.69E+04	9.93E+01	1.19E+04
Pu-239	1.68E+04	2.88E+01	3.80E+03
Pu-240	2.36E+04	4.04E+01	5.33E+03
Pu-241	3.52E+05	6.74E+02	4.62E+04
Pu-242	1.21E+02	2.08E–01	2.74E+01
Am-241	1.60E+05	2.73E+02	3.67E+04
Am-242m	3.07E+02	5.31E–01	6.60E+01
Am-243	2.29E+03	3.93E+00	5.18E+02
Cm-243	2.16E+02	3.91E–01	3.80E+01
Cm-244	7.60E+03	1.42E+01	1.13E+03
Cm-245	1.21E+01	2.07E–02	2.73E+00
Cm-246	3.20E+00	5.49E–03	7.24E–01

E37 O.12:D/O.12C:D/O.12S:D

E37.1 Description of the waste type

Waste types O.12:D, O.12C:D and O.12S:D are waste types adopted for low-level decommissioning waste in steel containers from OKG. O.12:D contains scrap metal or secondary waste. O.12C:D contains concrete and O.12S:D contains sand.

There is no approved waste type description for deposition of these waste types. Material quantities and activity have been calculated based on Larsson et al. (2013), supplemented with assumptions about secondary waste and material composition for the decommissioning waste, packaging and solidification material.

The acceptance criteria for BLA, described in Section E1.5.1, are assumed to be valid for these waste types.

E37.1.1 Waste

The scrap metal in O.12:D consists mainly of fittings and scrapped components. The secondary waste in O.12:D is assumed to consist of trash and scrap metal like waste type R.12 from operational waste. The waste in O.12C:D consists of concrete from the outer parts of the biological shield and contaminated concrete from the controlled area in the facility. The waste in O.12S:D consists of sand from the sand beds in the gas treatment system 341.

E37.1.2 Packaging

The waste is packed in ISO containers with dimensions 20-foot half height. The container is made of steel with a length of 6.06 m, a width of 2.5 m and a height of 1.3 m. The thickness of the walls and roof is normally 1.5 mm. An empty container weighs about 1,900 kg. Open containers are sealed with a lid.

The maximum permissible weight for a container including waste is 20 tonnes. The disposal volume is 20 m^3.

E37.1.3 Treatment

A container O.12:D is assumed to be filled with about 16,500 kg of scrap metal. The secondary waste is assumed to be treated like waste type R.12 from operational waste. A container O.12C:D is assumed to be filled with about 17,800 kg of concrete waste and O.12S:D is assumed to be filled with about 17,600 kg of sand.

E37.1.4 Activity determination of radionuclides

Before the waste is transported from OKG to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. The radioactivity in O.12S:D system 341 is during manufacturing of the waste package dominated by the long-lived noble gas daughters Sr-90, Cs-135 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The highest permissible surface dose rate is 2 mSv/h. The waste packages are assumed to be free from surface contamination.

E37.1.5 Production of the waste type

Table E37-1 lists the number of packages for SFR.

The waste will be deposited during the years 2032–2039 and 2045–2052. The waste vault given is according to the acceptance criteria for the waste types.

Table E37-1. Number of packages of the waste type.

Number of packages	Waste vault	O.12:D scrap metal	O.12:D secondary waste	O.12C:D	O.12S:D
Forecasted	(BLA)	382	75	160	37

E37.2 Average package for the waste type

E37.2.1 Material – waste, packaging and matrix

Table E37-2 gives values for an estimated average of the material content in waste types O.12:D, O.12C:D and O.12S:D. The material data refer to one container. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E37-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	O.12:D scrap metal	O.12:D secondary waste	O.12C:D	O.12S:D
Aluminium/zinc [kg]	Waste	–	100	–	–
Aluminium/zinc surface [m^2]	Waste	–	15	–	–
Aluminium/zinc thickness [mm]	Waste	–	5.0	–	–
Concrete [kg]	Waste	–	–	17,759	–
Cellulose [kg]	Waste	–	500	–	–
Iron/steel [kg]	Waste	16,455	4,500	–	–
Iron/steel surface [m^2]	Waste	844	229	–	–
Iron/steel thickness [mm]	Waste	5.0	5.0	–	–
Sand [kg]	Waste	–	–	–	17,605
Other inorganic [kg]	Waste	–	400	–	–
Other organic [kg]	Waste	–	3,000	–	–
Iron/steel [kg]	Packaging	1,900	1,900	1,900	1,900
Iron/steel surface [m^2]	Packaging	105	105	105	105
Iron/steel thickness [mm]	Packaging	1.5	1.5	1.5	1.5

E37.2.2 Radionuclide content

Table E37-3 provides values for a calculated average of the nuclide content in waste types O.12:D, O.12C:D and O.12S:D at the closure of SFR on 2075-12-31. Activity data refer to one container.

Table E37-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	O.12:D [Bq]	O.12C:D [Bq]	O.12S:D [Bq]
H-3	0.00E+00	2.66E+08	0.00E+00
Be-10	0.00E+00	5.31E–01	0.00E+00
C-14 org	3.44E+03	4.38E+03	0.00E+00
C-14 inorg	1.12E+05	2.05E+05	0.00E+00
C-14 ind	0.00E+00	7.60E+05	0.00E+00
Cl-36	1.74E+00	2.32E+04	0.00E+00
Ca-41	0.00E+00	2.28E+06	0.00E+00
Fe-55	3.45E+05	1.47E+05	0.00E+00
Co-60	2.27E+07	6.38E+06	0.00E+00
Ni-59	9.54E+06	6.69E+05	0.00E+00
Ni-63	9.99E+08	6.69E+07	0.00E+00
Se-79	1.93E–01	1.51E+00	0.00E+00
Sr-90	2.70E+06	3.50E+05	1.66E+08
Zr-93	5.06E+03	2.08E+03	0.00E+00
Nb-93m	7.51E+07	2.35E+07	0.00E+00
Nb-94	3.59E+05	1.42E+05	0.00E+00
Mo-93	2.57E+03	6.86E+03	0.00E+00
Tc-99	5.95E+03	1.96E+04	0.00E+00
Pd-107	3.01E–01	1.18E+00	0.00E+00
Ag-108m	3.56E+04	4.25E+05	0.00E+00
Cd-113m	8.04E+00	3.24E+02	0.00E+00
Sn-126	7.50E+01	5.58E+00	0.00E+00
Sb-125	2.74E+03	1.10E+02	0.00E+00
I-129	8.36E+01	7.70E+01	8.94E+03
Cs-134	1.34E+02	8.62E+02	0.00E+00
Cs-135	2.40E+03	6.40E+02	9.20E+05
Cs-137	9.39E+06	2.03E+07	2.95E+09
Ba-133	4.38E–03	1.83E+04	0.00E+00
Pm-147	3.25E+02	2.61E+02	0.00E+00
Sm-151	3.68E+04	3.96E+06	0.00E+00
Eu-152	6.32E+01	2.24E+07	0.00E+00
Eu-154	1.37E+04	3.52E+05	0.00E+00
Eu-155	4.29E+02	1.60E+04	0.00E+00

Nuclide	O.12:D [Bq]	O.12C:D [Bq]	O.12S:D [Bq]
Ho-166m	1.13E+00	5.50E+04	0.00E+00
U-232	6.80E–01	2.78E–02	0.00E+00
U-235	1.33E–03	1.51E–04	0.00E+00
U-236	3.89E+01	1.23E+00	0.00E+00
Np-237	4.65E+01	1.48E+00	0.00E+00
Pu-238	2.75E+05	1.48E+04	0.00E+00
Pu-239	4.12E+04	4.31E+03	0.00E+00
Pu-240	5.64E+04	3.47E+03	0.00E+00
Pu-241	1.12E+06	9.60E+04	0.00E+00
Pu-242	3.06E+02	9.49E+00	0.00E+00
Am-241	1.87E+05	1.43E+04	0.00E+00
Am-242m	1.13E+03	3.88E+01	0.00E+00
Am-243	4.17E+03	1.26E+02	0.00E+00
Cm-243	6.95E+02	3.71E+01	0.00E+00
Cm-244	8.42E+04	5.96E+03	0.00E+00
Cm-245	6.16E+01	1.79E+00	0.00E+00
Cm-246	2.47E+01	7.02E–01	0.00E+00

E38 O.16:D

E38.1 Description of the waste type

The waste type O.16:D is a waste type adopted for decommissioning waste from OKG. It consists of steel moulds containing cement-solidified intermediate-level ion exchange resins from system decontamination.

There is no approved waste type description for deposition of this waste type. Material quantities and activity have been calculated based on Larsson et al. (2013), supplemented with assumptions about packaging and solidification material.

The acceptance criteria for Silo, described in Section E1.1.1, are assumed to be valid for the waste type.

E38.1.1 Waste

The waste consists of ion exchange resins produced at system decontamination prior to decommissioning.

E38.1.2 Packaging

The waste is packed in steel moulds. The mould is a cubic box made of carbon steel with dimensions 1.2×1.2×1.2 m. The wall is 5 mm thick and the bottom is 6 mm thick. The mould weighs about 400 kg. The mould contains a stirrer of carbon steel. It weighs about 25 kg. The mould is also provided with a splash plate.

The maximum permissible weight for a mould including waste is 5,000 kg. The disposal volume is 1.728 m^3.

E38.1.3 Treatment

The mould is assumed to be filled with equal parts ion exchange resin and cement. The waste matrix is homogenised with the aid of the stirrer. The void of a package is assumed to be about 10%. The matrix is allowed to harden before a steel lid is placed on the mould.

E38.1.4 Activity determination of radionuclides

Before the waste is transported from OKG to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The highest permissible surface dose rate is 500 mSv/h. The waste packages are assumed to be free from surface contamination.

E38.1.5 Production of the waste type

Table E38-1 lists the number of packages for SFR.

The waste will be deposited during the years 2033, 2036, 2046 and 2051. The waste vault given is according to the acceptance criteria for the waste type.

Table E38-1. Number of packages of the waste type.

Number of packages	Waste vault	O.16:D
Forecasted	(Silo)	28

E38.2 Average package for the waste type

E38.2.1 Material – waste, packaging and matrix

Table E38-2 gives values for an estimated average of the material content in waste type O.16:D. The material data refer to one mould. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E38-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	O.16:D
Ion exchange resins [kg]	Waste	803
Iron/steel [kg]	Packaging	400
Iron/steel surface [m^2]	Packaging	17
Iron/steel thickness [mm]	Packaging	5.0–6.0
Cement [kg]	Matrix	1,836
Iron/steel [kg]	Matrix	25
Iron/steel surface [m^2]	Matrix	3.0
Iron/steel thickness [mm]	Matrix	5.0
Void [m^3]	Matrix	0.17

E38.2.2 Radionuclide content

Table E38-3 provides values for a calculated average of the nuclide content in waste type O.16:D at the closure of SFR on 2075-12-31. Activity data refer to one mould.

Table E38-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	O.16:D [Bq]	Nuclide	O.16:D [Bq]	Nuclide	O.16:D [Bq]	Nuclide	O.16:D [Bq]
H-3	0.00E+00	Nb-93m	1.18E+10	Pm-147	2.43E+04	Pu-242	5.94E+04
Be-10	0.00E+00	Nb-94	8.39E+07	Sm-151	7.12E+06	Am-241	4.02E+07
C-14 org	1.32E+04	Mo-93	5.30E+05	Eu-152	9.38E+03	Am-242m	3.21E+05
C-14 inorg	8.93E+05	Tc-99	1.07E+06	Eu-154	1.82E+06	Am-243	6.82E+05
C-14 ind	0.00E+00	Pd-107	4.08E+01	Eu-155	4.82E+04	Cm-243	1.25E+05
Cl-36	2.51E+02	Ag-108m	5.96E+06	Ho-166m	1.30E+02	Cm-244	1.11E+07
Ca-41	0.00E+00	Cd-113m	1.13E+03	U-232	1.28E+02	Cm-245	9.74E+03
Fe-55	2.10E+07	Sn-126	1.51E+04	U-235	3.40E-01	Cm-246	3.29E+03
Co-60	1.82E+09	Sb-125	1.88E+05	U-236	7.96E+03		
Ni-59	2.27E+09	I-129	1.64E+03	Np-237	9.39E+03		
Ni-63	2.26E+11	Cs-134	1.97E+04	Pu-238	5.10E+07		
Se-79	2.61E+01	Cs-135	1.98E+04	Pu-239	9.04E+06		
Sr-90	4.60E+08	Cs-137	3.93E+08	Pu-240	1.35E+07		
Zr-93	1.47E+06	Ba-133	5.79E-01	Pu-241	1.68E+08		

E39 O.23/O.23:9

E39.1 Description of the waste type

The waste type O.23 consists of concrete moulds containing concrete-solidified intermediate-level trash and scrap metal from OKG. Previously, waste from Clab has also been included in this waste type. In order to distinguish the waste owned by Clab, it is denoted with the letter C in the present report and is described in Chapter 16.

There is a variant of the waste type, O.23:9. The difference between O.23 and O.23:9 is mainly the design of the waste packaging. The differences are considered to be so small that the same data are used for O.23:9 as for O.23.

There are approved waste type descriptions for deposition of the waste type and the variant. Data are based on information in the waste type descriptions and Triumf NG v1.0.1.3.

The acceptance criteria for BMA, described in Section E1.3.1, apply for this waste type.

E39.1.1 Waste

The waste consists of trash and scrap metal that arise in connection with service and maintenance of active systems. This type of work is carried out continuously, but most of the waste is generated during the annual revision shutdowns. The trash consists of, among other things, plastics, rags, packaging, concrete and blasting sand. The scrap metal consists of e.g. valves, fittings, gaskets and filters.

E39.1.2 Packaging

The waste is packed in concrete moulds. The mould is a cubic box made of reinforced concrete with dimensions 1.2×1.2×1.2 m. The walls are normally 10 cm thick, but can in some exceptional cases be 25 cm thick. The reinforcement consists of 12 mm steel bars with a total weight of 274 kg. The 10-cm mould has an empty weight of about 1,600 kg. The 25-cm mould has an empty weight of about 3,100 kg.

The maximum permissible weight for a mould including waste is 5,000 kg. The disposal volume is 1.728 m^3.

E39.1.3 Treatment

The waste is placed directly in moulds and then solidified in concrete. The void in a package is assumed to be about 25%. After two days of hardening, the mould is provided with a lid. This is cast in place by further addition of concrete grout.

E39.1.4 Activity determination of radionuclides

Before the waste is transported from OKG to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The usually measured activity content is about 100–500 GBq. The highest permissible surface dose rate is 30 mSv/h, which is based on the manufacturing of the waste package. The waste packages are usually free from surface contamination.

E39.1.5 Production of the waste type

Table E39-1 lists the number of packages for SFR.

Deposited packages refer to packages in the existing SFR as of 2012-12-31. The waste type O.23 started being produced in 1981 and has been deposited since 1993. The variant O.23:9 started being produced in 1975 and has been deposited since 1993.

Forecasted packages refer to waste that has not yet been deposited and includes both waste that is interim-stored at the waste supplier and waste that has not yet been produced. Currently, there are 29 packages in interim storage and a production is planned of 5 packages per year up to 2037. The waste vault that is indicated for forecasted waste is according to the acceptance criteria for the waste type.

Table E39-1. Number of packages of the waste type.

Number of packages	Waste vault	O.23	O.23:9
Deposited	1BMA	324	131
Forecasted	(BMA)	154	0

E39.2 Average package for the waste type

E39.2.1 Material – waste, packaging and matrix

Table E39-2 gives values for an estimated average of the material content in waste type O.23. The material data refer to one mould. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E39-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	O.23/O.23:9
Aluminium/zinc [kg]	Waste	3.5
Aluminium/zinc surface [m^2]	Waste	0.5
Aluminium/zinc [mm]	Waste	5.0
Cellulose [kg]	Waste	30
Iron/steel [kg]	Waste	105
Iron/steel surface [m^2]	Waste	5.3
Iron/steel thickness [mm]	Waste	5.0
Sludge [kg]	Waste	53
Other inorganic [kg]	Waste	18
Other organic [kg]	Waste	67
Concrete [kg]	Packaging (including lid)	1,840
Iron/steel [kg]	Packaging (reinforcing bars)	274
Iron/steel surface [m^2]	Packaging (reinforcing bars)	12
Iron/steel thickness [mm]	Packaging (reinforcing bars)	12
Concrete [kg]	Matrix	565
Void [m^3]	Matrix	0.25

E39.2.2 Radionuclide content

Table E39-3 provides values for a calculated average of the nuclide content in waste type O.23 at the closure of SFR on 2075-12-31. Activity data refer to one mould.

Table E39-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	O.23/O.23:9 [Bq]
H-3	2.52E+04
Be-10	6.05E+00
C-14 org	0.00E+00
C-14 inorg	0.00E+00
Cl-36	6.05E+03
Fe-55	2.44E+04
Co-60	5.20E+06
Ni-59	7.39E+06
Ni-63	3.62E+08
Se-79	4.81E+03
Sr-90	4.63E+06
Zr-93	1.01E+04
Nb-93m	5.58E+05
Nb-94	1.01E+05
Mo-93	4.18E+04
Tc-99	1.87E+05
Pd-107	1.20E+03
Ag-108m	5.40E+05
Cd-113m	1.52E+04
Sn-126	6.02E+02
Sb-125	2.38E+02
I-129	3.47E+03
Cs-134	3.55E+00
Cs-135	5.82E+03
Cs-137	1.70E+08
Ba-133	1.44E+03

Nuclide	O.23/O.23:9 [Bq]
Pm-147	9.44E+02
Sm-151	1.84E+06
Eu-152	1.55E+03
Eu-154	3.94E+05
Eu-155	1.22E+04
Ho-166m	3.87E+04
U-232	8.49E–01
U-234	5.82E+01
U-235	1.17E+00
U-236	1.75E+01
U-238	2.33E+01
Np-237	2.78E+01
Pu-238	7.24E+04
Pu-239	2.42E+04
Pu-240	3.39E+04
Pu-241	2.82E+05
Pu-242	1.75E+02
Am-241	2.32E+05
Am-242m	4.08E+02
Am-243	3.30E+03
Cm-243	2.20E+02
Cm-244	6.67E+03
Cm-245	1.74E+01
Cm-246	4.61E+00

E40 O.4K23:D/O.4K23C:D/O.4K23S:D

E40.1 Description of the waste type

Waste types O.4K23:D, O.4K23C:D and O.4K23S:D are waste types adopted for low-level decommissioning waste from OKG. O.4K23:D consists of tetramoulds containing cement-solidified scrap metal. O.4K23C:D consists of tetramoulds containing concrete-solidified concrete and O.4K23S:D consists of tetramoulds containing concrete-solidified sand.

There is no approved waste type description for deposition of these waste types. Material quantities and activity have been calculated based on Larsson et al. (2013), supplemented with assumptions about material composition and packaging and solidification material.

The acceptance criteria for BMA, described in Section E1.3.1, are assumed to be valid for the waste types.

E40.1.1 Waste

The waste in O.4K23:D consists mainly of scrap metal in the form of fittings and scrapped components. The waste in O.4K23C:D consists of parts from the biological shield that have been close to the core. The waste in O.4K23S:D consists of sand from the sand beds in gas treatment system 341.

E40.1.2 Packaging

The waste is packed in tetramoulds. The tetramould is a mould of steel plate with outer dimensions 2.4×2.4×1.2 m. The thickness of the walls is 4 mm, the floor 8 mm and the lid 15 mm. The packaging weighs about 1,700 kg.

The maximum permissible weight for a tetramould including waste is 20 tonnes. The disposal volume is 6.912 m^3.

E40.1.3 Treatment

A tetramould O.4K23:D or O.4K23C:D is filled with about 5,500 kg of waste and a tetramould O.4K23S:D is filled with about 7,100 kg of waste. The waste is solidified with concrete. The void is estimated to be 25% of the inner volume of the packaging.

E40.1.4 Activity determination of radionuclides

Before the waste is transported from OKG to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The highest permissible surface dose rate is 100 mSv/h. The waste packages are assumed to be free from surface contamination.

E40.1.5 Production of the waste type

Table E40-1 lists the number of packages for SFR.

The waste will be deposited during the years 2032–2039 and 2045–2052. The waste vault given is to the acceptance criteria for the waste types.

Table E40-1. Number of packages of the waste type.

Number of packages	Waste vault	O.4K23:D	O.4K23C:D	O.4K23S:D
Forecasted	(BMA)	198	82	15

E40.2 Average package for the waste type

E40.2.1 Material – waste, packaging and matrix

Table E40-2 gives values for an estimated average of the material content in waste types O.4K23:D, O.4K23C:D and O.4K23S:D. The material data refer to one tetramould. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E40-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	O.4K23:D	O.4K23C:D	O.4K23S:D
Concrete [kg]	Waste	–	9,917	–
Iron/steel [kg]	Waste	5,534	–	–
Iron/steel surface [m^2]	Waste	284	–	–
Iron/steel thickness [mm]	Waste	5.0	–	–
Sand [kg]	Waste	–	–	7,073
Iron/steel [kg]	Packaging	1,722	1,722	1,722
Iron/steel surface [m^2]	Packaging	46	46	46
Iron/steel thickness [mm]	Packaging	4.0–15	4.0–15	4.0–15
Concrete [kg]	Matrix	9,997	1,783	675
Void [m^3]	Matrix	1.6	1.6	1.6

E40.2.2 Radionuclide content

Table E40-3 provides values for a calculated average of the nuclide content in waste types O.4K23:D, O.4K23C:D and O.4K23S:D at the closure of SFR on 2075-12-31. Activity data refer to one tetramould.

Table E40-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	O.4K23:D [Bq]	O.4K23C:D [Bq]	O.4K23S:D [Bq]
H-3	1.53E+02	9.35E+09	0.00E+00
Be-10	0.00E+00	2.13E+01	0.00E+00
C-14 org	4.18E+01	0.00E+00	0.00E+00
C-14 inorg	1.09E+03	0.00E+00	0.00E+00
C-14 ind	3.21E+05	3.10E+07	0.00E+00
Cl-36	5.58E+04	9.28E+05	0.00E+00
Ca-41	1.50E+06	9.13E+07	0.00E+00
Fe-55	2.90E+07	2.66E+06	0.00E+00
Co-60	1.94E+09	3.45E+07	0.00E+00
Ni-59	1.27E+09	6.39E+06	0.00E+00
Ni-63	1.30E+11	4.77E+08	0.00E+00
Se-79	3.02E+00	3.05E+01	0.00E+00
Sr-90	3.01E+08	1.40E+05	3.52E+08
Zr-93	7.50E+05	6.35E+02	0.00E+00
Nb-93m	8.12E+09	7.03E+06	0.00E+00
Nb-94	4.70E+07	3.11E+05	0.00E+00
Mo-93	3.29E+05	2.90E+05	0.00E+00
Tc-99	4.88E+05	5.56E+04	0.00E+00
Pd-107	4.51E+00	0.00E+00	0.00E+00
Ag-108m	3.98E+06	1.69E+07	0.00E+00
Cd-113m	4.28E+01	1.03E+04	0.00E+00
Sn-126	9.28E+03	0.00E+00	0.00E+00
Sb-125	2.25E+05	3.35E+01	0.00E+00
I-129	1.38E+03	0.00E+00	5.94E+03
Cs-134	9.58E+03	1.20E+03	0.00E+00
Cs-135	7.34E+03	0.00E+00	9.65E+05

Nuclide	O.4K23:D [Bq]	O.4K23C:D [Bq]	O.4K23S:D [Bq]
Cs-137	5.67E+08	1.53E+05	3.09E+09
Ba-133	1.04E–01	6.27E+05	0.00E+00
Pm-147	2.26E+04	6.26E+02	0.00E+00
Sm-151	4.34E+06	1.56E+08	0.00E+00
Eu-152	6.57E+03	7.98E+08	0.00E+00
Eu-154	1.30E+06	1.14E+07	0.00E+00
Eu-155	3.65E+04	4.19E+05	0.00E+00
Ho-166m	1.08E+02	2.20E+06	0.00E+00
U-232	8.17E+01	0.00E+00	0.00E+00
U-235	1.84E–01	2.57E–03	0.00E+00
U-236	4.89E+03	0.00E+00	0.00E+00
Np-237	5.67E+03	0.00E+00	0.00E+00
Pu-238	3.19E+07	0.00E+00	0.00E+00
Pu-239	5.23E+06	6.65E+04	0.00E+00
Pu-240	7.65E+06	0.00E+00	0.00E+00
Pu-241	1.17E+08	0.00E+00	0.00E+00
Pu-242	3.72E+04	0.00E+00	0.00E+00
Am-241	2.31E+07	0.00E+00	0.00E+00
Am-242m	1.54E+05	0.00E+00	0.00E+00
Am-243	4.68E+05	0.00E+00	0.00E+00
Cm-243	7.86E+04	0.00E+00	0.00E+00
Cm-244	8.37E+06	0.00E+00	0.00E+00
Cm-245	6.65E+03	0.00E+00	0.00E+00
Cm-246	2.48E+03	0.00E+00	0.00E+00

E41 O.24

E41.1 Description of the waste type

The waste type O.24 consists of concrete moulds containing cement-solidified intermediate-level solid waste in the form of components and scrap metal of steel from OKG.

There is no approved waste type description for deposition of this waste type. Data are based on information in Triumf NG v1.0.1.3.

The acceptance criteria for Silo, described in Section E1.1.1, apply for this waste type.

E41.1.1 Waste

The waste consists mainly of components and scrap metal of steel, steel alloys or other materials containing activity, e.g. valves or filters for clean-up of water and air.

E41.1.2 Packaging

The waste is packed in steel moulds. The mould is a cubic box of reinforced sheet metal with dimensions 1.2×1.2×1.2 m. The thickness is 5 mm in the walls and lid and 8 mm in the bottom. The mould has a bolted lid. The mould weighs about 575 kg and the lid about 45 kg.

The maximum permissible weight for a mould including waste is 5,000 kg. The disposal volume is 1.728 m^3.

E41.1.3 Treatment

When the mould has been filled with waste including any inner packaging, it is solidified with concrete. The void is assumed to be about 25% of the inner volume of a packaging.

E41.1.4 Activity determination of radionuclides

Before the waste is transported from OKG to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The measured activity content is normally 100–500 GBq. The highest permissible surface dose rate is 300 mSv/h. The limitation comes from the manufacturing of the waste package. The waste packages are usually free from surface contamination.

E41.1.5 Production of the waste type

Table E41-1 lists the number of packages for SFR.

There are no deposited packages in the existing SFR as of 2012-12-31.

Forecasted packages refer to waste that is interim-stored at the waste supplier. No future production of the waste type is planned. The waste vault given is according to the acceptance criteria for the waste type.

Table E41-1. Number of packages of the waste type.

Number of packages	Waste vault	O.24
Deposited	–	0
Forecasted	(Silo)	204

E41.2 Average package for the waste type

E41.2.1 Material – waste, packaging and matrix

Table E41-2 gives values for an estimated average of the material content in waste type O.24. The material data refer to one mould. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E41-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	O.24
Concrete [kg]	Waste	200
Cellulose [kg]	Waste	5.0
Iron/steel [kg]	Waste	1,500
Iron/steel surface [m^2]	Waste	76
Iron/steel thickness [mm]	Waste	5.0
Other inorganic [kg]	Waste	1,230
Other organic [kg]	Waste	15
Iron/steel [kg]	Packaging	620
Iron/steel surface [m^2]	Packaging	17
Iron/steel thickness [mm]	Packaging	5.0–8.0
Concrete [kg]	Matrix	1,356
Void [m^3]	Matrix	0.43

E41.2.2 Radionuclide content

Table E41-3 provides values for a calculated average of the nuclide content in waste type O.24 at the closure of SFR on 2075-12-31. Activity data refer to one mould.

Table E41-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	O.24 [Bq]
H-3	4.46E+05
Be-10	9.00E+01
C-14 org	0.00E+00
C-14 inorg	0.00E+00
Cl-36	9.00E+04
Fe-55	1.90E+04
Co-60	4.02E+07
Ni-59	1.10E+08
Ni-63	5.74E+09
Se-79	2.60E+05
Sr-90	8.17E+07
Zr-93	1.50E+05
Nb-93m	1.02E+07
Nb-94	1.50E+06
Mo-93	1.48E+05
Tc-99	6.95E+06
Pd-107	6.50E+04
Ag-108m	8.15E+06
Cd-113m	1.80E+06

Nuclide	O.24 [Bq]
Sn-126	3.25E+04
Sb-125	2.24E+03
I-129	1.95E+05
Cs-134	4.95E+01
Cs-135	6.50E+05
Cs-137	1.54E+10
Ba-133	2.43E+04
Pm-147	3.90E+03
Sm-151	1.20E+08
Eu-152	1.85E+05
Eu-154	4.19E+07
Eu-155	5.05E+05
Ho-166m	5.79E+05
U-232	1.38E+01
U-234	8.65E+02
U-235	1.73E+01
U-236	2.61E+02
U-238	3.46E+02
Np-237	4.01E+02

Nuclide	O.24 [Bq]
Pu-238	1.16E+06
Pu-239	3.60E+05
Pu-240	5.05E+05
Pu-241	5.12E+06
Pu-242	2.60E+03
Am-241	3.47E+06
Am-242m	6.36E+03
Am-243	4.91E+04
Cm-243	3.90E+03
Cm-244	1.21E+05
Cm-245	2.58E+02
Cm-246	6.86E+01

E42 O.99:1

E42.1 Description of the waste type

The waste type O.99 only exists in two variants, O.99:1, which is presented here, and O.99:3, which is presented in the subsequent section. That is, there is no waste type O.99 for deposition.

The variant O.99:1 consists of miscellaneous waste in the form of concrete-solidified ion exchange resins and filter aids from OKG. The waste is found in fractured concrete moulds inside Cortén boxes.

There is no approved waste type description for deposition of O.99:1. Data are based on information from the waste description for O.99:1 and Triumf NG v1.0.1.3.

The acceptance criteria for BTF, described in Section E1.4.1, apply for this waste type.

E42.1.1 Waste

The waste is well defined and consists of bead and powder resins and filter aids from the systems fuel storage pool clean-up (system 324), reactor water clean-up (system 331), waste facility clean-up (system 342) and condensate clean-up (system 332).

E42.1.2 Packaging

The waste is packed in Cortén boxes. The Cortén boxes are custom-made boxes of 5 mm Cortén steel with the dimension 1.5×1.5×1.5 m. The empty weight of a box is 960 kg. A lid of 5 mm Cortén steel is mechanically locked into the upper part of the box.

The maximum permissible weight for a Cortén box including waste is 5,000 kg. The disposal volume is 3.375 m^3.

E42.1.3 Treatment

The waste is solidified in cement in fractured concrete moulds. The fractured moulds are placed enclosed in PVC bags, but otherwise untreated, in Cortén boxes. After placement in the final repository, the Cortén boxes will be opened and the contents solidified in concrete. The weight of the Cortén box will then be about 7.5 tonnes. When the waste is solidified with concrete the void in a package is estimated to be 5% of the inner volume of the package.

E42.1.4 Activity determination of radionuclides

Before the waste is transported from OKG to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The total measured activity content for all moulds are 0.75 TBq. The highest permissible surface dose rate is 2 mSv/h, which is based on manufacturing. The waste packages are usually free from surface contamination.

E42.1.5 Production of the waste type

Table E42-1 lists the number of packages for SFR.

There are no deposited packages in the existing SFR as of 2012-12-31.

Forecasted packages refer to waste that is interim-stored at the waste supplier. No future production of the waste type is planned. The waste vault given is according to the acceptance criteria for the waste type.

Table E42-1. Number of packages of the waste type.

Number of packages	Waste vault	O.99:1
Deposited	–	0
Forecasted	(BTF)	40

E42.2 Average package for the waste type

E42.2.1 Material – waste, packaging and matrix

Table E42-2 gives values for an estimated average of the material content in waste type O.99:1. The material data refer to one Cortén box including drum. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E42-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	O.99:1
Concrete [kg]	Waste	1,340
Cellulose [kg]	Waste	21
Cement [kg]	Waste	1,540
Ion exchange resins [kg]	Waste	130
Iron/steel [kg]	Waste	290
Iron/steel surface [m^2]	Waste	15
Iron/steel thickness [mm]	Waste	5.0
Other organic [kg]	Waste	10
Iron/steel [kg]	Packaging	960
Iron/steel surface [m^2]	Packaging	28
Iron/steel thickness [mm]	Packaging	5.0
Concrete [kg]	Matrix	4,378
Void [m^3]	Matrix	0.18

E42.2.2 Radionuclide content

Table E42-3 provides values for a calculated average of the nuclide content in waste type O.99:1 at the closure of SFR on 2075-12-31. Activity data refer to one Cortén box including drum.

Table E42-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	O.99:1 [Bq]	Nuclide	O.99:1 [Bq]	Nuclide	O.99:1 [Bq]	Nuclide	O.99:1 [Bq]
H-3	4.84E+04	Tc-99	2.81E+06	Eu-155	2.15E+05	Am-243	8.62E+03
Be-10	9.78E+00	Pd-107	2.76E+04	Ho-166m	6.29E+04	Cm-243	6.18E+01
C-14 org	0.00E+00	Ag-108m	8.86E+05	U-232	3.30E+00	Cm-244	1.65E+04
C-14 inorg	0.00E+00	Cd-113m	7.65E+05	U-234	2.06E+02	Cm-245	6.16E+01
Cl-36	9.78E+03	Sn-126	1.38E+04	U-235	3.25E+00	Cm-246	1.64E+01
Fe-55	2.07E+03	Sb-125	2.43E+02	U-236	1.37E+02		
Co-60	4.37E+06	I-129	8.28E+04	U-238	4.40E+02		
Ni-59	1.45E+07	Cs-134	2.10E+01	Np-237	4.96E+02		
Ni-63	8.56E+08	Cs-135	2.76E+05	Pu-238	2.25E+05		
Se-79	1.10E+05	Cs-137	6.53E+09	Pu-239	8.59E+04		
Sr-90	2.73E+07	Ba-133	2.64E+03	Pu-240	1.20E+05		
Zr-93	1.63E+04	Pm-147	1.65E+03	Pu-241	1.22E+06		
Nb-93m	1.11E+06	Sm-151	5.11E+07	Pu-242	6.19E+02		
Nb-94	1.63E+05	Eu-152	7.86E+04	Am-241	8.16E+05		
Mo-93	1.61E+04	Eu-154	1.78E+07	Am-242m	1.52E+03		

E43 O.99:3

E43.1 Description of the waste type

The waste type O.99 only exists in two variants, O.99:1 which is presented in the previous section and O.99:3 which is presented here. That is, there is no waste type O.99 for deposition.

The variant O.99:3 consists of steel containers containing steel drums with miscellaneous waste from the years 1971–1981 from OKG.

There is no approved waste type description for deposition of this waste type. Data are based on information from the waste description for O.99:3 and Triumf NG v1.0.1.3.

The acceptance criteria for BLA, described in Section E1.5.1, apply for this waste type.

E43.1.1 Waste

The waste is well defined and consists of ion exchange resins, filter aids and sludge from the systems fuel storage pool clean-up (system 324), reactor water clean-up (system 331) and waste facility clean-up (system 342). Some drums contain blasting sand and some contain combustible and non-combustible trash and scrap metal.

E43.1.2 Packaging

The waste is packed in steel drums placed in ISO containers with dimensions 20-foot half height. On average, 69 drums fit in a container.

The steel drums are made of carbon steel with a thickness of about 1 mm and a diameter of 0.57 m and a height of 0.84 m. The empty weight of a drum is about 20 kg.

The container has a length of 6.1 m, a width of 2.5 m and a height of 2.6 m. The material of the walls and roof is normally about 1.5 mm of carbon steel. An empty container weighs about 2,200 kg.

The maximum permissible weight for a container including waste is 20 tonnes. The disposal volume is 40 m^3.

E43.1.3 Treatment

Ion-exchange resins, filter aids and sludge are solidified in cement in the drums. The cement solidification is done with a stirrer. Trash, scrap metal and blasting sand are untreated in the drums. The void in a package is assumed to be 16.5 m^3.

E43.1.4 Activity determination of radionuclides

Before the waste is transported from OKG to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The measured activity content is about 0.6 MBq/drum. The highest permissible surface dose rate is 2 mSv/h. The waste packages are usually free from surface contamination.

E43.1.5 Production of the waste type

Table E43-1 lists the number of packages for SFR.

There are no deposited packages in the existing SFR as of 2012-12-31.

Forecasted packages refer to waste that is interim-stored at the waste supplier. No future production of the waste type is planned. The waste vault given is according to the acceptance criteria for the waste type.

Table E43-1. Number of packages of the waste type.

Number of packages	Waste vault	O.99:3
Deposited	–	0
Forecasted	(BLA)	5

E43.2 Average package for the waste type

E43.2.1 Material – waste, packaging and matrix

Table E43-2 gives values for an estimated average of the material content in waste type O.99:3. The material data refer to one container including drums. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E43-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	O.99:3
Cement [kg]	Waste	15,000
Ion exchange resins [kg]	Waste	1,885
Iron/steel [kg]	Waste	2,580
Iron/steel surface [m^2]	Waste	369
Iron/steel thickness [mm]	Waste	1.0
Sludge [kg]	Waste	145
Other inorganic [kg]	Waste	600
Iron/steel [kg]	Packaging	2,200
Iron/steel surface [m^2]	Packaging	150
Iron/steel thickness [mm]	Packaging	1.5
Void [m^3]	Matrix	17

E43.2.2 Radionuclide content

Table E43-3 provides values for a calculated average of the nuclide content in waste type O.99:3 at the closure of SFR on 2075-12-31. Activity data refer to one container including drums.

Table E43-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	O.99:3 [Bq]	Nuclide	O.99:3 [Bq]	Nuclide	O.99:3 [Bq]
H-3	4.90E+00	Ag-108m	8.97E+01	U-235	3.29E-04
Be-10	9.90E-04	Cd-113m	1.01E+01	U-236	1.39E-02
C-14 org	0.00E+00	Sn-126	1.82E-01	U-238	4.45E-02
C-14 inorg	0.00E+00	Sb-125	2.46E-02	Np-237	5.02E-02
Cl-36	9.90E-01	I-129	1.09E+00	Pu-238	2.28E+01
Fe-55	2.09E-01	Cs-134	2.78E-04	Pu-239	8.69E+00
Co-60	4.43E+02	Cs-135	3.65E+00	Pu-240	1.22E+01
Ni-59	1.47E+03	Cs-137	8.63E+04	Pu-241	1.24E+02
Ni-63	8.66E+04	Ba-133	2.67E-01	Pu-242	6.27E-02
Se-79	1.46E+00	Pm-147	2.19E-02	Am-241	8.26E+01
Sr-90	2.76E+03	Sm-151	6.76E+02	Am-242m	1.54E-01
Zr-93	1.65E+00	Eu-152	1.04E+00	Am-243	8.72E-01
Nb-93m	1.12E+02	Eu-154	2.35E+02	Cm-243	6.26E-03
Nb-94	1.65E+01	Eu-155	2.84E+00	Cm-244	1.67E+00
Mo-93	1.63E+00	Ho-166m	6.37E+00	Cm-245	6.24E-03
Tc-99	4.14E+01	U-232	3.34E-04	Cm-246	1.66E-03
Pd-107	3.65E-01	U-234	2.09E-02		

E44 R.BWR:D

E44.1 Description of the waste type

The waste type R.BWR:D is a waste type adopted for reactor pressure vessels without internals from RAB (RPV R1).

There is no approved waste type description for deposition of this waste type. Data are based on Hansson et al. (2013).

Acceptance criteria for BRT are under development, see Section E1.2.1.

E44.1.1 Waste

The waste consists of surface contaminated and induced steel or steel alloys (C1070/SIS2333).

E44.1.2 Packaging

No waste packaging is used. The reactor pressure vessel is transported and stored intact, without packaging. The reactor pressure vessel R1 has a height of 20.2 m and an outer diameter of 6.2 m.

The disposal volume for a reactor pressure vessel is about 850 m^3 based on a cuboid with sides of 6.4 m and a length of 20.4 m, where the dimensions refer to reactor pressure vessel measurements including 0.1 m surrounding air.

E44.1.3 Treatment

Connections are sealed and radiation shielding is mounted as needed. No other treatment is planned, with the exception of covering with tarpaulin, painting or other surface treatment that can be done to avoid any surface contamination spreading.

The specified void is based on the inner volume of the reactor pressure vessel.

E44.1.4 Activity determination of radionuclides

The fully treated reactor pressure vessel is measured with respect to surface dose rate. The dominant gamma-emitting nuclide is Co-60. The highest permissible surface dose rate is 2 mSv/h.

The reactor pressure vessels are assumed to be free from surface contamination on the outside.

E44.1.5 Production of the waste type

Table E44-1 lists the number of packages for SFR.

The reactor pressure vessel R1 will be deposited during 2027. The waste vault given is according to the acceptance criteria for the waste type.

Table E44-1. Number of packages of the waste type.

Number of packages	Waste vault	R1
Forecasted	(BRT)	1

E44.2 Average package for the waste type

E44.2.1 Material – waste, packaging and matrix

Table E44-2 gives values for an estimated average of the material content in waste type R.BWR:D. The material data refer to one reactor pressure vessel. Besides weights, corrosion surface and thickness for metals and void in the waste package are given. The reactor pressure vessels are internally plated with a stainless layer of at least 3 mm.

Table E44-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	R1
Iron/steel [kg]	Waste	600,000
Iron/steel surface [m^2]	Waste	770
Iron/steel thickness [mm]	Waste	143
Void [m^3]	Matrix	500

E44.2.2 Radionuclide content

Table E44-3 provides values for a calculated average of the nuclide content in waste type R.BWR:D at the closure of SFR on 2075-12-31. Activity data refer to one reactor pressure vessel.

Table E44-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	R1 Induced activity [Bq]	Surface activity [Bq]
H-3	0.00E+00	0.00E+00
Be-10	0.00E+00	0.00E+00
C-14 org	0.00E+00	0.00E+00
C-14 inorg	0.00E+00	0.00E+00
C-14 ind	3.28E+09	0.00E+00
Cl-36	2.02E+06	0.00E+00
Ca-41	0.00E+00	0.00E+00
Fe-55	1.51E+08	2.19E+07
Co-60	6.81E+09	3.85E+09
Ni-59	1.63E+10	1.16E+10
Ni-63	1.18E+12	1.15E+12
Se-79	0.00E+00	0.00E+00
Sr-90	0.00E+00	3.14E+09
Zr-93	0.00E+00	5.76E+06
Nb-93m	3.17E+10	3.21E+11
Nb-94	2.14E+08	2.85E+09
Mo-93	1.11E+09	1.51E+08
Tc-99	1.67E+08	2.56E+07
Pd-107	0.00E+00	0.00E+00
Ag-108m	0.00E+00	2.22E+08
Cd-113m	0.00E+00	0.00E+00
Sn-126	0.00E+00	1.31E+05
Sb-125	1.36E+04	4.29E+04
I-129	0.00E+00	0.00E+00
Cs-134	0.00E+00	0.00E+00
Cs-135	0.00E+00	0.00E+00
Cs-137	0.00E+00	0.00E+00

Nuclide	R1 Induced activity [Bq]	Surface activity [Bq]
Ba-133	0.00E+00	0.00E+00
Pm-147	0.00E+00	3.93E+03
Sm-151	0.00E+00	5.00E+07
Eu-152	0.00E+00	6.59E+04
Eu-154	0.00E+00	6.86E+06
Eu-155	0.00E+00	6.21E+04
Ho-166m	0.00E+00	1.86E+03
U-232	0.00E+00	1.53E+03
U-235	0.00E+00	2.47E+00
U-236	0.00E+00	6.44E+04
Np-237	0.00E+00	8.65E+04
Pu-238	0.00E+00	5.60E+08
Pu-239	0.00E+00	6.10E+07
Pu-240	0.00E+00	8.70E+07
Pu-241	0.00E+00	9.12E+08
Pu-242	0.00E+00	5.79E+05
Am-241	0.00E+00	3.70E+08
Am-242m	0.00E+00	2.29E+06
Am-243	0.00E+00	9.53E+06
Cm-243	0.00E+00	1.11E+06
Cm-244	0.00E+00	1.31E+08
Cm-245	0.00E+00	2.32E+05
Cm-246	0.00E+00	7.45E+04

E45 R.01/R.01:9

E45.1 Description of the waste type

The waste type consists of concrete moulds containing cement-solidified intermediate-level ion exchange resins and filter aids from RAB. The waste type has been deposited in both BMA and in 1BTF, where the packages in 1BTF act as supportive walls for the steel drums deposited there.

There is a variant of the waste type, R.01:9. The differences between R.01 and R.01:9 mainly concern the design of packaging and small changes in embedding recipes. The differences are considered to be so small that the same data are used for R.01:9 as for R.01.

There are approved waste type descriptions for deposition of the waste type and the variant. Data are based on information in the waste type descriptions and Triumf NG v1.0.1.3.

The acceptance criteria for BMA, described in Section E1.3.1, apply for this waste type. For the packages deposited in BTF, the acceptance criteria in Section E1.4.1 apply.

E45.1.1 Waste

The waste is well defined and consists of bead and powder resins and filter aids from the systems reactor water clean-up (systems 331 and 334), condensate clean-up in BWR (system 333), treatment of fuel storage pool water in PWR and BWR (system 324), sampling (system 336), treatment of floor drainage (342), treatment of bottom blowing water (system 417) and condensate filter clean-up in BWR (system 332).

E45.1.2 Packaging

The waste is packed in concrete moulds. The mould is a cubic box made of reinforced concrete with dimensions 1.2×1.2×1.2 m. The reinforcement consists of 12 mm steel bars that weigh about 274 kg. The wall thickness is normally 10 cm, but can in some cases be 25 cm. The 10-cm mould weighs about 1,600 kg and the 25-cm mould weighs about 3,200 kg. The moulds are internally lined with polyethylene cellular plastic with a thickness of 20 mm in the 10-cm mould and 5 mm in the 25-cm mould. The plastic weighs about 10 kg. The moulds are provided with a disposable stirrer that is solidified with the waste and a splash plate with holes for filling tube and stirrer axle. The stirrer is made of carbon steel and weighs 16 kg.

The maximum permissible weight for a mould including waste is 5,000 kg. The disposal volume is 1.728 m^3.

E45.1.3 Treatment

Bead resins are metered directly in the mould while powder resins and filter aids are mixed and dewatered before metering. Cement and other additives are added during mixing. After completed cement dosage, mixing proceeds until a homogeneous matrix is obtained. The total fill volume of a package with expansion canister is about 67%. The matrix is allowed to harden for at least two days before a concrete lid is cast on.

E45.1.4 Activity determination of radionuclides

Before the waste is transported from RAB to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The maximum measured activity content is 1.0 TBq. The highest permissible surface dose rate is 30 mSv/h, which is based on the manufacturing of the waste package. The waste packages are usually free from surface contamination.

E45.1.5 Production of the waste type

Table E45-1 lists the number of packages for SFR.

Deposited packages refer to packages in the existing SFR as of 2012-12-31. The waste type R.01 started being produced in 1974 and has been deposited since 1988. The variant R.01:9 started being produced in 1975 and has been deposited since 1988.

Forecasted packages refer to waste that is interim-stored at the waste supplier. No future production of the waste type is planned. The waste vault that is indicated for forecasted waste is according to the acceptance criteria for the waste type.

Table E45-1. Number of packages of the waste type.

Number of packages	Waste vault	R.01	R.01:9
Deposited	1BMA	1,070	616
Deposited	1BTF	57	34
Forecasted	(BMA)	0	3

E45.2 Average package for the waste type

E45.2.1 Material – waste, packaging and matrix

Table E45-2 gives values for an estimated average of the material content in waste type R.01. The material data refer to one mould. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E45-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	R.01/R.01:9
Ion exchange resins [kg]	Waste	305
Concrete [kg]	Packaging (including lid)	1,840
Iron/steel [kg]	Packaging (reinforcing bars)	274
Iron/steel surface [m^2]	Packaging (reinforcing bars)	12
Iron/steel thickness [mm]	Packaging (reinforcing bars)	12
Other organic [kg]	Packaging	10
Cement [kg]	Matrix	1,400
Iron/steel [kg]	Matrix (stirrer)	16
Iron/steel surface [m^2]	Matrix (stirrer)	1.0
Iron/steel thickness [mm]	Matrix (stirrer)	5.0
Void [m^3]	Matrix	0.33

E45.2.2 Radionuclide content

Table E45-3 provides values for a calculated average of the nuclide content in waste type R.01 at the closure of SFR on 2075-12-31. Activity data refer to one mould.

Table E45-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	R.01/R.01:9 BMA [Bq]	R.01/R.01:9 1BTF [Bq]
H-3	4.89E+04	4.23E+03
Be-10	4.86E+01	5.40E+00
C-14 org	6.44E+07	7.19E+07
C-14 inorg	2.43E+08	2.71E+08
Cl-36	4.31E+04	4.80E+04
Fe-55	1.41E+01	3.24E-01
Co-60	5.56E+05	3.33E+04
Ni-59	8.99E+08	1.00E+08
Ni-63	6.28E+10	6.77E+09
Se-79	5.34E+04	9.88E+03
Sr-90	1.25E+08	1.25E+07
Zr-93	8.09E+04	9.00E+03
Nb-93m	1.62E+06	1.49E+05
Nb-94	8.07E+05	8.97E+04
Mo-93	9.97E+04	1.11E+04
Tc-99	1.90E+06	2.11E+05
Pd-107	1.34E+04	2.47E+03
Ag-108m	4.20E+06	4.64E+05
Cd-113m	8.29E+04	1.45E+04
Sn-126	6.68E+03	1.23E+03
Sb-125	1.30E+00	4.56E-02
I-129	4.21E+04	7.78E+03
Cs-134	2.26E-03	4.30E-05
Cs-135	3.86E+05	7.15E+04
Cs-137	1.56E+09	2.82E+08
Ba-133	2.03E+03	1.67E+02
Pm-147	6.25E-01	3.72E-02
Sm-151	1.95E+07	3.59E+06
Eu-152	8.00E+03	1.40E+03

Nuclide	R.01/R.01:9 BMA [Bq]	R.01/R.01:9 1BTF [Bq]
Eu-154	7.49E+05	1.25E+05
Eu-155	1.36E+03	1.97E+02
Ho-166m	3.07E+05	3.41E+04
U-232	3.04E+01	3.23E+00
U-234	2.53E+03	2.81E+02
U-235	5.07E+01	5.64E+00
U-236	7.64E+02	8.49E+01
U-238	1.01E+03	1.13E+02
Np-237	1.28E+03	1.43E+02
Pu-238	2.12E+06	2.27E+05
Pu-239	1.05E+06	1.17E+05
Pu-240	1.47E+06	1.63E+05
Pu-241	3.82E+06	3.43E+05
Pu-242	7.59E+03	8.44E+02
Am-241	1.03E+07	1.14E+06
Am-242m	1.62E+04	1.76E+03
Am-243	7.53E+04	8.37E+03
Cm-243	5.79E+03	5.79E+02
Cm-244	5.40E+04	5.06E+03
Cm-245	7.54E+02	8.38E+01
Cm-246	2.00E+02	2.22E+01

E46 R.02/R.02:9

E46.1 Description of the waste type

The waste type R.02 consists of concrete moulds containing cement-solidified intermediate-level ion exchange resins and filter aids from RAB.

There is a variant of the waste type, R.02:9. The differences between R.02 and R.02:9 mainly concern design of packaging and small changes in embedding recipes. The differences are considered to be so small that the same data are used for R.02:9 as for R.02.

There are approved waste type descriptions for deposition of the waste type and the variant. Data are based on information in the waste type descriptions and Triumf NG v1.0.1.3.

The acceptance criteria for Silo, described in Section E1.1.1, apply for this waste type.

E46.1.1 Waste

The waste is well defined and consists of bead and powder resins, as well as filter aids from the systems reactor water clean-up (systems 331 and 334), condensate clean-up in BWR (system 333), treatment of fuel storage pool water in PWR and BWR (system 324), sampling (system 336), treatment of floor drainage (342), treatment of bottom blowing water (system 417) and condensate clean-up in BWR (system 332).

E46.1.2 Packaging

The waste is packed in concrete moulds. The mould is a cubic box made of reinforced concrete with dimensions 1.2×1.2×1.2 m. The wall thickness is normally 10 cm, but can in some cases be 25 cm. The 10-cm mould weighs about 1,600 kg and the 25-cm mould weighs about 3,200 kg. The reinforcement consists of 12 mm steel bars with a total weight of 274 kg. The moulds are internally lined with polyethylene cellular plastic with a thickness of 20 mm in the 10-cm mould and 5 mm in the 25-cm mould. The plastic weighs about 10 kg. The moulds are provided with a disposable stirrer that is solidified with the waste and a splash plate with holes for filling tube and stirrer axle. The stirrer weighs about 16 kg.

The maximum permissible weight for a mould including waste is 5,000 kg. The disposal volume is 1.728 m^3.

E46.1.3 Treatment

Bead ion exchange resins are metered directly in the mould while powder resins and filter aids are mixed and dewatered before metering. Cement and other additives are added during mixing. After completed cement dosage, mixing proceeds until a homogeneous matrix is obtained. The total fill volume of a package with expansion canister is about 67%. The matrix is allowed to harden for at least two days before a concrete lid is cast on the package.

E46.1.4 Activity determination of radionuclides

Before the waste is transported from RAB to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The maximum measured activity content is 1.0 TBq for a mould. The highest permissible surface dose rate is 500 mSv/h. The waste packages are usually free from surface contamination.

E46.1.5 Production of the waste type

Table E46-1 lists the number of packages for SFR.

Deposited packages refer to packages in the existing SFR as of 2012-12-31. The waste type R.02 started being produced in 1984 and has been deposited since 1993. The variant R.02:9 started being produced in 1978 and has been deposited since 1993.

Forecasted packages refer to waste that is interim-stored at the waste supplier. No future production of the waste type is planned. The waste vault that is indicated for forecasted waste is according to the acceptance criteria for the waste type.

Table E46-1. Number of packages of the waste type.

Number of packages	Waste vault	R.02	R.02:9
Deposited	Silo	56	292
Forecasted	(Silo)	8	15

E46.2 Average package for the waste type

E46.2.1 Material – waste, packaging and matrix

Table E46-2 gives values for an estimated average of the material content in waste type R.02. The material data refer to one mould. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E46-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	R.02/R.02:9
Ion exchange resins [kg]	Waste	305
Concrete [kg]	Packaging (including lid)	1,840
Iron/steel [kg]	Packaging (reinforcing bars)	274
Iron/steel surface [m^2]	Packaging (reinforcing bars)	12
Iron/steel thickness [mm]	Packaging (reinforcing bars)	12
Other organic [kg]	Packaging	10
Cement [kg]	Matrix	1,400
Iron/steel [kg]	Matrix (stirrer)	16
Iron/steel surface [m^2]	Matrix (stirrer)	1.0
Iron/steel thickness [mm]	Matrix (stirrer)	5.0
Void [m^3]	Matrix	0.33

E46.2.2 Radionuclide content

Table E46-3 provides values for a calculated average of the nuclide content in waste type R.02 at the closure of SFR on 2075-12-31. Activity data refer to one mould.

Table E46-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	R.02/R.02:9 [Bq]	Nuclide	R.02/R.02:9 [Bq]	Nuclide	R.02/R.02:9 [Bq]	Nuclide	R.02/R.02:9 [Bq]
H-3	5.79E+04	Nb-94	7.74E+05	Pm-147	9.47E+01	Pu-239	1.01E+06
Be-10	4.66E+01	Mo-93	9.43E+04	Sm-151	6.79E+07	Pu-240	1.41E+06
C-14 org	8.25E+07	Tc-99	1.87E+06	Eu-152	2.95E+04	Pu-241	4.29E+06
C-14 inorg	2.82E+08	Pd-107	4.64E+04	Eu-154	3.24E+06	Pu-242	7.29E+03
Cl-36	4.67E+04	Ag-108m	4.04E+06	Eu-155	1.57E+04	Am-241	9.92E+06
Fe-55	6.77E+02	Cd-113m	3.04E+05	Ho-166m	2.95E+05	Am-242m	1.56E+04
Co-60	1.88E+06	Sn-126	2.32E+04	U-232	2.96E+01	Am-243	7.23E+04
Ni-59	8.63E+08	Sb-125	7.97E+01	U-234	2.43E+03	Cm-243	5.84E+03
Ni-63	6.06E+10	I-129	1.46E+05	U-235	4.87E+01	Cm-244	5.62E+04
Se-79	1.86E+05	Cs-134	4.49E-01	U-236	7.32E+02	Cm-245	7.23E+02
Sr-90	1.24E+08	Cs-135	1.31E+06	U-238	9.72E+02	Cm-246	1.92E+02
Zr-93	7.76E+04	Cs-137	5.45E+09	Np-237	1.22E+03		
Nb-93m	1.77E+06	Ba-133	2.59E+03	Pu-238	2.04E+06		

E47 R.02:D

E47.1 Description of the waste type

The waste type R.02:D is a waste type adopted for decommissioning waste from RAB. It consists of steel moulds containing cement-solidified intermediate-level ion exchange resins from system decontamination.

There is no approved waste type description for deposition of this waste type. Material quantities and activity have been calculated based on Hansson et al. (2013), supplemented with assumptions about packaging and solidification material.

The acceptance criteria for Silo, described in Section E1.1.1, are assumed to be valid for the waste type.

E47.1.1 Waste

The waste consists of ion exchange resins produced at system decontamination prior to decommissioning.

E47.1.2 Packaging

The waste is packed in steel moulds. The mould is a cubic box made of carbon steel with dimensions 1.2×1.2×1.2 m. The wall is 5 mm thick and the bottom is 6 mm thick. The mould weighs about 400 kg. The mould contains a stirrer of carbon steel. It weighs about 25 kg. The mould is also provided with a splash plate.

The maximum permissible weight for a mould including waste is 5,000 kg. The disposal volume is 1.728 m^3.

E47.1.3 Treatment

The mould is assumed to be filled with equal parts ion exchange resin and cement. The waste matrix is homogenised with the aid of the stirrer. The void is assumed to be about 10% of the inner volume of a packaging. The matrix is allowed to harden before a steel lid is placed on the mould.

E47.1.4 Activity determination of radionuclides

Before the waste is transported from RAB to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The highest permissible surface dose rate is 500 mSv/h. The waste packages are assumed to be free from surface contamination.

E47.1.5 Production of the waste type

Table E47-1 lists the number of packages for SFR.

The waste will be deposited during the years 2026–2027, 2042 and 2044. The waste vault given is according to the acceptance criteria for the waste type.

Table E47-1. Number of packages of the waste type.

Number of packages	Waste vault	R.02:D
Forecasted	(Silo)	42

E47.2 Average package for the waste type

E47.2.1 Material – waste, packaging and matrix

Table E47-2 gives values for an estimated average of the material content in waste type R.02:D. The material data refer to one mould. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E47-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	R.02:D
Ion exchange resins [kg]	Waste	803
Iron/steel [kg]	Packaging	400
Iron/steel surface [m^2]	Packaging	17
Iron/steel thickness [mm]	Packaging	5.0–6.0
Cement [kg]	Matrix	1,836
Iron/steel [kg]	Matrix	25
Iron/steel surface [m^2]	Matrix	3.0
Iron/steel thickness [mm]	Matrix	5.0
Void [m^3]	Matrix	0.17

E47.2.2 Radionuclide content

Table E47-3 provides values for a calculated average of the nuclide content in waste type R.02:D at the closure of SFR on 2075-12-31. Activity data refer to one mould.

Table E47-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	R.02:D [Bq]	Nuclide	R.02:D [Bq]	Nuclide	R.02:D [Bq]
H-3	0.00E+00	Tc-99	6.61E+07	U-232	9.81E+02
Be-10	0.00E+00	Pd-107	3.55E+05	U-235	1.54E+00
C-14 org	2.45E+07	Ag-108m	3.31E+09	U-236	4.81E+04
C-14 inorg	6.42E+07	Cd-113m	3.42E+05	Np-237	5.39E+04
C-14 ind	0.00E+00	Sn-126	1.77E+06	Pu-238	3.16E+08
Cl-36	4.24E+03	Sb-125	1.02E+07	Pu-239	4.00E+07
Ca-41	0.00E+00	I-129	1.04E+05	Pu-240	5.79E+07
Fe-55	1.05E+09	Cs-134	7.37E+05	Pu-241	8.20E+08
Co-60	2.71E+10	Cs-135	1.61E+06	Pu-242	3.04E+05
Ni-59	6.65E+09	Cs-137	1.84E+10	Am-241	2.31E+08
Ni-63	6.44E+11	Ba-133	2.29E+03	Am-242m	1.08E+06
Se-79	2.63E+05	Pm-147	1.25E+05	Am-243	4.42E+06
Sr-90	2.29E+09	Sm-151	3.71E+07	Cm-243	6.28E+05
Zr-93	6.29E+07	Eu-152	4.97E+04	Cm-244	6.87E+07
Nb-93m	2.01E+11	Eu-154	9.33E+06	Cm-245	8.79E+04
Nb-94	1.57E+09	Eu-155	2.08E+05	Cm-246	3.07E+04
Mo-93	1.37E+08	Ho-166m	6.46E+02		

E48 R.10

E48.1 Description of the waste type

The waste type R.10 consists of concrete moulds containing cement-solidified intermediate-level sludge from RAB. The waste type has been deposited in both BMA and in 1BTF, where the packages in 1BTF act as supportive walls for the steel drums deposited there.

There is an approved waste type description for deposition of this waste type. Data are based on information in the waste type description and Triumf NG v1.0.1.3.

The acceptance criteria for BMA, described in Section E1.3.1, apply for this waste type. For the packages deposited in BTF, the acceptance criteria in Section E1.4.1 apply.

E48.1.1 Waste

The waste is well defined and consists of sludge from chemical decontamination and sludge from cleaning of active tanks.

E48.1.2 Packaging

The waste is packed in concrete moulds. The mould is a cubic box made of reinforced concrete with dimensions 1.2×1.2×1.2 m. The reinforcement consists of 12 mm thick steel bars that weigh about 274 kg. The wall thickness is normally 10 cm, but can in some cases be 25 cm. The 10-cm mould weighs about 1,600 kg and the 25-cm mould weighs about 3,200 kg. The moulds are internally lined with polyethylene cellular plastic with a thickness of 20 mm in the 10-cm mould and 5 mm in the 25-cm mould. The plastic weighs about 10 kg. The moulds are provided with a disposable stirrer that is solidified with the waste and a splash plate with holes for filling tube and stirrer axle. The stirrer is made of carbon steel and weighs 16 kg.

The maximum permissible weight for a mould including waste is 5,000 kg. The disposal volume is 1.728 m^3.

E48.1.3 Treatment

Waste and cement are mixed in the mould during stirring. The total fill volume of a package with expansion canister is about 67%. After hardening for at least two days, a concrete lid is cast on the package.

E48.1.4 Activity determination of radionuclides

Before the waste is transported from RAB to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The maximum measured activity content at manufacturing is about 1.0 TBq. The highest permissible surface dose rate is 100 mSv/h. The waste packages are usually free from surface contamination.

E48.1.5 Production of the waste type

Table E48-1 lists the number of packages for SFR.

Deposited packages refer to packages in the existing SFR as of 2012-12-31. The waste type R.10 started being produced in 1977 and has been deposited since 1996.

Forecasted packages refer to waste that has not yet been deposited and includes both waste that is interim-stored at the waste supplier and waste that has not yet been produced. Currently, there are 5 packages in interim storage and a production is planned of one package per year up to 2044. The waste vault that is indicated for forecasted waste is according to the acceptance criteria for the waste type.

Table E48-1. Number of packages of the waste type.

Number of packages	Waste vault	R.10
Deposited	1BMA	84
Deposited	1BTF	4
Forecasted	(BMA)	37

E48.2 Average package for the waste type

E48.2.1 Material – waste, packaging and matrix

Table E48-2 gives values for an estimated average of the material content in waste type R.10. The material data refer to one mould. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E48-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	R.10
Sludge [kg]	Waste	425
Concrete [kg]	Packaging (including lid)	1,840
Iron/steel [kg]	Packaging (reinforcing bars)	274
Iron/steel surface [m^2]	Packaging (reinforcing bars)	12
Iron/steel thickness [mm]	Packaging (reinforcing bars)	12
Other organic [kg]	Packaging	10
Cement [kg]	Matrix	1,120
Iron/steel [kg]	Matrix (stirrer)	16
Iron/steel surface [m^2]	Matrix (stirrer)	1.0
Iron/steel thickness [mm]	Matrix (stirrer)	5.0
Void [m^3]	Matrix	0.33

E48.2.2 Radionuclide content

Table E48-3 provides values for a calculated average of the nuclide content in waste type R.10 at the closure of SFR on 2075-12-31. Activity data refer to one mould.

Table E48-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	R.10 BMA [Bq]	R.10 1BTF [Bq]
H-3	4.85E+04	4.19E+02
Be-10	1.58E+01	4.55E–01
C-14 org	0.00E+00	0.00E+00
C-14 inorg	0.00E+00	0.00E+00
Cl-36	1.58E+04	4.55E+02
Fe-55	1.59E+05	6.26E–02
Co-60	1.40E+07	4.22E+03
Ni-59	2.94E+08	8.45E+06
Ni-63	2.21E+10	5.79E+08
Se-79	6.21E+02	2.10E+01
Sr-90	1.82E+06	5.74E+04
Zr-93	2.64E+04	7.58E+02
Nb-93m	1.11E+06	1.42E+04
Nb-94	2.63E+05	7.56E+03
Mo-93	3.16E+04	9.33E+02
Tc-99	5.38E+05	1.78E+04
Pd-107	1.55E+02	5.25E+00
Ag-108m	1.40E+06	3.93E+04
Cd-113m	1.21E+03	3.44E+01
Sn-126	7.76E+01	2.63E+00
Sb-125	5.48E+02	1.17E–02
I-129	4.89E+02	1.66E+01
Cs-134	2.08E+00	1.42E–07
Cs-135	4.47E+03	1.52E+02
Cs-137	1.99E+07	6.32E+05
Ba-133	2.78E+03	1.71E+01
Pm-147	3.25E+01	1.35E–04
Sm-151	2.34E+05	7.77E+03
Eu-152	1.18E+02	3.33E+00
Eu-154	1.47E+04	3.17E+02
Eu-155	2.07E+02	5.69E–01
Ho-166m	1.01E+05	2.87E+03
U-232	0.00E+00	0.00E+00
U-234	0.00E+00	0.00E+00
U-235	0.00E+00	0.00E+00
U-236	0.00E+00	0.00E+00
U-238	0.00E+00	0.00E+00
Np-237	0.00E+00	0.00E+00
Pu-238	0.00E+00	0.00E+00
Pu-239	0.00E+00	0.00E+00
Pu-240	0.00E+00	0.00E+00
Pu-241	0.00E+00	0.00E+00
Pu-242	0.00E+00	0.00E+00
Am-241	0.00E+00	0.00E+00
Am-242m	0.00E+00	0.00E+00
Am-243	0.00E+00	0.00E+00
Cm-243	0.00E+00	0.00E+00
Cm-244	0.00E+00	0.00E+00
Cm-245	0.00E+00	0.00E+00
Cm-246	0.00E+00	0.00E+00

E49 R.12

E49.1 Description of the waste type

The waste type R.12 consists of steel containers containing low-level trash and scrap metal from RAB.

There is a variant of the waste type, R.12:1. The difference between R.12 and R.12:1 is that high-pressure-compacted drums are packed in the variant, providing a considerably larger amount of iron/steel than that included in R.12.

There are approved waste type descriptions for deposition of both the waste type and the variant. Data are based on information in the waste type descriptions and Triumf NG v1.0.1.3.

The acceptance criteria for BLA, described in Section E1.5.1, apply for this waste type.

E49.1.1 Waste

The waste consists of trash and scrap metal. The trash consists of compacted or non-compacted garbage bags containing e.g. textiles, paper, insulation, small pieces of aluminium, copper and plastics. The scrap metal consists of e.g. fittings, scrapped components, cables, suspensions and insulation. The mixture of waste materials has changed over time depending on the maintenance work, revisions or other work carried out.

E49.1.2 Packaging

The waste is packed in ISO containers of carbon steel with dimensions 20-foot half height or 20-foot full height.

The half-height container has a length of 6.1 m, a width of 2.5 m and a height of 1.3 m. The thickness of the walls and roof is normally about 1.5 mm. An empty container weighs about 1,900 kg.

The full-height container has a height of 2.6 m but otherwise the same geometry as the half-height container. It has an empty weight of 2,200 kg. The floor of the full-height container can consist of about 15–30 mm of plywood with load-bearing steel construction. The plywood floor weighs about 310 kg.

Open containers are sealed with a lid.

The maximum permissible weight for a container including waste is 20,000 tonnes. The disposal volumes are 20 m^3 and 40 m^3.

E49.1.3 Treatment

Combustible waste that cannot be burned due to high activity content is mixed with compactible waste and compacted and wrapped in plastic. Non-combustible and non-compactible waste is placed in plastic bags, steel drums or drum boxes or is placed directly in the container, without treatment.

As high a fill volume as possible should always be targeted; it can, however, vary widely depending on the nature of the waste. The void in a waste package is assumed to be 7.5 m^3 for both types of packaging, even though there is likely to be more void in the larger packaging.

E49.1.4 Activity determination of radionuclides

Before the waste is transported from RAB to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The usually measured activity content is about 25 GBq/container. The highest permissible surface dose rate is 2 mSv/h. The waste packages are usually free from surface contamination.

E49.1.5 Production of the waste type

Table E49-1 lists the number of packages for SFR.

Deposited packages refer to packages in the existing SFR as of 2012-12-31. The waste type R.12 started being produced in 1976 and has been deposited since 1991. The variant R.12:1 started being produced in 1975 and has been deposited since 1991.

Forecasted packages refer to waste that has not yet been deposited and includes both waste that is interim-stored at the waste supplier and waste that has not yet been produced. Currently, there are 28 packages in interim storage, of which 7 are R.12 half height and 21 are O.12 full height. A production is planned of 2.4 packages per year up to 2027 and then 17 packages per year up to 2044. The waste vault that is indicated for forecasted waste is according to the acceptance criteria for the waste type.

Table E49-1. Number of packages of the waste type.

Number of packages	Waste vault	R.12 half height	R.12 full height	R.12:1 half height
Deposited	1BLA	24	44	2
Forecasted	(BLA)	7	74	0

E49.2 Average package for the waste type

E49.2.1 Material – waste, packaging and matrix

Table E49-2 gives values for an estimated average of the material content in waste type R.12. The material data refer to one container. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E49-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	R.12 half height	R.12 full height	R.12:1 half height
Aluminium/zinc [kg]	Waste	100	100	100
Aluminium/zinc surface [m^2]	Waste	15	15	15
Aluminium/zinc thickness [mm]	Waste	5.0	5.0	5.0
Cellulose [kg]	Waste	500	500	500
Iron/steel [kg]	Waste	4,500	4,500	7,720
Iron/steel surface [m^2]	Waste	229	229	862
Iron/steel thickness [mm]	Waste	5.0	5.0	5.0
Other inorganic [kg]	Waste	400	400	–
Other organic [kg]	Waste	3,000	3,000	3,000
Cellulose [kg]	Packaging	–	310	–
Iron/steel [kg]	Packaging	1,900	2,200	1,900
Iron/steel surface [m^2]	Packaging	105	150	105
Iron/steel thickness [mm]	Packaging	1.5	1.5	1.5
Void [m^3]	Matrix	7.5	7.5	7.5

E49.2.2 Radionuclide content

Table E49-3 provides values for a calculated average of the nuclide content in waste type R.12 at the closure of SFR on 2075-12-31. Activity data refer to one container.

Table E49-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	R.12/R.12:1 half height [Bq]	R.12 full height [Bq]
H-3	4.43E+03	1.05E+04
Be-10	2.18E+00	1.56E+00
C-14 org	0.00E+00	0.00E+00
C-14 inorg	0.00E+00	0.00E+00
Cl-36	2.18E+03	1.55E+03
Fe-55	7.14E+01	3.34E+04
Co-60	2.02E+05	3.33E+06
Ni-59	4.10E+07	2.93E+07
Ni-63	3.08E+09	2.49E+09
Se-79	8.00E+03	2.68E+02
Sr-90	1.41E+06	4.95E+05
Zr-93	3.63E+03	2.59E+03
Nb-93m	1.22E+05	2.10E+05
Nb-94	3.62E+04	2.59E+04
Mo-93	4.34E+03	2.80E+03
Tc-99	7.67E+04	3.17E+04
Pd-107	2.00E+03	6.71E+01
Ag-108m	1.91E+05	6.53E+04
Cd-113m	1.88E+04	2.48E+03
Sn-126	9.99E+02	3.35E+01
Sb-125	6.57E-01	1.52E+03
I-129	6.30E+03	2.05E+02
Cs-134	1.22E-03	1.83E+01
Cs-135	5.73E+04	1.14E+03
Cs-137	2.83E+08	1.73E+07
Ba-133	2.13E+02	6.39E+02
Pm-147	1.95E+00	5.42E+02
Sm-151	3.12E+06	1.27E+05
Eu-152	1.84E+03	2.44E+04
Eu-154	2.23E+05	8.44E+04
Eu-155	7.90E+02	3.44E+03
Ho-166m	1.38E+04	1.00E+04
U-232	6.85E-01	2.30E-01
U-234	5.13E+01	1.62E+01
U-235	1.03E+00	3.24E-01
U-236	1.55E+01	4.88E+00
U-238	2.05E+01	6.48E+00
Np-237	2.75E+01	8.56E+00
Pu-238	7.47E+04	2.57E+04
Pu-239	2.13E+04	6.74E+03
Pu-240	3.06E+04	9.67E+03
Pu-241	1.39E+05	6.28E+04
Pu-242	1.54E+02	4.86E+01
Am-241	2.97E+05	9.58E+04
Am-242m	3.45E+02	1.12E+02
Am-243	1.53E+03	4.83E+02
Cm-243	1.53E+02	5.63E+01
Cm-244	1.73E+04	6.12E+03
Cm-245	1.53E+01	4.83E+00
Cm-246	4.06E+00	1.28E+00

E50 R.12:D/R.12C:D/R.12S:D

E50.1 Description of the waste type

Waste types R.12:D, R.12C:D and R.12S:D are waste types adopted for low-level decommissioning waste in steel containers from RAB. R.12:D contains scrap metal or secondary waste. R.12C:D contains concrete and R.12S:D contains sand.

There is no approved waste type description for deposition of the waste types. Material quantities and activity have been calculated based on Hansson et al. (2013), supplemented with assumptions on secondary waste, and material composition for the decommissioning waste, packaging and solidification material.

The acceptance criteria for BLA, described in Section E1.5.1, are assumed to be valid for these waste types.

E50.1.1 Waste

The scrap metal in R.12:D consists mainly of fittings and scrapped components. The secondary waste in R.12:D consists of trash and scrap metal like waste type R.12 from operational waste. The waste in R.12C:D consists of concrete from the outer parts of the biological shield and contaminated concrete from the controlled area in the facility. The waste in R.12S:D consists of sand from the sand beds in system 341.

E50.1.2 Packaging

The waste is packed in ISO containers of carbon steel with dimensions 20-foot half height. The container has a length of 6.06 m, a width of 2.5 m and a height of 1.3 m. The thickness of the walls and roof is normally 1.5 mm. An empty container weighs about 1,900 kg. Open containers are sealed with a lid.

The maximum permissible weight for a container including waste is 20 tonnes. The disposal volume is 20 m^3.

E50.1.3 Treatment

The scrap metal, concrete and sand are assumed to be packed with a packing degree of 1.1 tonnes/m^3. The secondary waste is assumed to be treated like waste type R.12 from operational waste.

E50.1.4 Activity determination of radionuclides

Before the waste is transported from RAB to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The highest permissible surface dose rate is 2 mSv/h. The waste packages are assumed to be free from surface contamination.

E50.1.5 Production of the waste type

Table E50-1 lists the number of packages for SFR.

Waste types R.12:D and R.12C:D are assumed to be deposited in 2025–2031 and 2041–2047. Waste type R.12S:D is assumed to be deposited during the years 2026–2031. The waste vault given is according to the acceptance criteria for the waste types.

Table E50-1. Number of packages of the waste type.

Number of packages	Waste vault	R.12:D scrap metal	R.12:D secondary waste	R.12C:D	R.12S:D
Forecasted	(BLA)	294	95	60	32

E50.2 Average package for the waste type

E50.2.1 Material – waste, packaging and matrix

Table E50-2 gives values for an estimated average of the material content in waste types R.12:D, R.12C:D and R.12S:D. The material data refer to one container. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E50-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	R.12:D scrap metal	R.12:D secondary waste	R.12C:D	R.12S:D
Aluminium/zinc [kg]	Waste	–	108	–	–
Aluminium/zinc surface [m^2]	Waste	–	16	–	–
Aluminium/zinc thickness [mm]	Waste	–	5.0	–	–
Concrete [kg]	Waste	–	–	16,500	–
Cellulose [kg]	Waste	–	541	–	–
Iron/steel [kg]	Waste	16,500	4,868	–	–
Iron/steel surface [m^2]	Waste	846	248	–	–
Iron/steel thickness [mm]	Waste	5.0	5.0	–	–
Sand [kg]	Waste	–	–	–	16,500
Other inorganic [kg]	Waste	–	433	–	–
Other organic [kg]	Waste	–	3,245	–	–
Iron/steel [kg]	Packaging	1,900	1,900	1,900	1,900
Iron/steel surface [m^2]	Packaging	105	105	105	105
Iron/steel thickness [mm]	Packaging	1.5	1.5	1.5	1.5
Void [m^3]	Matrix	13	6.8	8.1	4.3

E50.2.2 Radionuclide content

Table E50-3 provides values for a calculated average of the nuclide content in waste types R.12:D, R.12C:D and R.12S:D at the closure of SFR on 2075-12-31. Activity data refer to one container.

Table E50-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	R.12:D [Bq]	R.12C:D [Bq]	R.12S:D [Bq]
H-3	0.00E+00	0.00E+00	0.00E+00
Be-10	4.41E–07	0.00E+00	0.00E+00
C-14 org	1.41E+05	2.70E+06	0.00E+00
C-14 inorg	5.51E+05	7.45E+06	0.00E+00
C-14 ind	4.09E+04	0.00E+00	0.00E+00
Cl-36	2.84E+01	3.31E+02	0.00E+00
Ca-41	0.00E+00	0.00E+00	0.00E+00
Fe-55	2.94E+05	1.26E+04	0.00E+00
Co-60	7.87E+06	2.27E+06	0.00E+00
Ni-59	3.75E+06	4.74E+06	0.00E+00
Ni-63	3.61E+08	4.41E+08	0.00E+00
Se-79	3.31E+03	7.39E+04	0.00E+00
Sr-90	2.47E+06	1.62E+06	1.41E+08
Zr-93	3.71E+04	6.23E+04	0.00E+00
Nb-93m	1.17E+08	5.60E+07	0.00E+00
Nb-94	1.02E+06	5.13E+05	0.00E+00
Mo-93	7.74E+04	2.71E+04	0.00E+00
Tc-99	2.32E+05	1.48E+06	0.00E+00
Pd-107	2.35E+03	1.26E+04	0.00E+00
Ag-108m	1.64E+06	1.01E+06	0.00E+00
Cd-113m	3.55E+03	2.93E+04	0.00E+00
Sn-126	1.10E+04	5.91E+04	0.00E+00
Sb-125	4.14E+03	1.06E+03	0.00E+00
I-129	7.49E+02	4.10E+03	5.42E+03
Cs-134	2.17E+03	1.68E+02	0.00E+00
Cs-135	6.72E+04	4.65E+04	2.84E+05
Cs-137	1.16E+08	4.50E+08	1.49E+09
Ba-133	5.85E+00	2.45E–02	0.00E+00
Pm-147	1.30E+02	1.44E+01	0.00E+00

Nuclide	R.12:D [Bq]	R.12C:D [Bq]	R.12S:D [Bq]
Sm-151	2.80E+04	3.35E+04	0.00E+00
Eu-152	5.22E+01	7.24E+01	0.00E+00
Eu-154	7.89E+03	6.09E+03	0.00E+00
Eu-155	1.93E+02	8.97E+01	0.00E+00
Ho-166m	9.01E–01	2.91E+00	0.00E+00
U-232	1.33E+00	6.09E+00	0.00E+00
U-235	1.18E–03	1.42E–03	0.00E+00
U-236	5.74E+01	2.45E+02	0.00E+00
Np-237	3.75E+01	4.01E+01	0.00E+00
Pu-238	2.17E+05	1.02E+05	0.00E+00
Pu-239	2.84E+04	3.05E+04	0.00E+00
Pu-240	4.07E+04	4.44E+04	0.00E+00
Pu-241	5.06E+05	2.47E+05	0.00E+00
Pu-242	2.17E+02	2.36E+02	0.00E+00
Am-241	1.73E+05	1.14E+05	0.00E+00
Am-242m	7.59E+02	3.89E+02	0.00E+00
Am-243	3.24E+03	3.20E+03	0.00E+00
Cm-243	4.18E+02	1.74E+02	0.00E+00
Cm-244	4.82E+04	2.03E+04	0.00E+00
Cm-245	6.75E+01	5.55E+01	0.00E+00
Cm-246	2.36E+01	2.23E+01	0.00E+00

E51 R.15

E51.1 Description of the waste type

The waste type R.15 consists of concrete moulds containing cement-solidified intermediate-level ion exchange resins and filter aids from RAB.

There is an approved waste type description for deposition of the waste type. Data are based on information in the waste type description and Triumf NG v1.0.1.3.

The acceptance criteria for BMA, described in Section E1.3.1, apply for this waste type.

E51.1.1 Waste

The waste is well defined and consists of bead and powder resins, as well as filter aids from the systems reactor water clean-up (systems 331 and 334), condensate clean-up in BWR (system 333), treatment of fuel storage pool water in PWR and BWR (system 324), sampling (system 336), treatment of floor drainage (342), treatment of bottom blowing water (systems 417 and 337) and condensate clean-up in BWR (system 332).

E51.1.2 Packaging

The waste is packed in steel moulds. The mould is a cubic box made of carbon steel with dimensions 1.2×1.2×1.2 m. The wall is 5 mm thick, the bottom is 6 mm thick. The mould weighs about 400 kg. The mould contains a stirrer of carbon steel. It weighs about 25 kg. The mould is also provided with a splash plate.

The maximum permissible weight for a mould including waste is 5,000 kg. The disposal volume is 1.728 m^3.

E51.1.3 Treatment

Bead resins are metered directly in the mould while powder resins and filter aids are mixed and dewatered before metering. Cement and other additives are added during mixing. The void in a package is assumed to be about 5%. After hardening for at least two days, a concrete lid is cast on the package.

E51.1.4 Activity determination of radionuclides

Before the waste is transported from RAB to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The maximum measured activity content at manufacturing is about 1.0 TBq. The highest permissible surface dose rate is 100 mSv/h. The waste packages are usually free from surface contamination.

E51.1.5 Production of the waste type

Table E51-1 lists the number of packages for SFR.

Deposited packages refer to packages in the existing SFR as of 2012-12-31. The waste type R.15 started being produced in 1987 and has been deposited since 1992.

Forecasted packages refer to waste that has not yet been deposited and includes both waste that is interim-stored at the waste supplier and waste that has not yet been produced. Currently, there are 50 packages in interim storage and a production is planned of 2.5 packages per year up to 2044. The waste vault that is indicated for forecasted waste is according to the acceptance criteria for the waste type.

Table E51-1. Number of packages of the waste type.

Number of packages	Waste vault	R.15
Deposited	1BMA	124
Forecasted	(BMA)	130

E51.2 Average package for the waste type

E51.2.1 Material – waste, packaging and matrix

Table E51-2 gives values for an estimated average of the material content in waste type R.15. The material data refer to one mould. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E51-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	R.15
Filter aids [kg]	Waste	2.4
Ion exchange resins [kg]	Waste	700
Concrete [kg]	Packaging	500
Iron/steel [kg]	Packaging	400
Iron/steel surface [m^2]	Packaging	17
Iron/steel thickness [mm]	Packaging	5.0–6.0
Cement [kg]	Matrix	2,100
Iron/steel [kg]	Matrix (stirrer)	25
Iron/steel surface [m^2]	Matrix (stirrer)	3.0
Iron/steel thickness [mm]	Matrix (stirrer)	5.0
Void [m^3]	Matrix	0.085

E51.2.2 Radionuclide content

Table E51-3 provides values for a calculated average of the nuclide content in waste type R.15 at the closure of SFR on 2075-12-31. Activity data refer to one mould.

Table E51-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	R.15 [Bq]	Nuclide	R.15 [Bq]	Nuclide	R.15 [Bq]
H-3	3.16E+05	Cs-135	2.84E+05	Cm-243	1.19E+04
Be-10	5.35E+01	Cs-137	3.30E+09	Cm-244	1.72E+05
C-14 org	4.37E+07	Ba-133	1.96E+04	Cm-245	8.31E+02
C-14 inorg	1.65E+08	Pm-147	1.40E+05	Cm-246	2.21E+02
Cl-36	5.65E+04	Sm-151	2.63E+07		
Fe-55	1.34E+06	Eu-152	4.74E+04		
Co-60	1.18E+08	Eu-154	1.63E+07		
Ni-59	9.90E+08	Eu-155	7.90E+05		
Ni-63	8.08E+10	Ho-166m	3.43E+05		
Se-79	5.90E+04	U-232	4.25E+01		
Sr-90	2.53E+08	U-234	2.79E+03		
Zr-93	8.91E+04	U-235	5.58E+01		
Nb-93m	6.33E+06	U-236	8.39E+02		
Nb-94	8.89E+05	U-238	1.11E+03		
Mo-93	9.88E+04	Np-237	1.32E+03		
Tc-99	1.91E+06	Pu-238	2.79E+06		
Pd-107	1.47E+04	Pu-239	1.16E+06		
Ag-108m	4.80E+06	Pu-240	1.62E+06		
Cd-113m	4.52E+05	Pu-241	1.80E+07		
Sn-126	7.37E+03	Pu-242	8.36E+03		
Sb-125	1.47E+05	Am-241	1.13E+07		
I-129	4.53E+04	Am-242m	1.99E+04		
Cs-134	5.63E+02	Am-243	8.31E+04		

E52 R.16

E52.1 Description of the waste type

The waste type R.16 consists of concrete moulds containing cement-solidified intermediate-level ion exchange resins and filter aids from RAB.

There is an approved waste type description for deposition of the waste type. Data are based on information in the waste type description and Triumf NG v1.0.1.3.

The acceptance criteria for Silo, described in Section E1.1.1, apply for this waste type.

E52.1.1 Waste

The waste is well defined and consists of bead and powder resins, as well as filter aids from the systems reactor water clean-up (systems 331 and 334), condensate clean-up in BWR (system 333), treatment of fuel storage pool water in PWR and BWR (system 324), sampling (system 336), treatment of floor drainage (342), treatment of bottom blowing water (systems 417 and 337) and condensate clean-up in BWR (system 332).

E52.1.2 Packaging

The waste is packed in steel moulds. The mould is a cubic box made of carbon steel with dimensions 1.2×1.2×1.2 m. The wall is 5 mm thick, the bottom is 6 mm thick. The mould weighs about 400 kg. The mould contains a stirrer of carbon steel. It weighs about 25 kg. The mould is also provided with a splash plate.

The maximum permissible weight for a waste package including waste is 5,000 kg. The disposal volume for a concrete tank is 1.728 m^3.

E52.1.3 Treatment

Bead resins are metered directly in the mould while powder resins and filter aids are mixed and dewatered before metering. Cement and other additives are added during mixing. The void in a package is assumed to be about 5%. After hardening for at least two days, a concrete lid is cast on the package.

E52.1.4 Activity determination of radionuclides

Before the waste is transported from RAB to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The maximum measured activity content at manufacturing is about 1.0 TBq. The highest permissible surface dose rate is 500 mSv/h. The waste packages are usually free from surface contamination.

E52.1.5 Production of the waste type

Table E52-1 lists the number of packages for SFR.

Deposited packages refer to packages in the existing SFR as of 2012-12-31. The waste type R.16 started being produced in 1989 and has been deposited since 1995.

Forecasted packages refer to waste that has not yet been deposited and includes both waste that is interim-stored at the waste supplier and waste that has not yet been produced. Currently, there are 349 packages in interim storage and a production is planned of 60.1 packages per year up to 2027 and then 25 packages per year up to 2044. The waste vault that is indicated for forecasted waste is according to the acceptance criteria for the waste type.

Table E52-1. Number of packages of the waste type.

Number of packages	Waste vault	R.16
Deposited	Silo	1,164
Forecasted	(Silo)	1,675

E52.2 Average package for the waste type

E52.2.1 Material – waste, packaging and matrix

Table E52-2 gives values for an estimated average of the material content in waste type R.16. The material data refer to one mould. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E52-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	R.16
Ion exchange resins [kg]	Waste	700
Concrete [kg]	Packaging	500
Iron/steel [kg]	Packaging	400
Iron/steel surface [m^2]	Packaging	17
Iron/steel thickness [mm]	Packaging	5.0–6.0
Cement [kg]	Matrix	2,100
Iron/steel [kg]	Matrix (stirrer)	25
Iron/steel surface [m^2]	Matrix (stirrer)	3.0
Iron/steel thickness [mm]	Matrix (stirrer)	5.0
Void [m^3]	Matrix	0.085

E52.2.2 Radionuclide content

Table E52-3 provides values for a calculated average of the nuclide content in waste type R.16 at the closure of SFR on 2075-12-31. Activity data refer to one mould.

Table E52-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	R.16 [Bq]	Nuclide	R.16 [Bq]	Nuclide	R.16 [Bq]
H-3	6.31E+05	Cd-113m	8.16E+05	U-238	2.01E+03
Be-10	9.62E+01	Sn-126	1.20E+04	Np-237	2.35E+03
C-14 org	2.47E+08	Sb-125	1.37E+03	Pu-238	5.18E+06
C-14 inorg	8.45E+08	I-129	7.35E+04	Pu-239	2.09E+06
Cl-36	8.61E+04	Cs-134	1.01E+04	Pu-240	2.91E+06
Fe-55	2.14E+06	Cs-135	4.26E+05	Pu-241	3.61E+07
Co-60	2.09E+08	Cs-137	5.80E+09	Pu-242	1.50E+04
Ni-59	1.78E+09	Ba-133	3.87E+04	Am-241	2.04E+07
Ni-63	1.50E+11	Pm-147	2.01E+05	Am-242m	3.67E+04
Se-79	9.60E+04	Sm-151	4.42E+07	Am-243	1.50E+05
Sr-90	4.95E+08	Eu-152	2.38E+04	Cm-243	2.32E+04
Zr-93	1.60E+05	Eu-154	2.86E+07	Cm-244	3.44E+05
Nb-93m	1.27E+07	Eu-155	1.24E+06	Cm-245	1.50E+03
Nb-94	1.60E+06	Ho-166m	6.18E+05	Cm-246	3.97E+02
Mo-93	1.75E+05	U-232	7.97E+01		
Tc-99	3.24E+06	U-234	5.01E+03		
Pd-107	2.40E+04	U-235	1.00E+02		
Ag-108m	8.69E+06	U-236	1.51E+03		

E53 R.23

E53.1 Description of the waste type

The waste type R.23 consists of steel or concrete moulds containing concrete-solidified intermediate-level trash, scrap metal and sludge from RAB. The waste type has been deposited in both BMA and in 1BTF, where the packages in 1BTF act as supportive walls for the steel drums deposited there.

There is an approved waste type description for deposition of the waste type. Data are based on information in the waste type description and Triumf NG v1.0.1.3.

The acceptance criteria for BMA, described in Section E1.3.1, apply for this waste type. For the packages deposited in BTF, the acceptance criteria in Section E1.4.1 apply.

E53.1.1 Waste

The waste consists of trash, scrap metal and sludge. The trash consists of combustible and non-combustible material such as textiles, paper and plastics. The scrap metal consists of e.g. fittings, scrapped components, cables, suspensions and valves. The mixture of waste materials has varied over time depending on the maintenance work, revisions or other work carried out.

E53.1.2 Packaging

The waste is packed in steel and concrete moulds. The moulds are cubic boxes with dimensions 1.2×1.2×1.2 m. The steel mould is made of carbon steel and has a wall thickness of 5 mm and a bottom thickness of 6 mm. The mould contains press plates. The steel mould including press plates weighs about 660 kg. The concrete moulds are made of ready mixed concrete with reinforcement and are available in two designs with wall thicknesses of 10 cm and 25 cm.

The maximum permissible weight for a mould including waste is 5,000 kg. The disposal volume is 1.728 m^3.

E53.1.3 Treatment

The waste is placed directly in the mould. Compactible waste is compacted in both steel and concrete mould. Springback and floating to the surface at subsequent solidification is prevented by special press plates or reinforcing bars which are attached to the wall of the mould. When the mould is filled to the maximum, the waste is solidified with concrete. The void in a package is assumed to be about 25%. When the concrete has been hardened for two days, a lid with a thickness of at least 10 cm is cast in place.

E53.1.4 Activity determination of radionuclides

Before the waste is transported from RAB to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The maximum measured activity content is 1.0 TBq. The highest permissible surface dose rate is 100 mSv/h. The waste packages are usually free from surface contamination.

E53.1.5 Production of the waste type

Table E53-1 lists the number of packages for SFR.

Deposited packages refer to packages in the existing SFR as of 2012-12-31. The waste type R.23 started being produced in 1977 and has been deposited since 1993.

Forecasted packages refer to waste that has not yet been deposited and includes both waste that is interim-stored at the waste supplier and waste that has not yet been produced. Currently, there are 54 packages in interim storage and a production is planned of 4.5 packages per year up to 2027 and then 3 packages per year up to 2044. The waste vault that is indicated for forecasted waste is according to the acceptance criteria for the waste type.

Table E53-1. Number of packages of the waste type.

Number of packages	Waste vault	R.23 concrete mould	R.23 steel mould
Deposited	1BMA	338	96
Deposited	1BTF	21	0
Forecasted	(BMA)	0	172

E53.2 Average package for the waste type

E53.2.1 Material – waste, packaging and matrix

Table E53-2 gives values for an estimated average of the material content in waste type R.23. The material data refer to one mould. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E53-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	R.23 concrete mould	R.23 steel mould
Aluminium/zinc [kg]	Waste	1.0	4.0
Aluminium/zinc surface [m^2]	Waste	0.10	0.60
Aluminium/zinc thickness [mm]	Waste	5.0	5.0
Cellulose [kg]	Waste	11	44
Iron/steel [kg]	Waste	25	100
Iron/steel surface [m^2]	Waste	1.3	5.1
Iron/steel thickness [mm]	Waste	5.0	5.0
Other organic [kg]	Waste	25	100
Concrete [kg]	Packaging	1,840	500
Iron/steel [kg]	Packaging	274	661
Iron/steel surface [m^2]	Packaging	12	23
Iron/steel thickness [mm]	Packaging	12	5.0–6.0
Concrete [kg]	Matrix	565	1,356
Void [m^3]	Matrix	0.25	0.43

E53.2.2 Radionuclide content

Table E53-3 provides values for a calculated average of the nuclide content in waste type R.23 at the closure of SFR on 2075-12-31. Activity data refer to one mould.

Table E53-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	R.23 steel mould [Bq]	R.23 concrete mould [Bq]	R.23 concrete mould [Bq]
H-3	1.78E+05	1.75E+04	7.58E+02
Be-10	2.36E+01	1.48E+01	8.31E–01
C-14 org	0.00E+00	0.00E+00	0.00E+00
C-14 inorg	0.00E+00	0.00E+00	0.00E+00
Cl-36	2.36E+04	1.48E+04	8.31E+02
Fe-55	7.13E+05	1.01E+01	1.01E–01
Co-60	6.56E+07	2.72E+05	7.38E+03
Ni-59	4.45E+08	2.79E+08	1.57E+07
Ni-63	3.82E+10	1.97E+10	1.07E+09
Se-79	1.24E+03	2.73E+03	6.10E+01
Sr-90	2.49E+07	7.64E+06	3.85E+05
Zr-93	3.94E+04	2.47E+04	1.38E+03
Nb-93m	3.48E+06	5.58E+05	2.58E+04
Nb-94	3.93E+05	2.46E+05	1.38E+04
Mo-93	4.20E+04	3.04E+04	1.70E+03
Tc-99	3.86E+05	5.79E+05	3.24E+04
Pd-107	3.10E+02	6.83E+02	1.52E+01
Ag-108m	2.15E+06	1.29E+06	7.17E+04
Cd-113m	1.16E+04	5.68E+03	9.26E+01
Sn-126	1.55E+02	3.41E+02	7.62E+00
Sb-125	3.29E+03	1.29E+00	1.28E–02
I-129	9.46E+02	2.15E+03	4.80E+01
Cs-134	7.20E+00	3.29E–04	4.42E–07
Cs-135	5.13E+03	1.98E+04	4.41E+02
Cs-137	7.90E+07	9.11E+07	1.77E+06
Ba-133	1.11E+04	7.52E+02	3.08E+01
Pm-147	3.33E+03	1.16E–01	2.97E–04
Sm-151	5.81E+05	1.04E+06	2.23E+04
Eu-152	1.22E+03	5.56E+02	8.93E+00
Eu-154	4.23E+05	6.32E+04	8.18E+02
Eu-155	1.96E+04	1.79E+02	1.37E+00
Ho-166m	1.52E+05	9.39E+04	5.25E+03
U-232	9.13E+00	4.30E+00	2.31E–01
U-234	5.57E+02	3.50E+02	1.96E+01
U-235	1.12E+01	7.01E+00	3.93E–01
U-236	1.68E+02	1.05E+02	5.91E+00
U-238	2.23E+02	1.40E+02	7.84E+00
Np-237	2.79E+02	1.93E+02	1.10E+01
Pu-238	9.42E+05	4.73E+05	2.56E+04
Pu-239	2.32E+05	1.45E+05	8.14E+03
Pu-240	3.32E+05	2.08E+05	1.17E+04
Pu-241	4.53E+06	6.03E+05	2.72E+04
Pu-242	1.67E+03	1.05E+03	5.88E+01
Am-241	3.22E+06	2.01E+06	1.12E+05
Am-242m	4.13E+03	2.26E+03	1.24E+02
Am-243	1.66E+04	1.04E+04	5.83E+02
Cm-243	2.76E+03	8.49E+02	4.30E+01
Cm-244	4.25E+05	8.09E+04	3.82E+03
Cm-245	1.66E+02	1.04E+02	5.83E+00
Cm-246	4.42E+01	2.76E+01	1.55E+00

E54 R.23:D/R.4K23:D/R.4K23C:D

E54.1 Description of the waste type

Waste types R.23:D and R.4K23:D and R.4K23C:D are waste types adopted for low-level decommissioning waste from RAB. R.23:D consists of tetramoulds containing cement-solidified scrap metal. R.4K23:D consists of tetramoulds containing cement-solidified scrap metal and R.4K23C:D consists of tetramoulds containing concrete-solidified concrete.

There is no approved waste type description for deposition of the waste types. Material quantities and activity have been calculated based on Hansson et al. (2013), supplemented with assumptions on material composition and packaging and solidification material.

The acceptance criteria for BMA, described in Section E1.3.1, are assumed to be valid for these waste types.

E54.1.1 Waste

The waste in R.23:D and R.4K23:D consists mainly of scrap metal in the form of fittings and scrapped components. The waste in R.4K23C:D consists of parts from the reactor building and the biological shield.

E54.1.2 Packaging

The waste is packed in steel moulds or tetramoulds.

The steel mould is a cubic box with dimensions 1.2×1.2×1.2 m. It is made of carbon steel and has a wall thickness of 5 mm and a bottom thickness of 6 mm. The mould also contains press plates. The steel mould including press plates weighs about 660 kg.

The tetramould is a mould of steel plate with outer dimensions 2.4×2.4×1.2 m. The thickness of the walls is 4 mm, the floor 8 mm and the lid 15 mm. The packaging weighs about 1,700 kg.

The maximum permissible weight for a mould including waste is 5,000 kg and for a tetramould 20 tonnes. The disposal volume is 1.728 m^3 and 6.912 m^3.

E54.1.3 Treatment

The scrap metal is assumed to be packed with a packing degree of 1.1 tonnes/m^3. The waste is solidified with concrete. The void in the tetramoulds is estimated to be 25% of the inner volume of the packaging. The void in R.23:D is calculated to be about 26% of the inner volume of the packaging, in order to not exceed the maximum weight for a package.

E54.1.4 Activity determination of radionuclides

Before the waste is transported from RAB to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The highest permissible surface dose rate is 100 mSv/h. The waste packages are assumed to be free from surface contamination.

E54.1.5 Production of the waste type

Table E54-1 lists the number of packages for SFR.

The waste will be deposited during the years 2025–2031 and 2041–2047. The waste vault given is according to the acceptance criteria for the waste types.

Table E54-1. Number of packages of the waste type.

Number of packages	Waste vault	R.23:D	R.4K23:D	R.4K23C:D
Forecasted	(BMA)	153	314	149

E54.2 Average package for the waste type

E54.2.1 Material – waste, packaging and matrix

Table E54-2 gives values for an estimated average of the material content in waste types R.23:D, R.4K23:D and R.4K23C:D. The material data refer to one mould or tetramould. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E54-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	R.23:D	R.4K23:D	R.4K23C:D
Concrete [kg]	Waste	–	–	7,150
Iron/steel [kg]	Waste	1,870	7,150	–
Iron/steel surface [m^2]	Waste	96	367	–
Iron/steel thickness [mm]	Waste	5.0	5.0	–
Concrete [kg]	Packaging (lid)	500		
Iron/steel [kg]	Packaging	661	1,722	1,722
Iron/steel surface [m^2]	Packaging	23	46	46
Iron/steel thickness [mm]	Packaging	5.0–6.0	4.0–15	4.0–15
Concrete [kg]	Matrix	1,944	9,500	4,550
Void [m^3]	Matrix	0.44	1.63	1.63

E54.2.2 Radionuclide content

Table E54-3 provides values for a calculated average of the nuclide content in waste type R.23:D, R.4K23:D and R.4K23C:D at the closure of SFR on 2075-12-31. Activity data refer to one mould including drums.

Table E54-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	R.23:D [Bq]	R.4K23:D [Bq]	R.4K23C:D [Bq]
H-3	0.00E+00	1.33E+02	1.67E+09
Be-10	0.00E+00	2.78E–06	5.47E+00
C-14 org	0.00E+00	8.80E+03	3.60E+05
C-14 inorg	0.00E+00	1.15E+05	9.27E+05
C-14 ind	0.00E+00	1.53E+05	7.72E+06
Cl-36	0.00E+00	6.69E+04	2.30E+05
Ca-41	0.00E+00	1.87E+06	2.31E+07
Fe-55	1.61E+07	1.07E+07	4.23E+04
Co-60	4.06E+08	4.24E+08	2.20E+06
Ni-59	8.46E+07	5.84E+08	4.95E+05
Ni-63	8.20E+09	5.76E+10	4.73E+07
Se-79	0.00E+00	7.90E+01	4.91E+01
Sr-90	2.64E+07	1.66E+08	4.03E+04
Zr-93	8.56E+05	8.70E+05	4.20E+04
Nb-93m	2.59E+09	1.63E+10	4.88E+06
Nb-94	1.91E+07	1.44E+08	1.02E+05
Mo-93	1.85E+06	8.13E+06	3.39E+04
Tc-99	2.86E+05	1.65E+06	1.24E+04
Pd-107	0.00E+00	4.77E+03	4.76E+01
Ag-108m	4.69E+07	4.29E+07	5.81E+06
Cd-113m	0.00E+00	1.53E+03	4.23E+03
Sn-126	8.41E+02	2.82E+04	2.27E+02
Sb-125	1.48E+05	9.16E+04	7.59E+03
I-129	0.00E+00	2.01E+03	1.32E+01
Cs-134	0.00E+00	2.38E+02	7.48E+01
Cs-135	0.00E+00	1.17E+04	2.13E+02
Cs-137	0.00E+00	6.51E+08	2.37E+06
Ba-133	0.00E+00	1.04E+00	1.82E+05
Pm-147	1.40E+03	1.05E+03	1.55E+01
Sm-151	4.25E+05	2.67E+06	5.44E+07
Eu-152	5.48E+02	3.46E+03	1.46E+08
Eu-154	1.15E+05	3.99E+05	2.64E+06
Eu-155	2.54E+03	4.54E+03	4.65E+04
Ho-166m	5.37E+00	9.02E+01	8.76E+05
U-232	9.04E+00	7.77E+01	2.20E–02
U-235	1,68E–02	1.44E–01	2.72E–03
U-236	4.65E+02	3.36E+03	1.02E+00
Np-237	5.99E+02	4.54E+03	1.64E–01
Pu-238	3.55E+06	2.86E+07	4.29E+02
Pu-239	4.51E+05	3.19E+06	5.94E+04
Pu-240	6.49E+05	4.58E+06	1.38E+02
Pu-241	1.01E+07	4.92E+07	2.51E+03
Pu-242	3.29E+03	2.94E+04	7.39E–01
Am-241	2.47E+06	2.19E+07	3.65E+02
Am-242m	1.17E+04	1.15E+05	1.31E+00
Am-243	4.71E+04	4.77E+05	1.06E+01
Cm-243	7.18E+03	5.62E+04	1.07E+00
Cm-244	7.83E+05	6.55E+06	1.60E+02
Cm-245	9.08E+02	1.13E+04	2.43E–01
Cm-246	3.18E+02	3.69E+03	5.90E–02

E55 R.24

E55.1 Description of the waste type

The waste type consists of steel moulds containing concrete-solidified intermediate-level waste in the form of scrap metal, blasting agents, non-metallic scrap metal and ashes and slag from RAB.

There is no approved waste type description for deposition of the waste type. Data are based on information in the waste type description and Triumf NG v1.0.1.3.

The acceptance criteria for Silo, described in Section E1.1.1, apply for this waste type.

E55.1.1 Waste

The waste consists of scrap metal, non-metallic scrap, ingots, filters, blasting dust, ashes and slag. Organic material may only occur in limited amounts. The waste arises during operation and maintenance of active systems and at the exchange and waste treatment of components.

E55.1.2 Packaging

The waste is packed in steel moulds. The steel mould is a cubic box of reinforced sheet metal with dimensions 1.2×1.2×1.2 m. The thickness is 5 mm in the walls and lid and 8 mm in the bottom. The mould has a bolted lid. The mould weighs about 575 kg and the lid about 45 kg. Some waste requires inner packaging in the form of steel drums, steel cans or steel boxes.

The maximum permissible weight for a mould including waste is 5,000 kg. The disposal volume is 1.728 m^3.

E55.1.3 Treatment

The waste is sorted, segmented, compacted, melted, combusted or, when needed, treated in another way before it is packed for final disposal. Empty space in a mould is filled with a combination of foam glass and concrete. The void in a package is assumed to be about 25%.

E55.1.4 Activity determination of radionuclides

Before the waste is transported from RAB to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The maximum measured activity content is 2.0 TBq total activity of which a maximum of 0.5 TBq gamma-emitting nuclides for steel mould. The usually measured surface dose rate is less than 50 mSv/h. The highest permissible surface dose rate is 500 mSv/h. The waste packages are usually free from surface contamination.

E55.1.5 Production of the waste type

Table E55-1 lists the number of packages for SFR.

There are no deposited packages in the existing SFR as of 2012-12-31.

Forecasted packages refer to waste that has not yet been deposited and includes both waste that is interim-stored at the waste supplier and waste that has not yet been produced. Currently, there are 15 packages in interim storage and a production is planned of 2 packages per year from 2014 up to 2027 and then 1 package per year up to 2044. The waste vault that is indicated for forecasted waste is according to the acceptance criteria for the waste type.

Table E55-1. Number of packages of the waste type.

Number of packages	Waste vault	R.24
Deposited	–	0
Forecasted	(Silo)	60

E55.2 Average package for the waste type

E55.2.1 Material – waste, packaging and matrix

Table E55-2 gives values for an estimated average of the material content in waste type R.24. The material data refer to one mould. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E55-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	R.24
Concrete [kg]	Waste	200
Cellulose [kg]	Waste	5
Iron/steel [kg]	Waste	1,500
Iron/steel surface [m^2]	Waste	76
Iron/steel thickness [mm]	Waste	5.0
Other inorganic [kg]	Waste	1,230
Other organic [kg]	Waste	15
Iron/steel [kg]	Packaging	620
Iron/steel surface [m^2]	Packaging	17
Iron/steel thickness [mm]	Packaging	5.0–8.0
Concrete [kg]	Matrix	1,356
Void [m^3]	Matrix	0.43

E55.2.2 Radionuclide content

Table E55-3 provides values for a calculated average of the nuclide content in waste type R.24 at the closure of SFR on 2075-12-31. Activity data refer to one mould.

Table E55-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	R.24 [Bq]	Nuclide	R.24 [Bq]	Nuclide	R.24 [Bq]
H-3	1.22E+04	Ag-108m	1.10E+05	U-235	5.67E–01
Be-10	1.20E+00	Cd-113m	5.11E+04	U-236	8.52E+00
C-14 org	0.00E+00	Sn-126	5.00E+02	U-238	1.13E+01
C-14 inorg	0.00E+00	Sb-125	5.52E+03	Np-237	1.37E+01
Cl-36	1.20E+03	I-129	3.00E+03	Pu-238	5.10E+04
Fe-55	5.04E+04	Cs-134	1.49E+03	Pu-239	1.18E+04
Co-60	4.73E+06	Cs-135	1.00E+04	Pu-240	1.69E+04
Ni-59	2.26E+07	Cs-137	3.06E+08	Pu-241	3.04E+05
Ni-63	2.05E+09	Ba-133	7.77E+02	Pu-242	8.49E+01
Se-79	4.00E+03	Pm-147	1.58E+04	Am-241	1.63E+05
Sr-90	1.50E+06	Sm-151	2.01E+06	Am-242m	2.19E+02
Zr-93	2.00E+03	Eu-152	5.41E+03	Am-243	8.45E+02
Nb-93m	2.29E+05	Eu-154	2.00E+06	Cm-243	1.66E+02
Nb-94	2.00E+04	Eu-155	9.48E+04	Cm-244	2.74E+04
Mo-93	1.98E+03	Ho-166m	7.76E+03	Cm-245	8.45E+00
Tc-99	1.06E+05	U-232	5.03E–01	Cm-246	2.25E+00
Pd-107	1.00E+03	U-234	2.83E+01		

E56 R.29

E56.1 Description of the waste type

The waste type consists of concrete moulds containing cement-solidified intermediate-level evaporator concentrates from RAB.

There is no approved waste type description for deposition of the waste type. Data are based on information in the preliminary waste type description and Triumf NG v1.0.1.3.

The acceptance criteria for BMA, described in Section E1.3.1, apply for this waste type.

E56.1.1 Waste

The waste will primarily consist of system and drainage water from R1, with contribution of drainage water from R2-R4 and water from R2-R4 via the casting path 342 in connection with transport of ion exchange resins.

E56.1.2 Packaging

The waste is packed in concrete moulds. The concrete mould is a cubic box made of reinforced concrete with dimensions 1.2×1.2×1.2 m. The reinforcement consists of 12 mm thick steel bars that weigh about 274 kg. The wall thickness is 10 cm and the mould weighs about 1,600 kg. The mould is internally lined with polyethylene cellular plastic with a thickness of 20 mm and a weight of 10 kg. The mould is provided with a disposable stirrer that is solidified with the waste and a splash plate with holes for filling tube and stirrer axle. The stirrer is made of carbon steel and weighs 16 kg.

The maximum permissible weight for a mould including waste is 5,000 kg. The disposal volume is 1.728 m^3.

E56.1.3 Treatment

Water is gathered in evaporators for evaporation. The concentrate is then exposed to electrochemical oxidation to reduce the concentration of complexing agents. The concentrate is dried before it is mixed with cement during stirring directly in the mould. The total fill volume of a package with expansion canister is about 67%. After hardening for at least two days, a concrete lid is cast on the package.

E56.1.4 Activity determination of radionuclides

Before the waste is transported from RAB to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The usually measured activity content is about 1 GBq and the maximum measured activity content is 1.0 TBq. The usually measured surface dose rate is about 0.01–1 mSv/h. The highest permissible surface dose rate is 100 mSv/h. The waste packages are usually free from surface contamination.

E56.1.5 Production of the waste type

Table E56-1 lists the number of packages for SFR.

There are no deposited packages in the existing SFR as of 2012-12-31.

Forecasted packages refer to waste that has not yet been deposited. A production is planned of 20 packages per year from 2014 up to 2027 and then 5 packages per year up to 2044. The waste vault that is indicated for forecasted waste is according to the acceptance criteria for the waste type.

Table E56-1. Number of packages of the waste type.

Number of packages	Waste vault	R.29
Deposited	–	0
Forecasted	(BMA)	380

E56.2 Average package for the waste type

E56.2.1 Material – waste, packaging and matrix

Table E56-2 gives values for an estimated average of the material content in waste type R.29. The material data refer to one mould. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E56-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	R.29
Evaporator concentrates [kg]	Waste	700
Concrete [kg]	Packaging (including lid)	1,840
Iron/steel [kg]	Packaging	274
Iron/steel surface [m^2]	Packaging	12
Iron/steel thickness [mm]	Packaging	12
Other organic [kg]	Packaging	10
Cement [kg]	Matrix	1,600
Iron/steel [kg]	Matrix	16
Iron/steel surface [m^2]	Matrix	1.0
Iron/steel thickness [mm]	Matrix	5.0
Void [m^3]	Matrix	0.33

E56.2.2 Radionuclide content

Table E56-3 provides values for a calculated average of the nuclide content in waste type R.29 at the closure of SFR on 2075-12-31. Activity data refer to one mould.

Table E56-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	R.29 [Bq]	Nuclide	R.29 [Bq]	Nuclide	R.29 [Bq]
H-3	1.37E+03	Ag-108m	1.22E+04	U-235	1.20E–01
Be-10	1.32E–01	Cd-113m	1.41E+03	U-236	1.81E+00
C-14 org	0.00E+00	Sn-126	1.35E+01	U-238	2.40E+00
C-14 inorg	0.00E+00	Sb-125	4.97E+02	Np-237	2.74E+00
Cl-36	1.32E+02	I-129	8.10E+01	Pu-238	1.60E+04
Fe-55	4.53E+03	Cs-134	3.20E+01	Pu-239	2.50E+03
Co-60	4.83E+05	Cs-135	2.70E+02	Pu-240	3.52E+03
Ni-59	2.45E+06	Cs-137	8.42E+06	Pu-241	6.58E+04
Ni-63	2.24E+08	Ba-133	8.65E+01	Pu-242	1.80E+01
Se-79	1.08E+02	Pm-147	3.47E+02	Am-241	2.61E+04
Sr-90	8.00E+05	Sm-151	5.47E+04	Am-242m	4.67E+01
Zr-93	2.20E+02	Eu-152	1.49E+02	Am-243	1.79E+02
Nb-93m	2.58E+04	Eu-154	5.39E+04	Cm-243	3.60E+01
Nb-94	2.20E+03	Eu-155	2.33E+03	Cm-244	2.65E+03
Mo-93	2.18E+02	Ho-166m	8.54E+02	Cm-245	1.79E+00
Tc-99	3.36E+03	U-232	1.08E–01	Cm-246	4.76E–01
Pd-107	2.70E+01	U-234	6.00E+00		

E57 R.99:1

E57.1 Description of the waste type

The waste type R.99 exists only in a variant, R.99:1. That is, there is no waste type R.99 for deposition.

The variant R.99:1 consists of miscellaneous waste in the form of an old reactor pressure vessel lid from Ringhals 2.

There is an approved waste type description for deposition of R.99:1. Data are based on information in the waste type description and Triumf NG v1.0.1.3.

The acceptance criteria for BTF, described in Section E1.4.1, are followed in principle but some exceptions have been permitted since it is a miscellaneous waste type with only one package. The exceptions have been made for weight, geometry and surface dose rate.

E57.1.1 Waste

The waste consists of a reactor pressure vessel lid from Ringhals 2 made of carbon steel with stainless inner plating.

E57.1.2 Packaging

The lid has been provided with a protective hood on top. The hood is made of carbon steel with a thickness of 6 mm. The bottom has been sealed with a bottom plate in 20 mm carbon steel that is bolted on. The lid weighs about 65 tonnes, has a diameter of 4.7 m and a height of 3.2 m, including protective hood and bottom plate. The disposal volume is 100 m^3.

E57.1.3 Treatment

The lid has been externally sanitized and painted to prevent any surface contamination from spreading.

E57.1.4 Activity determination of radionuclides

Before the waste was transported from RAB to SFR, a measurement of gamma-emitting nuclides was made. The dominant gamma-emitting nuclide was Co-60. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The activity content is about 1.0 TBq. The highest permissible surface dose rate is 11.4 mSv/h (under the bottom plate). The waste package is free from surface contamination.

E57.1.5 Production of the waste type

Table E57-1 lists the number of packages for SFR.

The reactor pressure vessel lid R.99:1 was taken out of service in 1997 and deposited in SFR in 2001. No further production of the variant is planned.

Table E57-1. Number of packages of the waste type.

Number of packages	Waste vault	R.99:1
Deposited	1BTF	1
Forecasted	–	0

E57.2 Average package for the waste type

E57.2.1 Material – waste, packaging and matrix

Table E57-2 gives values for an estimated average of the material content in waste type R.99:1. The material data refer to one reactor pressure vessel lid. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E57-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	R.99:1
Iron/steel [kg]	Waste	65,000
Iron/steel surface [m^2]	Waste	150
Iron/steel thickness [mm]	Waste	160 (minimum)

E57.2.2 Radionuclide content

Table E57-3 provides values for a calculated average of the nuclide content in waste type R.99:1 at the closure of SFR on 2075-12-31. Activity data refer to one steel drum.

Table E57-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	R.99:1 [Bq]	Nuclide	R.99:1 [Bq]	Nuclide	R.99:1 [Bq]
H-3	3.39E+05	Cs-134	0.00E+00	Am-242m	0.00E+00
Be-10	1.68E+02	Cs-135	0.00E+00	Am-243	0.00E+00
C-14 org	0.00E+00	Cs-137	0.00E+00	Cm-243	0.00E+00
C-14 inorg	0.00E+00	Ba-133	1.58E+04	Cm-244	0.00E+00
Cl-36	1.68E+05	Pm-147	0.00E+00	Cm-245	0.00E+00
Fe-55	6.22E+02	Sm-151	0.00E+00	Cm-246	0.00E+00
Co-60	9.23E+06	Eu-152	0.00E+00		
Ni-59	3.11E+09	Eu-154	0.00E+00		
Ni-63	2.36E+11	Eu-155	0.00E+00		
Se-79	0.00E+00	Ho-166m	1.07E+06		
Sr-90	0.00E+00	U-232	0.00E+00		
Zr-93	2.79E+05	U-234	0.00E+00		
Nb-93m	9.59E+06	U-235	0.00E+00		
Nb-94	2.79E+06	U-236	0.00E+00		
Mo-93	3.45E+05	U-238	0.00E+00		
Tc-99	6.55E+06	Np-237	0.00E+00		
Pd-107	0.00E+00	Pu-238	0.00E+00		
Ag-108m	1.48E+07	Pu-239	0.00E+00		
Cd-113m	0.00E+00	Pu-240	0.00E+00		
Sn-126	0.00E+00	Pu-241	0.00E+00		
Sb-125	7.63E+01	Pu-242	0.00E+00		
I-129	0.00E+00	Am-241	0.00E+00		

E58 S.04

E58.1 Description of the waste type

The waste type S.04 consists of steel drums containing cement-solidified intermediate-level bead resins from SNAB and SVAFO. The package is produced by SNAB.

There is an approved waste type description for deposition of the waste type. Data are based on information in the waste type description and Triumf NG v1.0.1.3.

The acceptance criteria for Silo, described in Section E1.1.1, apply for this waste type.

E58.1.1 Waste

The waste is well defined and consists of bead resins from pool and primary circuit clean-up in the R2-reactor.

E58.1.2 Packaging

The waste is packed in standard 200-litre steel drums. The steel drum has a diameter of 0.57 m and a height of 0.84 m. The thickness of the steel plate is 1.5 mm in the casing and 3 mm in the bottom. The drum has an empty weight of about 60 kg. An inner stirrer, which weighs 10 kg, is mounted in the drum. The steel drum also has an inner lid with connection holes for waste, ventilation and stirrer and an outer lid which is applied after the solidification in cement.

The drums are positioned four by four on a drum tray of carbon steel. The drum tray has a bottom area of 1.2×1.2 m, a thickness of 4 mm and weighs 66.5 kg.

The maximum permissible weight for a drum including waste is 500 kg. The disposal volume for a drum on a drum tray is 0.324 m^3.

E58.1.3 Treatment

The ion exchange resins are pumped to a tank in the treatment facility. Stirrer and connections for ion exchange resins and ventilation are linked to the inner lid on a drum. The drum has then already been supplied with a weighed amount of cement. A set amount of water and ion exchange resins is added and mixed with the cement to a homogeneous mixture. The method with a so-called lost stirrer is used, which means that the stirrer remains in the waste package after mixing is completed and thereby acts as a reinforcement. The void is assumed to be about 90% of the inner volume of a packaging. The cement is hardened for at least 24 hours before the outer lid is put in place and attached.

E58.1.4 Activity determination of radionuclides

Before the waste is transported from SNAB to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are determined with measurement and calculation methods according to Appendix D.

The usually measured activity content is about 1–5 GBq/drum. The highest permissible surface dose rate is 50 mSv/h. This limitation comes from the manufacturing of the waste package. The waste packages are usually free from surface contamination.

E58.1.5 Production of the waste type

Table E58-1 lists the number of packages for SFR.

Deposited packages refer to packages in the existing SFR as of 2012-12-31. The waste type S.04 started being produced in 1999 and has been deposited since 2002.

Forecasted packages refer to waste that has not yet been deposited and includes both waste that is interim-stored at the waste supplier and waste that has not yet been produced. Currently, there are 22 packages in interim storage and a production is planned of 17.3 packages per year up to 2020, thereafter 10.5 packages per year up to 2040 and finally 10 packages per year up to 2045. The waste vault that is indicated for forecasted waste is according to the acceptance criteria for the waste type.

Table E58-1. Number of packages of the waste type.

Number of packages	Waste vault	S.04
Deposited	Silo	32
Forecasted	(Silo)	420

E58.2 Average package for the waste type

E58.2.1 Material – waste, packaging and matrix

Table E58-2 gives values for an estimated average of the material content in waste type S.04. The material data refer to one steel drum including a ¼ drum tray. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E58-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	S.04
Ion exchange resins [kg]	Waste	65
Iron/steel [kg]	Packaging	77
Iron/steel surface [m^2]	Packaging	4.7
Iron/steel thickness [mm]	Packaging (steel drum)	1.5–3.0
Iron/steel thickness [mm]	Packaging (drum tray)	4.0
Cement [kg]	Matrix	238
Iron/steel [kg]	Matrix	10
Iron/steel surface [m^2]	Matrix	0.50
Iron/steel thickness [mm]	Matrix	5.0
Void [m^3]	Matrix	0.021

E58.2.2 Radionuclide content

Table E58-3 provides values for a calculated average of the nuclide content in waste type S.04 at the closure of SFR on 2075-12-31. Activity data refer to one steel drum.

Table E58-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	S.04 [Bq]	Nuclide	S.04 [Bq]	Nuclide	S.04 [Bq]
H-3	3.55E+02	I-129	2.89E+03	Pu-242	6.50E+00
Be-10	2.96E–02	Cs-134	7.20E+01	Am-241	2.70E+04
C-14 org	7.98E+04	Cs-135	1.84E+04	Am-242m	1.69E+01
C-14 inorg	1.86E+05	Cs-137	1.56E+07	Am-243	6.47E+01
Cl-36	6.22E+03	Ba-133	2.34E+01	Cm-243	1.35E+01
Fe-55	2.25E+03	Pm-147	5.43E+02	Cm-244	1.28E+04
Co-60	1.81E+05	Sm-151	3.83E+04	Cm-245	6.47E–01
Ni-59	4.94E+04	Eu-152	7.00E+03	Cm-246	1.72E–01
Ni-63	6.73E+06	Eu-154	5.04E+04		
Se-79	7.50E+01	Eu-155	2.77E+03		
Sr-90	5.78E+05	Ho-166m	1.92E+02		
Zr-93	4.94E+01	U-232	3.93E–02		
Nb-93m	6.37E+03	U-234	2.21E+01		
Nb-94	4.93E+02	U-235	1.62E+01		
Mo-93	1.28E+04	U-236	6.53E–01		
Tc-99	7.22E+05	U-238	1.45E+01		
Pd-107	1.88E+01	Np-237	1.33E+00		
Ag-108m	2.73E+03	Pu-238	1.70E+05		
Cd-113m	3.37E+04	Pu-239	1.40E+03		
Sn-126	9.42E+00	Pu-240	2.37E+03		
Sb-125	2.47E+02	Pu-241	2.23E+04		

E59 S.11

E59.1 Description of the waste type

The waste type S.11 consists of steel moulds containing cement-solidified intermediate-level ion exchange resins and sludge from SNAB and SVAFO. The package is produced by SNAB.

There is an approved waste type description for deposition of the waste type. Data are based on information in the waste type description and Triumf NG v1.0.1.3.

The acceptance criteria for Silo, described in Section E1.1.1, apply for this waste type.

E59.1.1 Waste

The waste is well defined and consists of bead resins from R2, SNAB, institutions and the Ågesta reactor and sludge from the operation of SNAB.

E59.1.2 Packaging

The waste is packed in steel moulds. The steel mould is a cubic box made of steel with dimensions 1.2×1.2×1.2 m and a wall thickness of 5 mm. The mould weighs about 400 kg. The mould is provided with a stirrer and splash plate. The stirrer weighs about 25 kg.

The maximum permissible weight for a mould including waste is 5,000 kg. The disposal volume is 1.728 m^3.

E59.1.3 Treatment

The waste is homogenised before it is pumped to a mould. Cement is metered during stirring to obtain a homogeneous composition in the mould. The fill volume is assumed to be about 95%. After hardening for about two days a lid of cement is cast, which is also allowed to harden for two days, before the mould is relocated to interim storage. About 1,420 kg of cement is used for the matrix and about 400 kg for casting the top.

E59.1.4 Activity determination of radionuclides

Before the waste is transported from SNAB to SFR, a measurement of gamma-emitting nuclides is made.

The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The usually measured activity content for $\beta + \gamma$ is 127 MBq/kg and for α is 0.7 MBq/kg. The highest permissible surface dose rate is 5 mSv/h. This limitation comes from the manufacturing of the waste package. The waste packages are usually free from surface contamination.

E59.1.5 Production of the waste type

Table E59-1 lists the number of packages for SFR.

Deposited packages refer to packages in the existing SFR as of 2012-12-31. The waste type S.11 started being produced in 2000 and has been deposited since 2002.

Forecasted packages refer to waste that is interim-stored at the waste supplier. No future production of the waste type is planned. The waste vault that is indicated for forecasted waste is according to the acceptance criteria for the waste type.

Table E59-1. Number of packages of the waste type.

Number of packages	Waste vault	S.11
Deposited	Silo	96
Forecasted	(Silo)	10

E59.2 Average package for the waste type

E59.2.1 Material – waste, packaging and matrix

Table E59-2 gives values for an estimated average of the material content in waste type S.11. The material data refer to one mould. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E59-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	S.11
Cellulose [kg]	Waste	3.0
Ion exchange resins [kg]	Waste	667
Sludge [kg]	Waste	333
Iron/steel [kg]	Packaging	400
Iron/steel surface [m^2]	Packaging	17
Iron/steel thickness [mm]	Packaging	5.0
Cement [kg]	Matrix	1,820
Iron/steel [kg]	Matrix	25
Iron/steel surface [m^2]	Matrix	3.0
Iron/steel thickness [mm]	Matrix	5.0
Void [m^3]	Matrix	0.085

E59.2.2 Radionuclide content

Table E59-3 provides values for a calculated average of the nuclide content in waste type S.11 at the closure of SFR on 2075-12-31. Activity data refer to one mould.

Table E59-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	S.11 [Bq]	Nuclide	S.11 [Bq]	Nuclide	S.11 [Bq]	Nuclide	S.11 [Bq]
H-3	4.28E+03	Tc-99	7.34E+07	Eu-155	7.91E+03	Am-243	1.24E+06
Be-10	9.20E–01	Pd-107	1.84E+04	Ho-166m	5.18E+04	Cm-243	1.52E+05
C-14 org	0.00E+00	Ag-108m	4.25E+05	U-232	6.08E+02	Cm-244	7.23E+05
C-14 inorg	0.00E+00	Cd-113m	1.24E+05	U-234	4.16E+04	Cm-245	1.24E+04
Cl-36	1.32E+05	Sn-126	9.21E+03	U-235	8.32E+02	Cm-246	3.29E+03
Fe-55	1.70E+02	Sb-125	2.00E+01	U-236	1.25E+04		
Co-60	2.21E+04	I-129	2.31E+06	U-238	1.66E+04		
Ni-59	1.53E+06	Cs-134	4.20E–02	Np-237	1.99E+04		
Ni-63	1.82E+08	Cs-135	1.47E+07	Pu-238	3.24E+07		
Se-79	7.37E+04	Cs-137	1.13E+09	Pu-239	1.73E+07		
Sr-90	3.83E+08	Ba-133	2.04E+03	Pu-240	2.41E+07		
Zr-93	1.34E+04	Pm-147	8.12E+02	Pu-241	1.62E+08		
Nb-93m	9.91E+04	Sm-151	2.29E+07	Pu-242	1.25E+05		
Nb-94	1.53E+04	Eu-152	4.63E+04	Am-241	1.74E+08		
Mo-93	2.03E+05	Eu-154	1.50E+05	Am-242m	2.93E+05		

E60 S.12

E60.1 Description of the waste type

The waste type S.12 consists of steel containers containing low-level solid waste from SNAB and from industries, hospitals, institutions and SVAFO. The package is produced by SNAB.

There is an approved waste type description for deposition of the waste type. Data are based on information in the waste type description and Triumf NG v1.0.1.3.

The acceptance criteria for BLA, described in Section E1.5.1, apply for this waste type.

E60.1.1 Waste

The waste consists of ingots, slag, dust, smaller pieces of aluminium, furnace linings, lime, non-combustible trash and scrap. The scrap metal consists of e.g. fittings, scrapped components, cables, suspensions and insulation.

E60.1.2 Packaging

The waste is packed in ISO containers with dimensions 20-foot half height.

The container is made of carbon steel and has a length of 6.1 m, a width of 2.5 m and a height of 1.3 m. An empty container weighs about 1,900 kg. Material in the walls and roof is normally about 1.5 mm carbon steel. Open containers are sealed with a lid.

The maximum permissible weight for a container including waste is 20 tonnes. The disposal volume is 20 m^3.

E60.1.3 Treatment

Compactible waste is compacted and the bales are placed in steel drums or steel boxes prior being put in the container. Non-compactible waste can be placed in boxes/drums or directly in the container without further treatment. Cardboard boxes and wrapping are also used as inner packing.

As high a fill volume as possible should always be targeted; it can, however, vary widely depending on the nature of the waste. The void in a package is assumed to be 7.5 m^3.

E60.1.4 Activity determination of radionuclides

Before the waste is transported from SNAB to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

Measured activity content is normally less than 2 GBq, with a maximum measured activity content of 10 GBq. The usually measured surface dose rate is less than 0.5 mSv/h. The highest permissible surface dose rate is 2 mSv/h. The waste packages are usually free from surface contamination.

E60.1.5 Production of the waste type

Table E60-1 lists the number of packages for SFR.

There are no deposited packages in the existing SFR as of 2012-12-31.

Forecasted packages refer to waste that has not yet been deposited and includes both waste that is interim stored at the waste supplier and waste that has not yet been produced. Currently, there are 50 packages in interim storage and a production is planned of 7.4 packages per year up to year 2040 and then 0.4 packages per year up to year 2044. The waste vault that is indicated for forecasted waste is according to the acceptance criteria for the waste type.

Table E60-1. Number of packages of the waste type.

Number of packages	Waste vault	S.12
Deposited	–	0
Forecasted	(BLA)	260

E60.2 Average package for the waste type

E60.2.1 Material – waste, packaging and matrix

Table E60-2 gives values for an estimated average of the material content in waste type S.12. The material data refer to one container. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E60-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	S.12
Aluminium/zinc [kg]	Waste	100
Aluminium/zinc surface [m^2]	Waste	15
Aluminium/zinc thickness [mm]	Waste	5.0
Cellulose [kg]	Waste	500
Iron/steel [kg]	Waste	4,500
Iron/steel surface [m^2]	Waste	229
Iron/steel thickness [mm]	Waste	5.0
Other inorganic [kg]	Waste	400
Other organic [kg]	Waste	3,000
Iron/steel [kg]	Packaging	1,900
Iron/steel surface [m^2]	Packaging	105
Iron/steel thickness [mm]	Packaging	1.5
Void [m^3]	Matrix	7.5

E60.2.2 Radionuclide content

Table E60-3 provides values for a calculated average of the nuclide content in waste type S.12 at the closure of SFR on 2075-12-31. Activity data refer to one container.

Table E60-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	S.12 [Bq]	Nuclide	S.12 [Bq]	Nuclide	S.12 [Bq]
H-3	6.26E+03	Sb-125	1.82E+03	Pu-240	1.09E+06
Be-10	5.53E-01	I-129	6.22E+02	Pu-241	2.18E+07
C-14 org	0.00E+00	Cs-134	2.06E+02	Pu-242	6.47E+03
C-14 inorg	0.00E+00	Cs-135	2.07E+03	Am-241	8.22E+06
Cl-36	5.53E+02	Cs-137	6.60E+07	Am-242m	1.67E+04
Fe-55	2.65E+04	Ba-133	4.05E+02	Am-243	6.44E+04
Co-60	2.58E+06	Pm-147	3.43E+03	Cm-243	1.27E+04
Ni-59	9.22E+05	Sm-151	4.22E+05	Cm-244	9.03E+05
Ni-63	5.21E+07	Eu-152	1.44E+05	Cm-245	6.44E+02
Se-79	8.29E+02	Eu-154	1.42E+05	Cm-246	1.71E+02
Sr-90	6.28E+06	Eu-155	1.30E+04		
Zr-93	9.22E+02	Ho-166m	3.58E+03		
Nb-93m	1.15E+05	U-232	3.86E+01		
Nb-94	3.63E+05	U-234	2.16E+03		
Mo-93	9.14E+02	U-235	1.43E+06		
Tc-99	2.35E+04	U-236	6.49E+02		
Pd-107	2.07E+02	U-238	3.10E+05		
Ag-108m	1.55E+04	Np-237	9.58E+02		
Cd-113m	1.15E+04	Pu-238	5.75E+06		
Sn-126	1.04E+02	Pu-239	1.08E+06		

E61 S.12:D/S.12C:D

E61.1 Description of the waste type

Waste types S.12:D and S.12C:D are waste types adopted for low-level decommissioning waste in steel containers from SNAB. S.12:D contains scrap metal or secondary waste and S.12C:D contains concrete.

There is no approved waste type description for deposition of these waste types. Material quantities have been calculated based on data from SNAB, supplemented with assumptions on secondary waste, and material composition for the decommissioning waste, packaging and solidification material. Activity data are lacking for the waste.

The acceptance criteria for BLA, described in Section E1.5.1, are assumed to be valid for these waste types.

E61.1.1 Waste

The waste in S.12:D and S.12C:D consists of process equipment, building materials, slag, concrete, plastic tubes, fittings, drains and more. The scrap waste in S.12:D is assumed to consist mostly of scrap metal. The secondary waste in S.12:D is assumed to consist of trash and scrap similar to waste type R.12 from operational waste. The waste in S.12C:D contains concrete parts from plant buildings.

E61.1.2 Packaging

The waste is packed in ISO containers of carbon steel with dimensions 20-feet half height. The container has a length of 6.06 m, a width of 2.5 m and a height of 1.3 m. The thickness of the walls and roof is normally 1.5 mm. An empty container weighs about 1,900 kg. Open containers are sealed with a lid.

The maximum permissible weight for a container including waste is 20 tonnes. The disposal volume is 20 m^3.

E61.1.3 Treatment

The scrap metal is assumed to be packed with a packing degree of 1.1 tonnes/m^3. The secondary waste is assumed to be treated like waste type R.12 from operational waste. Concrete waste is assumed to be able to be packed with a packing degree of 1.5 tonnes/m^3 but is limited by the maximum weight capacity.

E61.1.4 Activity determination of radionuclides

Before the waste is transported from SNAB to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The highest permissible surface dose rate is 2 mSv/h. The waste packages are assumed to be free from surface contamination.

E61.1.5 Production of the waste type

Table E61-1 lists the number of packages for SFR.

The waste will be deposited during the years 2040–2043. The waste vault given is according to the acceptance criteria for the waste type.

Table E61-1. Number of packages of the waste type.

Number of packages	Waste vault	S.12:D scrap metal	S.12:D secondary waste	S.12C:D
Forecasted	(BLA)	49	14	26

E61.2 Average package for the waste type

E61.2.1 Material – waste, packaging and matrix

Table E61-2 gives values for an estimated average of the material content in waste types S.12:D and S.12C:D. The material data refer to one container. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E61-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	S.12:D scrap metal	S.12:D secondary waste	S.12C:D
Aluminium/zinc [kg]	Waste	–	100	–
Aluminium/zinc surface [m^2]	Waste	–	15	–
Aluminium/zinc thickness [mm]	Waste	–	5.0	–
Concrete [kg]	Waste	–	–	18,000
Cellulose [kg]	Waste	–	500	–
Iron/steel [kg]	Waste	16,500	4,500	–
Iron/steel surface [m^2]	Waste	846	229	–
Iron/steel thickness [mm]	Waste	5.0	5.0	–
Sand [kg]	Waste	–	–	–
Other inorganic [kg]	Waste	–	400	–
Other organic [kg]	Waste	–	3,000	–
Iron/steel [kg]	Packaging	1,900	1,900	1,900
Iron/steel surface [m^2]	Packaging	105	105	105
Iron/steel thickness [mm]	Packaging	1.5	1.5	1.5
Void [m^3]	Matrix	13	7.5	7.5

E61.2.2 Radionuclide content

There are no available data on the content of radionuclides in this waste type at present.

E62 S.13

E62.1 Description of the waste type

The waste type S.13 consists of steel drums containing concrete-solidified low-level ashes from SNAB, SVAFO and the nuclear power plants. The package is produced by SNAB.

There is a variant of the waste type, S.13:1. The difference between S.13 and S.13:1 is that in the variant the ashes are mixed with a small amount of pyrolysis residue. Only two drums of S.13:1 have been manufactured and no more will be made. It is judged that the same data can be used for S.13:1 as for S.13.

There are approved waste type descriptions for deposition of the waste type and the variant. Data are based on information in the waste type descriptions and Triumf NG v1.0.1.3.

The acceptance criteria for BTF, described in Section E1.4.1, apply for this waste type.

E62.1.1 Waste

The waste consists of combustion residue in the form of ashes, soot and slag. The waste can to a lesser degree also contain non-combustible material such as pieces of scrap and glass fibre material that followed with the combustible waste.

E62.1.2 Packaging

The waste is packed in 100-litre locking ring drums inside standard 200-litre steel drums.

The 100-litre drum weighs 10 kg and has a plate thickness of 1 mm. The diameter is 0.46 m and the height 0.70 m. The 200-litre drum weighs 20 kg and has a plate thickness of 1 mm. The diameter is 0.57 m and the height 0.84 m.

The maximum permissible weight for a drum including waste is 500 kg. The disposal volume is 0.324 m^3.

E62.1.3 Treatment

The combustion residue is gathered in a 100-litre drum directly after combustion. The drum is sealed with the aid of a locking ring seal with a lever arm. The drum is placed in a 200-litre drum and solidified with concrete. The void in a package is estimated to be 5% of the inner volume of the packaging.

E62.1.4 Activity determination of radionuclides

Before the waste is transported from SNAB to SFR, a measurement of gamma-emitting nuclides is made.

The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The usually measured activity content is about 50–200 MBq/drum. The maximum measured activity content is about 600–800 MBq/drum. The measured dose rate on 1 m is about 0.01–0.2 mSv/h. The highest permissible surface dose rate is 2 mSv/h. This limitation comes from the manufacturing. The waste packages are usually free from surface contamination.

E62.1.5 Production of the waste type

Table E62-1 lists the number of packages for SFR.

Deposited packages refer to packages in the existing SFR as of 2012-12-31. The waste type S.13 started being produced in 1981 and has been deposited since 1989. The variant S.13:1 has not yet been deposited.

Forecasted packages refer to waste that has not yet been deposited and includes both waste that is interim-stored at the waste supplier and waste that has not yet been produced. Currently, there are 1,711 packages in interim storage and a production is planned of 61.9 packages per year up to year 2020, thereafter 45.2 packages per year up to year 2040 and finally 41.2 packages per year up to year 2045. The waste vault that is indicated for forecasted waste is according to the acceptance criteria for the waste type.

Table E62-1. Number of packages of the waste type.

Number of packages	Waste vault	S.13/S.13:1
Deposited	1BTF	4,800
Forecasted	(BTF)	3,316

E62.2 Average package for the waste type

E62.2.1 Material – waste, packaging and matrix

Table E62-2 gives values for an estimated average of the material content in waste type S.13. Material data refer to one steel drum. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E62-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	S.13
Aluminium/zinc [kg]	Waste	6.5
Aluminium/zinc surface [m^2]	Waste	0.96
Aluminium/zinc thickness [mm]	Waste	5.0
Ashes [kg]	Waste	64
Iron/steel [kg]	Waste	10
Iron/steel surface [m^2]	Waste	2.7
Iron/steel thickness [mm]	Waste	1.0
Iron/steel [kg]	Packaging	20
Iron/steel surface [m^2]	Packaging	4.0
Iron/steel thickness [mm]	Packaging	1.0
Concrete [kg]	Matrix	240
Void [m^3]	Matrix	0.011

E62.2.2 Radionuclide content

Table E62-3 provides values for a calculated average of the nuclide content in waste type S.13 at the closure of SFR on 2075-12-31. Activity data refer to one steel drum.

Table E62-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	S.13 [Bq]
H-3	8.94E+01
Be-10	2.64E-02
C-14 org	0.00E+00
C-14 inorg	0.00E+00
Cl-36	2.64E+01
Fe-55	2.98E+02
Co-60	2.53E+04
Ni-59	1.61E+05
Ni-63	1.06E+07
Se-79	4.03E+01
Sr-90	2.23E+05
Zr-93	4.39E+01
Nb-93m	2.00E+03
Nb-94	3.51E+03
Mo-93	4.48E+03
Tc-99	2.24E+05
Pd-107	1.01E+01
Ag-108m	3.71E+04
Cd-113m	3.16E+02
Sn-126	5.03E+00
Sb-125	1.39E+01
I-129	1.42E+03
Cs-134	9.16E-01
Cs-135	8.92E+03
Cs-137	2.38E+06
Ba-133	5.16E+00
Pm-147	6.59E+01
Sm-151	1.84E+04
Eu-152	6.69E+03
Eu-154	2.05E+03
Eu-155	1.58E+00

Nuclide	S.13 [Bq]
Ho-166m	1.68E+02
U-232	1.47E+00
U-234	8.61E+01
U-235	2.27E+03
U-236	2.59E+01
U-238	3.44E+01
Np-237	4.42E+01
Pu-238	2.22E+05
Pu-239	4.30E+04
Pu-240	4.34E+04
Pu-241	7.04E+05
Pu-242	2.58E+02
Am-241	6.18E+05
Am-242m	6.54E+02
Am-243	2.57E+03
Cm-243	4.56E+02
Cm-244	3.05E+04
Cm-245	2.57E+01
Cm-246	6.83E+00

E63 S.14

E63.1 Description of the waste type

The waste type S.14 consists of steel drums containing concrete-solidified low-level trash and scrap metal from SNAB and from industries, hospitals, institutions, SVAFO and the nuclear power plants. The package is produced by SNAB.

There is a variant of the waste type, S.14:2. The variant is, however, only a container used as outer packaging for the drums. That is, all steel drums of waste type S.14 are placed in containers of the variant S.14:2. In this section, the drums and the container is treated as one waste type.

There are approved waste type descriptions for deposition of the waste type and the variant. Data are based on information in the waste type descriptions and Triumf NG v1.0.1.3.

The acceptance criteria for BLA, described in Section E1.5.1, apply for this waste type.

E63.1.1 Waste

The waste consists of trash and scrap metal in solid form. The waste consists mainly of iron, stainless steel and aluminium, but also of smaller quantities of lead, glass, insulation material and organic matter.

E63.1.2 Packaging

The waste is packed in 100-litre locking ring drums inside standard 200-litre steel drums, which are placed in ISO containers with dimensions 20-foot half height. On average, 38 200-litre drums fit in a container.

The 100-litre drum weighs 10 kg and has a plate thickness of 1 mm. The diameter is 0.46 m and the height 0.70 m. The 200-litre drum weighs 20 kg and has a plate thickness of 1 mm. The diameter is 0.57 m and the height 0.84 m.

The container has a length of 6.1 m, a width of 2.5 m and a height of 1.3 m. An empty container weighs about 1,900 kg. The material in the walls and roof is normally about 1.5 mm of carbon steel.

The maximum permissible weight for a 200-litre drum including waste is 500 kg. The maximum permissible weight for a container including waste is 20 tonnes. The disposal volume for a container is 20 m^3.

E63.1.3 Treatment

After any segmentation to suitable size, the waste is gathered in a 100-litre steel drum. The drum is sealed with the aid of a locking ring seal with a lever arm. The drum is placed in a 200-litre steel drum and solidified with concrete. After the solidification process, the concrete is allowed to harden for 3–5 days. The drums are placed in a container. The void in a container is assumed to be 7.5 m^3.

E63.1.4 Activity determination of radionuclides

Before the waste is transported from SNAB to SFR, a measurement of gamma-emitting nuclides is made.

The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The usually measured activity content is 0.3–100 MBq/package, the maximum measured activity content is less than 1 GBq. The highest permissible surface dose rate is 2 mSv/h. The waste packages are usually free from surface contamination.

E63.1.5 Production of the waste type

Table E63-1 lists the number of packages for SFR.

Deposited packages refer to packages in the existing SFR as of 2012-12-31. The waste type S.14 started being produced in 1982 and has been deposited since 1995.

Forecasted packages refer to waste that is interim-stored at the waste supplier. No future production of the waste type is planned. The waste vault that is indicated for forecasted waste is according to the acceptance criteria for the waste type.

Table E63-1. Number of packages of the waste type.

Number of packages	Waste vault	S.14
Deposited	1BLA	75
Forecasted	(BLA)	12

E63.2 Average package for the waste type

E63.2.1 Material – waste, packaging and matrix

Table E63-2 gives values for an estimated average of the material content in waste type S.14. The material data refer to one container including drums. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E63-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	S.14
Aluminium/zinc [kg]	Waste	190
Aluminium/zinc surface [m^2]	Waste	27
Aluminium/zinc thickness [mm]	Waste	5.0
Cellulose [kg]	Waste	303
Iron/steel [kg]	Waste	2,806
Iron/steel surface [m^2]	Waste	144
Iron/steel thickness [mm]	Waste	5.0
Other organic [kg]	Waste	152
Iron/steel [kg]	Packaging (steel drum)	1,138
Iron/steel surface [m^2]	Packaging (steel drum)	254
Iron/steel thickness [mm]	Packaging (steel drum)	1.0
Iron/steel [kg]	Packaging (container)	1,900
Iron/steel surface [m^2]	Packaging (container)	105
Iron/steel thickness [mm]	Packaging (container)	1.5
Concrete [kg]	Matrix	3,231
Void [m^3]	Matrix	7.5

E63.2.2 Radionuclide content

Table E63-3 provides values for a calculated average of the nuclide content in waste type S.14 at the closure of SFR in 2075-12-31. Activity data refer to one container including drums.

Table E63-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	S.14 [Bq]	Nuclide	S.14 [Bq]	Nuclide	S.14 [Bq]
H-3	2.62E+06	Cd-113m	4.19E+01	U-238	9.53E+06
Be-10	3.12E-01	Sn-126	1.87E+00	Np-237	2.76E+01
C-14 org	0.00E+00	Sb-125	4.36E-01	Pu-238	9.26E+04
C-14 inorg	0.00E+00	I-129	1.82E+03	Pu-239	2.33E+04
Cl-36	3.12E+02	Cs-134	2.08E-03	Pu-240	2.35E+04
Fe-55	3.63E+00	Cs-135	1.16E+04	Pu-241	7.73E+04
Co-60	1.40E+04	Cs-137	6.02E+05	Pu-242	1.40E+02
Ni-59	6.20E+05	Ba-133	2.15E+01	Am-241	3.27E+05
Ni-63	3.04E+07	Pm-147	3.49E-02	Am-242m	3.02E+02
Se-79	1.49E+01	Sm-151	5.87E+03	Am-243	1.39E+03
Sr-90	5.58E+04	Eu-152	1.22E+06	Cm-243	1.12E+02
Zr-93	5.21E+02	Eu-154	4.58E+04	Cm-244	4.64E+03
Nb-93m	1.40E+04	Eu-155	6.44E+01	Cm-245	1.39E+01
Nb-94	5.19E+03	Ho-166m	1.98E+03	Cm-246	3.69E+00
Mo-93	4.87E+05	U-232	5.72E-01		
Tc-99	2.46E+07	U-234	4.67E+01		
Pd-107	3.74E+00	U-235	2.87E+06		
Ag-108m	1.92E+06	U-236	1.41E+01		

E64 S.21

E64.1 Description of the waste type

The waste type S.21 consists of steel drums containing concrete-solidified intermediate-level trash and scrap metal from SNAB and from industries, hospitals and institutions. The package is produced by SNAB.

There is no approved waste type description for deposition of the waste type. Data are based on information in the preliminary waste type description and Triumf NG v1.0.1.3.

The acceptance criteria for BMA, described in Section E1.3.1, apply for this waste type.

E64.1.1 Waste

The waste consists of trash and scrap metal, mainly in the form of iron, stainless steel and aluminium. Small quantities of lead, glass, insulation material and organic matter also occur.

E64.1.2 Packaging

The waste is packed in 100-litre locking ring drums inside standard 200-litre steel drums.

The 100-litre drum weighs 10 kg and has a plate thickness of 1 mm. The diameter is 0.46 m and the height 0.70 m. The 200-litre drum weighs 20 kg and has a plate thickness of 1 mm. The diameter is 0.57 m and the height 0.84 m.

The drums are positioned four by four on a drum tray of steel that weighs 67 kg. The drum tray has the outer dimensions 1.2×1.2 m and a thickness of 4 mm.

The maximum permissible weight for a drum including waste is 500 kg. The disposal volume for a drum on a drum tray is 0.324 m^3.

E64.1.3 Treatment

After any segmentation to a suitable size, the waste is gathered in a 100-litre drum. The filled drum is sealed with a locking ring seal with a lever arm. The drum is placed in a 200-litre drum and solidified with concrete. The void in a package is assumed to be about 5%.

E64.1.4 Activity determination of radionuclides

Before the waste is transported from SNAB to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The usually measured activity content is about 1–100 MBq/package. The highest permissible surface dose rate is 2 mSv/h, which is based on the manufacturing of the waste package. The waste packages are usually free from surface contamination.

E64.1.5 Production of the waste type

Table E64-1 lists the number of packages for SFR.

There are no deposited packages in the existing SFR as of 2012-12-31.

Forecasted packages refer to waste that is interim-stored at the waste supplier. No future production of the waste type is planned. The waste vault indicated for forecasted waste is according to the acceptance criteria for the waste type.

Table E64-1. Number of packages of the waste type.

Number of packages	Waste vault	S.21
Deposited	–	0
Forecasted	(BMA)	488

E64.2 Average package for the waste type

E64.2.1 Material – waste, packaging and matrix

Table E64-2 gives values for an estimated average of the material content in waste type S.21. The material data refer to one steel drum including a ¼ drum tray. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E64-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	S.21
Aluminium/zinc [kg]	Waste	5.0
Aluminium/zinc surface [m^2]	Waste	0.7
Aluminium/zinc thickness [mm]	Waste	5.0
Cellulose [kg]	Waste	7.9
Iron/steel [kg]	Waste	74
Iron/steel surface [m^2]	Waste	6.5
Iron/steel thickness [mm]	Waste	5.0
Other organic [kg]	Waste	4.0
Iron/steel [kg]	Packaging	47
Iron/steel surface [m^2]	Packaging	7.4
Iron/steel thickness [mm]	Packaging (inner and outer drum)	1.0
Iron/steel thickness [mm]	Packaging (drum tray)	4.0
Cement [kg]	Matrix	96
Void [m^3]	Matrix	0.011

E64.2.2 Radionuclide content

Table E64-3 provides values for a calculated average of the nuclide content in waste type S.21 at the closure of SFR on 2075-12-31. Activity data refer to one steel drum.

Table E64-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	S.21 [Bq]	Nuclide	S.21 [Bq]	Nuclide	S.21 [Bq]
H-3	5.91E+01	Ag-108m	1.08E+03	U-235	3.77E+00
Be-10	1.19E-02	Cd-113m	5.21E+02	U-236	5.66E+01
C-14 org	0.00E+00	Sn-126	9.40E+00	U-238	7.52E+01
C-14 inorg	0.00E+00	Sb-125	7.93E-02	Np-237	1.23E+02
Cl-36	1.19E+01	I-129	5.64E+01	Pu-238	4.59E+05
Fe-55	2.53E+00	Cs-134	7.92E-04	Pu-239	9.38E+04
Co-60	5.34E+03	Cs-135	1.88E+02	Pu-240	9.48E+04
Ni-59	1.99E+04	Cs-137	4.45E+06	Pu-241	1.11E+06
Ni-63	1.03E+06	Ba-133	3.22E+00	Pu-242	5.64E+02
Se-79	7.52E+01	Pm-147	1.13E+00	Am-241	2.43E+06
Sr-90	4.17E+05	Sm-151	3.48E+04	Am-242m	1.38E+03
Zr-93	1.99E+01	Eu-152	2.62E+04	Am-243	5.61E+03
Nb-93m	1.35E+03	Eu-154	1.89E+03	Cm-243	8.47E+02
Nb-94	1.99E+02	Eu-155	5.55E+00	Cm-244	5.13E+04
Mo-93	1.97E+01	Ho-166m	7.68E+01	Cm-245	5.61E+01
Tc-99	1.94E+03	U-232	3.01E+00	Cm-246	1.49E+01
Pd-107	1.88E+01	U-234	1.88E+02		

E65 S.23

E65.1 Description of the waste type

The waste type S.23 consists of concrete moulds containing inserts of one or five concrete-solidified double lid drums with intermediate-level trash and scrap metal from SNAB and SVAFO and from industries, hospitals and institutions. The package is produced by SNAB.

There is no approved waste type description for deposition of the waste type. Data are based on information in the preliminary waste type description and Triumf NG v1.0.1.3.

The acceptance criteria for BMA, described in Section E1.3.1, apply for this waste type.

E65.1.1 Waste

The waste consists of metallic scrap and combustible and non-combustible trash in the form of e.g. rags, plastics, glass and rock wool.

E65.1.2 Packaging

The waste is packed in double lid drums placed in concrete moulds with a thick bottom.

The double lid drum is a steel drum with a diameter of 0.38 m, a height of 0.88 m and an empty weight of 20 kg. The drum with its content must not weigh more than 620 kg.

The concrete mould is a cubic box made of reinforced concrete with the dimensions 1.2×1.2×1.2 m and an empty weight of 1,600 kg. The wall thickness in the sides is 10 cm while the bottom and lid is 14 cm.

The maximum permissible weight for a mould including waste is 5,000 kg. The disposal volume is 1.728 m^3.

E65.1.3 Treatment

The waste is gathered in double lid drums after possible compaction and/or fragmentation. The lid of the drum is put in place and locked automatically according to the double lid principle. One or five double lid drums are placed in a concrete mould, one in the centre of the mould or one in the centre and four in positions inside the corners of the mould. The drums will then be solidified in concrete. The void in a package is assumed to be about 5%.

E65.1.4 Activity determination of radionuclides

Before the waste is transported from SNAB to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The usually measured activity content is about 1 TBq/package. The highest permissible surface dose rate is 100 mSv/h. The waste packages are usually free from surface contamination.

E65.1.5 Production of the waste type

Table E65-1 lists the number of packages for SFR.

There are no deposited packages in the existing SFR as of 2012-12-31.

Forecasted packages refer to waste that is not yet deposited. A production is planned of 22.5 packages per year up to year 2040 and then 17.6 packages per year up to year 2045. The waste vault that is indicated for forecasted waste is according to the acceptance criteria for the waste type.

Table E65-1. Number of packages of the waste type.

Number of packages	Waste vault	S.23
Deposited	–	0
Forecasted	(BMA)	718

E65.2 Average package for the waste type

E65.2.1 Material – waste, packaging and matrix

Table E65-2 gives values for an estimated average of the material content in waste type S.23. The material data refer to one mould. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E65-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	S.23
Aluminium/zinc [kg]	Waste	3.8
Aluminium/zinc surface [m^2]	Waste	0.60
Aluminium/zinc thickness [mm]	Waste	5.0
Cellulose [kg]	Waste	29
Iron/steel [kg]	Waste	113
Iron/steel surface [m^2]	Waste	5.8
Iron/steel thickness [mm]	Waste	5.0
Sludge [kg]	Waste	3.8
Other inorganic [kg]	Waste	139
Concrete [kg]	Packaging (including lid)	1,840
Iron/steel [kg]	Packaging	274
Iron/steel surface [m^2]	Packaging	12
Iron/steel thickness [mm]	Packaging	12
Concrete [kg]	Matrix	565
Void [m^3]	Matrix	0.067

E65.2.2 Radionuclide content

Table E65-3 provides values for a calculated average of the nuclide content in waste type S.23 at the closure of SFR on 2075-12-31. Activity data refer to one mould.

Table E65-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	S.23 [Bq]	Nuclide	S.23 [Bq]	Nuclide	S.23 [Bq]
H-3	1.38E+05	Ag-108m	9.47E+05	U-235	9.21E+01
Be-10	1.02E+01	Cd-113m	6.20E+04	U-236	1.38E+03
C-14 org	0.00E+00	Sn-126	4.70E+02	U-238	1.84E+03
C-14 inorg	0.00E+00	Sb-125	9.43E+04	Np-237	2.07E+03
Cl-36	1.02E+04	I-129	2.82E+03	Pu-238	1.27E+07
Fe-55	8.63E+05	Cs-134	3.02E+03	Pu-239	1.91E+06
Co-60	6.88E+07	Cs-135	9.40E+03	Pu-240	2.70E+06
Ni-59	1.70E+07	Cs-137	3.25E+08	Pu-241	6.34E+07
Ni-63	9.84E+08	Ba-133	9.19E+03	Pu-242	1.38E+04
Se-79	3.76E+03	Pm-147	3.01E+04	Am-241	1.97E+07
Sr-90	3.11E+07	Sm-151	1.97E+06	Am-242m	3.65E+04
Zr-93	1.70E+04	Eu-152	6.63E+03	Am-243	1.37E+05
Nb-93m	2.44E+06	Eu-154	2.78E+06	Cm-243	3.08E+04
Nb-94	1.70E+05	Eu-155	1.57E+05	Cm-244	2.43E+06
Mo-93	1.69E+04	Ho-166m	6.62E+04	Cm-245	1.37E+03
Tc-99	1.45E+05	U-232	8.64E+01	Cm-246	3.65E+02
Pd-107	9.40E+02	U-234	4.60E+03		

E66 S.23:D

E66.1 Description of the waste type

The waste type S.23:D is a waste type adopted for decommissioning waste from SNAB. It consists of steel moulds containing cement-solidified intermediate-level scrap metal from SNAB.

There is no approved waste type description for deposition of the waste type. Material quantities and activity have been calculated based on data from SNAB, supplemented with assumptions on material composition and packaging and solidification material.

The acceptance criteria for BMA, described in Section E1.3.1, are assumed to be valid for this waste type.

E66.1.1 Waste

The waste consists mainly of scrap metal in the form of process equipment.

E66.1.2 Packaging

The waste is packed in concrete moulds with thick bottom. The mould is a cubic box made of reinforced concrete with the dimensions 1.2×1.2×1.2 m and an empty weight of about 2,000 kg. The wall thickness in the sides is 10 cm while the bottom and lid is 14 cm.

The maximum permissible weight for a mould including waste is 5,000 kg. The disposal volume is 1.728 m^3.

E66.1.3 Treatment

The scrap metal is assumed to be packed with a packing degree of 1.1 tonnes/m^3. The waste is solidified with concrete. The void is estimated to be 25% of the inner volume of the packaging.

E66.1.4 Activity determination of radionuclides

Before the waste is transported from SNAB to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The highest permissible surface dose rate is 100 mSv/h. The waste packages are assumed to be free from surface contamination.

E66.1.5 Production of the waste type

Table E66-1 lists the number of packages for SFR.

The waste will be deposited during the years 2040–2043. The waste vault given is according to the acceptance criteria for the waste type.

Table E66-1. Number of packages of the waste type.

Number of packages	Waste vault	S.23:D
Forecasted	(BMA)	164

E66.2 Average package for the waste type

E66.2.1 Material – waste, packaging and matrix

Table E66-2 gives values for an estimated average of the material content in waste type S.23:D. The material data refer to one mould. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E66-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	S.23:D
Iron/steel [kg]	Waste	1,012
Iron/steel surface [m^2]	Waste	52
Iron/steel thickness [mm]	Waste	5.0
Iron/steel [kg]	Packaging	274
Iron/steel surface [m^2]	Packaging	12
Iron/steel thickness [mm]	Packaging	12
Concrete [kg]	Packaging	1,840
Concrete [kg]	Matrix	1,345
Void [m^3]	Matrix	0.23

E66.2.2 Radionuclide content

There are no available data on the content of radionuclides in this waste type at present.

E67 S.24/S.24:1

E67.1 Description of the waste type

The waste type S.24 consists of steel and concrete moulds containing concrete-solidified intermediate-level waste in the form of scrap metal, blasting agents, non-metallic scrap and ashes and slag from SNAB, SVAFO and RAB. The package is produced by SNAB.

There is a variant of the waste type, S.24:1, containing smoke detectors in steel moulds. Regarding material, the differences between S.24:1 and S.24 are considered so small that the same data are used but the activity will, however, be presented separately.

There is no approved waste type description for deposition of the waste type or the variant. Data are based on information in the preliminary waste type description and Triumf NG v1.0.1.3.

The acceptance criteria for Silo, described in Section E1.1.1, apply for this waste type.

E67.1.1 Waste

The waste consists of scrap metal, non-metallic scrap, ingots, filters, blasting dust, ashes and slag. Organic material may occur in limited amounts. The waste arises during operation and maintenance of active systems and at the exchange and waste treatment of components. The waste type S.24:1 contains smoke detectors.

E67.1.2 Packaging

The waste is packed in steel or concrete moulds.

The steel mould is a cubic box of reinforced sheet metal with dimensions 1.2×1.2×1.2 m. The thickness is 5 mm in the walls and lid and 8 mm in the bottom. The mould has a bolted lid. The mould weighs about 575 kg and the lid about 45 kg.

The concrete mould is a cubic box of reinforced concrete with the dimensions 1.2×1.2×1.2 m. The wall thickness is 115 mm in the upper parts of the side walls, 135 mm in the lower parts of the side walls towards the bottom and 140 mm in the bottom. The mould may have a bolted lid of steel plate. The mould weighs about 1,800 kg and the reinforced concrete lid about 500 kg. The reinforcing steel bars are about 12 mm in diameter.

Some waste requires inner packaging in the form of steel drums, steel cans or steel boxes.

The maximum permissible weight for a mould including waste is 5,000 kg. The disposal volume is 1.728 m^3.

E67.1.3 Treatment

The waste is sorted, segmented, compacted, melted, combusted or, when needed, treated in another way before it is packed for final disposal. Empty space in a mould is filled with a combination of foam glass and concrete. The remaining void in the package is assumed to be about 25%.

E67.1.4 Activity determination of radionuclides

Before the waste is transported from SNAB to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The maximum measured activity content is 2.0 TBq total activity, of which a maximum of 0.5 TBq of gamma-emitting nuclides for steel moulds and a maximum of 1.0 TBq gamma-emitting nuclides for concrete moulds. The usually measured surface dose rate is less than 50 mSv/h. The highest permissible surface dose rate is 500 mSv/h. The waste packages are usually free from surface contamination.

E67.1.5 Production of the waste type

Table E67-1 lists the number of packages for SFR.

There are no deposited packages in the existing SFR as of 2012-12-31.

Forecasted packages refer to waste that has not yet been deposited and includes both waste that is interim-stored at the waste supplier and waste that has not yet been produced. Currently, there are 17 packages in interim storage, of which one S.24 steel mould and 16 packages of the variant S.24:1. A production is planned of 71.3 packages of S.24 concrete moulds per year up to 2020 and then 5.3 packages per year up to 2044. A production is also planned of one S.24 steel mould per year and 3 packages of S.24:1 up to 2040. The waste vault that is indicated for forecasted waste is according to the acceptance criteria for the waste type.

Table E67-1. Number of packages of the waste type.

Number of packages	Waste vault	S.24 concrete mould	S.24 steel mould	S.24:1
Deposited	–	0	0	0
Forecasted	(Silo)	697	29	100

E67.2 Average package for the waste type

E67.2.1 Material – waste, packaging and matrix

Table E67-2 gives values for an estimated average of the material content in waste type S.24. The material data refer to one mould. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E67-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	S.24 concrete mould	S.24/S.24:1 steel mould
Aluminium/zinc [kg]	Waste	10	10
Aluminium/zinc surface [m^2]	Waste	1.5	1.5
Aluminium/zinc thickness [mm]	Waste	5.0	5.0
Concrete [kg]	Waste	235	200
Cellulose [kg]	Waste	5.0	5.0
Iron/steel [kg]	Waste	950	1,500
Iron/steel surface [m^2]	Waste	48	76
Iron/steel thickness [mm]	Waste	5.0	5.0
Other inorganic [kg]	Waste	830	1,230
Other organic [kg]	Waste	15	15
Concrete [kg]	Packaging	1,971	–
Iron/steel [kg]	Packaging	329	620
Iron/steel surface [m^2]	Packaging	14	17
Iron/steel thickness [mm]	Packaging	12	5.0–8.0
Concrete [kg]	Matrix	565	1,356
Void [m^3]	Matrix	0.25	0.43

E67.2.2 Radionuclide content

Table E67-3 provides values for a calculated average of the nuclide content in waste type S.24 at the closure of SFR on 2075-12-31. Activity data refer to one mould.

Table E67-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	S.24 concrete/ steel plate [Bq]	S.24:1 [Bq]
H-3	9.63E+03	0.00E+00
Be-10	1.20E+00	0.00E+00
C-14 org	0.00E+00	0.00E+00
C-14 inorg	0.00E+00	0.00E+00
Cl-36	1.20E+03	0.00E+00
Fe-55	2.35E+04	0.00E+00
Co-60	2.69E+06	0.00E+00
Ni-59	6.80E+06	0.00E+00
Ni-63	5.78E+08	0.00E+00
Se-79	4.00E+03	0.00E+00
Sr-90	1.92E+07	0.00E+00
Zr-93	2.00E+03	0.00E+00
Nb-93m	1.92E+05	0.00E+00
Nb-94	2.00E+04	0.00E+00
Mo-93	1.98E+03	0.00E+00
Tc-99	1.06E+05	0.00E+00
Pd-107	1.00E+03	0.00E+00
Ag-108m	1.10E+05	0.00E+00
Cd-113m	4.17E+04	0.00E+00
Sn-126	5.00E+02	0.00E+00
Sb-125	2.58E+03	0.00E+00
I-129	3.00E+03	0.00E+00
Cs-134	6.75E+02	0.00E+00
Cs-135	1.00E+04	0.00E+00
Cs-137	2.80E+08	0.00E+00
Ba-133	5.86E+02	0.00E+00
Pm-147	7.34E+03	0.00E+00
Sm-151	1.95E+06	0.00E+00
Eu-152	4.37E+03	0.00E+00
Eu-154	1.41E+06	0.00E+00
Eu-155	5.17E+04	0.00E+00
Ho-166m	7.75E+03	0.00E+00

Nuclide	S.24 concrete/ steel plate [Bq]	S.24:1 [Bq]
U-232	1.92E+02	0.00E+00
U-234	1.15E+04	0.00E+00
U-235	2.31E+02	0.00E+00
U-236	3.47E+03	0.00E+00
U-238	4.61E+03	0.00E+00
Np-237	5.37E+03	3.97E+06
Pu-238	2.90E+07	0.00E+00
Pu-239	4.79E+06	0.00E+00
Pu-240	6.76E+06	0.00E+00
Pu-241	8.41E+07	0.00E+00
Pu-242	3.45E+04	0.00E+00
Am-241	5.10E+07	2.30E+11
Am-242m	8.64E+04	0.00E+00
Am-243	3.44E+05	0.00E+00
Cm-243	5.73E+04	0.00E+00
Cm-244	3.72E+06	0.00E+00
Cm-245	3.44E+03	0.00E+00
Cm-246	9.13E+02	0.00E+00

E68 S.25:D

E68.1 Description of the waste type

The waste type S.25:D is a waste type adopted for decommissioning waste from SNAB. It consists of standard 200-litre steel drums containing concrete-solidified 100-litre drums with ashes.

There is no approved waste type description for deposition of the waste type. Material quantities and activity have been calculated based on data from SNAB, supplemented with assumptions on material composition and packaging and solidification material based on the waste type description for S.13 from operational waste.

The acceptance criteria for BMA, described in Section E1.3.1, are assumed to be valid for the waste type.

E68.1.1 Waste

The waste consists of ashes, dust and slag.

E68.1.2 Packaging

The waste is packed in 100-litre locking ring drums inside standard 200-litre steel drums.

The 100-litre drum weighs 10 kg and has a plate thickness of 1 mm. The diameter is 0.46 m and the height 0.70 m. The 200-litre drum weighs 20 kg and has a plate thickness of 1 mm. The diameter is 0.57 m and the height 0.84 m.

The steel drums are positioned four by four on a drum tray of steel. The drum tray has a bottom area of 1.2×1.2 m, a thickness of 4 mm and weighs 66.5 kg.

The maximum permissible weight for a drum including waste is 500 kg. The disposal volume for a drum on a drum tray is 0.324 m^3.

E68.1.3 Treatment

The waste is combusted and packed in the inner drum which is sealed by means of a locking ring seal with a lever arm. The inner drum is then placed in the outer drum for grouting in concrete. The outer drum, which has been prepared by filling the bottom with about 100 mm of concrete, has no lid. Prior to hardening, a metal tag with a unique identification number is placed on the concrete surface. The estimated void is 5% of the inner volume of the packaging.

E68.1.4 Activity determination of radionuclides

Before the waste is transported from SNAB to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The highest permissible surface dose rate is 100 mSv/h. The waste packages are assumed to be free from surface contamination.

E68.1.5 Production of the waste type

Table E68-1 lists the number of packages for SFR.

The waste will be deposited during the years 2040–2043. The waste vault given is according to the acceptance criteria for the waste type.

Table E68-1. Number of packages of the waste type.

Number of packages	Waste vault	S.25:D
Forecasted	(BMA)	2,384

E68.2 Average package for the waste type

E68.2.1 Material – waste, packaging and matrix

Table E68-2 gives values for an estimated average of the material content in waste type S.25:D. The material data refer to one steel drum including a ¼ drum tray. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E68-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	S.25:D
Aluminium/zinc [kg]	Waste	6.5
Aluminium/zinc surface [m^2]	Waste	1.0
Aluminium/zinc thickness [mm]	Waste	5.0
Ashes [kg]	Waste	64
Iron/steel [kg]	Packaging	47
Iron/steel surface [m^2]	Packaging	7.4
Iron/steel thickness [mm]	Packaging (inner and outer drum)	1.0
Iron/steel thickness [mm]	Packaging (drum tray)	4.0
Concrete [kg]	Matrix	240
Void [m^3]	Matrix	0.011

E68.2.2 Radionuclide content

There are no available data on the content of radionuclides in this waste type at present.

E69 V.12:D/V.12A:D/V.12C:D

E69.1 Description of the waste type

The waste types V.12:D, V.12A:D and V.12C:D are waste types adopted for low-level decommissioning waste in steel containers from SVAFO. V.12:D contains scrap metal or secondary waste. V.12A:D contains asphalt, soil and gravel and V.12C:D contains concrete.

There is no approved waste type description for deposition of the waste types. Material quantities have been calculated based on data from SVAFO, supplemented with assumptions on secondary waste, and material composition for the decommissioning waste, packaging and solidification material. There are no activity data for the waste.

The acceptance criteria for BLA, described in Section E1.5.1, are assumed to be valid for the waste types.

E69.1.1 Waste

The waste consists of material from the different plant units managed for demolition by SVAFO, such as buildings, handling units for waste and the reactor S-R2. The scrap in V.12:D is assumed to mainly consist of scrap metal. The secondary waste in V.12:D is assumed to consist of trash and scrap metal like waste type R.12 from operational waste. The waste in V.12C:D contains concrete and the waste in V.12A:D consists of soil, gravel and asphalt from sanitation around SVAFO's plant buildings.

E69.1.2 Packaging

The waste is packed in ISO containers with dimensions 20-foot half height. The container is made of steel with a length of 6.06 m, a width of 2.5 m and a height of 1.3 m. The thickness of the walls and roof is normally 1.5 mm. An empty container weighs about 1,900 kg. Open containers are sealed with a lid.

The maximum permissible weight for a container including waste is 20 tonnes. The disposal volume is 20 m^3.

E69.1.3 Treatment

The scrap metal is assumed to be packed with a packing degree of 1.1 tonnes/m^3. The secondary waste is assumed to be treated similar to waste type R.12 from operational waste. Asphalt, soil, gravel and concrete waste is assumed to be packed with a packing degree of 1.5 tonnes/m^3 but is limited by the maximum weight capacity.

E69.1.4 Activity determination of radionuclides

Before the waste is transported from SVAFO to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The highest permissible surface dose rate is 2 mSv/h. The waste packages are assumed to be free from surface contamination.

E69.1.5 Production of the waste type

Table E69-1 lists the number of packages for SFR.

The waste will be deposited during the years 2023, 2025, 2026 and 2045. The waste vault given is according to the acceptance criteria for the waste type.

Table E69-1. Number of packages of the waste type.

Number of packages	Waste vault	V.12:D scrap metal	V.12:D secondary waste	V.12A:D	V.12C:D
Forecasted	(BLA)	57	25	200	227

E69.2 Average package for the waste type

E69.2.1 Material – waste, packaging and matrix

Table E69-2 gives values for an estimated average of the material content in waste types V.12:D, V.12A:D and V.12C:D. The material data refer to one container. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E69-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	V.12:D scrap metal	V.12:D secondary waste	V.12A:D	V.12C:D
Aluminium/zinc [kg]	Waste	–	100	–	–
Aluminium/zinc surface [m^2]	Waste	–	15	–	–
Aluminium/zinc thickness [mm]	Waste	–	5.0	–	–
Asphalt, gravel, soil [kg]	Waste	–	–	18,000	–
Concrete [kg]	Waste	–	–	–	18,000
Cellulose [kg]	Waste	–	500	–	–
Iron/steel [kg]	Waste	16,500	4,500	–	–
Iron/steel surface [m^2]	Waste	846	229	–	–
Iron/steel thickness [mm]	Waste	5.0	5.0	–	–
Sand [kg]	Waste	–	–	–	–
Other inorganic [kg]	Waste	–	400	–	–
Other organic [kg]	Waste	–	3,000	–	–
Iron/steel [kg]	Packaging	1,900	1,900	1,900	1,900
Iron/steel surface [m^2]	Packaging	105	105	105	105
Iron/steel thickness [mm]	Packaging	1.5	1.5	1.5	1.5
Void [m^3]	Matrix	13	7.5	3.3	7.5

E69.2.2 Radionuclide content

There are no available data on the content of radionuclides in this waste type at present.

E70 Å.12:D/Å.12C:D

E70.1 Description of the waste type

Waste types Å.12:D and Å.12C:D are waste types adopted for low-level decommissioning waste in steel containers from Ågesta. Å.12:D contains scrap metal or secondary waste and Å.12C:D contains concrete.

There is no approved waste type description for deposition of the waste types. Material quantities and activity have been calculated based on Lindow (2012), supplemented with assumptions on secondary waste, and material composition for the decommissioning waste, packaging and solidification material.

The acceptance criteria for BLA, described in Section E1.5.1, are assumed to be valid for these waste types.

E70.1.1 Waste

The scrap metal in Å.12:D consists mainly of fittings and scrapped components. The secondary waste in Å.12:D is assumed to consist of trash and scrap metal like waste type R.12 from operational waste. The waste in Å.12C:D consists of concrete from the biological shield and concrete surfaces in the reactor building.

E70.1.2 Packaging

The waste is packed in ISO containers with dimensions 20-foot half height. The container is made of steel with a length of 6.06 m, a width of 2.5 m and a height of 1.3 m. The thickness of the walls and roof is normally 1.5 mm. An empty container weighs about 1,900 kg. Open containers are sealed with a lid.

The maximum permissible weight for a container including waste is 20 tonnes. The disposal volume is 20 m^3.

E70.1.3 Treatment

A container Å.12:D is assumed to be filled with about 16,300 kg of scrap metal. The secondary waste is assumed to be treated similar to waste type R.12 from operational waste. A container Å.12C:D is assumed to be filled with about 16,000 kg of concrete waste.

E70.1.4 Activity determination of radionuclides

Before the waste is transported from Ågesta to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137. Other nuclides that are not gamma-emitting are calculated according to Appendix D.

The highest permissible surface dose rate is 2 mSv/h. The waste packages are assumed to be free from surface contamination.

E70.1.5 Production of the waste type

Table E70-1 lists the number of packages for SFR.

The waste will be deposited during the years 2023–2025. The waste vault given is according to the acceptance criteria for the waste type.

Table E70-1. Number of packages of the waste type.

Number of packages	Waste vault	Å.12:D scrap metal	Å.12:D secondary waste	Å.12C:D
Forecasted	(BLA)	3	7	15

E70.2 Average package for the waste type

E70.2.1 Material – waste, packaging and matrix

Table E70-2 gives values for an estimated average of the material content in waste type Å.12:D and Å.12C:D. The material data refer to one container. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E70-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	Å.12:D scrap metal	Å.12:D secondary waste	Å.12C:D
Aluminium/zinc [kg]	Waste	–	175	–
Aluminium/zinc surface [m^2]	Waste	–	26	–
Aluminium/zinc thickness [mm]	Waste	–	5.0	–
Concrete [kg]	Waste	–	–	15,964
Cellulose [kg]	Waste	–	876	–
Iron/steel [kg]	Waste	16,332	7,887	–
Iron/steel surface [m^2]	Waste	838	402	–
Iron/steel thickness [mm]	Waste	5.0	5.0	–
Sand [kg]	Waste	–	–	–
Other inorganic [kg]	Waste	–	701	–
Other organic [kg]	Waste	–	5,258	–
Iron/steel [kg]	Packaging	1,900	1,900	1,900
Iron/steel surface [m^2]	Packaging	105	105	105
Iron/steel thickness [mm]	Packaging	1.5	1.5	1.5
Void [m^3]	Matrix	13	3.8	8.4

E70.2.2 Radionuclide content

Table E70-3 provides values for a calculated average of the nuclide content in waste type Å.12:D and Å.12C:D at the closure of SFR in 2075-12-31. Activity data refer to one container.

Table E70-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	Å.12:D [Bq]	Å.12C:D [Bq]	Nuclide	Å.12:D [Bq]	Å.12C:D [Bq]	Nuclide	Å.12:D [Bq]	Å.12C:D [Bq]
H-3	0.00E+00	3.65E+08	Cd-113m	6.40E+00	0.00E+00	Pu-239	6.30E+04	7.42E+04
Be-10	0.00E+00	0.00E+00	In115	0.00E+00	0.00E+00	Pu-240	4.16E+04	3.85E+04
C-14 org	6.54E+02	6.10E+02	Sn-126	0.00E+00	0.00E+00	Pu-241	1.62E+04	1.50E+04
C-14 inorg	1.68E+03	1.57E+03	Sb-125	1.23E–06	8.38E–07	Pu-242	0.00E+00	0.00E+00
C-14 ind	0.00E+00	2.40E+05	I-129	2.13E+00	1.99E+00	Am-241	8.46E+04	7.84E+04
Cl-36	1.03E+00	1.78E+03	Cs-134	3.30E–10	3.08E–10	Am-242m	0.00E+00	0.00E+00
Ca-41	0.00E+00	3.93E+06	Cs-135	3.02E+01	2.82E+01	Am-243	0.00E+00	0.00E+00
Fe-55	1.51E–05	5.75E–01	Cs-137	6.28E+05	5.86E+05	Cm-243	0.00E+00	0.00E+00
Co-60	2.14E+00	8.57E+03	Ba-133	0.00E+00	0.00E+00	Cm-244	1.87E+00	1.73E+00
Ni-59	2.37E+03	1.26E+04	Pm-147	0.00E+00	0.00E+00	Cm-245	0.00E+00	0.00E+00
Ni-63	1.45E+05	7.56E+05	Sm-151	1.52E+04	5.42E–01	Cm-246	0.00E+00	0.00E+00
Se-79	0.00E+00	0.00E+00	Eu-152	8.91E–01	1.54E+07			
Sr-90	3.53E+05	3.29E+05	Eu-154	1.48E+01	1.40E+05			
Zr-93	0.00E+00	0.00E+00	Eu-155	2.28E–02	2.94E–02			
Nb-93m	0.00E+00	0.00E+00	Ho-166m	0.00E+00	0.00E+00			
Nb-94	4.54E+02	5.33E–03	U-232	0.00E+00	0.00E+00			
Mo-93	0.00E+00	0.00E+00	U-235	3.48E–03	4.09E–03			
Tc-99	1.03E+03	9.61E+02	U-236	6.92E–02	6.41E–02			
Pd-107	0.00E+00	0.00E+00	Np-237	1.56E+00	1.45E+00			
Ag-108m	3.38E+02	5.69E–04	Pu-238	3.62E+03	3.36E+03			

E71 Å.4K23:D/Å.4K23C:D

E71.1 Description of the waste type

Waste types Å.4K23:D and Å.4K23C:D are waste types adopted for low-level decommissioning waste from Ågesta. Å.4K23:D consists of tetramoulds containing concrete-solidified scrap metal and Å.4K23C:D consists of tetramoulds containing concrete-solidified concrete.

There is no approved waste type description for deposition of the waste types. Material quantities and activity have been calculated based on Lindow (2012), supplemented with assumptions on material composition and packaging and solidification material.

The acceptance criteria for BMA, described in Section E1.3.1, are assumed to be valid for the waste types.

E71.1.1 Waste

The waste in Å.4K23:D consists mainly of scrap metal in the form of fittings and scrapped components. The waste in Å.4K23C:D consists of the containment and the biological shield.

E71.1.2 Packaging

The waste is packed in tetramoulds. The tetramould is a mould of steel plate with outer dimensions 2.4×2.4×1.2 m. The thickness of the walls is 4 mm, the floor 8 mm and the lid 15 mm. The packaging weighs about 1,700 kg.

The maximum permissible weight for a tetramould including waste is 20 tonnes. The disposal volume is 6.912 m^3.

E71.1.3 Treatment

A tetramould Å.4K23:D is assumed to be filled with about 7,300 kg of waste and Å.4KC:D with about 6,900 kg. The waste is solidified with concrete. The void is estimated to 25% of the inner volume of the packaging.

E71.1.4 Activity determination of radionuclides

Before the waste is transported from SNAB to SFR, a measurement of gamma-emitting nuclides is made. The dominant gamma-emitting nuclides are Co-60 and Cs-137.

The highest permissible surface dose rate is 100 mSv/h. The waste packages are assumed to be free from surface contamination.

E71.1.5 Production of the waste type

Table E71-1 lists the number of packages for SFR.

The waste will be deposited during the years 2023–2025. The waste vault given is according to the acceptance criteria for the waste type.

Table E71-1. Number of packages of the waste type.

Number of packages	Waste vault	Å.4K23:D	Å.4K23C:D
Forecasted	(BMA)	45	5

E71.2 Average package for the waste type

E71.2.1 Material – waste, packaging and matrix

Table E71-2 gives values for an estimated average of the material content in waste type Å.4K23:D and Å.4K23C:D. The material data refer to one tetramould. Besides weights, corrosion surface and thickness for metals and void in the waste package are given.

Table E71-2. Material content, corrosion surface and void in an estimated average package for the waste type.

Material	Origin	Å.4K23:D	Å.4K23C:D
Concrete [kg]	Waste	–	6,918
Iron/steel [kg]	Waste	7,340	–
Iron/steel surface [m^2]	Waste	376	–
Iron/steel thickness [mm]	Waste	5.0	–
Iron/steel [kg]	Packaging	1,722	1,722
Iron/steel surface [m^2]	Packaging	46	46
Iron/steel thickness [mm]	Packaging	4.0–15	4.0–15
Concrete [kg]	Matrix	9,442	4,782
Void [m^3]	Matrix	1.6	1.6

E71.2.2 Radionuclide content

Table E71-3 provides values for a calculated average of the nuclide content in waste type Å.4K23:D and Å.4K23C:D at the closure of SFR on 2075-12-31. Activity data refer to one tetramould.

Table E71-3. Radioactivity for a calculated average package of the waste type, at the closure of SFR in 2075.

Nuclide	Å.4K23:D [Bq]	Å.4K23C:D [Bq]
H-3	3.86E+10	3.28E+08
Be-10	0.00E+00	0.00E+00
C-14 org	2.02E+07	0.00E+00
C-14 inorg	5.20E+07	0.00E+00
C-14 ind	0.00E+00	1.60E+06
Cl-36	2.64E+01	1.19E+04
Ca-41	0.00E+00	2.62E+07
Fe-55	2.28E+01	3.84E+00
Co-60	2.77E+05	5.72E+04
Ni-59	2.21E+08	8.40E+04
Ni-63	1.33E+10	5.05E+06
Se-79	0.00E+00	0.00E+00
Sr-90	3.73E+07	0.00E+00
Zr-93	0.00E+00	0.00E+00
Nb-93m	0.00E+00	0.00E+00
Nb-94	2.34E+06	0.00E+00
Mo-93	0.00E+00	0.00E+00
Tc-99	2.52E+05	0.00E+00
Pd-107	0.00E+00	0.00E+00
Ag-108m	5.40E+07	0.00E+00
Cd-113m	2.94E+05	0.00E+00
In115	6.96E+03	0.00E+00
Sn-126	0.00E+00	0.00E+00
Sb-125	1.74E-03	0.00E+00
I-129	5.46E+01	0.00E+00
Cs-134	8.46E-09	0.00E+00
Cs-135	6.91E+03	0.00E+00
Cs-137	8.67E+07	0.00E+00
Ba-133	0.00E+00	0.00E+00
Pm-147	0.00E+00	0.00E+00
Sm-151	3.89E+05	3.62E+00
Eu-152	2.28E+01	1.03E+08
Eu-154	3.79E+02	9.34E+05
Eu-155	5.85E-01	5.37E-02
Ho-166m	0.00E+00	0.00E+00
U-232	0.00E+00	0.00E+00
U-235	2.20E-01	5.81E-03
U-236	4.37E+00	3.37E-06
Np-237	9.85E+01	9.44E-08
Pu-238	2.29E+05	6.78E-03
Pu-239	3.98E+06	1.05E+05
Pu-240	2.62E+06	2.02E+00
Pu-241	1.02E+06	1.92E-04
Pu-242	0.00E+00	0.00E+00
Am-241	5.34E+06	5.01E-03
Am-242m	0.00E+00	0.00E+00
Am-243	0.00E+00	0.00E+00
Cm-243	0.00E+00	0.00E+00
Cm-244	1.18E+02	0.00E+00
Cm-245	0.00E+00	0.00E+00
Cm-246	0.00E+00	0.00E+00

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