R-07-17

Low and intermediate level waste in SFR 1

Reference waste inventory 2007

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November 2007

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Preface

This document describes the method used when estimating future amounts of waste in SFR 1 and corresponding radionuclide inventory. Also, the resulting amounts and inventory used in the safety analysis for SFR are presented. This document constitutes one of the main references to SFR 1 SAR-08.

Lisa Almkvist, Vattenfall Power Consultant AB and Anna Gordon, SKB have compiled the report.

This document has been reviewed and all comments have been documented in accordance with SKIFS 2004:1.

Stockholm, November 2007

Agneta Innergård

Project manager, SFR 1 SAR-08

Summary

The objective with this report is to describe all the waste and the waste package that is expected to be deposited in SFR 1 at the time of closure. The report will form the basis for the release calculation in the safety analysis for SFR 1.

Three different scenarios are explored in this report; the waste inventory is based on an estimated operational lifetime of the Swedish nuclear power plants of 50 and 60 years and that closure of the SFR 1 repository will take place in 2040 or 2050 respectively. The third scenario is where the repository is full (one part where the activity adds up to 10^{16} Bq and one part where the repository is considered full regarding volume). In the report, data about geometries, weights, materials, chemicals and radionuclide are given. No chemotoxic material has been identified in the waste.

The inventory is estimated using the Prosit-interface which extracts information from the Triumf database. The inventory is based on so called "waste types" and the waste types' "reference waste package". The reference waste package combined with a prognosis of the number of waste packages to be delivered to SFR 1 gives the final waste inventory for SFR 1. All reference waste packages are thoroughly described in the appendices of this report. The reference waste packages are as far as possible based on actual experiences and measurements.

The radionuclide inventory is also based on actual measurements. The inventory is based on measurements of ^{60}Co and ^{137}Cs in waste packages and on measurements of ^{239}Pu and ^{240}Pu in reactor water. Other nuclides in the inventory are calculated with correlation factors.

Sammanfattning

Syftet med denna rapport är att beskriva det avfall och de avfallskollin som förväntas finnas deponerade i SFR 1 vid förslutning. Rapporten ligger till grund för de spridnings- och dosberäkningar som redovisas i säkerhetsanalysen för SFR 1.

Tre olika scenarion har beaktats: Avfallsinventariet bygger på att de svenska reaktorerna har en drifttid på 50 eller 60 år och att förslutning av SFR 1 sker 2040 eller 2050. Det tredje scenariot visar ett fullt förvar (ena delen med den totala aktiviteten som 10^{16} Bq och andra delen med fullt förvar volymmässigt). Det fulla inventariet kan användas som ett extremt fall för spridningsberäkningarna i säkerhetsrapporten. I rapporten ges uppgifter om geometrier, vikter, materialsammansättning, kemikalier och radionuklider. Inget kemotoxiskt material har identifierats i avfallet.

Inventariet har uppskattats med hjälp av användargränssnittet Prosit som hämtar data från Triumf-databasen. Inventariet är baserat på så kallade avfallstyper och avfallstypens 'normalkolli' (Reference Waste Package). Normalkollin kombinerat med en prognos för antalet avfallskollin ger det slutliga inventariet för SFR 1. Alla normalkollin beskrivs noggrant i rapportens bilagor. De normalkollin som redovisas är i så stor utsträckning som möjligt baserade på faktiska erfarenheter och mätningar.

Även radionuklidinnehållet i normalkollina är baserat på faktiska erfarenheter. Inventariet är uppbyggt på mätningar av ^{60}Co och ^{137}Cs på befintliga avfallskollin och på ^{239}Pu och ^{240}Pu mätningar på reaktorvatten. Baserat på dessa mätningar har resterande inventarier tagits fram med hjälp av korrelationsfaktorer.

Contents

1 General

1.1 Background

The SFR 1 repository has been in operation since 1988. In the final permit for operation the authorities demanded that the safety assessment of the repository be thoroughly updated at least every 10 years. A preliminary safety report was done in 1983; a final safety report was done in 1987. An update of the final report was published in 1991.

In 1997 a project named SAFE (**S**afety **A**ssessment of **F**inal Repository for Radioactive Operational Wast**e**) was initiated in order to make a complete update of the safety report and the safety assessment. A report (R-01-03) including a detailed prognosis of the waste in SFR 1 including materials and radionuclide content was produced for the purpose to serve as input to the release and dose calculations. This report is an update of the previous Reference Waste Inventory report (R-01-03).

Although the report includes very precise numbers one should keep in mind that this is only a prognosis. The assumptions made can sometimes be crude due to the uncertainties that inevitably lie in the future.

1.2 SFR 1

SFR 1 is situated in Forsmark in the northern part of Uppland, close to the Forsmark nuclear power plant. The storage vaults are located in the bedrock, approximately 60 m below the seabed, 1 km off the coast. The underground part of the repository is accessed through two tunnels.

SFR 1 is designed for final disposal of low and intermediate level radioactive waste from the Swedish nuclear power plants and the Central Interim Storage for Spent Nuclear Fuel (Clab) and for similar waste from other industries, research and medical care.

In total the SFR 1 was intended for 90,000 m^3 of waste. In the previous safety assessment /SKB 1993/ the total radioactivity in this waste was assumed to be 10^{16} Bq. Today (2006) the waste capacity in the existing parts of the facility is approximately 60,000 m^3 and approximately 31,000 m^3 of waste is disposed.

The repository is designed to isolate the waste from the biosphere in order to avoid harmful consequences to man and environment both under operation and after closure. This is accomplished by the emplacement in rock under the seabed and by the technical barriers surrounding the waste.

SFR 1 is divided into four types of rock vaults:

- The Silo.
- Rock Vault for Intermediate Level Waste (BMA).
- Rock Vaults for Concrete Tanks (BTF).
- Rock Vault for Low Level Waste (BLA).

The rock vaults are connected through a system of tunnels. The design of the first stage of SFR 1 is shown in Figure 1-1.

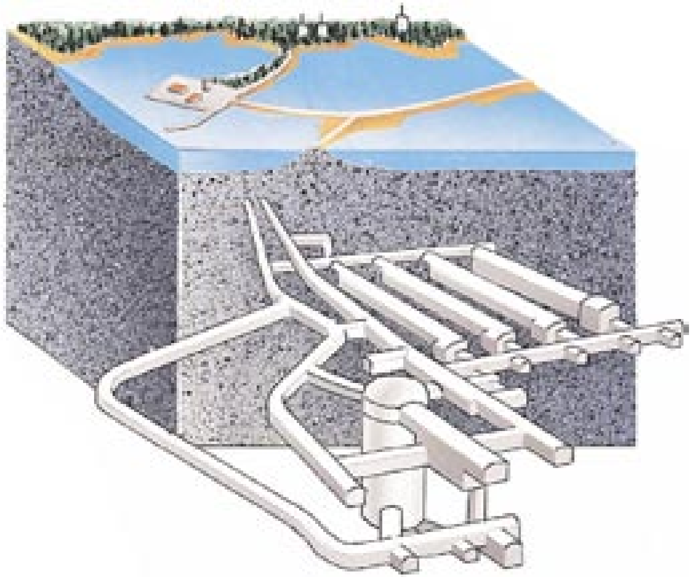

Figure 1-1. *SFR 1.*

1.2.1 The rock vault for low level waste (BLA)

The waste deposited in BLA is mainly low level scrap metal and refuses placed in standard steel containers. Some of the waste inside the containers is placed in steel drums and others in bales.

The rock vault is 160 m long and has a width of 15 m. Figure 1-2 shows BLA. There are no concrete structures in BLA except the floor.

The containers are placed two in a row and three full height containers or six half height containers in height. Most of the containers are half height. No backfill is planned.

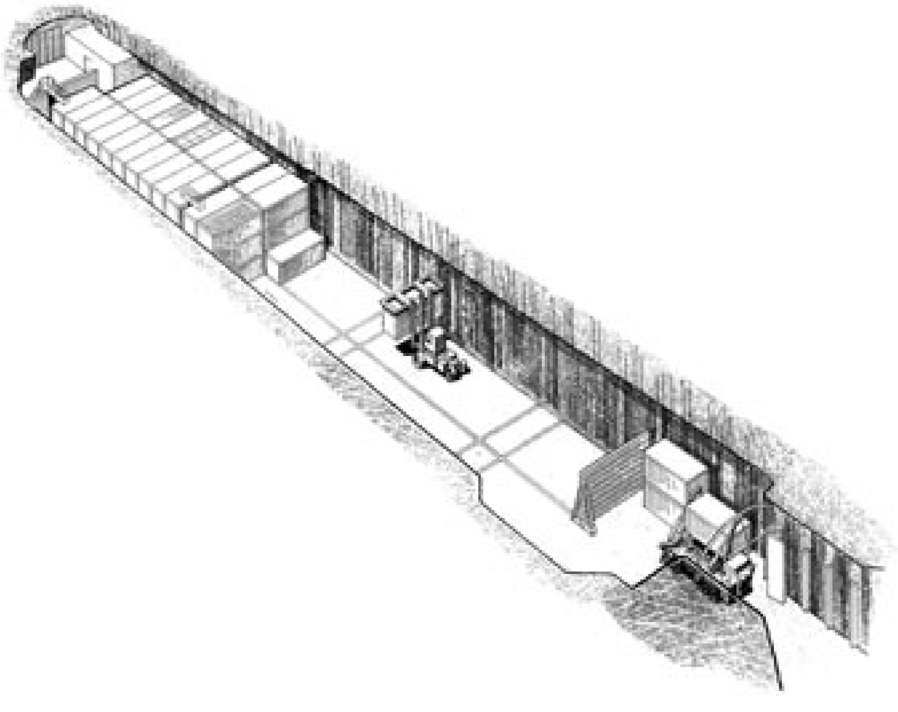

Figure 1-2. *The BLA.*

1.2.2 The rock vaults for concrete tanks (BTF)

In SFR 1 there are two rock vaults for concrete tanks, 1BTF and 2BTF. The waste placed in BTF is de-watered low-level ion exchange resin in concrete tanks. In addition, some drums with ashes are stored in 1BTF.

The rock vaults are 160 m long and width of 15 m. Figure 1-3 shows the principal picture of the vaults, Figure 1-4 shows a picture from 2BTF.

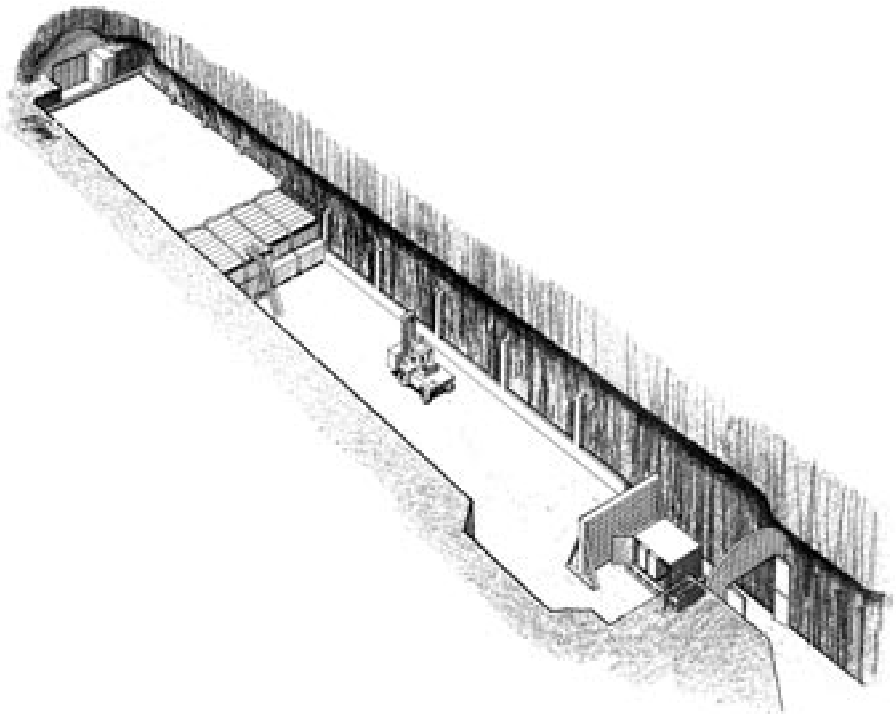

Figure 1-3. *1BTF and 2BTF.*

Figure 1-4. *Picture from 2BTF.*

The concrete tanks, each with a volume of 10 m^3, are piled in two levels with four tanks in each row. A concrete radiation protection lid is placed on top of the pile. The space between the different tanks is backfilled with concrete and the space between the tanks and the rock wall will be filled with, for example, sand stabilised in cement.

1.2.3 The rock vault for intermediate level waste (BMA)

The radioactivity in the waste deposited in the BMA is mainly lower than in the Silo waste. The waste consists of ion exchange resins, scrap metal and trash in a concrete or bitumen matrix. The waste packages are of the same type as in the Silo, i.e. moulds and drums.

The rock vault is 160 m long and has a width of 19.5 m. Figure 1-5 shows BMA. The concrete structure in the vault is divided into 15 compartments. The compartments are built like big boxes with concrete walls in between. The waste is piled on top of the concrete floor in a way that allows the concrete moulds to act as support for pre-fabricated concrete lids. The lids are put in position as soon as the compartments are filled. Finally a layer of concrete is casted on top of the lid. Figure 1-6 shows a view over the compartments.

Between the concrete structure and the rock wall there is a 2 m wide space. This space will be filled with sand. The space above the concrete structure could be left unfilled but could also be backfilled. Plugs will be placed in the two entrances to the vault when the repository is closed.

1.2.4 The Silo

The main part of the radioactivity in the waste in SFR 1 is intended for disposal in the Silo. The waste is mainly composed of ion-exchange resin in a concrete or bitumen matrix.

The Silo consists of a concrete cylinder with the height of approximately 50 m and a diameter of approximately 30 m. Figure 1-7 shows the Silo.

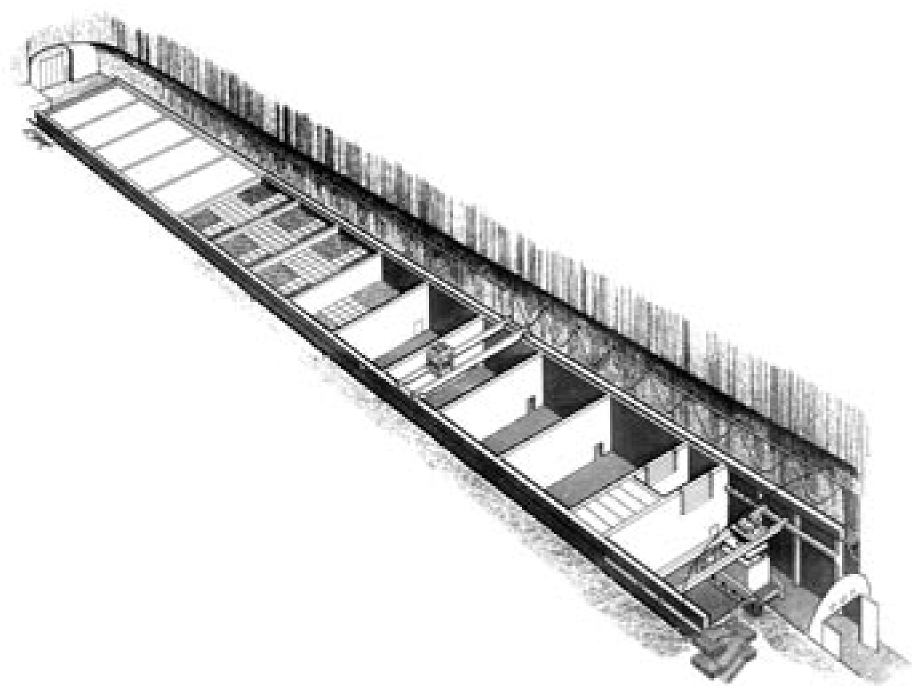

Figure 1-5. *The BMA.*

Figure 1-6. *View over BMA.*

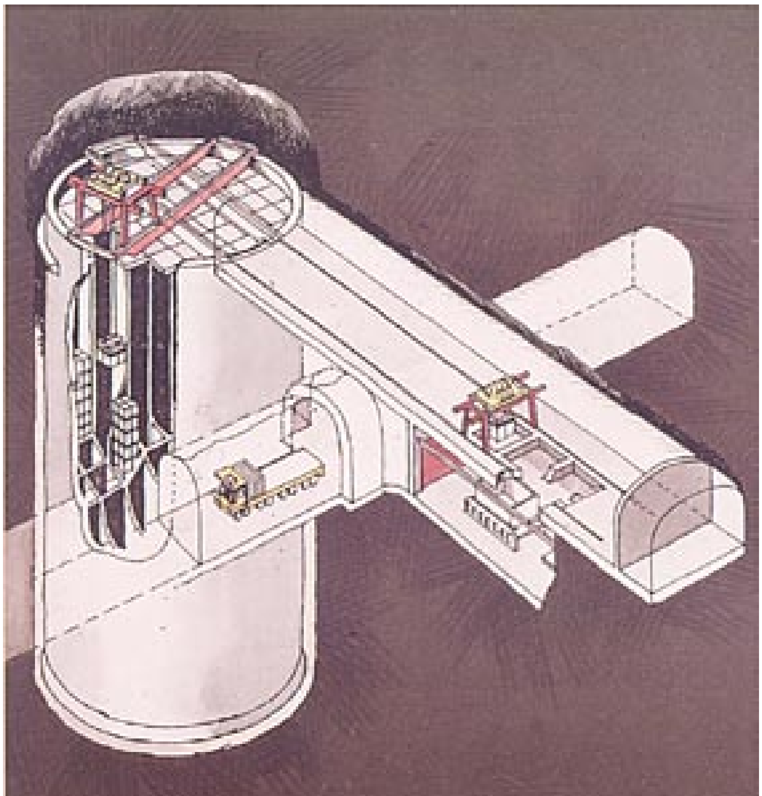

Figure 1-7. *The Silo including some adjacent tunnels.*

The Silo is divided into vertical shafts with intervening concrete walls. The waste packages are placed in the shafts, normally in levels with four moulds or 16 drums. The voids between the waste packages are gradually backfilled with porous concrete.

The walls of the Silo are made of reinforced concrete with a thickness of 0.8 m. In between the Silo walls and the surrounding rock there is a bentonite backfill, on average 1.2 m thick. The 1 m thick concrete floor in the bottom of the Silo is placed on a layer of 90/10 sand/bentonite mixture. According to present plans the top of the Silo will be a 1 m thick concrete lid. The top of the lid will be covered with a thin layer of sand and then 1.5 m of sand/bentonite mixture (90/10). The remaining void above the sand/bentonite in the top will be filled with sand or gravel or sand stabilised in cement.

1.3 Aim of the study and general assumptions

The aim of this study is to present a waste inventory to be used as input to radionuclide release calculations and other kinds of calculation.

Three different inventories are presented. The inventory is based on the assumption that Swedish reactors will operate for 50 or 60 years, except Barsebäck 1 and 2 which closed in 1999 and 2005 respectively. The third inventory presented is one of a full repository (one part where it is assumed to be full with reference to radioactivity and one to volume). It is also assumed that future production of waste is produced with the methods foreseen today. No consideration has been taken to the future power upgrades.

Waste in the form of large components or waste having some other feature differentiating it from ‘normal’ waste is expected in SFR 1. In this study this waste is predicted to be deposited in BLA, 1BTF, 2BTF and BMA.

Furthermore, it is assumed that the majority of the Swedish power plants continue to use shallow land burial for their very low-level waste. Today Forsmark NPP, Oskarshamn NPP, Ringhals NPP and the Studsvik research site uses this method.

1.4 Structure of the report

This report is divided in one main part and six appendices. The main part includes background, some common features for the waste and then a summary of all the waste, present and future, that is or will be deposited in the SFR 1.

The appendices are:

- Appendix A: A description of how the inventory regarding number of waste packages was calculated.
- Appendix B: A description of how the inventory regarding radionuclides was calculated.
- Appendix C: A thorough description of all waste types in the BLA rock cavern.
- Appendix D: A thorough description of all waste types in the BTF rock caverns.
- Appendix E: A thorough description of all waste types in the BMA rock cavern.
- Appendix F: A thorough description of all waste types in the Silo.

Appendix C–F includes geometry, materials, and radionuclide content for each waste type. Also, for each waste type a reference waste type is defined which is used in all kinds of calculations.

2 Waste packages in SFR 1 – general

2.1 The code system of waste in SFR – abbreviations

The first Swedish commercial nuclear power plant (Oskarshamn 1) has been in operation since the early 70's and consequently has produced waste since then. From the period before that, radioactive waste was produced during research and from the first test reactors. Most of that waste was dumped in the sea. The Swedish programme for final storage of the waste was not formed from the beginning, consequently the different power plants used different forms of treatment and also different forms of geometries, although the waste have some common features. In order to bring some systematic classification into the waste treatment, different waste types have been defined and a code system for these types has been developed. The system is very useful for data transfer between the power plants and the SFR facility.

The code system consists of one letter which denominates the producing plant, and two numbers giving information of what kind of raw waste, treatment method, geometry and in which part of the SFR the waste should be deposited. A complementary number (given after a ':') could also be used to give information about some feature that differentiates this waste from others of the same kind.

For example the code R.01:9 means ion-exchange resins from the Ringhals NPP solidified in cement in a concrete package (mould) with the side 1.2×1.2×1.2 m. The waste is meant for disposal in the rock cavern for intermediate level waste (BMA). The ':9' means that it is of an older type. In Table 2-1 and in Table 2-2 there are explanations of the different abbreviations.

Regarding the complimentary number the meaning is defined for each type, e.g. for B.05:2 the ':2' means that it is drums in bad condition placed in a steel box. The only complementary number generally defined is ':9' which means that it is of an older kind.

For each waste type there is a 'waste type description'. This document includes descriptions of origin of the raw waste, the treatment process, interim storage at the site, transportation, handling in SFR and disposal in SFR. The document also includes the demands defined for each waste package regarding general features like ID-number and chemical, physical, radiological and mechanistic features and how to achieve them. This document is written by the waste producers and then approved first by SKB and then by the Swedish authorities. Before the document is approved, no disposal in SFR is allowed.

Table 2-1. Abbreviations in the code system for the nuclear facilities in Sweden.

Aberration	Nuclear power plant
B	Barsebäck NPP
C	Clab (central interim fuel storage)
F	Forsmark NPP
O	Oskarshamn NPP
R	Ringhals NPP
S	Studsvik Research Site

Table 2-2. Abbreviations in the code system for treatment etc.

Aberration	Disposal in	Raw waste	Geometry	Treatment
01	BMA	Ion-exchange resin	Concrete mould	Cement solidification
02	Silo	Ion-exchange resin	Concrete mould	Cement solidification
03	–			
04	Silo	Ion-exchange resin	Steel drum	Cement solidification
05	BMA	Ion-exchange resin	Steel drum	Bitumen stabilisation
06	Silo	Ion-exchange resin	Steel drum	Bitumen stabilisation
07	BTF	Ion-exchange resin	Concrete tank	De-watering
08	–	–	–	–
09	BMA	Sludge	Steel drum	Cement solidification
10	BMA	Sludge	Concrete mould	Cement solidification
11	Silo	Sludge	Concrete mould	Cement solidification
12	BLA	Scrap metal and refuse	ISO-container	–
13	BTF	Ashes	Steel drum	Cement stabilisation
14	BLA	Scrap metal and refuse	ISO-container	Cement stabilisation
15	BMA	Ion-exchange resin	Steel mould	Cement solidification
16	Silo	Ion-exchange resin	Steel mould	Cement solidification
17	BMA	Ion-exchange resin	Steel mould	Bitumen stabilisation
18	Silo	Ion-exchange resin	Steel mould	Bitumen stabilisation
19	BTF	Graphite	Steel box	–
20	BLA	Ion-exchange resins	ISO-container	Bitumen stabilised drums
21	BMA	Scrap metal and refuse	Steel drum	Cement stabilisation
22	Silo	Scrap metal and refuse	Steel drum	Cement stabilisation
23	BMA	Scrap metal and refuse	Mould	Cement stabilisation
24	Silo	Scrap metal and refuse	Concrete mould	Cement stabilisation
29	BMA	Evaporator concentrate	Concrete mould	Cement solidification
99	All caverns	Odd waste	Differs	Differs

2.2 Waste containers and matrices

As seen in the previous part, the waste packages in SFR are of many different kinds, but the containers and the waste matrices, i.e. treatment methods, are quite similar.

The containers are basically of six different kinds:

- Steel drums. Standard 200-litre drums. The measures differ a bit but the drums are approximately 90 cm high and have a diameter of 60 cm. In the BMA and the silo the drums are handled four by four put on a steel plate or in a steel box. Both types are custom made for the system. In the BTF the drums are handled one by one.
- Concrete moulds. A concrete cube with the side 1.2 m. The walls usually have a thickness of 10 cm, but 25 cm and 35 cm can occur. The moulds are deposited in the BMA and in the Silo. Moulds with low dose rate are used to build stabilisation walls in 1BTF.
- Steel moulds. Steel cubes with the same outer dimensions as the concrete mould but with just 5 or 6 mm thick walls. The steel moulds have room for more than 70% more waste than the concrete ones but with considerably less radiation shielding properties. The steel moulds are used in BMA and the Silo.
- Concrete tanks. These tanks have the length of 3.3 m, width of 1.3 m and height of 2.3 m. The walls are 15 cm thick. The container has a de-watering system in the bottom of the tank. The tank is used in 1BTF and 2BTF.

- ISO-containers. Standard containers, usually with the dimensions 6.4×2.4×1.3 m but other dimensions can also be used. The containers can hold drums, boxes or bales. There can also be no inner package, just piled scrap metal. The containers are used in the BLA.
- Steel boxes. These boxes are primarily used inside the ISO-containers.
- 'Odd' Waste. Large components as heat exchangers etc could be of interest to dispose of in SFR 1. In this report the packages are assumed to be made of steel and concrete where the geometry is like an ISO-container, a concrete tank or a mould depending on which rock vault they are supposed to be placed.

The different treatment methods are:

- Cement solidification. Ion-exchange resins or sludge are mixed with concrete in drums or moulds.
- Cement stabilisation. Scrap metal and refuse are placed in moulds and cement is poured over it. Types 13 and 14 are a bit different. The waste is placed in 100-litre drums, which are then put inside standard 200-litre drums. Concrete is then poured in-between the drums.
- Bitumen stabilisation. Ion-exchange resins are dried and mixed with bitumen and then poured into moulds or drums.
- De-watering. Wet ion-exchange resin is pumped into a concrete tank and water is removed by suction.

3 Description of waste in BLA

This chapter is a short summary of Appendix C.

3.1 Waste packages in BLA

The BLA is designed mainly to contain scrap metals and refuse. All waste in BLA is handled in ISO-containers. As stated in the previous chapter the containers can hold different smaller packages like drums, boxes and bales. One special type is type 20 (B.20 and F.20). These containers hold drums with bituminised ion-exchange resins.

3.2 Amount of waste in BLA

The number of the different waste types used in the inventory calculations is presented in Table 3-1. How the numbers has been estimated is presented in Appendix A.

From the number of waste packages and the reference waste type one can derive the summarised amounts of different materials in the BLA, see Table 3-2a–c.

Table 3-1. Number of different waste packages in BLA at the time of closure for the four scenarios.

Waste type	Number of packages Year 2040	Number of packages Year 2050	Number of packages Full Repository*
B.12	238	238	238
B.20	12	12	12
F.12	39	44	42
F.20	15	15	15
O.12	34	44	39
O.99	5	5	5
R.12	137	167	152
S.12	45	45	45
S.14	75	75	75

*) With respect to volume.

Table 3-2a. Summarised amounts of different materials in BLA at the time of closure, year 2040.

Material	Weight (kg)	Area (m^2)*
Iron/Steel	3.78E+06	2.20E+05
Aluminium	5.58E+04	8.36E+03
Cellulose (wood, paper, cloth)	4.33E+05	
Other organic (plastics, rubber, cable)	1.48E+06	
Other inorganic (insulation etc)	1.05E+05	
Ion-exchange Resin	9.77E+04	
Bitumen	1.17E+05	

* 'Area' means the area exposed to ground water after closure of SFR 1.

Table 3-2b. Summarised amounts of different materials in BLA at the time of closure, year 2050.

Material	Weight (kg)	Area (m^2)*
Iron/Steel	4.07E+06	2.36E+05
Aluminium	6.73E+04	9.95E+03
Cellulose (wood, paper, cloth)	4.68E+05	
Other organic (plastics, rubber, cable)	1.62E+06	
Other inorganic (insulation etc)	1.05E+05	
Ion-exchange Resin	9.77E+04	
Bitumen	1.17E+05	

* 'Area' means the area exposed to ground water after closure of SFR 1.

Table 3-2c. Summarised amounts of different materials in BLA at the time of closure, full inventory.**

Material	Weight (kg)	Area (m^2)*
Iron/Steel	3.92E+06	2.28E+05
Aluminium	6.50E+04	9.60E+03
Cellulose (wood, paper, cloth)	4.50E+05	
Other organic (plastics, rubber, cable)	1.55E+06	
Other inorganic (insulation etc)	1.05E+05	
Ion-exchange Resin	9.77E+04	
Bitumen	1.17E+05	

* 'Area' means the area exposed to ground water after closure of SFR 1.

** Regarding volume.

3.3 Radionuclide inventory in BLA

The inventory of radionuclides in BLA is presented in Table 3-3. How the amounts of different radionuclides have been calculated is presented in Appendix B.

Table 3-3. Summarised amounts of different radionuclides in BLA at the time of closure for the three scenarios.

Nuclide	Activity (Bq) Year 2040	Activity (Bq) Year 2050	Activity (Bq) Full inventory*
H-3	2.2E+07	1.9E+07	1.5E+08
Be-10	8.1E+02	9.2E+02	5.6E+03
C-14 org	1.2E+09	1.4E+09	8.0E+09
C-14 inorg	2.7E+09	3.3E+09	1.9E+10
C-14 tot	3.9E+09	4.7E+09	2.7E+10
Cl-36	1.0E+06	1.2E+06	7.2E+06
Fe-55	4.2E+09	4.0E+09	2.9E+10
Ni-59	5.3E+09	6.1E+09	3.7E+10
Co-60	3.6E+10	3.1E+10	2.5E+11
Ni-63	6.6E+11	7.7E+11	4.6E+12
Se-79	4.9E+05	5.1E+05	3.4E+06
Sr-90	4.7E+09	3.9E+09	3.3E+10
Mo-93	2.0E+07	2.7E+07	1.4E+08
Nb-93m	3.2E+08	3.0E+08	2.2E+09
Zr-93	4.3E+06	5.1E+06	3.0E+07
Nb-94	1.3E+07	1.5E+07	9.2E+07
Tc-99	5.0E+08	6.4E+08	3.5E+09
Ru-106	3.1E+03	3.1E+03	2.2E+04
Pd-107	1.2E+05	1.3E+05	8.6E+05
Ag-108m	4.2E+08	4.2E+08	2.9E+09
Cd-113m	1.1E+07	7.3E+06	7.4E+07
Sb-125	3.1E+08	2.9E+08	2.1E+09
Sn-126	6.2E+04	6.3E+04	4.3E+05
I-129	3.1E+05	3.3E+05	2.2E+06
Ba-133	1.7E+06	1.5E+06	1.2E+07
Cs-134	2.5E+07	1.8E+07	1.7E+08
Cs-135	2.0E+06	2.1E+06	1.4E+07
Cs-137	4.8E+10	4.1E+10	3.4E+11
Pm-147	1.0E+08	7.4E+07	7.0E+08
Sm-151	2.7E+08	2.6E+08	1.9E+09
Eu-152	1.1E+09	6.5E+08	7.5E+09
Eu-154	7.5E+08	4.4E+08	5.2E+09
Eu-155	8.1E+07	4.6E+07	5.6E+08
Ho-166m	1.7E+07	2.0E+07	1.2E+08
U-232	6.0E+02	5.7E+02	4.1E+03
U-234	3.0E+04	3.1E+04	2.1E+05
U-235	4.3E+08	4.3E+08	3.0E+09
U-236	9.0E+03	9.4E+03	6.2E+04
Np-237	2.7E+04	2.8E+04	1.9E+05
Pu-238	4.7E+07	4.6E+07	3.3E+08
U-238	1.4E+09	1.4E+09	9.9E+09

Nuclide	Activity (Bq) Year 2040	Activity (Bq) Year 2050	Activity (Bq) Full inventory*
Pu-239	1.3E+07	1.3E+07	9.1E+07
Pu-240	2.6E+07	2.7E+07	1.8E+08
Pu-241	4.8E+08	3.6E+08	3.4E+09
Am-241	5.0E+07	5.3E+07	3.5E+08
Am-242m	2.4E+05	2.4E+05	1.7E+06
Pu-242	9.0E+04	9.4E+04	6.2E+05
Am-243	9.0E+05	9.3E+05	6.2E+06
Cm-243	2.3E+05	2.0E+05	1.6E+06
Cm-244	2.2E+07	1.9E+07	1.5E+08
Cm-245	9.0E+03	9.3E+03	6.2E+04
Cm-246	2.4E+03	2.5E+03	1.7E+04
Tot	7.7E+11	8.7E+11	5.3E+12

* Regarding activity.

4 Description of waste in BTF

This chapter is a short summary of Appendix D.

4.1 Waste packages in BTF

The BTF is designed mainly to contain de-watered ion-exchange resins, but solidified resins and ashes are present too. All waste in BTF is handled in concrete tanks, moulds or drums.

When the drums containing ashes is placed in the 1BTF-cavern some sort of stabilising walls are necessary. Concrete tanks are placed alongside the rock walls and drums are then piled lying down between them. When six rows of drums have been piled, moulds are placed across the rock vault, see Figure 4-1 and Figure 4-2. Concrete is then poured over the drums in order to stabilise them.

4.2 Amount of waste in BTF

The number of the different waste types used in the inventory calculations is presented in Table 4-1a and b. How the numbers have been estimated is presented in Appendix A.

There are two rock caverns in SFR named BTF, i.e. 1BTF and 2BTF. There is a limited freedom to choose how to place the waste in the two caverns.

Figure 4-1. *Inplacement of drums in 1BTF.*

Figure 4-2. *Wall of moulds across 1BTF.*

Table 4-1a. Number of different waste packages in BTF 1 at the time of closure for the three scenarios.

Waste type	Number of packages Year 2040	Number of packages Year 2050	Number of packages Full Repository*
B.07	29	29	29
O.01	28	28	28
O.07	119	159	459
O.99	20	20	20
R.01	91	91	91
R.10	4	4	4
R.23	21	21	21
R.99	1	1	1
S.13	5,055	5,130	5,700

* With respect to volume.

Table 4-2b. Number of different waste packages in BTF 2 at the time of closure for the three scenarios.

Waste type	Number of packages Year 2040	Number of packages Year 2050	Number of packages Full Repository*
B.07	194	194	194
F.99	18	18	18
O.07	537	577	532
O.99	20	20	20
S.13	399	474	390

* With respect to volume.

In order to perform detailed calculations an estimated distribution of the waste packages is made. The main principles are:

- Waste already deposited has been distributed as they are placed with a small exception of some odd waste.
- 2BTF contains concrete tanks with de-watered ion-exchange resins and old steam separators from Forsmark.
- 1BTF contains ashes, moulds with solidified waste and concrete tanks.
- As a crude simplification it is assumed that all future waste is divided evenly between the two caverns. This renders an uneven distribution between the two caverns since they do not contain the same amount of waste today, hence there is a somewhat askew distribution of both volume and radionuclide contents in these caluculations.

From the number of waste packages and the reference waste type one can derive the summarised amounts of different materials in the BTF, see Table 4-3a–c and Table 4-4a–c.

Table 4-3a. Summarised amounts of different materials in 1BTF at the time of closure, year 2040.

Material	Weight (kg)	Area (m^2)*
Iron/Steel	4.20E+05	4.54E+04
Aluminium	3.00E+04	1.19E+04
Ashes	1.48E+05	
Ion-exchange resin	1.77E+05	
Other organic	1.31E+04	
Other inorganic	4.65E+03	
Sludge	9.31E+03	
Cement and concrete	1.98E+06	
Cellulose	1.68E+03	

* 'Area' means the area exposed to ground water after closure of SFR 1.

Table 4-3b. Summarised amounts of different materials in 1BTF at the time of closure, year 2050.

Material	Weight (kg)	Area (m^2)*
Iron/Steel	4.62E+05	4.78E+04
Aluminium	3.05E+04	1.21E+04
Ashes	1.53E+05	
Ion-exchange resin	2.17E+05	
Other organic	1.57E+04	
Other inorganic	6.15E+03	
Sludge	1.17E+04	
Cement and concrete	2.41E+06	
Cellulose	1.68E+03	

* 'Area' means the area exposed to ground water after closure of SFR 1.

Table 4-3c. Summarised amounts of different materials in 1BTF at the time of closure, full inventory.**

Material	Weight (kg)	Area (m^2)*
Iron/Steel	7.79E+05	6.67E+04
Aluminium	3.42E+04	1.42E+04
Ashes	1.89E+05	
Ion-exchange resin	5.21E+05	
Other organic	3.58E+04	
Sludge	3.00E+04	
Other inorganic	1.76E+04	
Cement and concrete	5.68E+06	
Cellulose	1.68E+03	

* 'Area' means the area exposed to ground water after closure of SFR 1.

** regarding volume.

Table 4-4a. Summarised amounts of different materials in 2BTF at the time of closure, year 2040.

Material	Weight (kg)	Area (m^2)*
Iron/Steel	6.39E+05	3.72E+04
Aluminium	2.59E+03	1.44E+03
Ashes	2.53E+04	
Ion-exchange resin	8.11E+05	
Other organic	6.03E+04	
Other inorganic	4.65E+03	
Sludge	4.38E+04	
Cement and concrete	3.75E+06	
Cellulose	1.66E+02	

* 'Area' means the area exposed to ground water after closure of SFR 1.

Table 4-4b. Summarised amounts of different materials in 2BTF at the time of closure, year 2050.

Material	Weight (kg)	Area (m^2)*
Iron/Steel	6.81E+05	3.96E+04
Aluminium	3.08E+03	1.71E+03
Ashes	3.01E+04	
Ion-exchange resin	8.51E+05	
Other organic	6.29E+04	
Other inorganic	6.15E+03	
Sludge	4.62E+04	
Cement and concrete	4.18E+06	
Cellulose	1.66E+02	

* 'Area' means the area exposed to ground water after closure of SFR 1.

Table 4-4c. Summarised amounts of different materials in 2BTF at the time of closure, full inventory.**

Material	Weight (kg)	Area (m^2)*
Iron/Steel	6.34E+05	3.69E+04
Aluminium	2.54E+03	1.40E+03
Ashes	2.48E+04	
Ion-exchange resin	8.06E+05	
Other organic	6.00E+04	
Sludge	4.35E+04	
Other inorganic	4.47E+03	
Cement and concrete	3.70E+06	
Cellulose	1.66E+02	

* 'Area' means the area exposed to ground water after closure of SFR 1.

** Regarding volume.

4.3 Radionuclide inventory in BTF

The inventories of radionuclides in BTF 1 and BTF 2 are presented in Table 4-5 and Table 4-6. Table 4-7 shows a summarised amount of different radionuclides in 1BTF and 2BTF as a sum. How the amounts of different radionuclides have been calculated is presented in Appendix B.

Table 4-5. Summarised amounts of different radionuclides in 1BTF at the time of closure for the three different scenarios.

Nuclide	Activity (Bq) Year 2040	Activity (Bq) Year 2050	Activity (Bq) Full inventory*
H-3	1.6E+08	1.7E+08	1.1E+09
Be-10	5.1E+03	6.4E+03	3.5E+04
C-14 org	7.4E+09	9.5E+09	5.1E+10
C-14 inorg	1.7E+10	2.2E+10	1.2E+11
C-14 tot	2.5E+10	3.2E+10	1.7E+11
Cl-36	1.1E+07	1.2E+07	7.3E+07
Fe-55	2.9E+10	2.9E+10	2.0E+11
Ni-59	2.1E+10	2.3E+10	1.4E+11
Co-60	2.9E+11	2.9E+11	2.0E+12
Ni-63	2.3E+12	2.4E+12	1.6E+13
Se-79	5.9E+06	6.7E+06	4.1E+07
Sr-90	6.4E+10	6.4E+10	4.5E+11
Mo-93	4.4E+07	5.2E+07	3.1E+08
Nb-93m	2.3E+09	2.5E+09	1.6E+10
Zr-93	1.5E+07	1.8E+07	1.0E+08
Nb-94	8.5E+07	1.1E+08	5.9E+08
Tc-99	7.0E+09	7.4E+09	4.8E+10
Ru-106	2.6E+06	2.6E+06	1.8E+07
Pd-107	1.5E+06	1.7E+06	1.0E+07
Ag-108m	4.8E+08	6.0E+08	3.4E+09
Cd-113m	1.8E+08	1.6E+08	1.2E+09
Sb-125	6.1E+09	6.1E+09	4.2E+10
Sn-126	7.4E+05	8.4E+05	5.2E+06
I-129	3.1E+06	3.6E+06	2.1E+07

Nuclide	Activity (Bq) Year 2040	Activity (Bq) Year 2050	Activity (Bq) Full inventory*
Ba-133	1.3E+07	1.3E+07	8.8E+07
Cs-134	6.1E+08	6.1E+08	4.3E+09
Cs-135	1.6E+07	1.8E+07	1.1E+08
Cs-137	6.7E+11	6.6E+11	4.6E+12
Pm-147	2.1E+09	2.1E+09	1.5E+10
Sm-151	3.4E+09	3.7E+09	2.3E+10
Eu-152	1.5E+08	1.3E+08	1.1E+09
Eu-154	1.2E+10	1.0E+10	8.3E+10
Eu-155	1.7E+09	1.4E+09	1.2E+10
Ho-166m	5.8E+07	7.1E+07	4.1E+08
U-232	9.6E+03	9.6E+03	6.7E+04
U-234	5.0E+05	5.3E+05	3.5E+06
U-235	1.9E+07	1.9E+07	1.3E+08
U-236	1.9E+05	2.1E+05	1.3E+06
Np-237	4.1E+05	4.7E+05	2.8E+06
Pu-238	3.5E+08	3.7E+08	2.4E+09
U-238	3.3E+05	3.9E+05	2.3E+06
Pu-239	1.5E+08	1.6E+08	1.0E+09
Pu-240	3.0E+08	3.1E+08	2.1E+09
Pu-241	7.5E+09	6.4E+09	5.2E+10
Am-241	8.4E+08	8.6E+08	5.8E+09
Am-242m	4.0E+06	4.1E+06	2.8E+07
Pu-242	1.5E+06	1.6E+06	1.0E+07
Am-243	1.5E+07	1.6E+07	1.0E+08
Cm-243	3.0E+06	2.5E+06	2.1E+07
Cm-244	1.0E+08	8.3E+07	7.1E+08
Cm-245	1.5E+05	1.6E+05	1.0E+06
Cm-246	4.0E+04	4.2E+04	2.7E+05
Tot	3.5E+12	3.6E+12	2.4E+13

* Regarding activity.

Table 4-6. Summarised amounts of different radionuclides in 2BTF at the time of closure for the three different scenarios.

Nuclide	Activity (Bq) Year 2040	Activity (Bq) Year 2050	Activity (Bq) Full inventory*
H-3	4.0E+08	3.0E+08	2.8E+09
Be-10	2.5E+04	2.6E+04	1.7E+05
C-14 org	5.5E+10	5.8E+10	3.8E+11
C-14 inorg	1.3E+11	1.3E+11	9.0E+11
C-14 tot	1.8E+11	1.9E+11	1.3E+12
Cl-36	3.1E+07	3.2E+07	2.1E+08
Fe-55	2.9E+10	2.9E+10	2.0E+11
Ni-59	4.2E+10	4.4E+10	2.9E+11
Co-60	3.7E+11	3.1E+11	2.6E+12
Ni-63	2.8E+12	2.8E+12	1.9E+13
Se-79	2.4E+07	2.4E+07	1.6E+08
Sr-90	2.1E+11	1.8E+11	1.4E+12
Mo-93	8.9E+07	9.7E+07	6.2E+08
Nb-93m	6.6E+09	5.3E+09	4.6E+10

Nuclide	Activity (Bq) Year 2040	Activity (Bq) Year 2050	Activity (Bq) Full inventory*
Zr-93	4.8E+07	5.1E+07	3.3E+08
Nb-94	4.2E+08	4.4E+08	2.9E+09
Tc-99	7.6E+09	8.1E+09	5.3E+10
Ru-106	2.6E+06	2.6E+06	1.8E+07
Pd-107	5.9E+06	6.1E+06	4.1E+07
Ag-108m	2.3E+09	2.4E+09	1.6E+10
Cd-113m	3.4E+08	2.6E+08	2.3E+09
Sb-125	6.2E+09	6.1E+09	4.3E+10
Sn-126	2.9E+06	3.0E+06	2.0E+07
I-129	1.9E+07	1.9E+07	1.3E+08
Ba-133	2.8E+07	2.1E+07	1.9E+08
Cs-134	6.1E+08	6.1E+08	4.3E+09
Cs-135	1.1E+08	1.1E+08	7.6E+08
Cs-137	1.8E+12	1.6E+12	1.3E+13
Pm-147	2.1E+09	2.1E+09	1.5E+10
Sm-151	1.2E+10	1.1E+10	8.2E+10
Eu-152	1.1E+08	1.0E+08	7.4E+08
Eu-154	1.7E+10	1.2E+10	1.2E+11
Eu-155	1.8E+09	1.5E+09	1.2E+10
Ho-166m	1.9E+08	2.0E+08	1.3E+09
U-232	1.1E+04	1.0E+04	7.4E+04
U-234	5.4E+05	5.7E+05	3.7E+06
U-235	7.5E+05	8.5E+05	5.2E+06
U-236	3.7E+05	3.9E+05	2.6E+06
Np-237	1.9E+06	2.0E+06	1.3E+07
Pu-238	6.0E+08	6.0E+08	4.2E+09
U-238	8.8E+05	9.4E+05	6.1E+06
Pu-239	2.0E+08	2.1E+08	1.4E+09
Pu-240	4.0E+08	4.1E+08	2.8E+09
Pu-241	8.6E+09	7.0E+09	6.0E+10
Am-241	4.9E+08	5.1E+08	3.4E+09
Am-242m	4.4E+06	4.5E+06	3.0E+07
Pu-242	1.6E+06	1.7E+06	1.1E+07
Am-243	1.6E+07	1.7E+07	1.1E+08
Cm-243	2.0E+06	1.8E+06	1.4E+07
Cm-244	1.1E+08	9.0E+07	7.8E+08
Cm-245	1.6E+05	1.7E+05	1.1E+06
Cm-246	4.3E+04	4.5E+04	3.0E+05
Tot	5.5E+12	5.2E+12	3.8E+13

* Regarding activity.

Table 4-7. Summarised amounts of different radionuclides in BTF (sum of 1BTF and 2BTF) at the time of closure for the three different scenarios.

Nuclide	Activity (Bq) Year 2040	Activity (Bq) Year 2050	Activity (Bq) Full inventory*
H-3	5.6E+08	4.8E+08	3.9E+09
Be-10	3.0E+04	3.3E+04	2.1E+05
C-14 org	6.3E+10	6.7E+10	4.4E+11
C-14 inorg	1.5E+11	1.6E+11	1.0E+12

Nuclide	Activity (Bq) Year 2040	Activity (Bq) Year 2050	Activity (Bq) Full inventory*
C-14 tot	2.1E+11	2.2E+11	1.5E+12
Cl-36	4.1E+07	4.4E+07	2.9E+08
Fe-55	5.8E+10	5.7E+10	4.0E+11
Ni-59	6.2E+10	6.7E+10	4.3E+11
Co-60	6.6E+11	5.9E+11	4.6E+12
Ni-63	5.1E+12	5.2E+12	3.6E+13
Se-79	2.9E+07	3.1E+07	2.0E+08
Sr-90	2.7E+11	2.4E+11	1.9E+12
Mo-93	1.3E+08	1.5E+08	9.2E+08
Nb-93m	8.9E+09	7.8E+09	6.2E+10
Zr-93	6.3E+07	7.0E+07	4.4E+08
Nb-94	5.0E+08	5.4E+08	3.5E+09
Tc-99	1.5E+10	1.5E+10	1.0E+11
Ru-106	5.2E+06	5.2E+06	3.6E+07
Pd-107	7.4E+06	7.8E+06	5.1E+07
Ag-108m	2.8E+09	3.0E+09	1.9E+10
Cd-113m	5.1E+08	4.1E+08	3.6E+09
Sb-125	1.2E+10	1.2E+10	8.5E+10
Sn-126	3.7E+06	3.9E+06	2.6E+07
I-129	2.2E+07	2.3E+07	1.5E+08
Ba-133	4.0E+07	3.4E+07	2.8E+08
Cs-134	1.2E+09	1.2E+09	8.5E+09
Cs-135	1.3E+08	1.3E+08	8.7E+08
Cs-137	2.5E+12	2.2E+12	1.7E+13
Pm-147	4.3E+09	4.1E+09	3.0E+10
Sm-151	1.5E+10	1.5E+10	1.1E+11
Eu-152	2.6E+08	2.3E+08	1.8E+09
Eu-154	2.9E+10	2.2E+10	2.0E+11
Eu-155	3.5E+09	2.9E+09	2.4E+10
Ho-166m	2.5E+08	2.7E+08	1.7E+09
U-232	2.0E+04	2.0E+04	1.4E+05
U-234	1.0E+06	1.1E+06	7.2E+06
U-235	2.0E+07	2.0E+07	1.4E+08
U-236	5.6E+05	6.0E+05	3.9E+06
Np-237	2.3E+06	2.5E+06	1.6E+07
Pu-238	9.6E+08	9.7E+08	6.6E+09
U-238	1.2E+06	1.3E+06	8.4E+06
Pu-239	3.5E+08	3.6E+08	2.4E+09
Pu-240	7.0E+08	7.3E+08	4.8E+09
Pu-241	1.6E+10	1.3E+10	1.1E+11
Am-241	1.3E+09	1.4E+09	9.2E+09
Am-242m	8.4E+06	8.6E+06	5.8E+07
Pu-242	3.1E+06	3.3E+06	2.2E+07
Am-243	3.1E+07	3.3E+07	2.1E+08
Cm-243	5.1E+06	4.3E+06	3.5E+07
Cm-244	2.1E+08	1.7E+08	1.5E+09
Cm-245	3.1E+05	3.3E+05	2.1E+06
Cm-246	8.2E+04	8.7E+04	5.7E+05
Tot	9.0E+12	8.8E+12	6.2E+13

* Regarding activity.

5 Description of waste in BMA

This chapter is a short summary of Appendix E.

5.1 Waste packages in BMA

The BMA is designed to contain intermediate level waste which have a lower dose rate or waste that of some kind of reason is not suitable to deposit in the Silo. The waste contains solidified (bitumen or cement) ion-exchange resins and stabilised scrap metal and refuses. All waste in BMA is handled in moulds or drums.

5.2 Amount of waste in BMA

The number of the different waste types is presented in Table 5-1. How the numbers has been estimated is presented in Appendix A.

From the number of waste packages and the reference waste type one can derive the summarised amounts of different materials in the BMA, see Table 5-2a–c.

Table 5-1. Number of different waste types in BMA at the time of closure for the three scenarios.

Waste type	Number of packages Year 2040	Number of packages Year 2050	Number of packages Full repository
B.05	4,188	4,188	4,188
B.23	30	30	30
C.01	68	68	68
C.23	162	202	167
F.05	1,712	1,712	1,712
F.15	11	11	11
F.17	1,305	1,605	1,343
F.23	389	469	399
F.99	2	2	2
O.01	675	675	675
O.23	570	640	579
R.01	1,686	1,686	1,686
R.10	126	146	129
R.15	153	163	154
R.23	558	608	564
R.29	660	960	698
S.09	880	1,080	905
S.23	340	440	353

Table 5-2a. Summarised amounts of different materials in BMA at the time of closure, year 2040.

Material	Weight (kg)	Area (m^2)*
Iron/Steel	2.68E+06	1.24E+05
Aluminium	7.09E+03	1.02E+03
Cellulose (wood, paper, cloth)	1.44E+05	
Other organic (plastic, rubber, cable)	2.85E+05	
Other inorganic	3.43E+04	
Ion-exchange resin	1.63E+06	
Sludge	5.86E+04	
Evaporator bottoms	6.20E+05	
Bitumen	1.76E+06	
Cement and concrete	1.25E+07	

* 'Area' means the area exposed to ground water after closure of SFR 1.

Table 5-2b. Summarised amounts of different materials in BMA at the time of closure, year 2050.

Material	Weight (kg)	Area (m^2)*
Iron/Steel	3.03E+06	1.38E+05
Aluminium	8.43E+03	1.22E+03
Cellulose (wood, paper, cloth)	1.65E+05	
Other organic (plastic, rubber, cable)	3.33E+05	
Other inorganic	4.32E+04	
Ion-exchange resin	1.83E+06	
Sludge	6.95E+04	
Evaporator bottoms	8.66E+05	
Bitumen	2.01E+06	
Cement and concrete	1.41E+07	

* 'Area' means the area exposed to ground water after closure of SFR 1.

Table 5-2c. Summarised amounts of different materials in BMA at the time of closure, full inventory.**

Material	Weight (kg)	Area (m^2)*
Iron/Steel	2.76E+06	1.26E+05
Aluminium	7.26E+03	1.05E+03
Cellulose (wood, paper, cloth)	1.47E+05	
Other organic (plastic, rubber, cable)	2.91E+05	
Other inorganic	3.54E+04	
Ion-exchange resin	1.65E+06	
Sludge	6.00E+04	
Evaporator bottoms	6.51E+05	
Bitumen	1.79E+06	
Cement and concrete	1.27E+07	

* 'Area' means the area exposed to ground water after closure of SFR 1.

** With regard to volume.

5.3 Radionuclide inventory in BMA

The inventory of radionuclides is presented in Table 5-3. How the amounts of different radionuclides have been calculated is presented in Appendix B.

Table 5-3. Summarised amounts of different radionuclides in BMA at the time of closure for the three different scenarios.

Nuclide	Activity (Bq) Year 2040	Activity (Bq) Year 2050	Activity (Bq) Full inventory*
H-3	3.8E+09	3.4E+09	2.6E+10
Be-10	2.2E+05	2.3E+05	1.5E+06
C-14 org	3.2E+11	3.4E+11	2.2E+12
C-14 inorg	7.4E+11	8.0E+11	5.2E+12
C-14 tot	1.1E+12	1.1E+12	7.4E+12
Cl-36	2.3E+08	2.5E+08	1.6E+09
Fe-55	1.5E+12	1.5E+12	1.0E+13
Ni-59	2.1E+12	2.1E+12	1.4E+13
Co-60	7.0E+12	6.8E+12	4.9E+13
Ni-63	2.6E+14	2.6E+14	1.8E+15
Se-79	2.1E+08	2.3E+08	1.5E+09
Sr-90	1.7E+12	1.5E+12	1.2E+13
Mo-93	6.8E+08	7.5E+08	4.7E+09
Nb-93m	5.9E+10	5.3E+10	4.1E+11
Zr-93	4.2E+08	4.5E+08	2.9E+09
Nb-94	3.6E+09	3.9E+09	2.5E+10
Tc-99	3.7E+10	4.4E+10	2.6E+11
Ru-106	1.2E+08	1.2E+08	8.1E+08
Pd-107	5.4E+07	5.6E+07	3.7E+08
Ag-108m	2.0E+10	2.1E+10	1.4E+11
Cd-113m	3.7E+09	3.2E+09	2.5E+10
Sb-125	1.5E+11	1.5E+11	1.1E+12
Sn-126	2.7E+07	2.8E+07	1.9E+08
I-129	1.8E+08	1.9E+08	1.3E+09
Ba-133	2.8E+08	2.6E+08	2.0E+09
Cs-134	7.0E+10	7.0E+10	4.8E+11
Cs-135	1.0E+09	1.1E+09	7.2E+09
Cs-137	2.0E+13	1.9E+13	1.4E+14
Pm-147	1.6E+11	1.6E+11	1.1E+12
Sm-151	1.1E+11	1.1E+11	7.6E+11
Eu-152	4.0E+08	3.4E+08	2.8E+09
Eu-154	2.3E+11	2.1E+11	1.6E+12
Eu-155	4.7E+10	4.6E+10	3.3E+11
Ho-166m	1.6E+09	1.8E+09	1.1E+10
U-232	1.5E+05	1.5E+05	1.0E+06
U-234	7.7E+06	8.3E+06	5.3E+07
U-235	2.8E+06	3.2E+06	2.0E+07
U-236	3.0E+06	3.2E+06	2.1E+07
Np-237	2.5E+07	2.9E+07	1.7E+08
Pu-238	9.6E+09	9.5E+09	6.7E+10
U-238	6.2E+06	6.8E+06	4.3E+07

Nuclide	Activity (Bq) Year 2040	Activity (Bq) Year 2050	Activity (Bq) Full inventory*
Pu-239	2.0E+09	2.0E+09	1.4E+10
Pu-240	4.0E+09	4.1E+09	2.7E+10
Pu-241	1.4E+11	1.3E+11	9.7E+11
Am-241	6.9E+09	7.7E+09	4.8E+10
Am-242m	6.2E+07	6.5E+07	4.3E+08
Pu-242	2.3E+07	2.5E+07	1.6E+08
Am-243	2.3E+08	2.5E+08	1.6E+09
Cm-243	5.5E+07	5.3E+07	3.8E+08
Cm-244	1.4E+09	1.2E+09	9.4E+09
Cm-245	2.3E+06	2.5E+06	1.6E+07
Cm-246	6.1E+05	6.6E+05	4.2E+06
Tot	2.9E+14	3.0E+14	2.0E+15

* Regarding activity.

5.4 Inplacement of waste in BMA

The BMA cavern is the most complex regarding different waste, materials, placement of packages and so on. The cavern also contains a fair part (approximately 6%) of the total inventory of radionuclides.

The distribution as of December 2006 is shown as it is. Future deposition will be made to get an even distribution of radioactivity throughout the entire cavern.

Table 5-5 shows the distribution of the different waste types in the different compartments in the BMA until 31 December 2006.

Table 5-5. Distribution of waste in the different compartments in the BMA until 31 December 2006.

Waste type	Package	Matrix	Compartment															
			1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	Total
B.05	drum	bitumen			1,168		2,000	176										3,344
B.05:02	drum*	bitumen		382	270		96	96										844
B.23	mould	cement								30								30
C.01	mould	cement				20			10	1	21							52
C.23	mould	cement								8	16	1						25
F.05	drum	cement		1,712														1,712
F.15	mould	cement				8		3										11
F.17	mould	bitumen			144		28	259										431
F.23	mould	cement				64			12	92								168
F.99	mould	cement						2										2
O.01	mould	cement				56			254	29	184	163						686
O.23	mould	cement				36			36	134	131	28						365
R.01	mould	cement	576	148	145	143	144	144	144	146	88			8				1,686
R.10	mould	cement								16	48							64
R.15	mould	cement				124												124
R.23	mould	cement				124			120	90	68			4				406
R.29	mould	cement																0
S.09	drum	cement																0
S.23	drum	cement																0
Total			576	2,242	1,727	575	2,268	680	576	546	556	192	0	12	0	0	0	9,950

* Four drums are placed in one mould.

6 Description of waste in the Silo

This chapter is a short summary of Appendix F.

6.1 Waste packages in the Silo

The Silo is designed to contain intermediate level waste. The waste contains solidified (bitumen or cement) ion-exchange resins and some small amount of stabilised scrap metals and refuse. All waste in the Silo is handled in moulds or drums.

6.2 Amount of waste in Silo

The number of the different waste types is presented in Table 6-1. How the numbers has been estimated is presented in Appendix A.

One should note that when the prognosis of waste was made it was concluded that only approximately 63% of the silo were filled. Hence, the prognosticated number has been multiplied with a factor 100% / 63% = 1.59 to have a completely filled silo to make calculations on a fully used repository.

From the number of waste packages and the reference waste type one can derive the summarised amounts of different materials in the Silo, see Table 6-2a–c.

Table 6-1. Number of different waste types in the Silo at the time of closure for the three scenarios.

Waste type	Number of packages Year 2040	Number of packages Year 2050	Number of packages Full repository
B.04	1,000	1,000	1,000
B.06	1,840	1,840	1,840
C.02	1,210	1,460	1,636
C.24	154	194	222
F.18	792	992	1,132
O.02	1,942	2,342	2,623
R.02	371	371	371
R.16	2,737	3,437	3,928
S.11	60	60	60
S.24	104	134	155
S.04	181	221	249

Table 6-2a. Summarised amounts of different materials in Silo at the time of closure, year 2040.

Material	Weight (kg)	Area (m^2) *
Iron/Steel	2.79E+06	1.22E+05
Aluminium	4.02E+02	7.60E+01
Cellulose (wood, paper, cloth)	5.57E+03	
Other organics (plastic, rubber, cable)	3.22E+04	
Other inorganics	1.91E+04	
Ion-exchange resins	1.80E+06	
Sludge	2.00E+04	
Bitumen	1.04E+06	
Cement and concrete	1.56E+07	

* 'Area' means the area exposed to ground water after closure of SFR 1.

Table 6-2b. Summarised amounts of different materials in Silo at the time of closure, year 2050.

Material	Weight (kg)	Area (m^2) *
Iron/Steel	3.40E+06	1.48E+05
Aluminium	5.18E+02	9.80E+01
Cellulose (wood, paper, cloth)	6.97E+03	
Other organics (plastic, rubber, cable)	3.87E+04	
Other inorganics	2.42E+04	
Ion-exchange resins	2.18E+06	
Sludge	2.00E+04	
Bitumen	1.23E+06	
Cement and concrete	1.91E+07	

* 'Area' means the area exposed to ground water after closure of SFR 1.

Table 6-2c. Summarised amounts of different materials in Silo at the time of closure, full inventory.**

Material	Weight (kg)	Area (m^2) *
Iron/Steel	3.83E+06	1.66E+05
Aluminium	5.99E+02	1.13E+02
Cellulose (wood, paper, cloth)	7.95E+03	
Other organics (plastic, rubber, cable)	4.32E+04	
Other inorganics	2.78E+04	
Ion-exchange resins	2.45E+06	
Sludge	2.00E+04	
Bitumen	1.36E+06	
Cement and concrete	2.16E+07	

* 'Area' means the area exposed to ground water after closure of SFR 1.

** Regarding volume.

6.3 Radionuclide inventory in Silo

The inventory of radionuclides is presented in Table 6-3. How the amounts of different radionuclides have been calculated is presented in Appendix B.

Table 6-3. Summarised amounts of different radionuclides in Silo at the time of closure for the three different scenarios.

Nuclide	Activity (Bq) Year 2040	Activity (Bq) Year 2050	Activity (Bq) Full inventory*
H-3	3.5E+10	3.6E+10	2.4E+11
Be-10	9.7E+05	1.2E+06	6.7E+06
C-14 org	1.4E+12	1.7E+12	9.6E+12
C-14 inorg	3.2E+12	4.0E+12	2.2E+13
C-14 tot	4.6E+12	5.8E+12	3.2E+13
Cl-36	1.1E+09	1.3E+09	7.5E+09
Fe-55	2.0E+13	2.0E+13	1.4E+14
Ni-59	7.3E+12	9.0E+12	5.1E+13
Co-60	8.5E+13	8.5E+13	5.9E+14
Ni-63	8.9E+14	1.1E+15	6.2E+15
Se-79	1.0E+09	1.2E+09	7.1E+09
Sr-90	1.1E+13	1.2E+13	7.5E+13
Mo-93	2.9E+09	3.2E+09	2.0E+10
Nb-93m	4.8E+11	5.0E+11	3.3E+12
Zr-93	1.8E+09	2.2E+09	1.2E+10
Nb-94	1.6E+10	2.0E+10	1.1E+11
Tc-99	3.6E+11	4.6E+11	2.5E+12
Ru-106	1.4E+09	1.4E+09	1.0E+10
Pd-107	2.6E+08	3.0E+08	1.8E+09
Ag-108m	9.2E+10	1.1E+11	6.4E+11
Cd-113m	3.1E+10	3.2E+10	2.1E+11
Sb-125	2.1E+12	2.1E+12	1.5E+13
Sn-126	1.3E+08	1.5E+08	8.9E+08
I-129	8.1E+08	9.3E+08	5.6E+09
Ba-133	2.8E+09	2.9E+09	2.0E+10
Cs-134	8.9E+11	8.9E+11	6.2E+12
Cs-135	4.0E+09	4.4E+09	2.8E+10
Cs-137	1.1E+14	1.2E+14	7.8E+14
Pm-147	1.7E+12	1.7E+12	1.2E+13
Sm-151	5.7E+11	6.5E+11	4.0E+12
Eu-152	3.6E+09	3.7E+09	2.5E+10
Eu-154	2.5E+12	2.5E+12	1.7E+13
Eu-155	5.5E+11	5.5E+11	3.8E+12
Ho-166m	7.0E+09	8.4E+09	4.9E+10
U-232	5.7E+05	6.7E+05	4.0E+06
U-234	2.6E+07	3.2E+07	1.8E+08
U-235	1.4E+07	1.7E+07	9.8E+07
U-236	1.0E+07	1.2E+07	7.2E+07
Np-237	1.3E+08	1.6E+08	9.0E+08
Pu-238	3.8E+10	4.5E+10	2.6E+11
U-238	2.8E+07	3.4E+07	1.9E+08

Nuclide	Activity (Bq) Year 2040	Activity (Bq) Year 2050	Activity (Bq) Full inventory*
Pu-239	8.9E+09	1.1E+10	6.2E+10
Pu-240	1.8E+10	2.2E+10	1.2E+11
Pu-241	6.6E+11	6.8E+11	4.6E+12
Am-241	4.9E+11	4.8E+11	3.4E+12
Am-242m	2.2E+08	2.7E+08	1.6E+09
Pu-242	7.9E+07	9.6E+07	5.5E+08
Am-243	8.4E+08	1.0E+09	5.8E+09
Cm-243	2.3E+08	2.5E+08	1.6E+09
Cm-244	8.5E+09	9.2E+09	5.9E+10
Cm-245	7.9E+06	9.6E+06	5.5E+07
Cm-246	2.1E+06	2.5E+06	1.5E+07
Tot	1.1E+15	1.4E+15	7.9E+15

* Regarding activity.

6.4 Inplacement of waste in Silo

The Silo contains the majority (approximately 92%) of the total inventory of radionuclides.

In order to do so some kind of distribution of the waste packages had to be made. The main principles has been:

- Waste that actually have been deposited have of course been distributed as they are placed.
- Bitumenised waste is placed in the centre of the silo.

7 Summary

This chapter summarises the waste amounts and the radionuclide inventory from Chapter 3 to 6.

7.1 Amount of waste in SFR 1

The number of the different waste types is presented in Table 7-1. How the numbers have been estimated is presented in Appendix A.

7.2 Radionuclide inventory in SFR 1

The inventory of radionuclides is presented in Table 7-2, 7-3 and 7-4. How the amounts of different radionuclides have been calculated is presented in Appendix B.

Table 7-1. Number of different waste types in SFR 1 for the three different scenarios.

Waste type	Rock cavern	Number of packages Year 2040	Number of packages Year 2050	Number of packages Full inventory*
B.04	Silo	1,000	1,000	1,000
B.05	BMA	4,188	4,188	4,188
B.06	Silo	1,840	1,840	1,840
B.07	1BTF/2BTF	29/194	29/194	29/194
B.12	BLA	238	238	238
B.20	BLA	12	12	12
B.23	BMA	30	30	30
F.05	BMA	1,712	1,712	1,712
F.12	BLA	39	44	42
F.15	BMA	11	11	11
F.17	BMA	1,305	1,605	1,343
F.18	Silo	792	992	1,132
F.20	BLA	15	15	15
F.23	BMA	389	469	399
F.99	BMA/2BTF	2/18	2/18	2/18
O.01	BMA/1BTF	675/28	675/28	675/28
O.02	Silo	1,942	2,342	2,623
O.07	1BTF/2BTF	118/536	159/577	459/532
O.12	BLA	34	44	39
O.23	BMA	570	640	579
O.99	1BTF/2BTF/BLA	20/20/5	20/20/5	20/20/5
C.01	BMA	68	68	68
C.02	Silo	1,210	1,460	1,636
C.23	BMA	162	202	167
C.24	Silo	154	194	222
R.01	BMA/1BTF	1,686/91	1,686/91	1,686/91
R.02	Silo	371	371	371
R.10	BMA/1BTF	126/4	146/4	129/4
R.12	BLA	137	167	152

Waste type	Rock cavern	Number of packages Year 2040	Number of packages Year 2050	Number of packages Full inventory*
R.15	BMA	153	163	154
R.16	Silo	2,737	3,437	3,928
R.23	BMA/1BTF	558/21	608/21	564/21
R.29	BMA	660	960	697
R.99	1BTF	1	1	1
S.04	Silo	181	221	249
S.09	BMA	880	1,080	905
S.11	Silo	60	60	60
S.12	BLA	45	45	45
S.13	1BTF/2BTF	5,055/399	5,130/474	5,700/390
S.14	BLA	75	75	75
S.19	BTF	0	0	0
S.21	BMA	0	0	0
S.22	Silo	0	0	0
S.23	BMA	340	440	352
S.24	Silo	104	134	155
Total		31,100	34,200	35,000

Table 7-2. Summarised amounts of different radionuclides in SFR 1 at the time of closure (2040).

Nuclide	Total (Bq)	Silo (Bq)	BMA (Bq)	1BTF (Bq)	2BTF (Bq)	BLA (Bq)
H-3	3.9E+10	3.5E+10	3.8E+09	1.6E+08	4.0E+08	2.2E+07
Be-10	1.2E+06	9.7E+05	2.2E+05	5.1E+03	2.5E+04	8.1E+02
C-14 org	1.8E+12	1.4E+12	3.2E+11	7.4E+09	5.5E+10	1.2E+09
C-14 inorg	4.1E+12	3.2E+12	7.4E+11	1.7E+10	1.3E+11	2.7E+09
C-14 tot	5.9E+12	4.6E+12	1.1E+12	2.5E+10	1.8E+11	3.9E+09
Cl-36	1.4E+09	1.1E+09	2.3E+08	1.1E+07	3.1E+07	1.0E+06
Fe-55	2.1E+13	2.0E+13	1.5E+12	2.9E+10	2.9E+10	4.2E+09
Ni-59	9.5E+12	7.3E+12	2.1E+12	2.1E+10	4.2E+10	5.3E+09
Co-60	9.3E+13	8.5E+13	7.0E+12	2.9E+11	3.7E+11	3.6E+10
Ni-63	1.2E+15	8.9E+14	2.6E+14	2.3E+12	2.8E+12	6.6E+11
Se-79	1.3E+09	1.0E+09	2.1E+08	5.9E+06	2.4E+07	4.9E+05
Sr-90	1.3E+13	1.1E+13	1.7E+12	6.4E+10	2.1E+11	4.7E+09
Mo-93	3.7E+09	2.9E+09	6.8E+08	4.4E+07	8.9E+07	2.0E+07
Nb-93m	5.5E+11	4.8E+11	5.9E+10	2.3E+09	6.6E+09	3.2E+08
Zr-93	2.3E+09	1.8E+09	4.2E+08	1.5E+07	4.8E+07	4.3E+06
Nb-94	2.0E+10	1.6E+10	3.6E+09	8.5E+07	4.2E+08	1.3E+07
Tc-99	4.1E+11	3.6E+11	3.7E+10	7.0E+09	7.6E+09	5.0E+08
Ru-106	1.6E+09	1.4E+09	1.2E+08	2.6E+06	2.6E+06	3.1E+03
Pd-107	3.2E+08	2.6E+08	5.4E+07	1.5E+06	5.9E+06	1.2E+05
Ag-108m	1.2E+11	9.2E+10	2.0E+10	4.8E+08	2.3E+09	4.2E+08
Cd-113m	3.5E+10	3.1E+10	3.7E+09	1.8E+08	3.4E+08	1.1E+07
Sb-125	2.3E+12	2.1E+12	1.5E+11	6.1E+09	6.2E+09	3.1E+08
Sn-126	1.6E+08	1.3E+08	2.7E+07	7.4E+05	2.9E+06	6.2E+04
I-129	1.0E+09	8.1E+08	1.8E+08	3.1E+06	1.9E+07	3.1E+05
Ba-133	3.1E+09	2.8E+09	2.8E+08	1.3E+07	2.8E+07	1.7E+06
Cs-134	9.6E+11	8.9E+11	7.0E+10	6.1E+08	6.1E+08	2.5E+07
Cs-135	5.1E+09	4.0E+09	1.0E+09	1.6E+07	1.1E+08	2.0E+06

Nuclide	Total (Bq)	Silo (Bq)	BMA (Bq)	1BTF (Bq)	2BTF (Bq)	BLA (Bq)
Cs-137	1.3E+14	1.1E+14	2.0E+13	6.7E+11	1.8E+12	4.8E+10
Pm-147	1.9E+12	1.7E+12	1.6E+11	2.1E+09	2.1E+09	1.0E+08
Sm-151	7.0E+11	5.7E+11	1.1E+11	3.4E+09	1.2E+10	2.7E+08
Eu-152	5.4E+09	3.6E+09	4.0E+08	1.5E+08	1.1E+08	1.1E+09
Eu-154	2.8E+12	2.5E+12	2.3E+11	1.2E+10	1.7E+10	7.5E+08
Eu-155	6.0E+11	5.5E+11	4.7E+10	1.7E+09	1.8E+09	8.1E+07
Ho-166m	8.9E+09	7.0E+09	1.6E+09	5.8E+07	1.9E+08	1.7E+07
U-232	7.5E+05	5.7E+05	1.5E+05	9.6E+03	1.1E+04	6.0E+02
U-234	3.5E+07	2.6E+07	7.7E+06	5.0E+05	5.4E+05	3.0E+04
U-235	4.7E+08	1.4E+07	2.8E+06	1.9E+07	7.5E+05	4.3E+08
U-236	1.4E+07	1.0E+07	3.0E+06	1.9E+05	3.7E+05	9.0E+03
Np-237	1.6E+08	1.3E+08	2.5E+07	4.1E+05	1.9E+06	2.7E+04
Pu-238	4.9E+10	3.8E+10	9.6E+09	3.5E+08	6.0E+08	4.7E+07
U-238	1.5E+09	2.8E+07	6.2E+06	3.3E+05	8.8E+05	1.4E+09
Pu-239	1.1E+10	8.9E+09	2.0E+09	1.5E+08	2.0E+08	1.3E+07
Pu-240	2.3E+10	1.8E+10	4.0E+09	3.0E+08	4.0E+08	2.6E+07
Pu-241	8.2E+11	6.6E+11	1.4E+11	7.5E+09	8.6E+09	4.8E+08
Am-241	5.0E+11	4.9E+11	6.9E+09	8.4E+08	4.9E+08	5.0E+07
Am-242m	3.0E+08	2.2E+08	6.2E+07	4.0E+06	4.4E+06	2.4E+05
Pu-242	1.1E+08	7.9E+07	2.3E+07	1.5E+06	1.6E+06	9.0E+04
Am-243	1.1E+09	8.4E+08	2.3E+08	1.5E+07	1.6E+07	9.0E+05
Cm-243	2.9E+08	2.3E+08	5.5E+07	3.0E+06	2.0E+06	2.3E+05
Cm-244	1.0E+10	8.5E+09	1.4E+09	1.0E+08	1.1E+08	2.2E+07
Cm-245	1.0E+07	7.9E+06	2.3E+06	1.5E+05	1.6E+05	9.0E+03
Cm-246	2.8E+06	2.1E+06	6.1E+05	4.0E+04	4.3E+04	2.4E+03
Tot	1.4E+15	1.1E+15	2.9E+14	3.5E+12	5.5E+12	7.7E+11

Table 7-3. Summarised amounts of different radionuclides in SFR 1 at the time of closure (2050).

Nuclide	Total (Bq)	Silo (Bq)	BMA (Bq)	1BTF (Bq)	2BTF (Bq)	BLA (Bq)
H-3	4.0E+10	3.6E+10	3.4E+09	1.7E+08	3.0E+08	1.9E+07
Be-10	1.4E+06	1.2E+06	2.3E+05	6.4E+03	2.6E+04	9.2E+02
C-14 org	2.1E+12	1.7E+12	3.4E+11	9.5E+09	5.8E+10	1.4E+09
C-14 inorg	5.0E+12	4.0E+12	8.0E+11	2.2E+10	1.3E+11	3.3E+09
C-14 tot	7.1E+12	5.8E+12	1.1E+12	3.2E+10	1.9E+11	4.7E+09
Cl-36	1.6E+09	1.3E+09	2.5E+08	1.2E+07	3.2E+07	1.2E+06
Fe-55	2.1E+13	2.0E+13	1.5E+12	2.9E+10	2.9E+10	4.0E+09
Ni-59	1.1E+13	9.0E+12	2.1E+12	2.3E+10	4.4E+10	6.1E+09
Co-60	9.3E+13	8.5E+13	6.8E+12	2.9E+11	3.1E+11	3.1E+10
Ni-63	1.4E+15	1.1E+15	2.6E+14	2.4E+12	2.8E+12	7.7E+11
Se-79	1.5E+09	1.2E+09	2.3E+08	6.7E+06	2.4E+07	5.1E+05
Sr-90	1.3E+13	1.2E+13	1.5E+12	6.4E+10	1.8E+11	3.9E+09
Mo-93	4.2E+09	3.2E+09	7.5E+08	5.2E+07	9.7E+07	2.7E+07
Nb-93m	5.7E+11	5.0E+11	5.3E+10	2.5E+09	5.3E+09	3.0E+08
Zr-93	2.7E+09	2.2E+09	4.5E+08	1.8E+07	5.1E+07	5.1E+06
Nb-94	2.4E+10	2.0E+10	3.9E+09	1.1E+08	4.4E+08	1.5E+07
Tc-99	5.2E+11	4.6E+11	4.4E+10	7.4E+09	8.1E+09	6.4E+08
Ru-106	1.6E+09	1.4E+09	1.2E+08	2.6E+06	2.6E+06	3.1E+03
Pd-107	3.7E+08	3.0E+08	5.6E+07	1.7E+06	6.1E+06	1.3E+05

Nuclide	Total (Bq)	Silo (Bq)	BMA (Bq)	1BTF (Bq)	2BTF (Bq)	BLA (Bq)
Ag-108m	1.4E+11	1.1E+11	2.1E+10	6.0E+08	2.4E+09	4.2E+08
Cd-113m	3.5E+10	3.2E+10	3.2E+09	1.6E+08	2.6E+08	7.3E+06
Sb-125	2.3E+12	2.1E+12	1.5E+11	6.1E+09	6.1E+09	2.9E+08
Sn-126	1.8E+08	1.5E+08	2.8E+07	8.4E+05	3.0E+06	6.3E+04
I-129	1.1E+09	9.3E+08	1.9E+08	3.6E+06	1.9E+07	3.3E+05
Ba-133	3.2E+09	2.9E+09	2.6E+08	1.3E+07	2.1E+07	1.5E+06
Cs-134	9.7E+11	8.9E+11	7.0E+10	6.1E+08	6.1E+08	1.8E+07
Cs-135	5.6E+09	4.4E+09	1.1E+09	1.8E+07	1.1E+08	2.1E+06
Cs-137	1.4E+14	1.2E+14	1.9E+13	6.6E+11	1.6E+12	4.1E+10
Pm-147	1.9E+12	1.7E+12	1.6E+11	2.1E+09	2.1E+09	7.4E+07
Sm-151	7.7E+11	6.5E+11	1.1E+11	3.7E+09	1.1E+10	2.6E+08
Eu-152	4.9E+09	3.7E+09	3.4E+08	1.3E+08	1.0E+08	6.5E+08
Eu-154	2.7E+12	2.5E+12	2.1E+11	1.0E+10	1.2E+10	4.4E+08
Eu-155	6.0E+11	5.5E+11	4.6E+10	1.4E+09	1.5E+09	4.6E+07
Ho-166m	1.0E+10	8.4E+09	1.8E+09	7.1E+07	2.0E+08	2.0E+07
U-232	8.4E+05	6.7E+05	1.5E+05	9.6E+03	1.0E+04	5.7E+02
U-234	4.1E+07	3.2E+07	8.3E+06	5.3E+05	5.7E+05	3.1E+04
U-235	4.7E+08	1.7E+07	3.2E+06	1.9E+07	8.5E+05	4.3E+08
U-236	1.6E+07	1.2E+07	3.2E+06	2.1E+05	3.9E+05	9.4E+03
Np-237	1.9E+08	1.6E+08	2.9E+07	4.7E+05	2.0E+06	2.8E+04
Pu-238	5.5E+10	4.5E+10	9.5E+09	3.7E+08	6.0E+08	4.6E+07
U-238	1.5E+09	3.4E+07	6.8E+06	3.9E+05	9.4E+05	1.4E+09
Pu-239	1.3E+10	1.1E+10	2.0E+09	1.6E+08	2.1E+08	1.3E+07
Pu-240	2.7E+10	2.2E+10	4.1E+09	3.1E+08	4.1E+08	2.7E+07
Pu-241	8.3E+11	6.8E+11	1.3E+11	6.4E+09	7.0E+09	3.6E+08
Am-241	4.9E+11	4.8E+11	7.7E+09	8.6E+08	5.1E+08	5.3E+07
Am-242m	3.4E+08	2.7E+08	6.5E+07	4.1E+06	4.5E+06	2.4E+05
Pu-242	1.2E+08	9.6E+07	2.5E+07	1.6E+06	1.7E+06	9.4E+04
Am-243	1.3E+09	1.0E+09	2.5E+08	1.6E+07	1.7E+07	9.3E+05
Cm-243	3.1E+08	2.5E+08	5.3E+07	2.5E+06	1.8E+06	2.0E+05
Cm-244	1.1E+10	9.2E+09	1.2E+09	8.3E+07	9.0E+07	1.9E+07
Cm-245	1.2E+07	9.6E+06	2.5E+06	1.6E+05	1.7E+05	9.3E+03
Cm-246	3.3E+06	2.5E+06	6.6E+05	4.2E+04	4.5E+04	2.5E+03
Tot	1.7E+15	1.4E+15	3.0E+14	3.6E+12	5.2E+12	8.7E+11

Table 7-4. Summarised amounts of different radionuclides in SFR 1 at the time of closure (2040) at a full inventory regarding activity.

Nuclide	Total (Bq)	Silo (Bq)	BMA (Bq)	1BTF (Bq)	2BTF (Bq)	BLA (Bq)
H-3	2.7E+11	2.4E+11	2.6E+10	1.1E+09	2.8E+09	1.5E+08
Be-10	8.4E+06	6.7E+06	1.5E+06	3.5E+04	1.7E+05	5.6E+03
C-14 org	1.2E+13	9.6E+12	2.2E+12	5.1E+10	3.8E+11	8.0E+09
C-14 inorg	2.9E+13	2.2E+13	5.2E+12	1.2E+11	9.0E+11	1.9E+10
C-14 tot	4.1E+13	3.2E+13	7.4E+12	1.7E+11	1.3E+12	2.7E+10
Cl-36	9.4E+09	7.5E+09	1.6E+09	7.3E+07	2.1E+08	7.2E+06
Fe-55	1.5E+14	1.4E+14	1.0E+13	2.0E+11	2.0E+11	2.9E+10
Ni-59	6.6E+13	5.1E+13	1.4E+13	1.4E+11	2.9E+11	3.7E+10
Co-60	6.5E+14	5.9E+14	4.9E+13	2.0E+12	2.6E+12	2.5E+11
Ni-63	8.0E+15	6.2E+15	1.8E+15	1.6E+13	1.9E+13	4.6E+12

Nuclide	Total (Bq)	Silo (Bq)	BMA (Bq)	1BTF (Bq)	2BTF (Bq)	BLA (Bq)
Se-79	8.8E+09	7.1E+09	1.5E+09	4.1E+07	1.6E+08	3.4E+06
Sr-90	8.9E+13	7.5E+13	1.2E+13	4.5E+11	1.4E+12	3.3E+10
Mo-93	2.6E+10	2.0E+10	4.7E+09	3.1E+08	6.2E+08	1.4E+08
Nb-93m	3.8E+12	3.3E+12	4.1E+11	1.6E+10	4.6E+10	2.2E+09
Zr-93	1.6E+10	1.2E+10	2.9E+09	1.0E+08	3.3E+08	3.0E+07
Nb-94	1.4E+11	1.1E+11	2.5E+10	5.9E+08	2.9E+09	9.2E+07
Tc-99	2.9E+12	2.5E+12	2.6E+11	4.8E+10	5.3E+10	3.5E+09
Ru-106	1.1E+10	1.0E+10	8.1E+08	1.8E+07	1.8E+07	2.2E+04
Pd-107	2.2E+09	1.8E+09	3.7E+08	1.0E+07	4.1E+07	8.6E+05
Ag-108m	8.0E+11	6.4E+11	1.4E+11	3.4E+09	1.6E+10	2.9E+09
Cd-113m	2.4E+11	2.1E+11	2.5E+10	1.2E+09	2.3E+09	7.4E+07
Sb-125	1.6E+13	1.5E+13	1.1E+12	4.2E+10	4.3E+10	2.1E+09
Sn-126	1.1E+09	8.9E+08	1.9E+08	5.2E+06	2.0E+07	4.3E+05
I-129	7.0E+09	5.6E+09	1.3E+09	2.1E+07	1.3E+08	2.2E+06
Ba-133	2.2E+10	2.0E+10	2.0E+09	8.8E+07	1.9E+08	1.2E+07
Cs-134	6.7E+12	6.2E+12	4.8E+11	4.3E+09	4.3E+09	1.7E+08
Cs-135	3.6E+10	2.8E+10	7.2E+09	1.1E+08	7.6E+08	1.4E+07
Cs-137	9.4E+14	7.8E+14	1.4E+14	4.6E+12	1.3E+13	3.4E+11
Pm-147	1.3E+13	1.2E+13	1.1E+12	1.5E+10	1.5E+10	7.0E+08
Sm-151	4.9E+12	4.0E+12	7.6E+11	2.3E+10	8.2E+10	1.9E+09
Eu-152	3.7E+10	2.5E+10	2.8E+09	1.1E+09	7.4E+08	7.5E+09
Eu-154	1.9E+13	1.7E+13	1.6E+12	8.3E+10	1.2E+11	5.2E+09
Eu-155	4.2E+12	3.8E+12	3.3E+11	1.2E+10	1.2E+10	5.6E+08
Ho-166m	6.2E+10	4.9E+10	1.1E+10	4.1E+08	1.3E+09	1.2E+08
U-232	5.2E+06	4.0E+06	1.0E+06	6.7E+04	7.4E+04	4.1E+03
U-234	2.4E+08	1.8E+08	5.3E+07	3.5E+06	3.7E+06	2.1E+05
U-235	3.2E+09	9.8E+07	2.0E+07	1.3E+08	5.2E+06	3.0E+09
U-236	9.7E+07	7.2E+07	2.1E+07	1.3E+06	2.6E+06	6.2E+04
Np-237	1.1E+09	9.0E+08	1.7E+08	2.8E+06	1.3E+07	1.9E+05
Pu-238	3.4E+11	2.6E+11	6.7E+10	2.4E+09	4.2E+09	3.3E+08
U-238	1.0E+10	1.9E+08	4.3E+07	2.3E+06	6.1E+06	9.9E+09
Pu-239	7.8E+10	6.2E+10	1.4E+10	1.0E+09	1.4E+09	9.1E+07
Pu-240	1.6E+11	1.2E+11	2.7E+10	2.1E+09	2.8E+09	1.8E+08
Pu-241	5.7E+12	4.6E+12	9.7E+11	5.2E+10	6.0E+10	3.4E+09
Am-241	3.5E+12	3.4E+12	4.8E+10	5.8E+09	3.4E+09	3.5E+08
Am-242m	2.1E+09	1.6E+09	4.3E+08	2.8E+07	3.0E+07	1.7E+06
Pu-242	7.3E+08	5.5E+08	1.6E+08	1.0E+07	1.1E+07	6.2E+05
Am-243	7.6E+09	5.8E+09	1.6E+09	1.0E+08	1.1E+08	6.2E+06
Cm-243	2.0E+09	1.6E+09	3.8E+08	2.1E+07	1.4E+07	1.6E+06
Cm-244	7.0E+10	5.9E+10	9.4E+09	7.1E+08	7.8E+08	1.5E+08
Cm-245	7.3E+07	5.5E+07	1.6E+07	1.0E+06	1.1E+06	6.2E+04
Cm-246	1.9E+07	1.5E+07	4.2E+06	2.7E+05	3.0E+05	1.7E+04
Tot	1.0E+16	7.9E+15	2.0E+15	2.4E+13	3.8E+13	5.3E+12

7.3 Toxic material

No toxic material has been identified in SFR 1. Only small amounts of toxic material is allowed, e.g. lead, epoxi, asbestos.

8 Uncertainties

Three major uncertainties have been identified at a general level. Uncertainties in different waste types etc are described in the text in Appendices A–F.

The actual number of different waste packages is the first one. As in all prognosis there is a major uncertainty in what lays in the future. Not only the number of packages is uncertain there could also be other waste types or changes in the described types that occur in the future.

The amount of materials is calculated with a simple model, a reference waste package times a prognosis of waste packages. This means that the uncertainties are multiplied. One should keep in mind that the most materials come from waste types that are relatively common and therefore are the uncertainties smaller. No uncertainty analysis has been performed since it has been judged that what is in future waste overshadows all other uncertainties. The future uncertainty can not be quantified.

The radionuclides are estimated with correlation factors. The factors themselves include quite large uncertainties, but the method of correlation has weaknesses as well. For more discussion about correlation factors see Appendix A and references therein. A statistical quantification regarding the uncertainties for the correlation factors can be found in /Cronstrand 2005/. Further updates have been made on the subject and are presented in /Torstenfelt 2007/.

9 References

This reference list only includes references from the main report. References in the appendices are listed after each appendix (Appendices A–F).

SKB, 1993. Slutlig säkerhetsrapport för SFR 1 (in Swedish). Svensk Kärnbränslehantering AB, May 1993.

Cronstrand, 2005. Assessment of uncertainty to correlation factors. SKB R-05-76, Svensk Kärnbränslehantering AB.

Torstenfelt, 2007. Osäkerheter för radionuklider för deponering i SFR 1. Dok ID 1086802, version 3.0.

Appendix A

Calculation of waste amounts

1 Method of calculation

1.1 Aim of calculations – restrictions

To know the amounts of waste or rather the number of waste packages is not important in it self when you perform a safety assessment for a repository for radioactive waste. However, the number of waste packages is a fundamental parameter if you are to calculate the total amount of materials, radionuclides and chemicals. The aim of the calculations of waste amounts is to get a realistic estimate of the amounts of radionuclides and materials and also their distribution in different parts of the SFR 1 repository.

There are always restrictions to a calculation; in this case it is of course the geometry of the repository. Also, as a starting point of the analysis we have assumed two different scenarios; that the Swedish reactors have a life span of 50 or 60 years and that they during these years produce waste that can and should be disposed of in SFR 1. Also we have assumed that the repository will be sealed in year 2040 or 2050 corresponding to a reactor life span of 50 and 60 years respectively.

1.2 Available information sources

Radioactive waste has been produced in the Swedish nuclear power plants since the early 70's. In the research facility Studsvik and other places production of radioactive waste took place earlier than the 70's. The waste amounts produced through research to be disposed of in SFR 1 are small, if any, from the period before 1980. The SFR repository has been in operation since 1988. Hence, today there is an extensive experience regarding waste handling in Sweden. The sources to get information and data in this study are relatively few since the ones that are used are to be seen as thorough collections of data.

Information about waste that has already been disposed of in SFR has been taken from the operational database /Triumf 2007/ in SFR.

Regarding future amounts of different waste types the nuclear power plants and Clab and Studsvik have reported data for annual production /Johansson 2007/ and waste packages stored at the site. This information is stored in Prosit /Prosit 2007/.

1.3 Method of calculation

All calculations are performed by Prosit. In order to systemise the vast amounts of different sorts of waste Prosit uses the waste types described in the main report Section 2.1.

The calculations of the number of waste packages can be separated in two categories, existing waste and future waste.

Concerning existing waste Prosit extracts data from the operational database in SFR, Triumf. The number of existing waste packages in this inventory is the number of deposited packages 31 December 2006.

Concerning future waste, Prosit calculates the number of future packages of each waste type by multiplying the annual production with the number of remaining years of nuclear power plant operation and adding the packages stored on site.

The remaining years of operation has been estimated for the two different scenarios; 50 and 60 years operation. In the prognosis it is assumed that waste produced one year is delivered to SFR that same year. Since it is almost impossible to back track the different waste streams to a single reactor, an average of end of waste production per site has been calculated. The remaining time (in years) is then calculated as "average year" minus "present year" (2006). In Table 1-1 the assumed years of operation for the two scenarios are presented.

Four nuclear facilities differ from the described calculation above, Barsebäck 1, Barsebäck 2, Studsvik and Clab. Barsebäck 1 was closed in November 1999 and Barsebäck 2 in May 2005. Clab will be in operation until the last spent fuel is deposited in the deep repository, therefore it is assumed that Clab produces waste until SFR 1 is closed. Studsvik has been estimated to produce waste until their quota is full. Regarding volume, BLA will be the only cavern which limits waste production for reactor life span of 50 and 60 years respectively for Studsvik according to the prognosis made. Regarding activity there are some nuclides that will render limitations on waste production. For these nuclides the upper limit has been put in as a restriction and it is assumed that Studsvik will produce waste that will fit within these limitations. Hence Studsvik is assumed to be able to produce some waste types longer than others due to these limitations. This estimate is very uncertain.

Only waste types that are in production today or are in the process of being licensed have been included in the calculations. Possible changes in the production of the different waste types are ignored due to lack of data.

Table 1-1. Estimated end of operational waste production in the Swedish nuclear facilities.

Reactor	Start of operation*	Estimated end of operation* (reactor life span 50/60 years)	Average (per site) end of waste production (reactor life span 50/60 years)
Barsebäck 1	1 July 1975	30 November 1999	
Barsebäck 2	1 July 1977	31 May 2005	
Forsmark 1	10 December 1980	9 December 2030/2040	
Forsmark 2	7 July 1981	6 July 2031/2041	
Forsmark 3	22 August 1985	21 August 2035/2045	2032/2042
Oskarshamn 1	6 February 1972	5 February 2022/2032	
Oskarshamn 2	15 December 1974	14 December 2024/2034	
Oskarshamn 3	15 August 1985	14 August 2035/2042	2027/2037
Ringhals 1	1 January 1976	31 December 2025/2035	
Ringhals 2	1 May 1975	30 April 2015	
Ringhals 3	9 September 1981	8 September 2031/2041	
Ringhals 4	21 November 1983	20 November 2033/2043	2029/2039
Clab			2039/2049
Studsvik			2039/2049

* This is an estimate of end of operation for nuclear facilities in Sweden. The purpose is to estimate waste amounts in SFR 1. Actual end of operation could differ significantly from this table. For Studsvik it is assumed that the average end of waste production is 2039/2049, although due to the limitations in activity some of the waste types will not be able to be produced the entire waste production cycle.

In SFR some odd waste packages that have odd geometry, chemical composition or some other feature that differ from the ordinary waste types (numbered X:99) are deposited and some are currently stored on site waiting for transportation to SFR. In this prognosis only already deposited waste and waste stored on site is included. No consideration to future production of odd waste packages is taken.

A schematic picture of the calculation is shown in Figure 1-1.

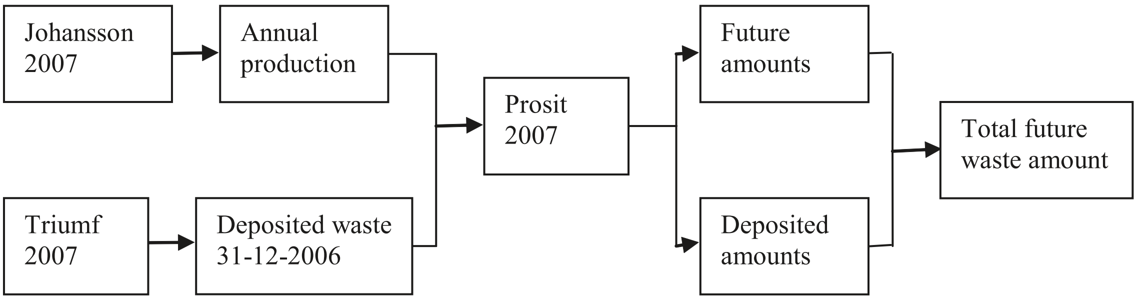

Figure 1-1. *Schematic structure of calculations.*

2 Waste amounts

2.1 Present amount

At present, i.e. 31 December 2006, there are approximately 20,000 waste packages of different kinds disposed of in SFR 1. Data for the packages are presented in Table 2-1. The data is quantified in waste types according to the types presented in Appendix C to Appendix F. Note that for waste that is stored on site and supposed to be disposed of in BTF the waste packages are split between 1BTF and 2BTF in this report.

Table 2-1. Number of waste packages disposed of in SFR 1 in 31 January 2006.

Waste type	Rock cavern in SFR	Waste packages 31 December 2006	Waste packages stored on site
B.04	Silo	0	1,000
B.05:0/B.05:9	BMA	288/3,056	0
B.05:2	BMA	844	0
B.06	Silo	1,680	160
B.07	1BTF/2BTF	24/189	5/5
B.12	BLA	215	23
B.20	BLA	12	0
B.23	BMA	0	30
F.05	BMA	1,712	0
F.12	BLA	24	2
F.15	BMA	11	0
F.17	BMA	431	94
F.18	Silo	212	60
F.20	BLA	15	0
F.23	BMA	168	13
F.99	BMA/2BTF	2/18	0/0
O.01	BMA/1BTF	670/28	5/0
O.02	Silo	694	408
O.07	1BTF/2BTF	16/434	19/19
O.12	BLA	2	11
O.23	BMA	365	58
O.99	1BTF/2BTF/BLA	0	20/20/5
C.01	BMA	68	0
C.02	Silo	150	300
C.23	BMA	25	19
C.24	Silo	0	36
R.01	BMA/1BTF	1,683/91	3/0
R.02	Silo	150	23
R.10	BMA/1BTF	64/4	16/0
R.12	BLA	50	18
R.15	BMA	124	6
R.16	Silo	936	191
R.23	BMA/1BTF	406/21	37/0
R.29	BMA	0	0
R.99	1BTF	1	0

S.04	Silo	32	13
S.09	BMA	0	200
S.11	Silo	60	0
S.12	BLA	0	28
S.13	1BTF/2BTF	4,656/0	144/144
S.14	BLA	75	0
S.23	BMA	0	0
S.24	Silo	0	0

2.2 Future amount

In Table 2-2 estimated future annual production rates are presented. The data is quantified in waste types according to the types presented in Appendix C to Appendix F. Note that for waste that is supposed to be disposed of in BTF the annual production is split between 1BTF and 2BTF in this report.

Table 2-2. Future amounts of Waste in SFR 1.

Waste type	Rock cavern	Annual production	Future operational years (reactor life span 50/60 years)
B.04	Silo	0	
B.05	BMA	0	
B.06	Silo	0	
B.07	1BTF/2BTF	0	
B.12	BLA	0	
B.20	BLA	0	
B.23	BMA	0	
F.05	BMA	0	26/36
F.12	BLA	0.5	
F.15	BMA	0	
F.17	BMA	30	
F.18	Silo	20	
F.20	BLA	0	
F.23	BMA	8	
F.99	BMA/2BTF	0/0	
O.01	BMA/1BTF	0/0	21/31
O.02	Silo	40	
O.07	1BTF/2BTF	4/4	
O.12	BLA	1	
O.23	BMA	7	
O.99	1BTF/2BTF/BLA	0/0/0	
C.01	BMA	0	34/44
C.02	Silo	15 (25*)	
C.23	BMA	2 (4*)	
C.24	Silo	2 (4*)	
R.01	BMA/1BTF	0/0	23/33
R.02	Silo	0	
R.10	BMA/1BTF	2/0	
R.12	BLA	3	
R.15	BMA	1	
R.16	Silo	70	

R.23	BMA/1BTF	5/0	
R.29	BMA	30	
R.99	1BTF	0	
S.04	Silo	4	**
S.09	BMA	20	**
S.11	Silo	0	**
S.12	BLA	1	**
S.13	1BTF/2BTF	7.5/7.5	**
S.14	BLA	0	**
S.23	BMA	10	**
S.24	Silo	3	**

* After the year 2015.

** The operational years will depend on how Studsvik will be able to produce waste that fits with the nuclide and volume restrictions, hence no specific number is given here.

2.3 Total future amount

If you add the present and future waste amounts you will find that the repository does not get filled to 100%, see Table 2-3. In Silo there is quite a lot of space left even after the assumed year 2050. The volume in BLA is already at the year 2040 somewhat limited. Regarding BTF the volume between 1BTF and 2BTF is somewhat distorted. This is due to a simplification that all future waste will be split between the two caverns regardless of the present content. No consideration has been taken whether the caverns gets full (with the exception of waste from Studsvik where there are limitations for each cavern in an agreement between SKB and Studsvik) while conducting the activity and material prognosis. It is assumed that should one of the caverns get full there is the possibility of compacting waste to render more free volume.

The numbers that are presented in is the total amount of waste packages in SFR 1 with an estimated closure the year 2040 and 2050 respectively. These numbers are the foundation in the safety assessment for all calculations of materials, radionuclides etc that has its origin in the waste in SFR.

Table 2-3. Percent filling regarding volume in the SFR 1 according to calculations.

	Silo	1BTF*	2BTF*	BMA	BLA	Total
Year 2040	74%	48%	101%	99%	96%	87%
Year 2050	89%	54%	107%	111%	105%	98%

* Note that a simplification has been made regarding future waste in BTF. It is assumed that all future waste is split evenly between 1BTF and 2BTF, while the waste that is already in store is not supposed to be touched. This leads to the somewhat distorted distribution in volume between the two caverns.

Table 2-4. Total number of waste packages in SFR 1 in 2040 and 2050.

Waste type	Rock cavern	Number of waste packages 2040	Number of waste packages 2050
B.04	Silo	1,000	1,000
B.05	BMA	4,188	4,188
B.06	Silo	1,840	1,840
B.07	1BTF/2BTF	29/194	29/194
B.12	BLA	238	238
B.20	BLA	12	12

Waste type	Rock cavern	Number of waste packages 2040	Number of waste packages 2050
B.23	BMA	30	30
F.05	BMA	1,712	1,712
F.12	BLA	39	44
F.15	BMA	11	11
F.17	BMA	1,305	1,605
F.18	Silo	792	992
F.20	BLA	15	15
F.23	BMA	389	469
F.99	BMA/2BTF	2/18	2/18
O.01	BMA/1BTF	675/28	675/28
O.02	Silo	1,942	2,342
O.07	1BTF/2BTF	119/537	159/577
O.12	BLA	34	44
O.23	BMA	570	640
O.99	1BTF/2BTF/BLA	20/20/5	20/20/5
C.01	BMA	68	68
C.02	Silo	1,210	1,460
C.23	BMA	162	202
C.24	Silo	154	194
R.01	BMA/1BTF	1,686/91	1,686/91
R.02	Silo	371	371
R.10	BMA/1BTF	126/4	146/4
R.12	BLA	137	167
R.15	BMA	153	163
R.16	Silo	2,737	3,437
R.23	BMA/1BTF	558/21	608/21
R.29	BMA	660	960
R.99	1BTF	1	1
S.04	Silo	181	221
S.09	BMA	880	1,080
S.11	Silo	60	60
S.12	BLA	45	45
S.13	1BTF/2BTF	5,055/399	5,130/474
S.14	BLA	75	75
S.23	BMA	340	440
S.24	Silo	104	134
Total		31,057	34,167

3 Uncertainties

The identified uncertainties are:

- The annual production of waste varies from year to year. The numbers are originally based on experience and knowledge of the different waste producers. Over a period of time the uncertainties are smaller.
- New waste types will certainly be licensed during the time until 2040/2050.
- The operational life of the reactors could vary a lot because of economical, political and/or technical issues.

In total the uncertainties are relatively large. But one has to remember the purpose of the calculations is to form a best estimate of waste amounts.

4 References

Johansson, 2007. Prognos över nyttjande av SFR 1, Dok ID 1069693, version 1.0.

Triumf, 2007. Operational database for SFR, SKB [accessed February 2007].

Prosit, 2007. Prognosis tool for SFR, SKB [accessed February 2007].

Appendix B

Calculation of radionuclide inventory

1 Method of calculation

1.1 Aim of calculations – restrictions

The primary issue in a safety assessment of a repository for radioactive waste is almost always to calculate the release of radionuclides to the environment. A nuclide specific estimate of the radionuclide content in the repository is then fundamental. This is also true for the safety assessment of SFR 1.

The most accurate method to estimate the radionuclides is almost always to make direct measurements on the waste. Unfortunately this is never feasible in reality for all waste, due to a number of reasons, e.g. the waste is not produced yet, the geometry, the shielding of activity (high detection limit), the measurements are not correct etc. The solution is to make some kind of estimate.

For the safety assessment of SFR 1 estimates are used since the facility still is in operation and waste are constantly being supplied to the repository. Also, there are no possibilities to make measurements on the waste packages other than on gamma emitting nuclides.

The license for SFR 1 allows a maximum of 10^{16} Bq to be deposited by the time of sealing. In this report we present the inventory for three different scenarios: two inventories for a lifetime of the nuclear power plants of 50 and 60 years and one scenario where the inventory reaches 10^{16} Bq.

1.2 Available information sources

In the work estimating a radionuclide inventory we have aspired to use the best available data for this kind of waste. The sources we have used are

- /Triumf 2007/, a database for waste deposited in SFR 1. The database includes detailed information for the approximately 20,000 deposited waste packages regarding gamma emitting nuclides. The data has been restricted to waste deposited before December 2006.
- /Prosit 2007/, a prognosis tool linked to the Triumf database. Prosit entails correlation factors, data for reference waste types, data on future production of waste, transuranic activities etc.
- Data concerning future waste is taken from /Johansson 2007/.
- Correlation factors are firstly taken from SKB report R-07-05 "Correlation factors for C-14, Cl-36, Ni-59, Ni-63, Mo-93, Tc-99, I-129 and Cs-135 in operational waste for SFR 1" /Lindgren et al. 2007/ and secondarily from "Low and Intermediate Level Waste in SFL 3-5: Reference Inventory" /Lindgren et al. 1998/.
- For a handful radionuclides a new and improved method of assessing the amount has been used. The nuclides at hand are Pu-239/240, C-14, Cl-36, Ni-59, Ni-63, Mo-93, Tc-99, I-129 and Cs-135, see Section 1.3.6.
- Half-lives for nuclides have been taken from /Firestone 1998/.

1.3 Method of calculation

1.3.1 Correlation factors

The nuclides that have been judged to exist in the waste and have such a long half-life that it is meaningful to perform release calculations on them are usually not directly measurable on the packages. In this report, nuclides with a half-life longer than one year are included. The method used to make an estimate is based on correlation factors. The "key-nuclides" used are Co-60, Cs-137 and Pu-239/240.

Note that no exclusion is made dependent on the amount of the nuclide in this phase of the safety assessment. In the calculations of radionuclide release in the safety assessment some nuclides were judged as negligible.

For the waste packages that already have measured values for the nuclides these values take precedence over the values rendered with the correlation factors.

The nuclides and the correlation factors are presented in Table 1-1 to 1-3.

Table 1-1. Nuclides correlated to Co-60.

Nuclide	Half-life (yr)	Correlation factor
H-3	12.3	$1\cdot10^{-4}$
Be-10	$1.51\cdot10^{6}$	$6\cdot10^{-10}$
Cl-36	$3.01\cdot10^{5}$	$6\cdot10^{-7}$
Fe-55	2.73	1
Ni-59 (BWR)	$7.60\cdot10^{4}$	$1\cdot10^{-3}$
Ni-59 (PWR)	$7.60\cdot10^{4}$	$3\cdot10^{-2}$
Co-60	**5.27**	**1**
Ni-63 (BWR)	100	$8\cdot10^{-2}$
Ni-63 (PWR)	100	4
Mo-93	$4.00\cdot10^{3}$	$1\cdot10^{-6}$
Zr-93	$1.53\cdot10^{6}$	$1\cdot10^{-6}$
Nb-93m	16.1	$1\cdot10^{-3}$
Nb-94	$2.03\cdot10^{4}$	$1\cdot10^{-5}$
Tc-99*	$2.11\cdot10^{5}$	$3\cdot10^{-6}$
Ag-108m	418	$6\cdot10^{-5}$
Sb-125	2.76	$1\cdot10^{-1}$
Ba-133	10.5	$1\cdot10^{-5}$
Ho-166m	$1.20\cdot10^{3}$	$4\cdot10^{-6}$

* Tc-99 is correlated with both Co-60 and Cs-137.

Table 1-2. Nuclides correlated to Cs-137.

Nuclide	Half-life (yr)	Correlation factor
Se-79	$1.13\cdot10^{6}$	$4\cdot10^{-6}$
Sr-90	28.8	$1\cdot10^{-1}$
Tc-99*	$2.11\cdot10^{5}$	$9\cdot10^{-4}$
Ru-106	1.02	$5\cdot10^{-3}$
Pd-107	$6.50\cdot10^{6}$	$1\cdot10^{-6}$
Cd-113m	14.1	$6\cdot10^{-4}$
Sn-126	$1.00\cdot10^{5}$	$5\cdot10^{-7}$
I-129	$1.57\cdot10^{7}$	$3\cdot10^{-6}$
Cs-134	2.07	1
Cs-135	$2.30\cdot10^{6}$	$1\cdot10^{-5}$
Cs-137	**30.1**	**1**
Pm-147	2.62	$9\cdot10^{-1}$
Sm-151	90.0	$3\cdot10^{-3}$
Eu-152	13.5	$7\cdot10^{-5}$
Eu-154	8.59	$1\cdot10^{-1}$
Eu-155	4.76	$7\cdot10^{-2}$

* Tc-99 is correlated with both Co-60 and Cs-137.

Table 1-3. Nuclides correlated to Pu-239 and Pu-240.

Nuclide	Half-life (yr)	Correlation factor
U-232	68.9	$3 \cdot 10^{-5}$
U-234	$2.46 \cdot 10^{5}$	$1 \cdot 10^{-3}$
U-235	$7.04 \cdot 10^{8}$	$2 \cdot 10^{-5}$
U-236	$2.34 \cdot 10^{7}$	$3 \cdot 10^{-4}$
U-238	$4.47 \cdot 10^{9}$	$4 \cdot 10^{-4}$
Np-237	$2.14 \cdot 10^{6}$	$4 \cdot 10^{-4}$
Pu-238	87.7	4
Pu-239	**$2.41 \cdot 10^{4}$**	**1**
Pu-240	**$6.56 \cdot 10^{3}$**	**1**
Pu-241	14.4	$1 \cdot 10^{2}$
Pu-242	$3.73 \cdot 10^{5}$	$3 \cdot 10^{-3}$
Am-241	$4.32 \cdot 10^{2}$	1
Am-242m	$1.41 \cdot 10^{2}$	$1 \cdot 10^{-2}$
Am-243	$7.37 \cdot 10^{3}$	$3 \cdot 10^{-2}$
Cm-243	29.1	$2 \cdot 10^{-2}$
Cm-244	18.1	3
Cm-245	$8.50 \cdot 10^{3}$	$3 \cdot 10^{-4}$
Cm-246	$4.73 \cdot 10^{3}$	$8 \cdot 10^{-5}$

To render the amount of C-14 no correlation factor has been used. In the future the annual amount of C-14 and Cl-36 deposited in SFR 1 will be correlated to produced electric energy. For further information, see Section 1.3.6.

As seen in Table 1-1 the correlation factor differs for Ni-59 and Ni-63 depending if the origin of the waste is BWR or PWR. For Ringhals the distribution of C-60 has been set to 65% originating from BWR and 35% originating from PWR. This distribution renders from a mean value of Co-60 distribution in ion-exchange resins at Ringhals between BWR and PWR between 1988 and 2005, see Figure 1-1. Each separate year might give rise to some questions, but the average over the years has been assumed to be true.

The value of Co-60TD is then used with the above mentioned distribution along with the respective correlation factors to calculate the nuclide content of the waste.

1.3.2 Calculation of nuclides correlated to Co-60 or Cs-137

Prosit calculates activities for nuclides correlated to Co-60 and Cs-137 according to the following method: In the Triumf waste database all activity data concerning measurements of Co-60 and Cs-137 is collected together with dates for waste production and activity measurements. This data is extracted by Prosit and using the measured activity and the dates, the activity is recalculated to the time of production of each of the waste packages (denominated Co-60TD and Cs-137TD).

For all waste packages already deposited in SFR 1, calculated values for Co-60TD and Cs-137TD are used to correlate values for all nuclides listed in Table 1-1 and 1-2. In this way a nuclide vector is created.

For all future waste packages prognosticated for SFR 1 an average value for each nuclide in the nuclide vector is estimated; for each waste type an average activity is calculated based on the waste packages already deposited. Specific activity data used for future waste packages is documented in the reference waste type descriptions in Appendices C–F.

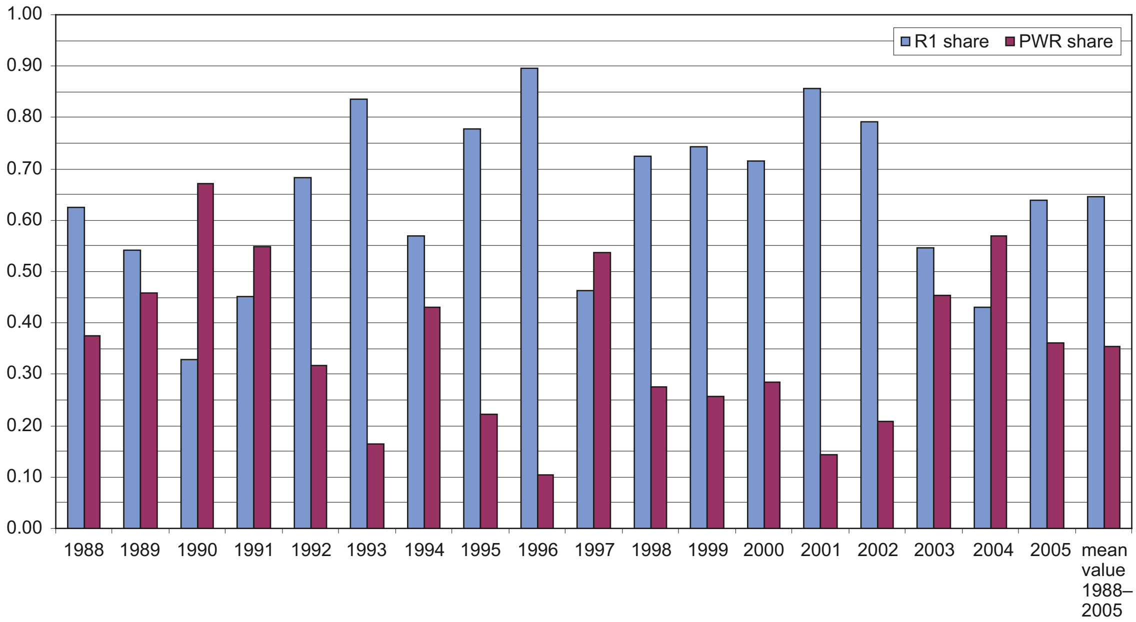

Figure 1-1. *Distribution of Co-60 in ion-exchange resins at Ringhals between BWR and PWR.*

The future activity is estimated using the prognosis for number of waste packages of each waste type and the waste type specific nuclide vector. By summing up the results for the waste packages already deposited and the activity for the future waste packages the total prognosticated inventory is calculated.

1.3.3 Calculation of nuclides correlated to Pu-239 and Pu-240

In some cases data for the transuranic nuclides are missing in the reference waste type description due to lack of data (all packages was produced before 1988 when transuranic data was of poor quality). For these nuclides the key nuclide is Pu-239 + Pu-240.

The method used to estimate TRU is based on measurements on reactor water and water in the fuel basins according to a method used by SKB and accepted by the authorities. The method assumes 100% efficiency in the ion-exchange resins and that all TRU is delivered to SFR the same year as it is produced.

The nuclear power plants report measured TRU to SKB every year and this data is stored in Prosit. For each reactor plutonium is measured in the reactor water and an average per year for the whole plant is calculated. An amount of Pu in the scrap metal and refuse is also calculated. In Prosit, TRU values for the period up to 2006 is calculated. The quality of data before 1988 has been judged to be too poor.

The data that is extractable from Prosit is plant specific, not waste type specific. The total TRU from each site is distributed on the waste types produced at that site using a method described below. To distribute the transuranic elements on each waste type the distribution of Co-60 in the different waste types is used. One could argue that Cs-137 should be a better correlation to Pu since it is a fission product compared to the activation product Co-60, but experience shows that cobalt is a better key-nuclide, probably due to transport properties.

The chain of calculation in Prosit is as follows:

1. The total amount of Co-60 activity in each waste type is calculated by multiplying the total number of waste packages for a specific waste type with the amount of Co-60 in the reference waste package.

2. The waste types were divided in types containing ion exchange resins and types containing scrap and refuse.
3. The percentage of Co-60 in each waste type, i.e. percentage regarding power plant and type of waste resins or refuse. This was done for both existing and prognosticated packages separately.
   Example for existing waste in Barsebäck: 100% of all scrap and refuse are in type B.12. Then all Co-60 activity in scrap is in the B.12. The ion-exchange resins are divided in type B.05, B.06; B.07 and B.20 with a Co-60 distribution of 6.9%, 78.8%, 14.5% and < 0.1% (= 100%).
4. The amount of Pu in resins and in refuse for each plant is distributed between the different types according to the percentages of Co-60.
   Example for existing waste in Barsebäck: Pu in scrap and refuse is all in B.12. The Pu distribution follows the percentages of Co-60, hence 6.9% in B.05, 78.8% in B.06 and so on.
5. The other nuclides correlated to Pu-239/240 are then calculated with correlation factors.

Waste packages from Studsvik must be calculated in a different way since no analysis of reactor water is possible. Studsvik has a percentage of the nuclide inventory in SFR 1 which is regulated through a contract. When the inventory is considered full regarding activity the quota that Studsvik has been awarded is put to the maximum. The activity inventory for a plant lifetime of 50 and 60 years respectively is based on calculations from Cs-137 for Studsvik.

Some exceptions exist;

- S.11; 50 m^3 of sludge and ion-exchange resins to be solidified, which has been thoroughly investigated regarding among other parameters transuranic activity. The measured Pu-activity has been used as "key-activity".
- S.24:01; one or two moulds containing smoke detectors. This waste contains only Am-241. The amount is estimated to $5 \cdot 10^{11}$ Bq.
- S.24:00; prognosticated with an estimated waste package.

1.3.4 Radioactive decay and summing up

In order to estimate the radionuclide inventory by the time of closure of the repository the different half-lives have to be regarded.

The waste packages can be divided into two kinds; already existing and future packages. For the ones that already exist, decay can easily be calculated since all necessary data is available (nuclide, half life ($t_{1/2}$), activity (A_0) at a specific time (T_0), and time of closure of the repository (T)). Decay takes place according to the well know formula:

$$A(T) = A_0 \cdot e^{-\lambda \cdot (T - T_0)} \ ; \ \lambda = \ln 2 / t_{1/2}$$

This formula is also used for waste packages still to be produced. T_0 is set by using the annual production rates specified in Appendix A. Each year a batch is produced, which then decays to the time of closure.

1.3.5 Inventory at the time of closure

The result of these calculations are the inventories for a lifetime of the nuclear power plants of 50 and 60 years. For all these options the inventory adds up to approximately $1.5 \cdot 10^{15}$ Bq.

In order to create a full inventory (with respect to nuclide content) the nuclide inventory from 50 years lifetime for the nuclear power plants was used as a base. This inventory was then up scaled to the limit of $1 \cdot 10^{16}$ Bq for SFR. This is a very rough estimate and does not take into account any nuclide decay after 2040. It does not either take into account the allowed nuclide inventory for each separate rock cavern, see Table 1-4.

Table 1-4. Allowed nuclide inventory according to the license for SFR 1.

Rock cavern	Activity (Bq)
Silo	$9.3 \cdot 10^{15}$
BTF	$1.4 \cdot 10^{14}$
BMA	$5.9 \cdot 10^{14}$
BLA	$1.2 \cdot 10^{13}$
Total	$1.0 \cdot 10^{16}$

1.3.6 Nuclides requiring alternative methods of quantification

Pu-239/240

Pu-239/240 are difficult to separate at measurements. Measurements from waste packages are therefore presented in Triumf sometimes as a summarised value for one of the nuclides. The distribution is according to experience 1/3 Pu-239 and 2/3 Pu-240. The nuclide inventory must hence be adjusted so that it concurs with reality.

C-14

In /Magnusson et al. 2007/ the amount of C-14 based on a life-time of the nuclear power plants of 40 years is assumed to be 5.0 TBq. In the report there is also an estimated annual production of each nuclear reactor.

The year 2007 has been used as a staring-point for the other prognoses. To render the amount of C-14 at 2007 the annual production of each nuclear reactor between the year 2007 and the year of closure for a 40 year life-time has been deducted from the assumed total amount of 5.0 TBq. The render the distribution between the different rock caverns the distribution of Co-60TD for 2007 has been used.

The render the amount for 2040 and 2050 the annual production for each nuclear power plant between 2007 and the year of closure for 50 and 60 years life-time respectively has been calculated. This addition has been distributed between the rock caverns according to the distribution of Co-60TD for 2007 until 2040 and 2050 respectively. Finally the two amounts (until 2007 and from 2007 and onwards) have been added to get the total amount of C-14 for the different rock caverns.

It is assumed that of the total amount of C-14, approximately 30% is organic C-14 /Magnusson et al. 2007/.

In the future the annual amount of C-14 deposited in SFR 1 will be correlated to produced electric energy.

Cl-36

The total amount of Cl-36 up to 2005 is estimated to 530 MBq /Johansson 2007/. The distribution between the rock caverns is assumed to follow the distribution of Co-60TD up to 2005. The amount of Cl-36 from 2006 and onward is calculated using Prosit and correlation against Co-60 according to Table 1-1.

In the future the annual amount of Cl-36 deposited in SFR 1 will be correlated to produced electric energy.

Ni-59, Ni-63

Ni-59 and Ni-63 are calculated according to Section 1.3.1.

Mo-93

The total amount of Mo-93 up to 2004 is estimated to 1,200 MBq /Johansson 2007/. The distribution between the rock caverns is assumed to follow the distribution of Co-60TD up to 2004. The amount of Mo-93 from 2005 and onward is calculated using Prosit and correlation against Co-60 according to Table 1-1.

Tc-99

The total amount of Tc-99 up to 2004 is estimated to 17,500 MBq /Johansson 2007/. The distribution between the rock caverns is assumed to follow the distribution of Co-60TD up to 2004. The amount of Tc-99 from 2005 and onward is calculated using Prosit and correlation against Co-60 and Cs-137 through the equation;

$$A_{Tc-99} = 3 \cdot 10^{-6} \times A_{Co-60} + 1 \cdot 10^{-4} \times A_{Cs-137}$$

I-129

The total amount of I-129 up to 2004 is estimated to 630 MBq /Johansson 2007/. The distribution between the rock caverns is assumed to follow the distribution of Co-60TD up to 2004. The amount of I-129 from 2005 and onward is calculated using Prosit and correlation against Cs-137 according to Table 1-2.

Cs-135

The total amount of Cs-135 up to 2004 is estimated to 3,800 MBq /Johansson 2007/. The distribution between the rock caverns is assumed to follow the distribution of Co-60TD up to 2004. The amount of Cs-135 from 2005 and onward is calculated using Prosit and correlation against Cs-137 according to Table 1-2.

1.3.7 Assumptions

To be able to calculate the nuclide inventory some assumptions have been made.

No limit on volume has been taken into consideration. It is assumed that should a cavern become full there is always a possibility to compress the waste. The exception is waste rendering from Studsvik where the prognosticated waste is somewhat limited for BLA to not deposit a greater volume than allowed.

Studsvik also has specific limits on nuclides in the different caverns. In the case where a specific nuclide is prognosticated to become higher than the limit, the maximum allowed amount is set on this specific nuclide. This leads to the case where some of the prognosticated waste will in reality not be able to be deposited since the allowed nuclides limits a specific waste type.

The prognosticated waste that is supposed to be put in BTF is not divided into 1BTF and 2BTF in Prosit, but all future waste is put in 1BTF. Since this is not going to be the case a rough estimation is made that all the waste that is already deposited in the different caverns are not moved. The waste that is supposed to be deposited is quite simply split between 1BTF and 2BTF. This leads to a somewhat distorted distribution between the two caverns, and in reality the distribution will be more even.

2 Nuclide inventory – NPP life-time 50 years

The inventory for a life-time of the nuclear power plants of 50 years has been calculated according to the methods described in Section 1.3. The result is an inventory that consists of approximately $1.5 \cdot 10^{15}$ Bq at the time of closure (31 December 2040). In Table 2-1 the inventory is presented for all the nuclides.

Table 2-1. Inventory 31 December 2040.

Nuclide	Total	Silo	BMA	1BTF	2BTF	BLA
Correlated to Co-60:						
Co-60	9.3E+13	8.5E+13	7.0E+12	2.9E+11	3.7E+11	3.6E+10
Cl-36	1.4E+09	1.1E+09	2.3E+08	1.1E+07	3.1E+07	1.0E+06
Ag-108m	1.2E+11	9.2E+10	2.0E+10	4.8E+08	2.3E+09	4.2E+08
Ba-133	3.1E+09	2.8E+09	2.8E+08	1.3E+07	2.8E+07	1.7E+06
Be-10	1.2E+06	9.7E+05	2.2E+05	5.1E+03	2.5E+04	8.1E+02
C-14 org.	1.8E+12	1.4E+12	3.2E+11	7.4E+09	5.5E+10	1.2E+09
C-14 inorg.	4.1E+12	3.2E+12	7.4E+11	1.7E+10	1.3E+11	2.7E+09
C-14 tot	5.9E+12	4.6E+12	1.1E+12	2.5E+10	1.8E+11	3.9E+09
Fe-55	2.1E+13	2.0E+13	1.5E+12	2.9E+10	2.9E+10	4.2E+09
H-3	3.9E+10	3.5E+10	3.8E+09	1.6E+08	4.0E+08	2.2E+07
Ho-166m	8.9E+09	7.0E+09	1.6E+09	5.8E+07	1.9E+08	1.7E+07
Mo-93	3.7E+09	2.9E+09	6.8E+08	4.4E+07	8.9E+07	2.0E+07
Nb-93m	5.5E+11	4.8E+11	5.9E+10	2.3E+09	6.6E+09	3.2E+08
Nb-94	2.0E+10	1.6E+10	3.6E+09	8.5E+07	4.2E+08	1.3E+07
Ni-59	9.5E+12	7.3E+12	2.1E+12	2.1E+10	4.2E+10	5.3E+09
Ni-63	1.2E+15	8.9E+14	2.6E+14	2.3E+12	2.8E+12	6.6E+11
Sb-125	2.3E+12	2.1E+12	1.5E+11	6.1E+09	6.2E+09	3.1E+08
Zr-93	2.3E+09	1.8E+09	4.2E+08	1.5E+07	4.8E+07	4.3E+06
Correlated to Cs-137:						
Cs-137	1.3E+14	1.1E+14	2.0E+13	6.7E+11	1.8E+12	4.8E+10
Cd-113m	3.5E+10	3.1E+10	3.7E+09	1.8E+08	3.4E+08	1.1E+07
Cs-134	9.6E+11	8.9E+11	7.0E+10	6.1E+08	6.1E+08	2.5E+07
Cs-135	5.1E+09	4.0E+09	1.0E+09	1.6E+07	1.1E+08	2.0E+06
Eu-152	5.4E+09	3.6E+09	4.0E+08	1.5E+08	1.1E+08	1.1E+09
Eu-154	2.8E+12	2.5E+12	2.3E+11	1.2E+10	1.7E+10	7.5E+08
Eu-155	6.0E+11	5.5E+11	4.7E+10	1.7E+09	1.8E+09	8.1E+07
I-129	1.0E+09	8.1E+08	1.8E+08	3.1E+06	1.9E+07	3.1E+05
Pd-107	3.2E+08	2.6E+08	5.4E+07	1.5E+06	5.9E+06	1.2E+05
Pm-147	1.9E+12	1.7E+12	1.6E+11	2.1E+09	2.1E+09	1.0E+08
Ru-106	1.6E+09	1.4E+09	1.2E+08	2.6E+06	2.6E+06	3.1E+03
Se-79	1.3E+09	1.0E+09	2.1E+08	5.9E+06	2.4E+07	4.9E+05
Sm-151	7.0E+11	5.7E+11	1.1E+11	3.4E+09	1.2E+10	2.7E+08
Sn-126	1.6E+08	1.3E+08	2.7E+07	7.4E+05	2.9E+06	6.2E+04
Sr-90	1.3E+13	1.1E+13	1.7E+12	6.4E+10	2.1E+11	4.7E+09
Tc-99*	4.1E+11	3.6E+11	3.7E+10	7.0E+09	7.6E+09	5.0E+08

Nuclide	Total	Silo	BMA	1BTF	2BTF	BLA
Correlated to Pu-239 + Pu-240:						
Pu-239	1.1E+10	8.9E+09	2.0E+09	1.5E+08	2.0E+08	1.3E+07
Pu-240	2.3E+10	1.8E+10	4.0E+09	3.0E+08	4.0E+08	2.6E+07
U-232	7.5E+05	5.7E+05	1.5E+05	9.6E+03	1.1E+04	6.0E+02
U-234	3.5E+07	2.6E+07	7.7E+06	5.0E+05	5.4E+05	3.0E+04
U-235	4.7E+08	1.4E+07	2.8E+06	1.9E+07	7.5E+05	4.3E+08
U-236	1.4E+07	1.0E+07	3.0E+06	1.9E+05	3.7E+05	9.0E+03
U-238	1.5E+09	2.8E+07	6.2E+06	3.3E+05	8.8E+05	1.4E+09
Np-237	1.6E+08	1.3E+08	2.5E+07	4.1E+05	1.9E+06	2.7E+04
Pu-238	4.9E+10	3.8E+10	9.6E+09	3.5E+08	6.0E+08	4.7E+07
Pu-241	8.2E+11	6.6E+11	1.4E+11	7.5E+09	8.6E+09	4.8E+08
Pu-242	1.1E+08	7.9E+07	2.3E+07	1.5E+06	1.6E+06	9.0E+04
Am-241	5.0E+11	4.9E+11	6.9E+09	8.4E+08	4.9E+08	5.0E+07
Am-242m	3.0E+08	2.2E+08	6.2E+07	4.0E+06	4.4E+06	2.4E+05
Am-243	1.1E+09	8.4E+08	2.3E+08	1.5E+07	1.6E+07	9.0E+05
Cm-243	2.9E+08	2.3E+08	5.5E+07	3.0E+06	2.0E+06	2.3E+05
Cm-244	1.0E+10	8.5E+09	1.4E+09	1.0E+08	1.1E+08	2.2E+07
Cm-245	1.0E+07	7.9E+06	2.3E+06	1.5E+05	1.6E+05	9.0E+03
Cm-246	2.8E+06	2.1E+06	6.1E+05	4.0E+04	4.3E+04	2.4E+03
Total	1.4E+15	1.2E+15	3.0E+14	3.5E+12	5.5E+12	7.8E+11

* Tc-99 has in effect been correlated towards both Co-60 and Cs-137.

3 Nuclide inventory – NPP life-time 60 years

The inventory for a life-time of the nuclear power plants of 60 years has been calculated according to the methods described in Section 1.3. The result is an inventory that consists of approximately $1.7 \cdot 10^{15}$ Bq at the time of closure (31 December 2050). In Table 3-1 the inventory is presented for all the nuclides.

Table 3-1. Inventory 31 December 2050.

Nuclide	Total	Silo	BMA	1BTF	2BTF	BLA
Correlated to Co-60:						
Co-60	9.3E+13	8.5E+13	6.8E+12	2.9E+11	3.1E+11	3.1E+10
Cl-36	1.6E+09	1.3E+09	2.5E+08	1.2E+07	3.2E+07	1.2E+06
Ag-108m	1.4E+11	1.1E+11	2.1E+10	6.0E+08	2.4E+09	4.2E+08
Ba-133	3.2E+09	2.9E+09	2.6E+08	1.3E+07	2.1E+07	1.5E+06
Be-10	1.4E+06	1.2E+06	2.3E+05	6.4E+03	2.6E+04	9.2E+02
C-14 org.	2.1E+12	1.7E+12	3.4E+11	9.5E+09	5.8E+10	1.4E+09
C-14 inorg.	5.0E+12	4.0E+12	8.0E+11	2.2E+10	1.3E+11	3.3E+09
C-14 tot	7.1E+12	5.8E+12	1.1E+12	3.2E+10	1.9E+11	4.7E+09
Fe-55	2.1E+13	2.0E+13	1.5E+12	2.9E+10	2.9E+10	4.0E+09
H-3	4.0E+10	3.6E+10	3.4E+09	1.7E+08	3.0E+08	1.9E+07
Ho-166m	1.0E+10	8.4E+09	1.8E+09	7.1E+07	2.0E+08	2.0E+07
Mo-93	4.2E+09	3.2E+09	7.5E+08	5.2E+07	9.7E+07	2.7E+07
Nb-93m	5.7E+11	5.0E+11	5.3E+10	2.5E+09	5.3E+09	3.0E+08
Nb-94	2.4E+10	2.0E+10	3.9E+09	1.1E+08	4.4E+08	1.5E+07
Ni-59	1.1E+13	9.0E+12	2.1E+12	2.3E+10	4.4E+10	6.1E+09
Ni-63	1.4E+15	1.1E+15	2.6E+14	2.4E+12	2.8E+12	7.7E+11
Sb-125	2.3E+12	2.1E+12	1.5E+11	6.1E+09	6.1E+09	2.9E+08
Zr-93	2.7E+09	2.2E+09	4.5E+08	1.8E+07	5.1E+07	5.1E+06
Correlated to Cs-137:						
Cs-137	1.4E+14	1.2E+14	1.9E+13	6.6E+11	1.6E+12	4.1E+10
Cd-113m	3.5E+10	3.2E+10	3.2E+09	1.6E+08	2.6E+08	7.3E+06
Cs-134	9.7E+11	8.9E+11	7.0E+10	6.1E+08	6.1E+08	1.8E+07
Cs-135	5.6E+09	4.4E+09	1.1E+09	1.8E+07	1.1E+08	2.1E+06
Eu-152	4.9E+09	3.7E+09	3.4E+08	1.3E+08	1.0E+08	6.5E+08
Eu-154	2.7E+12	2.5E+12	2.1E+11	1.0E+10	1.2E+10	4.4E+08
Eu-155	6.0E+11	5.5E+11	4.6E+10	1.4E+09	1.5E+09	4.6E+07
I-129	1.1E+09	9.3E+08	1.9E+08	3.6E+06	1.9E+07	3.3E+05
Pd-107	3.7E+08	3.0E+08	5.6E+07	1.7E+06	6.1E+06	1.3E+05
Pm-147	1.9E+12	1.7E+12	1.6E+11	2.1E+09	2.1E+09	7.4E+07
Ru-106	1.6E+09	1.4E+09	1.2E+08	2.6E+06	2.6E+06	3.1E+03
Se-79	1.5E+09	1.2E+09	2.3E+08	6.7E+06	2.4E+07	5.1E+05
Sm-151	7.7E+11	6.5E+11	1.1E+11	3.7E+09	1.1E+10	2.6E+08
Sn-126	1.8E+08	1.5E+08	2.8E+07	8.4E+05	3.0E+06	6.3E+04
Sr-90	1.3E+13	1.2E+13	1.5E+12	6.4E+10	1.8E+11	3.9E+09
Tc-99*	5.2E+11	4.6E+11	4.4E+10	7.4E+09	8.1E+09	6.4E+08

Nuclide	Total	Silo	BMA	1BTF	2BTF	BLA
Correlated to Pu-239 + Pu-240:						
Pu-239	1.3E+10	1.1E+10	2.0E+09	1.6E+08	2.1E+08	1.3E+07
Pu-240	2.7E+10	2.2E+10	4.1E+09	3.1E+08	4.1E+08	2.7E+07
U-232	8.4E+05	6.7E+05	1.5E+05	9.6E+03	1.0E+04	5.7E+02
U-234	4.1E+07	3.2E+07	8.3E+06	5.3E+05	5.7E+05	3.1E+04
U-235	4.7E+08	1.7E+07	3.2E+06	1.9E+07	8.5E+05	4.3E+08
U-236	1.6E+07	1.2E+07	3.2E+06	2.1E+05	3.9E+05	9.4E+03
U-238	1.5E+09	3.4E+07	6.8E+06	3.9E+05	9.4E+05	1.4E+09
Np-237	1.9E+08	1.6E+08	2.9E+07	4.7E+05	2.0E+06	2.8E+04
Pu-238	5.5E+10	4.5E+10	9.5E+09	3.7E+08	6.0E+08	4.6E+07
Pu-241	8.3E+11	6.8E+11	1.3E+11	6.4E+09	7.0E+09	3.6E+08
Pu-242	1.2E+08	9.6E+07	2.5E+07	1.6E+06	1.7E+06	9.4E+04
Am-241	4.9E+11	4.8E+11	7.7E+09	8.6E+08	5.1E+08	5.3E+07
Am-242m	3.4E+08	2.7E+08	6.5E+07	4.1E+06	4.5E+06	2.4E+05
Am-243	1.3E+09	1.0E+09	2.5E+08	1.6E+07	1.7E+07	9.3E+05
Cm-243	3.1E+08	2.5E+08	5.3E+07	2.5E+06	1.8E+06	2.0E+05
Cm-244	1.1E+10	9.2E+09	1.2E+09	8.3E+07	9.0E+07	1.9E+07
Cm-245	1.2E+07	9.6E+06	2.5E+06	1.6E+05	1.7E+05	9.3E+03
Cm-246	3.3E+06	2.5E+06	6.6E+05	4.2E+04	4.5E+04	2.5E+03
Total	1.7E+15	1.4E+15	3.0E+14	3.6E+12	5.1E+12	8.8E+11

* Tc-99 has in effect been correlated towards both Co-60 and Cs-137.

4 Full nuclide inventory

The full nuclide inventory is calculated according to Section 1.3.5. The result is an inventory that consists of $1 \cdot 10^{16}$ Bq at the time of closure at 2040 but with a theoretical continued waste production with no decay taken into account. In Table 4-1 the inventory is presented for all the nuclides. No consideration is taken to the limit in Table 1-4.

Table 4-1. Full inventory 2040 for SFR 1 regarding activity.

Nuclide	Total	Silo	BMA	1BTF	2BTF	BLA
Correlated to Co-60:						
Co-60	6.5E+14	5.9E+14	4.9E+13	2.0E+12	2.6E+12	2.5E+11
Cl-36	9.4E+09	7.5E+09	1.6E+09	7.3E+07	2.1E+08	7.2E+06
Ag-108m	8.0E+11	6.4E+11	1.4E+11	3.4E+09	1.6E+10	2.9E+09
Ba-133	2.2E+10	2.0E+10	2.0E+09	8.8E+07	1.9E+08	1.2E+07
Be-10	8.4E+06	6.7E+06	1.5E+06	3.5E+04	1.7E+05	5.6E+03
C-14 org.	1.2E+13	9.6E+12	2.2E+12	5.1E+10	3.8E+11	8.0E+09
C-14 inorg.	2.9E+13	2.2E+13	5.2E+12	1.2E+11	9.0E+11	1.9E+10
C-14 tot	4.1E+13	3.2E+13	7.4E+12	1.7E+11	1.3E+12	2.7E+10
Fe-55	1.5E+14	1.4E+14	1.0E+13	2.0E+11	2.0E+11	2.9E+10
H-3	2.7E+11	2.4E+11	2.6E+10	1.1E+09	2.8E+09	1.5E+08
Ho-166m	6.2E+10	4.9E+10	1.1E+10	4.1E+08	1.3E+09	1.2E+08
Mo-93	2.6E+10	2.0E+10	4.7E+09	3.1E+08	6.2E+08	1.4E+08
Nb-93m	3.8E+12	3.3E+12	4.1E+11	1.6E+10	4.6E+10	2.2E+09
Nb-94	1.4E+11	1.1E+11	2.5E+10	5.9E+08	2.9E+09	9.2E+07
Ni-59	6.6E+13	5.1E+13	1.4E+13	1.4E+11	2.9E+11	3.7E+10
Ni-63	8.0E+15	6.2E+15	1.8E+15	1.6E+13	1.9E+13	4.6E+12
Sb-125	1.6E+13	1.5E+13	1.1E+12	4.2E+10	4.3E+10	2.1E+09
Zr-93	1.6E+10	1.2E+10	2.9E+09	1.0E+08	3.3E+08	3.0E+07
Correlated to Cs-137:						
Cs-137	9.4E+14	7.8E+14	1.4E+14	4.6E+12	1.3E+13	3.4E+11
Cd-113m	2.4E+11	2.1E+11	2.5E+10	1.2E+09	2.3E+09	7.4E+07
Cs-134	6.7E+12	6.2E+12	4.8E+11	4.3E+09	4.3E+09	1.7E+08
Cs-135	3.6E+10	2.8E+10	7.2E+09	1.1E+08	7.6E+08	1.4E+07
Eu-152	3.7E+10	2.5E+10	2.8E+09	1.1E+09	7.4E+08	7.5E+09
Eu-154	1.9E+13	1.7E+13	1.6E+12	8.3E+10	1.2E+11	5.2E+09
Eu-155	4.2E+12	3.8E+12	3.3E+11	1.2E+10	1.2E+10	5.6E+08
I-129	7.0E+09	5.6E+09	1.3E+09	2.1E+07	1.3E+08	2.2E+06
Pd-107	2.2E+09	1.8E+09	3.7E+08	1.0E+07	4.1E+07	8.6E+05
Pm-147	1.3E+13	1.2E+13	1.1E+12	1.5E+10	1.5E+10	7.0E+08
Ru-106	1.1E+10	1.0E+10	8.1E+08	1.8E+07	1.8E+07	2.2E+04
Se-79	8.8E+09	7.1E+09	1.5E+09	4.1E+07	1.6E+08	3.4E+06
Sm-151	4.9E+12	4.0E+12	7.6E+11	2.3E+10	8.2E+10	1.9E+09
Sn-126	1.1E+09	8.9E+08	1.9E+08	5.2E+06	2.0E+07	4.3E+05
Sr-90	8.9E+13	7.5E+13	1.2E+13	4.5E+11	1.4E+12	3.3E+10
Tc-99*	2.9E+12	2.5E+12	2.6E+11	4.8E+10	5.3E+10	3.5E+09

Nuclide	Total	Silo	BMA	1BTF	2BTF	BLA
Correlated to Pu-239 + Pu-240:						
Pu-239	7.8E+10	6.2E+10	1.4E+10	1.0E+09	1.4E+09	9.1E+07
Pu-240	1.6E+11	1.2E+11	2.7E+10	2.1E+09	2.8E+09	1.8E+08
U-232	5.2E+06	4.0E+06	1.0E+06	6.7E+04	7.4E+04	4.1E+03
U-234	2.4E+08	1.8E+08	5.3E+07	3.5E+06	3.7E+06	2.1E+05
U-235	3.2E+09	9.8E+07	2.0E+07	1.3E+08	5.2E+06	3.0E+09
U-236	9.7E+07	7.2E+07	2.1E+07	1.3E+06	2.6E+06	6.2E+04
U-238	1.0E+10	1.9E+08	4.3E+07	2.3E+06	6.1E+06	9.9E+09
Np-237	1.1E+09	9.0E+08	1.7E+08	2.8E+06	1.3E+07	1.9E+05
Pu-238	3.4E+11	2.6E+11	6.7E+10	2.4E+09	4.2E+09	3.3E+08
Pu-241	5.7E+12	4.6E+12	9.7E+11	5.2E+10	6.0E+10	3.4E+09
Pu-242	7.3E+08	5.5E+08	1.6E+08	1.0E+07	1.1E+07	6.2E+05
Am-241	3.5E+12	3.4E+12	4.8E+10	5.8E+09	3.4E+09	3.5E+08
Am-242m	2.1E+09	1.6E+09	4.3E+08	2.8E+07	3.0E+07	1.7E+06
Am-243	7.6E+09	5.8E+09	1.6E+09	1.0E+08	1.1E+08	6.2E+06
Cm-243	2.0E+09	1.6E+09	3.8E+08	2.1E+07	1.4E+07	1.6E+06
Cm-244	7.0E+10	5.9E+10	9.4E+09	7.1E+08	7.8E+08	1.5E+08
Cm-245	7.3E+07	5.5E+07	1.6E+07	1.0E+06	1.1E+06	6.2E+04
Cm-246	1.9E+07	1.5E+07	4.2E+06	2.7E+05	3.0E+05	1.7E+04
Total	1.0E+16	7.9E+15	2.0E+15	2.4E+13	3.8E+13	5.3E+12

* Tc-99 has in effect been correlated towards both Co-60 and Cs-137.

5 Uncertainties

The one uncertainty that overshadows all others is the concept of correlation factors.

Large uncertainties are involved in the use of general correlation factors to estimate the activity content in specific waste types. The correlation factors evaluated for specific radionuclides vary within orders of magnitude, which gives an indication of the uncertainty interval. In addition, no statistical evaluations have been made in this study to prove that correlations between the key nuclides (Co-60, Cs-137 and Pu-239+240) and the other nuclides exist.

The outcome of the safety assessment will, however, give indications on which radionuclides are of major concern to reduce the uncertainties in the reference inventory.

Another important uncertainty is related to the estimated amount of future waste. The life-span of the reactors is unknown and thus the amount of waste.

6 References

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Appendix C

Waste types in BLA

1 General

1.1 Waste in BLA

In SFR in the BLA – rock cavern for low-level waste – short-lived waste from the nuclear power plants and the Studsvik research site are disposed off. The BLA cavern is approximately 160 m long, 15 m wide and has a height of 12.5 m. The cavern is very simple to its construction, basically there is only a concrete floor on which containers are placed. During the operational phase a ceiling has been placed over the waste in order to minimise water dripping on to the waste. This roof will be dismantled when the repository is closed.

The waste materials are metal scrap (iron/steel, aluminium); cellulose (e.g. wood, textile, paper), other organic materials (e.g. plastics, cables) and other waste like insulation (e.g. rock wool).

The dose rate allowed on the waste packages is maximum 2 mSv/h at the surface of the packages. The amounts of radionuclides are low, dominating nuclide are Co-60. Other criteria for acceptance are:

- Construction, geometry, dimensions and weight must be suited to the handling system in SFR and the transportation system in general and in BLA in particular. The strength of the packages must be sufficient to withstand normal and abnormal handling.
- Each package must have an individual identity and the radionuclide content of gamma emitting nuclides shall be known.
- Surface contamination should not exceed 40 kBq/m^2 for gamma and beta emitting nuclides and 4 kBq/m^2 for alpha emitting nuclides.
- The internal dose rates and integrated dose must not affect the barrier properties in the repository.
- The chemical and physical properties and the structure of the waste should be known.
- Chemical aggressive or explosive material, pressurised gas and free liquid are not allowed.
- Gas production from the waste must not be too big.
- The waste shall have enough resistance towards corrosion that the integrity of the package is kept until sealing of the repository.
- Burnable waste must not be subject to self-ignition.
- Complexing agents should if possible be avoided.

All waste is normally packed in ISO containers. Inside the ISO container the waste could also have inner packages e.g. steel drums, bales, boxes etc. Some of the waste has been compacted.

1.2 Definitions

Surface area is defined as the area subject to anaerobic corrosion and gas production after sealing of the repository.

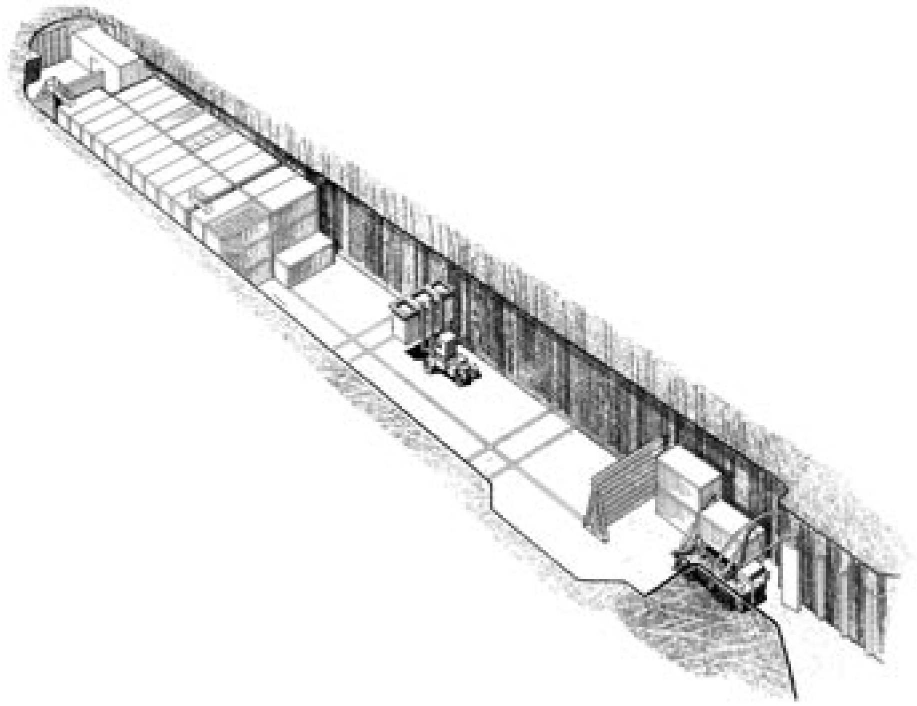

Figure 1-1. BLA.

1.3 Uncertainties – general

- The presented waste types describe the types that have been, are or can be foreseen to be produced in Sweden. New waste types may be present in the future.
- Almost all data is based on literature values, smaller changes may not been properly documented.
- All future production is based on a relative simple prognosis model. This model, described in Appendix A, was chosen since the uncertainties about processes etc are large.
- No regard has been taken to improvements in processes etc.
- A reference package has been chosen based on best estimates. Actual packages may differ considerably from this reference package, the uncertainty for the whole population of packages is judged to be smaller.
- Big components as heat exchangers etc could in the future be of interest to dispose off in BLA. See Chapter 10 in this appendix.
- For the radionuclide inventory for each waste package the amount presented is either based on the amount received from correlation factors or actual measurements when applicable. When there is a difference between different sub types of waste packages within one waste type the inventory presented in this appendix is the most conservative one.
- Void presented in this report is based on best estimates, the estimates can be quite crude.
- Tc-99 is correlated against both Co-60 and Cs-137 when quantifying the inventories, according to Appendix B. In the tables "Radionuclide composition of a reference waste package" however Tc-99 is presented as only correlated towards Cs-137.

2 B.12

2.1 Waste type description

2.1.1 Waste package

The B.12 waste type consists of a standard ISO-container containing iron/steel, aluminium, cellulose and other organic materials generated at the Barsebäck nuclear power plant. The material is classified as low level waste.

The physical properties and chemical conditions are well known. The following restrictions are always applicable:

- The general restrictions listed in Section 1.1.
- Since the package contains burnable substances the restrictions regarding self-ignition etc are of special importance.
- The contents of toxic material should fall within the limits for SFR 1. That means avoidance of toxic material. Small amounts of lead and PVC could follow this waste stream.

Packaging

The ISO container is made of steel with the length of 6.1 m, width of 2.4 m and height of 1.3 m. The container is often referred to as '20-foot half-height container'. The weight of steel is approximately 1,900 kg and the surface area of the container is approximately 104 m^2. The floor of the container could be of plywood or steel. The plywood floor weighs approximately 310 kg. The variation of weight between containers is between 1,800–2,300 kg.

Other containers are possible, 10-foot full- or half-height and 20-foot full-height could be used.

Treatment and conditioning

The raw waste is collected from controlled areas from various places in the power plant. Some coarse sorting is made at each collection point before transported to central treatment. The waste is normally packed in plastic bags or steel drums and then placed in the ISO container. A few containers include super compacted drums.

The void varies but is estimated to 7.5 m^3 in a container.

2.1.2 Materials – chemical composition

The amounts of different materials in a B.12 container are given in Table 2-1. Data comes from /Berntsson 1990/ and (Jan-Eric Jönsson, Barsebäck Kraft, 2000, pers. comm.). The mix of different waste materials has changed from time to time depending if incineration is used, on different maintenance work or other reasons.

The weight of the raw waste material is normally 10,000 kg per package including the weight of the container /Johansson 2000/. The weight on the waste material can vary from approximately 5,000–15,000 kg depending on different waste materials.

Table 2-1. Amount of different material in a typical package B.12.

Material	Weight (kg)	Area (m²)	Weight (%)*
Iron/steel	0–10,000	0–508	0–100
Aluminium	0–200	0–30	0–2
Cellulose (including wood, paper, textiles, absorbed water)	0–5,000		0–50
Other organic material (including plastics, rubber, cable)	0–3,500		0–35
Other material (including insulation like mineral wool etc)	0–2,500		0–25

* Upper limit calculated as per cent of the ratio between mass of material and 10,000 kg.

2.1.3 Radionuclide inventory

Before the waste is transported from Barsebäck NPP to SFR 1 a measurement of gamma emitting nuclides is performed. Dominating nuclides are Co-60 and Cs-137. The amounts of these nuclides are between 0 and $5.4 \cdot 10^9$ Bq for Co-60 and between 0 and $3.2 \cdot 10^8$ Bq for Cs-137. Other nuclides than gamma emitting ones must be calculated according to Appendix B.

The dose rate limit for this package is 2 mSv/h but normally it is between 0–1.5 mSv/h on 1 m. The surface contamination should not exceed 40 kBq/m² for gamma and beta and 4 kBq/m² for alpha emitting nuclides. The waste packages are usually free of contamination.

2.1.4 Waste production

The waste type was in production from 1991. No more production is foreseen, see Appendix A.

2.2 Reference waste type description

2.2.1 Waste package and material

Table 2-2 shows the content of different materials and steel surface areas of the packaging in typical package from Barsebäck.

The assumed composition of the waste in a reference waste package of B.12 is given in Table 2-3. The surface area on a metal components in the waste are estimated assuming planar plates with a thickness of 5 mm, and assuming a density of 7,860 kg/m³ for carbon and stainless steel and 2,700 kg/m³ for aluminium.

The estimated volumes of the waste, the packaging materials and the void volume inside the package for a reference package are given in Table 2-5. The waste volume is estimated from the weights of the waste components in Table 2-3 and approximate densities of the different material. Average void is estimated to 50% in a package.

Void and waste volume is specified in Table 2-4.

Table 2-2. Estimated reference packaging composition and surface area and thickness of components in an ISO container of B.12.

Component	Material	Weight (kg)	Surface area (m²)	Thickness (mm)
ISO container	Carbon steel	1,900	104	1.5
Plywood	Cellulose	310		

Table 2-3. Estimated reference waste composition and surface area and thickness of components in one B.12.

Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Iron/steel	4,500	229	5
Aluminium	100	15	5
Cellulose (including wood, paper, textiles, absorbed water)	500 (2,000)		
Other organic material (including plastics, rubber, cable)	3,000		
Other material (including insulation like mineral wool etc)	400		

Table 2-4. Volumes in one waste packages of B.12.

	Volume in one package (m^3)
Waste	11.5
Void	7.5

2.2.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 2-5.

A reference waste package of the type B.12 has a surface dose rate of 2 mSv/h and no surface contamination.

Table 2-5. Radionuclide composition of a reference waste package of the type B.12.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	9.7E+08	Pu-239	1.8E+01
Cl-36	5.8E+02	Pu-240	3.5E+01
Ag-108m	5.8E+04	U-232	1.6E–03
Ba-133	9.7E+03	U-234	5.3E–02
Be-10	5.8E-01	U-235	1.1E–03
C-14	2.9E+06	U-236	2.9E–03
Fe-55	9.7E+08	U-238	2.1E–02
H-3	9.7E+04	Np-237	2.1E–02
Ho-166m	3.9E+03	Pu-238	4.0E+01
Mo-93	9.7E+02	Pu-241	5.3E+03
Nb-93m	9.7E+05	Pu-242	1.6E–01
Nb-94	9.7E+03	Am-241	2.1E+01
Ni-59	9.7E+05	Am-242m	5.3E–01
Ni-63	7.8E+07	Am-243	1.6E+00
Sb-125	2.6E+07	Cm-243	1.1E+00
Zr-93	9.7E+02	Cm-244	2.6E+01
		Cm-245	1.6E–02

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Cs-137		Cm-246	4.2E–03
Cs-137	3.0E+07		
Cd-113m	1.8E+04		
Cs-134	2.6E+07		
Cs-135	3.0E+02		
Eu-152	2.1E+03		
Eu-154	3.0E+06		
Eu-155	2.1E+06		
I-129	9.1E+01		
Pd-107	3.0E+01		
Pm-147	2.7E+07		
Ru-106	1.5E+05		
Se-79	1.2E+02		
Sm-151	9.1E+04		
Sn-126	1.5E+01		
Sr-90	3.0E+06		
Tc-99	2.7E+04		

2.3 Uncertainties

- General uncertainties are described in Section 1.3.

3 B.20

3.1 Waste type description

3.1.1 Waste package

The B.20 waste type consists of a standard ISO-container containing 200 l drums of steel with bitumen conditioned ion-exchange resins generated at the Barsebäck nuclear power plant. Each container includes 36 drums. The material is classified as low level waste.

The physical properties and chemical conditions are well known. The following restrictions are always applicable:

- The general restrictions listed in Section 1.1.
- Since the package contains burnable substances the restrictions regarding self-ignition etc are of special importance.

Packaging

The ISO container is made of steel with the length of 6.1 m, width of 2.4 m and height of 1.3 m. The container is often referred to as '20-foot half-height container'. The weight of steel is approximately 1,900 kg and the surface area of the container is approximately 104 m^2. The floor of the container could be of plywood or steel. The plywood floor weighs approximately 310 kg. The variation of weight between containers is between 1,800–2,300 kg.

The drums are exactly like waste type B.05 (see Appendix E) or B.06 (see Appendix F) except the amount of radionuclides and dose rates. For all details of the drums about composition, treatment etc see the Appendices E and F. The steel weight of 36 drums is 828 kg.

Treatment and conditioning

36 drums are placed in a container.

The void varies but is estimated to 7.4 m^3 in a container.

3.1.2 Materials – chemical composition

The amounts of different materials in a B.20 container are given in Table 3-1. Data comes from /Berntsson 1991/.

The weight of a container and waste is between 9,600 kg and 9,800 kg /Johansson 2000/. The weight on the waste material can vary but weighs approximately 200 kg per drum.

Table 3-1. Amount of different materials in a typical package B.20.

Material	Weight (kg)	Weight (%)
Ion exchange resin	1,800	25
Bitumen	5,400	75

3.1.3 Radionuclide inventory

Before the waste is transported from Barsebäck NPP to SFR 1 a measurement of gamma emitting nuclides is performed. Dominating nuclides are Co-60 and Cs-137. The amounts of these nuclides are between 0 and $6.3 \cdot 10^6$ Bq for Co-60 and between 0 and $1.5 \cdot 10^4$ Bq for Cs-137. Other nuclides than gamma emitting ones must be calculated according to Appendix B.

The dose rate limit for this package is 2 mSv/h but normally it is between 0–0.4 mSv/h on 1 m. The surface contamination should not exceed 40 kBq/m^2 for gamma and beta and 4 kBq/m^2 for alpha emitting nuclides. The waste packages are usually free of contamination.

3.1.4 Waste production

The waste type was in production until 1985. The number of packages produced was 12. No more production is foreseen, see Appendix A.

3.2 Reference waste type description

3.2.1 Waste package and material

Table 3-2 shows the content of different materials and steel surface areas of the packaging in typical package from Barsebäck.

The assumed composition of the waste in a reference waste package of B.20 is given in Table 3-3.

The estimated volumes of the waste, the packaging materials and the void volume inside the package for a reference package are given in Table 3-4. The waste volume is estimated from the weights of the waste components in Table 3-3 and approximate densities of the different material. Average void is estimated to 45% in a package.

Void and waste volume is specified in Table 3-4.

Table 3-2. Estimated reference packaging composition and surface area and thickness of components in an ISO container of B.20.

Component	Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
ISO container	Carbon steel	1,900	104	1.5
Plywood	Cellulose	310		
Steel drums (36)	Carbon steel	828	109	1.25

Table 3-3. Estimated reference waste composition and surface area and thickness of components in one B.20.

Material	Weight (kg)
Ion-exchange resin	1,800
Bitumen	5,400

Table 3-4. Volumes in one waste packages of B.20 (including drums).

	Volume in one package (m^3)
Waste	11.7
Void	7.4

3.2.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 3-5.

A reference waste package of the type B.20 has a surface dose rate of 2 mSv/h and no surface contamination.

Table 3-5. Radionuclide composition of a reference waste package of the type B.20.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	2.6E+06	Pu-239	4.5E–01
Cl-36	1.6E+00	Pu-240	8.9E–01
Ag-108m	1.6E+02	U-232	4.0E–05
Ba-133	2.6E+01	U-234	1.3E–03
Be-10	1.6E-03	U-235	4.3E–02
C-14	7.9E+03	U-236	3.7E–02
Fe-55	2.6E+06	U-238	3.2E–02
H-3	2.6E+02	Np-237	2.2E–01
Ho-166m	1.1E+01	Pu-238	1.4E+00
Mo-93	2.6E+00	Pu-241	1.3E+02
Nb-93m	2.6E+03	Pu-242	4.0E–03
Nb-94	2.6E+01	Am-241	1.0E+00
Ni-59	2.6E+03	Am-242m	1.3E–02
Ni-63	2.1E+05	Am-243	4.0E–02
Sb-125	2.8E+04	Cm-243	2.7E–02
Zr-93	2.6E+00	Cm-244	8.2E–01
		Cm-245	4.0E–04
Correlated to Cs-137		Cm-246	1.1E–04
Cs-137	4.9E+03		
Cd-113m	3.0E+00		
Cs-134	4.9E+03		
Cs-135	4.9E–02		
Eu-152	3.5E–01		
Eu-154	4.9E+02		
Eu-155	3.5E+02		
I-129	1.5E–02		
Pd-107	4.9E–03		
Pm-147	4.4E+03		
Ru-106	2.5E+01		
Se-79	2.0E–02		
Sm-151	1.5E+01		
Sn-126	2.5E–03		
Sr-90	4.9E+02		
Tc-99	4.4E+00		

3.3 Uncertainties

- General uncertainties are described in Section 1.3.

4 F.12

4.1 Waste type description

4.1.1 Waste package

The F.12 waste type consists of a standard ISO-container containing iron/steel, aluminium, cellulose and other organic materials generated at the Forsmark nuclear power plant. The material is classified as low level waste.

The physical properties and chemical conditions are well known. The following restrictions are always applicable:

- The general restrictions listed in Section 1.1.
- Since the package contains burnable substances the restrictions regarding self-ignition etc are of special importance.
- The contents of toxic material should fall within the limits set up earlier for SFR 1. That means avoidance of toxic material. Small amounts of lead and PVC could follow this waste stream.

Packaging

The ISO container is made of steel with the length of 6.1 m, width of 2.4 m and height of 1.3 m. The container is often referred to as '20-foot half-height container'. The weight of steel is approximately 1,900 kg and the surface area of the container is approximately 104 m^2. The floor of the container could be of plywood or steel. The plywood floor weighs approximately 310 kg. The variation of weight between containers is between 1,800–2,300 kg.

Other containers are possible, 10-foot full- or half-height and 20-foot full-height could be used.

Treatment and conditioning

The raw waste is collected from controlled areas from various places in the power plant. Some coarse sorting is made at each collection point before transported to central treatment.

The waste is normally packed in plastic bags or steel drums and then placed in the ISO container. A few containers include super compacted drums with steel waste.

The void varies but is estimated to 7.5 m^3 in a container.

4.1.2 Materials – chemical composition

The amounts of different materials in a F.12 container are given in Table 4-1. Data comes from /Meijer 1987, Lindberg and Malmkvist 1992/ and (Tomas Larsson, Forsmarks Kraftgrupp AB, 2000, pers. comm.). The mix of different waste materials has changed from time to time depending if incineration is used, on different maintenance work or other reasons.

The weight of the raw waste material is normally 10,000 kg per package including the weight of the container /Johansson 2000/. The weight on the waste material can vary from approximately 5,000–15,000 kg depending on different waste materials.

Table 4-1. Amount of different material in a typical package F.12.

Material	Weight (kg)	Area (m^2)	Weight (%)
Iron/steel	0–10,000	0–508	0–100
Aluminium	0–200	0–30	0–2
Cellulose (including wood, paper, textiles, absorbed water)	0–5,000		0–50
Other organic material (including plastics, rubber, cable)	0–3,500		0–35
Other material (including insulation like mineral wool etc)	0–2,500		0–25

* Upper limit calculated as per cent of the ratio between mass of material and 10,000 kg.

4.1.3 Radionuclide inventory

Before the waste is transported from Forsmark NPP to SFR 1 a measurement of gamma emitting nuclides is performed. Dominating nuclides are Co-60 and Cs-137. The amounts of these nuclides are between 0 and $1.7 \cdot 10^{10}$ Bq for Co-60 and between 0 and $9.1 \cdot 10^{8}$ Bq for Cs-137. Other nuclides than gamma emitting ones must be calculated according to Appendix B.

The dose rate limit for this package is 2 mSv/h but normally it is between 0–0.8 mSv/h on 1 m. The surface contamination should not exceed 40 kBq/m^2 for gamma and beta and 4 kBq/m^2 for alpha emitting nuclides. The waste packages are usually free of contamination.

4.1.4 Waste production

The waste type has been in production since 1991 and is still in production. The annual production is estimated to 0.5, see Appendix A.

4.2 Reference waste type description

4.2.1 Waste package and material

Table 4-2 shows the content of different materials and steel surface areas of the packaging in typical package from Forsmark.

The assumed composition of the waste in a reference waste package of F.12 is given in Table 4-3. The surface area on a metal components in the waste are estimated assuming planar plates with a thickness of 5 mm, and assuming a density of 7,860 kg/m^3 for carbon and stainless steel and 2,700 kg/m^3 for aluminium.

The estimated volumes of the waste, the packaging materials and the void volume inside the package for a reference package are given in Table 4-4. The waste volume is estimated from the weights of the waste components in Table 4-3 and approximate densities of the different material. Average void is estimated to 50% in a package.

Void and waste volume is specified in Table 4-4.

Table 4-2. Estimated reference packaging composition and surface area and thickness of components in an ISO container of F.12.

Component	Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
ISO container	Carbon steel	1,900	104	1.5
Plywood	Cellulose	310		

Table 4-3. Estimated reference waste composition and surface area and thickness of components in one F.12.

Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Iron/steel	4,500	229	5
Aluminium	100	15	5
Cellulose (including wood, paper, textiles, absorbed water)	500 (2,000)		
Other organic material (including plastics, rubber, cable)	3,000		
Other material (including insulation like mineral wool etc)	400		

Table 4-4. Volumes in one reference waste packages of F.12.

	Volume in one package (m^3)
Waste	11.5
Void	7.5

4.2.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 4-5.

A reference waste package of the type F.12 has a surface dose rate of 2 mSv/h and no surface contamination.

Table 4-5. Radionuclide composition of a reference waste package of the type F.12.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	7.3E+09	Pu-239	2.0E+03
Cl-36	4.4E+03	Pu-240	4.0E+03
Ag-108m	4.4E+05	U-232	1.8E–01
Ba-133	7.3E+04	U-234	6.0E+00
Be-10	4.4E+00	U-235	4.1E+01
C-14	2.2E+07	U-236	1.8E+00
Fe-55	7.3E+09	U-238	3.1E+01
H-3	7.3E+05	Np-237	2.0E+02
Ho-166m	2.9E+04	Pu-238	9.3E+03
Mo-93	7.3E+03	Pu-241	6.0E+05
Nb-93m	7.3E+06	Pu-242	1.8E+01
Nb-94	7.3E+04	Am-241	2.7E+03
Ni-59	7.3E+06	Am-242m	6.0E+01
Ni-63	5.8E+08	Am-243	1.8E+02
Sb-125	5.7E+08	Cm-243	2.7E–02
Zr-93	7.3E+03	Cm-244	3.5E+03
		Cm-245	1.8E+00

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Cs-137		Cm-246	4.8E–01
Cs-137	1.8E+08		
Cd-113m	1.1E+05		
Cs-134	1.2E+08		
Cs-135	1.8E+03		
Eu-152	1.3E+04		
Eu-154	1.8E+07		
Eu-155	1.3E+07		
I-129	5.4E+02		
Pd-107	1.8E+02		
Pm-147	1.6E+08		
Ru-106	9.0E+05		
Se-79	7.2E+02		
Sm-151	5.4E+05		
Sn-126	9.0E+01		
Sr-90	1.8E+07		
Tc-99	1.6E+05		

4.3 Uncertainties

- General uncertainties are described in Section 1.3.

5 F.20

5.1 Waste type description

5.1.1 Waste package

The F.20 waste type consists of a standard ISO-container containing 200 l drums of steel with bitumen conditioned ion-exchange resins generated at the Forsmark nuclear power plant. Each container includes 36 drums. The material is classified as low level waste.

The physical properties and chemical conditions are well known. The following restrictions are always applicable:

- The general restrictions listed in Section 1.1.
- Since the package contains burnable substances the restrictions regarding self-ignition etc are of special importance.

Packaging

The ISO container is made of steel with the length of 6.1 m, width of 2.4 m and height of 1.3 m. The container is often referred to as '20-foot half-height container'. The weight of steel is approximately 1,900 kg and the surface area of the container is approximately 104 m^2. The floor of the container could be of plywood or steel. The plywood floor weighs approximately 310 kg. The variation of weight between containers is between 1,800–2,300 kg.

The drums are exactly like waste type F.05 (see Appendix E) except the amount of radionuclides and the dose rates. For all details of the drums about composition, treatment etc see the Appendix E. The steel weight of 36 drums is 900 kg.

Treatment and conditioning

36 drums are placed in a container.

The void varies but is estimated to 7 m^3 in a container.

5.1.2 Materials – chemical composition

The amounts of different materials in a F.20 container are given in Table 5-1. Data comes from /Lindberg and Malmkvist 1991/.

The weight of a container and waste is between 6,600 kg and 7,200 kg /Johansson 2000/. The weight on the waste material can vary but weighs approximately 200 kg per drum.

Table 5-1. Amount of different materials in a typical package F.20.

Material	Weight (kg)	Weight (%)
Ion exchange resin	4,680	60
Bitumen	3,420	40

5.1.3 Radionuclide inventory

Before the waste is transported from Forsmark NPP to SFR 1 a measurement of gamma emitting nuclides is performed. Dominating nuclides are Co-60 and Cs-137. The amounts of these nuclides are between 0 and $8.4 \cdot 10^9$ Bq for Co-60 and between 0 and $4.2 \cdot 10^8$ Bq for Cs-137. Other nuclides than gamma emitting ones must be calculated according to Appendix B.

The dose rate limit for this package is 2 mSv/h but normally it is between 0–0.1 mSv/h on 1 m. The surface contamination should not exceed 40 kBq/m^2 for gamma and beta and 4 kBq/m^2 for alpha emitting nuclides. The waste packages are usually free of contamination.

5.1.4 Waste production

The waste type was in production until 1982. The number of packages produced was 15. No more production is foreseen, see Appendix A.

5.2 Reference waste type description

5.2.1 Waste package and material

Table 5-2 shows the content of different materials and steel surface areas of the packaging in typical package from Forsmark.

The assumed composition of the waste in a reference waste package of F.20 is given in Table 5-3.

The estimated volumes of the waste, the packaging materials and the void volume inside the package for a reference package are given in Table 5-4. The waste volume is estimated from the weights of the waste components in Table 5-3 and approximate densities of the different material. Average void is estimated to 45% in a package.

Void and waste volume is specified in Table 5-4.

Table 5-2. Estimated reference packaging composition and surface area and thickness of components in an ISO container of F.20.

Component	Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
ISO container	Carbon steel	1,900	104	1.5
Plywood	Cellulose	310		
Steel drums (36)	Carbon steel	900	90	1.25

Table 5-3. Estimated reference waste composition and surface area and thickness of components in one F.20.

Material	Weight (kg)
Ion-exchange resin	4,680
Bitumen	3,450

Table 5-4. Volumes in one reference waste packages of F.20 (including drums).

	Volume in one package (m^3)
Waste	12
Void	7

5.2.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 5-5.

A reference waste package of the type F.20 has a surface dose rate of 2 mSv/h and no surface contamination.

Table 5-5. Radionuclide composition of a reference waste package of the type F.20.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	2.8E+09	Pu-239	4.5E+03
Cl-36	1.7E+03	Pu-240	9.1E+03
Ag-108m	1.7E+05	U-232	4.1E–01
Ba-133	2.8E+04	U-234	1.4E+01
Be-10	1.7E+00	U-235	5.3E+01
C-14	8.2E+06	U-236	4.1E+00
Fe-55	2.8E+09	U-238	5.8E+01
H-3	2.8E+05	Np-237	4.8E+02
Ho-166m	1.1E+04	Pu-238	2.6E+04
Mo-93	2.8E+03	Pu-241	1.4E+06
Nb-93m	2.8E+06	Pu-242	4.1E+01
Nb-94	2.8E+04	Am-241	7.4E+03
Ni-59	2.8E+06	Am-242m	1.4E+02
Ni-63	2.2E+08	Am-243	4.1E+02
Sb-125	2.3E+08	Cm-243	1.4E+02
Zr-93	2.8E+03	Cm-244	2.5E+04
		Cm-245	4.1E+00
Correlated to Cs-137		Cm-246	1.1E+00
Cs-137	1.1E+08		
Cd-113m	6.7E+04		
Cs-134	3.2E+07		
Cs-135	1.1E+03		
Eu-152	7.8E+03		
Eu-154	1.1E+07		
Eu-155	7.8E+06		
I-129	3.4E+02		
Pd-107	1.1E+02		
Pm-147	1.0E+08		
Ru-106	5.6E+05		
Se-79	4.5E+02		
Sm-151	3.4E+05		
Sn-126	5.6E+01		
Sr-90	1.1E+07		
Tc-99	1.0E+05		

5.3 Uncertainties

- General uncertainties are described in Section 1.3.

6 O.12

6.1 Waste type description

6.1.1 Waste package

The O.12 waste type consists of a standard ISO-container containing iron/steel, aluminium, cellulose and other organic materials generated at the Oskarshamn nuclear power plant. The material is classified as low level waste.

The physical properties and chemical conditions are well known. The following restrictions are always applicable:

- The general restrictions listed in Section 1.1.
- Since the package contains burnable substances the restrictions regarding self-ignition etc are of special importance.
- The contents of toxic material should fall within the limits set up earlier for SFR 1. That means avoidance of toxic material. Small amounts of lead and PVC could follow this waste stream.

Packaging

The ISO container is made of steel with the length of 6.1 m, width of 2.4 m and height of 1.3 m. The container is often referred to as '20-foot half-height container'. The weight of steel is approximately 1,900 kg and the surface area of the container is approximately 104 m^2. The floor of the container could be of plywood or steel. The plywood floor weighs approximately 310 kg. The variation of weight between containers is between 1,800–2,300 kg.

Other containers are possible, 10-foot full- or half-height and 20-foot full-height could be used.

Treatment and conditioning

The raw waste is collected from controlled areas from various places in the power plant. Some coarse sorting is made at each collection point before transported to central treatment.

The waste is normally packed in plastic bags or steel drums and then placed in the ISO container. A few containers include super compacted drums with steel waste.

The void varies but is estimated to 7.5 m^3 in a container.

6.1.2 Materials – chemical composition

The amounts of different materials in an O.12 container are given in Table 6-1. Data comes from /Meijer 1987, Ingemansson 1999/ and (Karl-Erik Ingemansson, Oskarshamns Kraftgrupp, 2000, pers. comm.). The mix of different waste materials has changed from time to time depending if incineration is used, on different maintenance work or other reasons.

The weight of the raw waste material is normally 10,000 kg per package including the weight of the container /Johansson 2000/. The weight on the waste material can vary from approximately 5,000–15,000 kg depending on different waste materials.

Table 6-1. Amount of different material in a typical package O.12.

Material	Weight (kg)	Area (m^2)	Weight (%)
Iron/steel	0–10,000	0–508	0–100
Aluminium	0–200	0–30	0–2
Cellulose (including wood, paper, textiles, absorbed water)	0–5,000		0–50
Other organic material (including plastics, rubber, cable)	0–3,500		0–35
Other material (including insulation like mineral wool etc)	0–2,500		0–25

* Upper limit calculated as per cent of the ratio between mass of material and 10,000 kg.

6.1.3 Radionuclide inventory

Before the waste is transported from Oskarshamn NPP to SFR 1 a measurement of gamma emitting nuclides is performed. Dominating nuclides are Co-60 and Cs-137. The amounts of these nuclides are between 0 and $6.2 \cdot 10^{10}$ Bq for Co-60 and between 0 and $5.1 \cdot 10^{11}$ Bq for Cs-137 (based on types B.12, F.12 and R.12). Other nuclides than gamma emitting ones must be calculated according to Appendix B.

The dose rate limit for this package is 2 mSv/h. No information is available on normal dose rates, but probably the dose rates are like the normal dose rates for types B.12, F.12 and R.12. The surface contamination should not exceed 40 kBq/m^2 for gamma and beta and 4 kBq/m^2 for alpha emitting nuclides. The waste packages are usually free of contamination.

6.1.4 Waste production

The annual production is estimated to 1, see Appendix A.

6.2 Reference waste type description

6.2.1 Waste package and material

Table 6-2 shows the content of different materials and steel surface areas of the packaging in typical package from Oskarshamn.

The assumed composition of the waste in a reference waste package of O.12 is given in Table 6-3. The surface area on a metal components in the waste are estimated assuming planar plates with a thickness of 5 mm, and assuming a density of 7,860 kg/m^3 for carbon and stainless steel and 2,700 kg/m^3 for aluminium.

The estimated volumes of the waste, the packaging materials and the void volume inside the package for a reference package are given in Table 6-4. The waste volume is estimated from the weights of the waste components in Table 6-3 and approximate densities of the different material. Average void is estimated to 50% in a package.

Void and waste volume is specified in Table 6-4.

Table 6-2. Estimated reference packaging composition and surface area and thickness of components in a ISO container of O.12.

Component	Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
ISO container	Carbon steel	1,900	104	1.5
Plywood	Cellulose	310		

Table 6-3. Estimated reference waste composition and surface area and thickness of components in one O.12.

Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Iron/steel	4,500	229	5
Aluminium	100	15	5
Cellulose (including wood, paper, textiles, absorbed water)	500 (2,000)		
Other organic material (including plastics, rubber, cable)	3,000		
Other material (including insulation like mineral wool etc)	400		

Table 6-4. Volumes in one waste packages of O.12.

	Volume in one package (m^3)
Waste	11.5
Void	7.5

6.2.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 6-5.

A reference waste package of the type O.12 has a surface dose rate of 2 mSv/h and no surface contamination.

Table 6-5. Radionuclide composition of a reference waste package of the type O.12.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	7.0E+09	Pu-239	5.1E+03
Cl-36	8.1E+02	Pu-240	1.0E+04
Ag-108m	4.2E+05	U-232	2.3E–01
Ba-133	7.0E+04	U-234	7.7E+00
Be-10	4.2E+00	U-235	1.5E–01
C-14	7.0E+06	U-236	2.3E+00
Fe-55	7.0E+09	U-238	3.1E+00
H-3	7.0E+05	Np-237	3.1E+00
Ho-166m	2.8E+05	Pu-238	3.1E+04
Mo-93	3.5E+05	Pu-241	7.7E+05
Nb-93m	7.0E+06	Pu-242	2.3E+01
Nb-94	7.0E+04	Am-241	7.7E+03
Ni-59	7.0E+06	Am-242m	7.7E+01
Ni-63	1.4E+09	Am-243	2.3E+02
Sb-125	7.0E+08	Cm-243	1.5E+02
Zr-93	7.0E+04	Cm-244	2.3E+04
		Cm-245	2.3E+00

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Cs-137		Cm-246	6.2E–01
Cs-137	8.1E+07		
Cd-113m	4.9E+03		
Cs-134	8.1E+07		
Cs-135	4.1E+02		
Eu-152	5.7E+03		
Eu-154	8.1E+06		
Eu-155	5.7E+06		
I-129	2.4E+01		
Pd-107	8.1E+01		
Pm-147	7.3E+07		
Ru-106	4.1E+05		
Se-79	3.2E+02		
Sm-151	2.4E+05		
Sn-126	4.1E+01		
Sr-90	8.1E+06		
Tc-99	4.1E+05		

6.3 Uncertainties

- General uncertainties are described in Section 1.3.

7 R.12

7.1 Waste type description

7.1.1 Waste package

The R.12 waste type consists of a standard ISO-container containing iron/steel, aluminium, cellulose and other organic materials generated at the Ringhals nuclear power plant. The material is classified as low level waste.

The physical properties and chemical conditions are well known. The following restrictions are always applicable:

- The general restrictions listed in Section 1.1.
- Since the package contains burnable substances the restrictions regarding self-ignition etc are of special importance.
- The contents of toxic material should fall within the limits set up earlier for SFR 1. That means avoidance of toxic material. Small amounts of lead and PVC could follow this waste stream.

Packaging

The ISO container is made of steel with the length of 6.1 m, width of 2.4 m and height of 1.3 m. The container is often referred to as '20-foot half-height container'. The weight of steel is approximately 1,900 kg and the surface area of the container is approximately 104 m^2. The floor of the container could be of plywood or steel. The plywood floor weighs approximately 310 kg. The variation of weight between containers is between 1,800–2,300 kg.

Other containers are possible, 10-foot full- or half-height and 20-foot full-height could be used.

Treatment and conditioning

The raw waste is collected from controlled areas from various places in the power plant. Some coarse sorting is made at each collection point before transported to central treatment.

The waste is normally packed in plastic bags or steel drums and then placed in the ISO container. A few containers include super compacted drums with steel waste.

The void varies but is estimated to 7.5 m^3 in a container.

7.1.2 Materials – chemical composition

The amounts of different materials in a R.12 container are given in Table 7-1. Data comes from /Meijer 1987, Ahlqvist 1990/ and (Tommy Hansson, Ringhals AB, 2000, pers. comm.). The mix of different waste materials has changed from time to time depending if incineration is used, on different maintenance work or other reasons.

The weight of the raw waste material is normally 10,000 kg per package including the weight of the container /Johansson 2000/. The weight on the waste material can vary from approximately 5,000–15,000 kg depending on different waste materials.

Table 7-1. Amount of different material in a typical package R.12.

Material	Weight (kg)	Area (m^2)	Weight (%)
Iron/steel	0–10,000	0–508	0–100
Aluminium	0–200	0–30	0–2
Cellulose (including wood, paper, textiles, absorbed water)	0–5,000		0–50
Other organic material (including plastics, rubber, cable)	0–3,500		0–35
Other material (including insulation like mineral wool etc)	0–2,500		0–25

* Upper limit calculated as per cent of the ratio between mass of material and 10,000 kg.

7.1.3 Radionuclide inventory

Before the waste is transported from Ringhals NPP to SFR 1 a measurement of gamma emitting nuclides is performed. Dominating nuclides are Co-60 and Cs-137. The amounts of these nuclides are between 0 and $6.2 \cdot 10^{11}$ Bq for Co-60 and between 0 and $5.1 \cdot 10^{10}$ Bq for Cs-137. Other nuclides than gamma emitting ones must be calculated according to Appendix B.

The dose rate limit for this package is 2 mSv/h but normally it is between 0–1.5 mSv/h on 1 metre. The surface contamination should not exceed 40 kBq/m^2 for gamma and beta and 4 kBq/m^2 for alpha emitting nuclides. The waste packages are usually free of contamination.

7.1.4 Waste production

The waste type has been in production since 1991 and is still in production. The annual production is estimated to 3, see Appendix A.

7.2 Reference waste type description

7.2.1 Waste package and material

Table 7-2 shows the content of different materials and steel surface areas of the packaging in typical package from Ringhals.

The assumed composition of the waste in a reference waste package of R.12 is given in Table 7-3. The surface area on a component in the waste is estimated assuming planar plates with a thickness of 5 mm, and assuming a density of 7,860 kg/m^3 for carbon and stainless steel and 2,700 kg/m^3 for aluminium.

The estimated volumes of the waste, the packaging materials and the void volume inside the package for a reference package are given in Table 7-4. The waste volume is estimated from the weights of the waste components in Table 7-3 and approximate densities of the different material. Average void is estimated to 50% in a package.

Void and waste volume is specified in Table 7-4.

Table 7-2. Estimated reference packaging composition and surface area and thickness of components in a ISO container of R.12.

Component	Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
ISO container	Carbon steel	1,900	104	1.5
Plywood	Cellulose	310		

Table 7-3. Estimated reference waste composition and surface area and thickness of components in one R.12.

Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Iron/steel	4,500	229	5
Aluminium	100	15	5
Cellulose (including wood, paper, textiles, absorbed water)	500 (2,000)		
Other organic material (including plastics, rubber, cable)	3,000		
Other material (including insulation like mineral wool etc)	400		

Table 7-4. Volumes in one waste packages of R.12.

	Volume in one package (m^3)
Waste	11.5
Void	7.5

7.2.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 7-5.

A reference waste package of the type R.12 has a surface dose rate of 2 mSv/h and no surface contamination.

Table 7-5. Radionuclide composition of a reference waste package of the type R.12.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	2.5E+09	Pu-239	1.5E+04
Cl-36	1.5E+03	Pu-240	2.9E+04
Ag-108m	1.5E+05	U-232	1.3E+00
Ba-133	2.5E+04	U-234	4.4E+01
Be-10	1.5E+00	U-235	8.7E–01
C-14	7.4E+06	U-236	1.3E+01
Fe-55	2.5E+09	U-238	1.8E+01
H-3	2.5E+05	Np-237	1.8E+01
Ho-166m	9.8E+03	Pu-238	1.2E+05
Mo-93	2.5E+03	Pu-241	4.4E+06
Nb-93m	2.5E+06	Pu-242	1.3E+02
Nb-94	2.5E+04	Am-241	1.2E+05
Ni-59	2.5E+06	Am-242m	4.4E+02
Ni-63	2.0E+08	Am-243	1.3E+03
Sb-125	1.2E+08	Cm-243	8.7E+02
Zr-93	2.5E+03	Cm-244	2.9E+05
		Cm-245	1.3E+01

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Cs-137		Cm-246	3.5E+00
Cs-137	3.6E+07		
Cd-113m	2.1E+04		
Cs-134	2.6E+07		
Cs-135	3.6E+02		
Eu-152	2.5E+03		
Eu-154	3.6E+06		
Eu-155	2.5E+06		
I-129	1.1E+02		
Pd-107	3.6E+01		
Pm-147	3.2E+07		
Ru-106	7.4E+05		
Se-79	1.4E+02		
Sm-151	1.1E+05		
Sn-126	1.8E+01		
Sr-90	4.3E+06		
Tc-99	3.2E+04		

7.3 Uncertainties

- General uncertainties are described in Section 1.3.

8 S.12

8.1 Waste type description

8.1.1 Waste package

The S.12 waste type consists of a standard ISO-container containing iron/steel, aluminium, cellulose and other organic materials generated at the Studsvik Research site. The material is classified as low level waste.

The physical properties and chemical conditions are well known. The following restrictions are always applicable:

- The general restrictions listed in Section 1.1.
- Since the package contains burnable substances the restrictions regarding self-ignition etc are of special importance.
- The contents of toxic material should fall within the limits set up earlier for SFR 1. That means avoidance of toxic material. Small amounts of lead and PVC could follow this waste stream.

Packaging

The ISO container is made of steel with the length of 6.1 m, width of 2.4 m and height of 1.3 m. The container is often referred to as ’20-foot half-height container’. The weight of steel is approximately 1,900 kg and the surface area of the container is approximately 104 m^2. The floor of the container could be of plywood or steel. The plywood floor weighs approximately 310 kg. The variation of weight between containers is between 1,800–2,300 kg.

Other containers are possible, 10-foot full- or half-height and 20-foot full-height could be used.

Treatment and conditioning

The raw waste is collected from controlled areas from various places on the site. Some coarse sorting is made at each collection point before transported to central treatment.

The waste is normally packed in plastic bags or steel drums and then placed in the ISO container. A few containers include super compacted drums with steel waste.

The void varies but is estimated to 7.5 m^3 in a container.

8.1.2 Materials – chemical composition

The amounts of different materials in a S.12 container are given in Table 8-1. Data comes from /Meijer 1987, Öberg 1990/ and (Jan Chyssler, Studsvik RadWaste, 2000, pers. comm.). The mix of different waste materials has changed from time to time depending if incineration is used, on different maintenance work or other reasons.

The weight of the raw waste material is normally 10,000 kg per package including the weight of the container /Johansson 2000/. The weight on the waste material can vary from approximately 5,000–15,000 kg depending on different waste materials.

Table 8-1. Amount of different material in a typical package S.12.

Material	Weight (kg)	Area (m^2)	Weight (%)
Iron/steel	0–10,000	0–508	0–100
Aluminium	0–200	0–30	0–2
Cellulose (including wood, paper, textiles, absorbed water)	0–5,000		0–50
Other organic material (including plastics, rubber, cable)	0–3,500		0–35
Other material (including insulation like mineral wool etc)	0–2,500		0–25

* Upper limit calculated as per cent of the ratio between mass of material and 10,000 kg.

8.1.3 Radionuclide inventory

Before the waste is transported from Studsvik to SFR 1 a measurement of gamma emitting nuclides is performed. Dominating nuclides are Co-60 and Cs-137. The amounts of these nuclides are between 0 and $6.2 \cdot 10^{10}$ Bq for Co-60 and between 0 and $5.1 \cdot 10^{11}$ Bq for Cs-137 (based on types B.12, F.12 and R.12). Other nuclides than gamma emitting ones must be calculated according to Appendix B.

The dose rate limit for this package is 2 mSv/h. No information is available on normal dose rates, but probably the dose rates are like the normal dose rates for types B.12, F.12 and R.12. The surface contamination should not exceed 40 kBq/m^2 for gamma and beta and 4 kBq/m^2 for alpha emitting nuclides. The waste packages are usually free of contamination.

8.1.4 Waste production

The waste type is not yet licensed to be deposited in SFR 1. The annual production is estimated to 1, see Appendix A.

8.2 Reference waste type description

8.2.1 Waste package and material

Table 8-2 shows the content of different materials and steel surface areas of the packaging in typical package from Studsvik.

The assumed composition of the waste in a reference waste package of S.12 is given in Table 8-3. The surface area on a components in the waste are estimated assuming planar plates with a thickness of 5 mm, and assuming a density of 7,860 kg/m^3 for carbon and stainless steel and 2,700 kg/m^3 for aluminium.

The estimated volumes of the waste, the packaging materials and the void volume inside the package for a reference package are given in Table 8-4. The waste volume is estimated from the weights of the waste components in Table 8-3 and approximate densities of the different material. Average void is estimated to 50% in a package.

Void and waste volume is specified in Table 8-4.

Table 8-2. Estimated reference packaging composition and surface area and thickness of components in an ISO container of S.12.

Component	Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
ISO container	Carbon steel	1,900	104	1.5

Table 8-3. Estimated reference waste composition and surface area and thickness of components in one S.12.

Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Iron/steel	4,500	229	5
Aluminium	100	15	5
Cellulose (including wood, paper, textiles, absorbed water)	500 (2,000)		
Other organic material (including plastics, rubber, cable)	3,000		
Other material (including insulation like mineral wool etc)	400		

Table 8-4. Volumes in one waste packages of S.12.

	Volume in one package (m^3)
Waste	11.5
Void	7.5

8.2.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 8-5.

A reference waste package of the type S.12 has a surface dose rate of 2 mSv/h and no surface contamination.

Table 8-5. Radionuclide composition of a reference waste package of the type S.12.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	4.3E+09	Pu-239	1.3E+05
Cl-36	8.1E+03	Pu-240	2.7E+05
Ag-108m	2.6E+05	U-232	6.0E+00
Ba-133	4.3E+04	U-234	2.0E+02
Be-10	2.6E+00	U-235	4.0E+00
C-14	4.3E+06	U-236	6.0E+01
Fe-55	4.3E+09	U-238	8.0E+01
H-3	4.3E+05	Np-237	8.0E+01
Ho-166m	1.7E+05	Pu-238	8.0E+05
Mo-93	2.2E+05	Pu-241	2.0E+07
Nb-93m	4.3E+06	Pu-242	6.0E+02
Nb-94	4.3E+04	Am-241	2.0E+05
Ni-59	4.3E+06	Am-242m	2.0E+03
Ni-63	8.6E+08	Am-243	6.0E+03
Sb-125	4.3E+08	Cm-243	4.0E+03
Zr-93	4.3E+04	Cm-244	6.0E+05
		Cm-245	6.0E+01

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Cs-137		Cm-246	1.6E+01
Cs-137	8.1E+08		
Cd-113m	4.9E+05		
Cs-134	8.1E+08		
Cs-135	4.1E+03		
Eu-152	5.7E+04		
Eu-154	8.1E+07		
Eu-155	5.7E+07		
I-129	2.4E+02		
Pd-107	8.1E+02		
Pm-147	7.3E+08		
Ru-106	4.1E+06		
Se-79	3.2E+03		
Sm-151	2.4E+06		
Sn-126	4.1E+02		
Sr-90	8.1E+07		
Tc-99	4.1E+06		

8.3 Uncertainties

- General uncertainties are described in Section 1.3.

9 S.14

9.1 Waste type description

9.1.1 Waste package

The S.14 waste type consists of a standard ISO-container containing 200 l drums of steel with scrap metal and refuse generated at the Studsvik research site. The waste type is nearly identical to S.21 except that S.21 has somewhat different radionuclide content and dose rates. The description of S.21 is assumed to be covered in this description, although S.21 is placed in BMA and not BLA like S.14.

The waste is placed in smaller drums, which is placed in the 200 l drum. Concrete is used between the two drums. Each container normally includes 36 drums. The material is classified as low level waste.

The physical properties and chemical conditions are well known. The following restrictions are always applicable:

- The general restrictions listed in Section 1.1
- The content of toxic material should fall within the limits set up earlier for SFR 1. That means in principle avoidance of toxic material. Small amounts of lead and PVC could follow this waste stream.

Packaging

The ISO container is made of steel with the length of 6.1 m, width of 2.4 m and height of 1.3 m. The container is often referred to as '20-foot half-height container'. The weight of steel is approximately 1,900 kg and the surface area of the container is approximately 104 m^2. The floor of the container could be of plywood or steel. The plywood floor weighs approximately 310 kg. The variation of weight between containers is between 1,800–2,300 kg.

Treatment and conditioning

The raw waste is after eventual fragmentation packed in the inner drum (100 l). Inner drums are then placed inside the outer drum (200 l). Free void between the two drums is filled with concrete. Normally 36 conditioned drums are placed in an ISO container, but it could differ from 33 to 36.

The void varies but is estimated to 6.4 m^3 in a container.

9.1.2 Materials – chemical composition

The amounts of different materials in a S.14 container are given in Table 9-1. Data comes from reference /Meijer 1987, Aggeryd et al. 1994/. At present no information is available of normal variation of materials a S.14 package.

The weight of a container and waste is normally 10,000 kg /Johansson 2000/ but can be up to 17,500 kg. The weight on the waste material can vary but weighs approximately 200 kg per drum.

Table 9-1. Amount of different materials in a typical package S.14.

Material	Weight (kg)	Surface area (m^2)	Weight (%)
Iron/steel	2,664	133.2	81
Aluminium	180	25.2	6
Cellulose (including wood, paper, textiles, absorbed water)	284.4		9
Other organic material (including plastics, rubber, cable)	144		4

9.1.3 Radionuclide inventory

Before the waste is transported from Studsvik to SFR 1 a measurement of gamma emitting nuclides is performed. Dominating nuclides are Co-60 and Cs-137. The amounts of these nuclides for 1 drum are between 0 and $1.2 \cdot 10^9$ for Co-60 and between 0 and $1.6 \cdot 10^7$ for Cs-137. Other nuclides than gamma emitting ones must be calculated according to Appendix B.

The dose rate limit for this package is 2 mSv/h but normally it is between 0–0.2 mSv/h on 1 m. The surface contamination should not exceed 40 kBq/m^2 for gamma and beta and 4 kBq/m^2 for alpha emitting nuclides. The waste packages are usually free of contamination.

9.1.4 Waste production

The waste type has been in production since 1980 and is still in production. For the annual production, see Appendix A.

9.2 Reference waste type description

9.2.1 Waste package and material

Table 9-2 shows the content of different materials and steel surface areas of the packaging in typical package from Studsvik.

The assumed composition of the waste in a reference waste package of S.14 is given in Table 9-3.

The estimated volumes of the waste, the packaging materials and the void volume inside the package for a reference package are given in Table 9-4. The waste volume is estimated from the weights of the waste components in Table 9-3 and approximate densities of the different material. Average void is estimated to 47% in a package.

Void and waste volume is specified in Table 9-4.

Table 9-2. Estimated reference packaging composition and surface area and thickness of components in an ISO container of S.14.

Component	Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
ISO container	Carbon steel	1,900	104	1.5
Plywood	Cellulose	310		
36 inner and outer drums	Carbon steel	1,080	256	1

Table 9-3. Estimated reference waste composition and surface area and thickness of components in one S.14.

Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Iron/steel	2,664	133	5
Aluminium	180	25	5
Cellulose (including wood, paper, textiles, absorbed water)	284		
Other organic material (including plastics, rubber, cable)	144		

Table 9-4. Volumes in one waste packages of S.14 (including drums).

	Volume in one package (m^3)
Waste	12.7
Void	6.4

9.2.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 9-5a, b and c. Table 9-5d presents the radionuclide inventory for S.21 as a comparison. The transuranic elements for the container have been calculated in a different way and it is not possible to calculate a reference waste type.

A reference waste package of the type S.14 has a surface dose rate of 2 mSv/h and no surface contamination.

Table 9-5a. Radionuclide composition of a reference waste package of the type S.14 from Studsvik in a steel drum.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	1.1E+07	Pu-239	1.6E+03
Cl-36	6.8E+00	Pu-240	3.3E+03
Ag-108m	6.8E+02	U-232	1.5E–01
Ba-133	1.1E+02	U-234	4.9E+00
Be-10	6.8E-03	U-235	4.7E+04
C-14	3.4E+04	U-236	1.5E+00
Fe-55	1.1E+07	U-238	4.9E+04
H-3	1.1E+03	Np-237	2.0E+00
Ho-166m	4.5E+01	Pu-238	2.5E+03
Mo-93	1.1E+01	Pu-241	4.9E+05
Nb-93m	1.1E+04	Pu-242	1.5E+01
Nb-94	1.1E+02	Am-241	9.8E+03
Ni-59	1.1E+04	Am-242m	4.9E+01
Ni-63	9.1E+05	Am-243	1.5E+02
Sb-125	1.2E+06	Cm-243	9.8E+01
Zr-93	1.1E+01	Cm-244	2.5E+03
		Cm-245	1.5E+00

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Cs-137		Cm-246	3.9E–01
Cs-137	7.0E+04		
Cd-113m	4.2E+01		
Cs-134	8.2E+04		
Cs-135	7.0E–01		
Eu-152	3.5E+06		
Eu-154	1.4E+06		
Eu-155	4.9E+05		
I-129	2.1E–01		
Pd-107	7.0E–02		
Pm-147	6.3E+04		
Ru-106	5.3E+04		
Se-79	2.8E–01		
Sm-151	2.1E+02		
Sn-126	3.5E–02		
Sr-90	7.0E+03		
Tc-99	6.3E+01		

Table 9-5b. Radionuclide composition of a reference waste package of the type S.14 from Svafo in a steel drum.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	1.7E+07	Pu-239	1.8E+03
Cl-36	1.0E+01	Pu-240	3.5E+03
Ag-108m	1.1E+05	U-232	1.6E–01
Ba-133	1.7E+02	U-234	5.3E+00
Be-10	1.0E-02	U-235	9.7E+04
C-14	5.1E+04	U-236	1.6E+00
Fe-55	1.7E+07	U-238	4.0E+05
H-3	1.7E+03	Np-237	2.1E+00
Ho-166m	6.8E+01	Pu-238	2.7E+03
Mo-93	1.7E+01	Pu-241	5.3E+05
Nb-93m	1.7E+04	Pu-242	1.6E+01
Nb-94	1.7E+02	Am-241	1.1E+04
Ni-59	1.7E+04	Am-242m	5.3E+01
Ni-63	1.4E+06	Am-243	1.6E+02
Sb-125	1.8E+06	Cm-243	1.1E+02
Zr-93	1.7E+01	Cm-244	2.6E+03
		Cm-245	1.6E+00

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Cs-137		Cm-246	4.3E–01
Cs-137	1.2E+05		
Cd-113m	7.1E+01		
Cs-134	7.0E+05		
Cs-135	1.2E+00		
Eu-152	1.3E+06		
Eu-154	4.8E+05		
Eu-155	4.7E+04		
I-129	3.5E–01		
Pd-107	1.2E–01		
Pm-147	1.1E+05		
Ru-106	5.8E+05		
Se-79	4.7E–01		
Sm-151	3.5E+02		
Sn-126	5.9E–02		
Sr-90	1.2E+04		
Tc-99	1.1E+02		

Table 9-5c. Radionuclide composition of a reference waste package of the type S.14 in a container.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Cs-137	
Co-60	2.8E+08	Cs-137	3.0E+06
Cl-36	1.7E+02	Cd-113m	1.8E+03
Ag-108m	2.4E+06	Cs-134	1.4E+06
Ba-133	2.8E+03	Cs-135	3.0E+01
Be-10	1.7E–01	Eu-152	6.9E+07
C-14	8.4E+05	Eu-154	2.5E+07
Fe-55	2.8E+08	Eu-155	5.5E+06
H-3	2.8E+04	I-129	8.9E+00
Ho-166m	1.1E+03	Pd-107	3.0E+00
Mo-93	2.8E+02	Pm-147	2.7E+06
Nb-93m	2.8E+05	Ru-106	3.0E+06
Nb-94	2.8E+03	Se-79	1.2E+01
Ni-59	2.8E+05	Sm-151	8.9E+03
Ni-63	2.2E+07	Sn-126	1.5E+00
Sb-125	2.4E+07	Sr-90	3.0E+05
Zr-93	2.8E+02	Tc-99	2.7E+03

Table 9-5d. Radionuclide composition of a reference waste package of the type S.21.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	4.9E+07	Pu-239	4.4E+03
Cl-36	7.7E+01	Pu-240	8.8E+03
Ag-108m	2.9E+03	U-232	1.3E–01
Ba-133	4.9E+02	U-234	4.4E+00
Be-10	2.9E–02	U-235	8.8E–02
C-14	4.9E+04	U-236	1.3E+00
Fe-55	4.9E+07	U-238	1.8E+00
H-3	4.9E+03	Np-237	1.8E+00
Ho-166m	2.0E+03	Pu-238	1.8E+04
Mo-93	2.5E+03	Pu-241	4.4E+05
Nb-93m	4.9E+04	Pu-242	1.3E+01
Nb-94	4.9E+02	Am-241	4.4E+03
Ni-59	4.9E+04	Am-242m	4.4E+01
Ni-63	9.8E+06	Am-243	1.3E+02
Sb-125	4.9E+06	Cm-243	8.8E+01
Zr-93	4.9E+02	Cm-244	1.3E+04
		Cm-245	1.3E+00
Correlated to Cs-137		Cm-246	3.5E–01
Cs-137	7.7E+06		
Cd-113m	4.6E+03		
Cs-134	7.7E+06		
Cs-135	3.9E+01		
Eu-152	5.4E+02		
Eu-154	7.7E+05		
Eu-155	5.4E+05		
I-129	2.3E+00		
Pd-107	7.7E+00		
Pm-147	6.9E+06		
Ru-106	3.9E+04		
Se-79	3.1E+01		
Sm-151	2.3E+04		
Sn-126	3.9E+00		
Sr-90	7.7E+05		
Tc-99	3.9E+04		

9.3 Uncertainties

- General uncertainties are described in Section 1.3.
- Waste volume in individual packages depends on fragmentation and the possibility to compact in the inner of the two steel drums.

10 O.99:3

10.1 Waste type description

10.1.1 Waste package

General

The O.99:3 type consists of a standard ISO-container containing 200-liter drums of steel with ion-exchange resins, filters and sludge in a concrete matrix. Some of the drums contain gravel for blast engines and some odd packages contain combustible and non combustible waste and scrap metal. The material is classified as low level waste. All data comes from /Björnberg 2000, Triumf 2007/.

The following restrictions are always applicable:

- The general restrictions listed in Section 1.1.

Packaging

The ISO container is made of steel with the length of 6.1 m, width of 2.4 m and height of 2.6 m. The container is often referred to as '20-foot full-height container'. The weight of steel is approximately 2,280 kg and the surface area of the container is approximately 414 m^2. The floor of the container could be of plywood or steel. The plywood floor weighs approximately 15–30 mm thick.

The maximum allowed weight of a waste package is 20,000 kg.

Treatment and conditioning

The waste has been put in 345 steel drums that have been placed in 5 containers. Normally 70 conditioned drums are placed in an ISO container, but this amount could differ.

The void is estimated to 16.5 m^3 in a container.

10.1.2 Materials – chemical composition

The amounts of different materials in a O.99:3 container are given in Table 10-1. The waste is fairly well defined and consists mainly of ion-exchange resins, filter aids, sludge and blast sand. The weight of the raw waste material varies quite a lot but it is approximated to 1,500 kg waste per package, see Table 10-1.

Table 10-1. Amounts of different material in waste type O.99:3 (per package).

Material	Weight (kg)*	Area (m^2)	Weight (%)
Other organic material	320		21.8
Iron/steel	3,870	414	72.2
Ion exchange resins	1,170		6.0

* Variation of amounts of material is not available in referred documents.

10.1.3 Radionuclide inventory

Before the waste is transported from Oskarshamn to SFR 1 a measurement of gamma emitting nuclides is performed. Dominating nuclides are Co-60 and Cs-137. The amounts of these nuclides are approximately $1.6 \cdot 10^6$ Bq for Co-60 and $3.7 \cdot 10^5$ Bq for Cs-137. Other nuclides than gamma emitting ones must be calculated according to Appendix B.

Dose rate limit for this package is 2 mSv/h but normally it is around 0.1 mSv/h on 1 m. The surface contamination should not exceed 40 kBq/m^2 for gamma and beta and 4 kBq/m^2 for alpha emitting nuclides. The waste packages are usually free of contamination.

10.1.4 Waste production

The waste type was in production between 1971 and 1981. No further production is foreseen, see Appendix A.

10.2 Reference waste type description

10.2.1 Waste package and material

Table 10-2 shows the content of different materials and the steel surface areas of the packaging in the reference packages.

The assumed composition of the waste in the reference waste packages of O.99:3 is given in Table 10-3.

The estimated volumes of the waste, the packaging materials and the void volume inside the package for a reference package are given in Table 10-4. The waste volume is estimated from the weights of the waste components in Table 10-3 and approximate densities of the different material. Average void is estimated to 40% in a package.

Void and waste volume is specified in Table 10-4.

Table 10-2. Estimated reference packaging composition, surface area and thickness of components in O.99:3.

Component	Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
ISO container	Carbon steel	2,280	104	1.5
Steel drums (70)	Carbon steel	1,400	90	1

Table 10-3. Estimated reference waste composition and surface area and thickness of components in one O.99:3.

Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Other organic material	320		
Iron/steel	190	414	1
Ion exchange resin	1,170		

Table 10-4. Volumes in one reference waste packages of O.99:3.

	Volume one package (m^3)
Waste	23.5
Void	16.5

10.2.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 10-6.

A reference waste package of the type O.99:3 has a maximum surface dose rate of less than 2 mSv/h and no surface contamination.

Table 10-6. Radionuclide composition of a reference waste package of the type O.99:3.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	1.7E+06	Pu-239	1.4E+03
Cl-36	3.7E+00	Pu-240	2.8E+03
Ag-108m	9.9E+01	U-232	6.3E–02
Ba-133	1.7E+01	U-234	2.1E+00
Be-10	9.9E-04	U-235	4.2E–02
C-14	1.7E+03	U-236	6.3E–01
Fe-55	1.7E+06	U-238	8.4E–01
H-3	1.7E+02	Np-237	8.4E–01
Ho-166m	6.6E+01	Pu-238	8.4E+03
Mo-93	8.3E+01	Pu-241	2.1E+05
Nb-93m	1.7E+03	Pu-242	6.3E+00
Nb-94	1.7E+01	Am-241	2.1E+03
Ni-59	1.7E+03	Am-242m	2.1E+01
Ni-63	3.3E+05	Am-243	6.3E+01
Sb-125	1.7E+05	Cm-243	4.2E+01
Zr-93	1.7E+01	Cm-244	6.3E+03
		Cm-245	6.3E–01
Correlated to Cs-137		Cm-246	1.7E–01
Cs-137	3.7E+05		
Cd-113m	2.1E+01		
Cs-134	3.7E+05		
Cs-135	1.8E+00		
Eu-152	2.6E+01		
Eu-154	3.7E+04		
Eu-155	2.6E+04		
I-129	1.1E–01		
Pd-107	3.7E–01		
Pm-147	3.3E+05		
Ru-106	1.8E+03		
Se-79	1.5E+00		
Sm-151	1.1E+03		
Sn-126	1.8E–01		
Sr-90	3.7E+04		
Tc-99	1.8E+03		

10.3 Uncertainties

- General uncertainties are described in Section 1.3.

11 Other waste types

Since SFR 1 still is in operation there is always the possibility that new waste types are needed. It is not possible to define these types per se.

12 References

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Appendix D

Waste types in BTF

1 General

1.1 Waste in BTF

In SFR, in the BTF – rock cavern for concrete tanks – short-lived waste from the nuclear power plants and Studsvik research site are disposed off. BTF includes two different rock caverns, 1BTF and 2BTF. The BTF caverns are approximately 160 m long, 14.8 m wide and have a height of 9.5 m.

The waste in 1BTF consist mainly of ash drums and concrete tanks containing ion-exchange resins and filter aids. In the 2BTF only concrete tanks containing ion-exchange resins and filter aids. Moreover, some big components of metal e.g. steam separators or rector vessel lids may be disposed of in the caverns.

A few waste packages of waste types that are designed for BMA, type O.01, C.01, R.01, R.10, R.23 are used in 1BTF to build supportive walls for ash drums. These waste types are discussed in Appendix E.

The dose rates allowed on packages are maximum 10 mSv/h. The amount of radionuclides are fairly low, dominating nuclide are Co-60 and Cs-137. Some transuranic elements are also present in the waste streams to BTF. Other criteria for acceptance are:

- Construction, geometry, dimensions and weight must be suited to the handling system in SFR and the transportation system in general and in BTF in particular. The strength of the packages must be sufficient to withstand normal and abnormal handling.
- Each package must have an individual identity and the radionuclide content of gamma emitting nuclides shall be known.
- Surface contamination should not exceed 40 kBq/m^2 for gamma and beta emitting nuclides and 4 kBq/m^2 for alpha emitting nuclides.
- The internal dose rates and integrated dose must not affect the barrier properties in the repository.
- The chemical and physical properties and the structure of the waste should be known.
- Chemical aggressive or explosive material, pressurised gas and free liquid are not allowed.
- Gas production from the waste must not be too big.
- The waste shall have enough resistance towards corrosion that the integrity of the package is kept until sealing of the repository.
- Burnable waste must not be subject to self-ignition.
- Complexing agents should if possible be avoided.

All waste in 1BTF is normally packed in steel drums, concrete or, in a few cases, steel tanks. Waste disposed off in BTF-2 is packed in concrete tanks.

1.2 Definitions

Surface area is defined as the area subject to anaerobic corrosion and gas production after sealing of the repository.

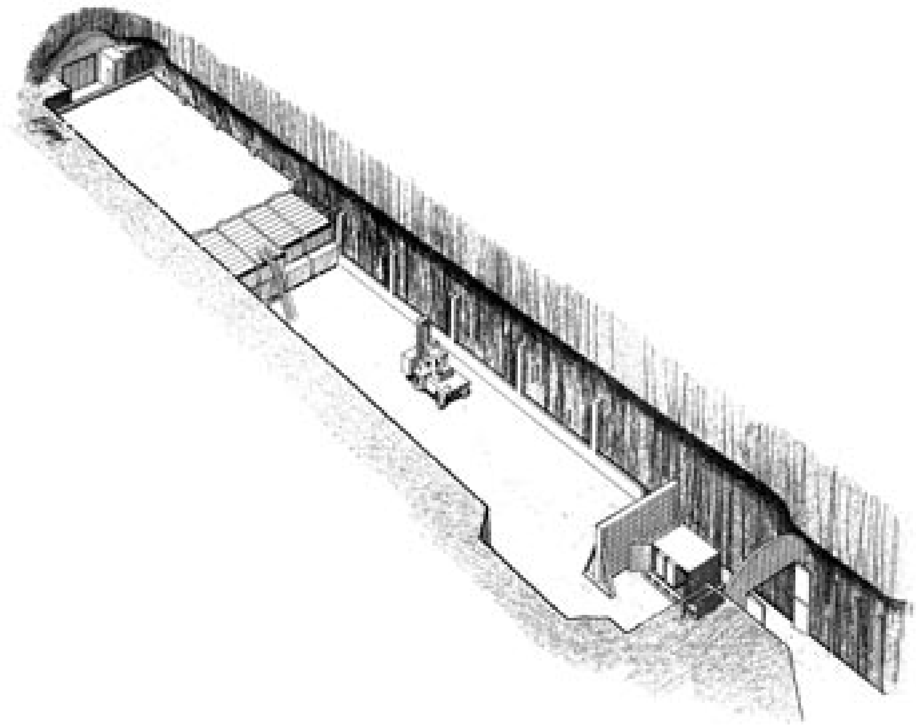

Figure 1-1. The BTF rock caverns.

1.3 Uncertainties

- The presented waste types describe the types that have been, are or can be foreseen to be produced in Sweden. New waste types may be present in the future.
- Almost all data is based on literature values, smaller changes may not been properly documented.
- All future production is based on a relative simple prognosis model. This model, described in Appendix A, was chosen since the uncertainties about processes etc are large.
- No regard has been taken to improvements in processes etc.
- A reference package has been chosen. Actual packages may differ considerably from this reference package, the uncertainty for the whole population of packages is judged to be smaller.
- Void presented in this report is based on best estimates, the estimates can be quite crude.
- For the radionuclide inventory for each waste package the amount presented is either based on the amount received from correlation factors or actual measurements when applicable. When there is a difference between different sub types of waste packages within one waste type the inventory presented in this appendix is the most conservative one.
- Big components as heat exchangers etc could in the future be of interest to dispose off in BLA. See Chapter 8 in this appendix.
- Tc-99 is correlated against both Co-60 and Cs-137 when quantifying the inventories, according to Appendix B. In the tables "Radionuclide composition of a reference waste package" however Tc-99 is presented as only correlated towards Cs-137.

2 B.07

2.1 Waste type description

2.1.1 Waste package

General

The B.07 waste type consists of a concrete tank containing de-watered ion-exchange resins. The raw waste material consists of ion-exchange resins, filter aids and sludge generated at the Barsebäck nuclear power plant. The material is classified as low level waste. All data comes from /Berntsson 1991, Johansson 2000, Triumf 2007/.

The physical properties and chemical conditions can vary a lot, but the following restrictions are always applicable:

- The general restrictions listed in Section 1.1.

Packaging

The tank is a concrete container with the outer dimensions of 3.3×1.3×2.3 m, see Figure 2-1. The thickness of the concrete wall is 0.15 m. The concrete weight is approximately 10,300 kg and the weight of steel in the package is approximately 647 kg. The reinforcement bars have a thickness of 8 mm which means that the surface of steel is approximately 40 m^2. The tank is lined with a 2 mm thick butyl rubber on the inside. The lining weighs 50 kg. The total weight of package is approximately 11,000 kg. The variations between packages are minimal. No major changes in design have been made since the start of production of this package.

Treatment and conditioning

The ion-exchange resin containing metal hydroxides and other substances are pumped into the concrete tank. Then the resins are de-watered by suction through a filter system in the bottom of the tank. The filling and suction cycle is repeated two to three times in order to fill the container with waste. The weight of a tank including waste could be up to 15,000 kg.

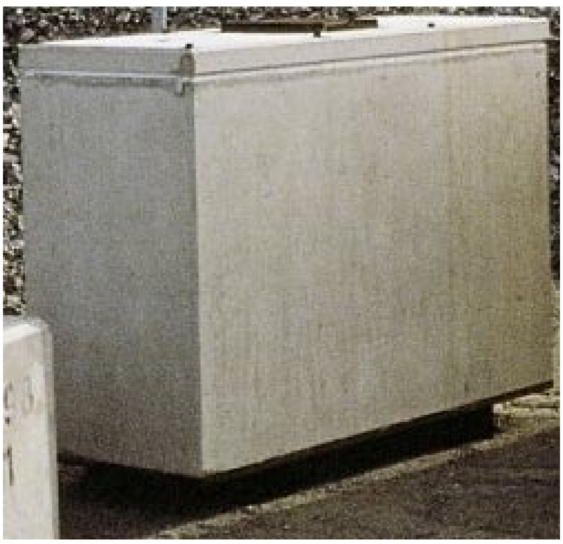

Figure 2-1. *A concrete tank.*

In order to save space in the repository the operators tries to fill the containers as much as possible, but there will always be a void in the top of the tank. This void is fairly constant and is estimated to 0.5 m^3 per tank. Additional void is to be found in the waste itself. The porosity of the waste is also fairly constant and is estimated to 60%, which gives a total void of approximately 3.8 m^3 in a tank.

2.1.2 Materials – chemical composition

The amounts of different materials in a B.07 tank are given in Table 2-1. The waste contains ion-exchange resins used for filtering metal hydroxides and other substances and other substances from the condensate cleaning system (system 332). The weight of the raw waste material can differ between 4,000 and 5,000 including water, normally it is 1,400 kg of ion-exchange resin per package. In the waste there could also be sludges and other organic matter. The amount is between 0 and 200 kg but it is as an average approximately 60 kg sludge and 66 kg organic material.

Table 2-1. Amounts of different material in waste type B.07 (per package).

Material	Weight (kg)	Weight (%)
Ion-exchange resin	1,200–1,600	86–100
Sludge	0–100	0–7
Other organic material	0–100	0–7

2.1.3 Radionuclide inventory

Before the waste is transported from Barsebäck NPP to SFR 1 a measurement of gamma emitting nuclides is performed. Dominating nuclides are Co-60 and Cs-137. The amounts of these nuclides are between 0 and $8.7 \cdot 10^{10}$ Bq for Co-60 and between 0 and $9.3 \cdot 10^{9}$ Bq for Cs-137. Other nuclides than gamma emitting ones must be calculated according to Appendix B.

The dose rate limit for this package is 3 mSv/h on the surface, but usually the dose rate is between 0–1 mSv/h. Contamination on surface should not exceed 40 kBq/m^2 for gamma and beta emitting nuclides and 4 kBq/m^2 for alpha emitting nuclides on the package. Usually there is no surface contamination.

2.1.4 Waste production

The waste type was production since 1983 and a few of the total amount were produced before 1983. No further production is foreseen, see Appendix A.

2.2 Reference waste type description

2.2.1 Waste package and material

Table 2-2 shows the content of different materials and the surface areas of steel in the packaging in reference tank from Barsebäck NPP.

The assumed composition of the waste in a reference waste package of B.07 is given in Table 2-3.

Void and waste volume is specified in Table 2-4.

Table 2-2. Typical packaging composition and surface area and thickness of components in a concrete tank.

Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Steel in reinforcement	647	40	8
Rubber lining	50		2
Concrete	10,300		150

Table 2-3. Estimated reference waste composition in one concrete tank.

Material	Weight (kg)	Weight (%)
Ion-exchange resin	1,400	92
Sludge	60	4
Other organic material	66	4

Table 2-4. Volumes in one of B.07 concrete tanks.

	Volume one package (m^3)
Waste	2.2
Void	3.8

2.2.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 2-5.

A reference waste package of the type B.07 has a surface dose rate of 3 mSv/h and no surface contamination.

Table 2-5. Radionuclide composition of a reference waste package of the type B.07.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	3.3E+10	Pu-239	5.7E+03
Cl-36	2.0E+04	Pu-240	1.1E+04
Ag-108m	2.0E+06	U-232	5.1E–01
Ba-133	3.3E+05	U-234	1.7E+01
Be-10	2.0E+01	U-235	5.5E+02
C-14	9.8E+07	U-236	4.7E+02
Fe-55	3.3E+10	U-238	4.1E+02
H-3	3.3E+06	Np-237	2.8E+03
Ho-166m	1.3E+05	Pu-238	1.8E+04
Mo-93	3.3E+04	Pu-241	1.7E+06
Nb-93m	3.3E+07	Pu-242	5.1E+01
Nb-94	3.3E+05	Am-241	1.3E+04

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Ni-59	3.3E+07	Am-242m	1.7E+02
Ni-63	2.6E+09	Am-243	5.1E+02
Sb-125	3.1E+09	Cm-243	3.4E+02
Zr-93	3.3E+04	Cm-244	1.0E+04
		Cm-245	5.1E+00
Correlated to Cs-137		Cm-246	1.4E+00
Cs-137	2.0E+09		
Cd-113m	1.2E+06		
Cs-134	7.7E+08		
Cs-135	2.0E+04		
Eu-152	1.4E+05		
Eu-154	2.0E+08		
Eu-155	1.4E+08		
I-129	6.0E+03		
Pd-107	2.0E+03		
Pm-147	1.8E+09		
Ru-106	9.9E+06		
Se-79	7.9E+03		
Sm-151	6.0E+06		
Sn-126	9.9E+02		
Sr-90	2.0E+08		
Tc-99	1.8E+06		

2.3 Uncertainties

- General uncertainties are described in Section 1.3.
- Void in concrete tank according to porosity is estimated to be in the range 60–75%.

3 O.07

3.1 Waste type description

3.1.1 Waste package

General

The O.07 waste type consists of a concrete tank containing de-watered ion-exchange resins. The raw waste material consists of ion-exchange resins, filter aids and sludge generated at the Oskarshamn nuclear power plant. The material is classified as low level waste. All data comes from /Ingemansson 1999, Johansson 2000, Triumf 2007/.

The physical properties and chemical conditions can vary a lot, but the following restrictions are always applicable:

- The general restrictions listed in Section 1.1.

Packaging

The tank is a concrete container with the outer dimensions of 3.3×1.3×2.3 m, see Figure 3-1. The thickness of the concrete wall is 0.15 m. The concrete weight is approximately 10,300 kg and the weight of steel in the package is approximately 647 kg. The reinforcement bars have a thickness of 8 mm which gives the surface of steel to 40 m^2. The tank is lined with a 2 mm thick butyl rubber on the inside. The lining weighs 50 kg. The total weight of package is approximately 11,000 kg. The variations between packages are minimal. No major changes in design have been made since the start of production of this package.

Treatment and conditioning

The ion-exchange resin containing metal hydroxides and other substances are pumped into the concrete tank. Then the resins are de-watered by suction through a filter system in the bottom of the tank. The filling and suction cycle is repeated two to three times in order to fill the container with waste. The weight of a tank including waste could be up to 15,000 kg.

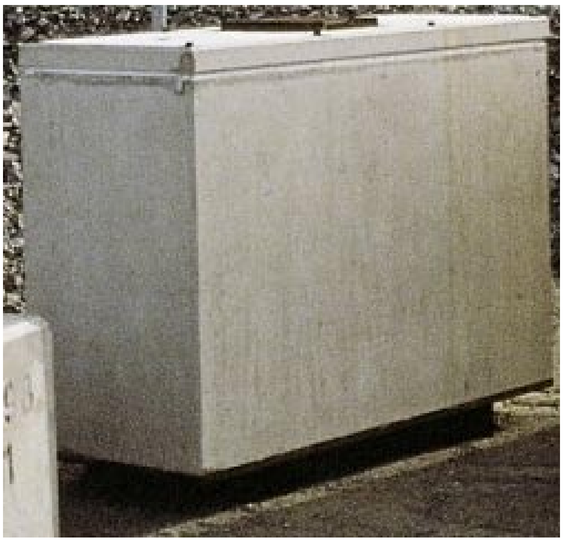

Figure 3-1. *A concrete tank.*

In order to save space in the repository the operators tries to fill the containers as much as possible, but there will always be a void in the top of the tank. This void is fairly constant and is estimated to 0.5 m^3 per tank. Additional void is to be found in the waste itself. The porosity of the waste is also fairly constant and is estimated to 60%, which gives a total void of approximately 3.8 m^3 in a tank.

3.1.2 Materials – chemical composition

The amounts of different materials in a O.07 tank are given in Table 3-1. The waste contains ion-exchange resins used for filtering metal hydroxides and other substances from the condensate cleaning system (system 332). The weight of the raw waste material can differ between 3,400 and 4,700 including water but it is normally 1,000 kg of ion-exchange resin per package. In the waste there could also be sludges and other organic matter. The amount is between 0 and 200 kg but it is as an average approximately 60 kg sludge and 66 kg organic material. Some trace amounts of complexing agents from detergents etc could be present in this waste type.

Table 3-1. Amounts of different material in waste type O.07 (per package).

Material	Weight (kg)	Weight (%)
Ion-exchange resin	850–1,200	81–100
Sludge	0–100	0–9
Other organic material	0–100	0–9

3.1.3 Radionuclide inventory

Before the waste is transported from Oskarshamn NPP to SFR 1 a measurement of gamma emitting nuclides is performed. Dominating nuclides are Co-60 and Cs-137. The amounts of these nuclides are between 0 and $4.9 \cdot 10^{11}$ Bq for Co-60 and between 0 and $6.1 \cdot 10^{11}$ Bq for Cs-137. Other nuclides than gamma emitting ones must be calculated according to Appendix B.

The dose rate limit for this package is 10 mSv/h on the surface, but usually the dose rate is between 0 and 4 mSv/h. Contamination on surface should not exceed 40 kBq/m^2 for gamma and beta emitting nuclides and 4 kBq/m^2 for alpha emitting nuclides on the package. Usually there is no surface contamination.

3.1.4 Waste production

The waste type has been in production since before 1988 and is still in production. The annual production is estimated to 8, see Appendix A.

3.2 Reference waste type description

3.2.1 Waste package and material

Table 3-2 shows the content of different materials and the surface areas of steel in the packaging in reference tank from Oskarshamn NPP.

The assumed composition of the waste in a reference waste package of O.07 is given in Table 3-3.

Void and waste volume is specified in Table 3-4.

Table 3-2. Typical packaging composition and surface area and thickness of components in a concrete tank.

Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Steel in reinforcement	647	40	8
Rubber lining	50		2
Concrete	10,300		150

Table 3-3. Estimated reference waste composition in one concrete tank.

Material	Weight (kg)	Weight (%)
Ion-exchange resin	1,000	89
Sludge	60	5
Other organic material	66	6

Table 3-4. Volumes in one typical O.07 concrete tanks.

	Volume one package (m^3)
Waste	2.2
Void	3.8

3.2.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 3-5.

A reference waste package of the type O.07 has a surface dose rate of 3 mSv/h and no surface contamination.

Table 3-5. radionuclide composition of a reference waste package of the type O.07.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	4.9E+10	Pu-239	1.1E–05
Cl-36	2.9E+04	Pu-240	1.1E–04
Ag-108m	2.9E+06	U-232	5.5E+02
Ba-133	4.9E+05	U-234	3.8E+02
Be-10	2.9E+01	U-235	1.2E+03
C-14	1.5E+08	U-236	1.3E+03
Fe-55	4.9E+10	U-238	9.2E+05
H-3	4.9E+06	Np-237	5.5E+07
Ho-166m	1.9E+05	Pu-238	1.6E+03
Mo-93	4.9E+04	Pu-241	2.6E+05
Nb-93m	4.9E+07	Pu-242	5.5E+03
Nb-94	4.9E+05	Am-241	1.6E+04

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Ni-59	4.9E+07	Am-242m	7.6E+02
Ni-63	3.9E+09	Am-243	4.3E+05
Sb-125	6.3E+09	Cm-243	1.6E+02
Zr-93	4.9E+04	Cm-244	4.4E+01
		Cm-245	3.8E–04
Correlated to Cs-137		Cm-246	5.5E+05
Cs-137	1.7E+10		
Cd-113m	1.0E+07		
Cs-134	8.4E+09		
Cs-135	1.7E+05		
Eu-152	1.2E+06		
Eu-154	1.7E+09		
Eu-155	1.2E+09		
I-129	5.0E+04		
Pd-107	1.7E+04		
Pm-147	1.5E+10		
Ru-106	8.3E+07		
Se-79	6.6E+04		
Sm-151	5.0E+07		
Sn-126	8.3E+03		
Sr-90	1.8E+05		
Tc-99	3.7E+05		

3.3 Uncertainties

- Void in concrete tank according to porosity is estimated to be in the range 60–75%.

4 S.13

4.1 Waste type description

4.1.1 Waste package

General

The S.13 waste type consists of a ordinary 200 litre steel drums with ashes from the Studsvik incineration facility. The interior of the 200 litre drum consists of a 100 litre drum and concrete filling between the two drums. The waste material consists of ashes which is a residue after incineration of low level waste as textiles, plastics and wood. The waste could also contain some fragments of scrap metals. The raw waste comes from the operations in Studsvik but also from all of the Swedish nuclear power plants. Some raw waste comes from medical and non-nuclear industrial sources. The material is classified as low level waste /Andersson and Aggeryd 1990, Johansson 2000, Triumf 2007/.

Following restrictions are always applicable on this waste:

- The general restrictions listed in Section 1.1.

Packaging

The drum is a standard 200 litre (more precise 218 l) steel drum with the outer dimensions height 0.84 m and diameter 0.57 m. The thickness of the walls are 1 mm and it weighs 20 kg. The inner drum is a 100 litre (precise 113 l) drum with the height 0.70 m and the diameter 0.46 m. The thickness of the walls is 1 mm and it weighs 10 kg. The space between the drums are filled with concrete, the content of cement in the concrete is 20% by weight. A picture of a 200 l steel drum can be seen in Figure 4-1.

The variations between packages are minimal. No major changes in design has been made since the start of production of this package. In the beginning of production of this waste type second-hand drums were used as outer package, which has resulted in small differences in the dimensions of the packages. The variations are a few centimetres on height and diameter. Nowadays only new drums are used in the process.

Figure 4-1. *View of a S.13 outer drum 218 l.*

Treatment and conditioning

The Waste material is packed in the inner drum after incineration. The inner drum is then placed inside the outer drum, which is already fitted with the concrete lining. Wet concrete is then poured on top of the inner drum as a lid.

Average void is estimated to 5% in a package including inner and outer steel drums.

4.1.2 Materials – chemical composition

The amounts of different materials in a S.13 drum are given in Table 4-1. Even though the waste is ashes the composition can change from time to time depending on what kind of origin and from where the waste are coming from. The content of waste in an individual package can vary compared to an average package. The amount of aluminium and other metals have been analysed in a batch sample of ash with different size of particles /Lidberg-Berg and Chyssler 2000/.

The weight of ash in the waste matrix is normally 70 kg per conditioned drum. Weight of the total package varies between 300 and 450 kg. About 6.5 kg of these average 70 kg ashes is approximately aluminium according to the analysis. The variations are huge depending on origin of the raw waste.

Table 4-1. Amounts of different material in waste type S.13 (per package).

Material	Weight (kg)	Surface area (m^2)	Weight (%)
Ashes	60–100		19–30
Aluminium	0–10	0–3.6	0–3
Concrete	230–250		70–75
Steel	30–40	7.1–12.7	9–14

4.1.3 Radionuclide inventory

Before the waste is transported from Studsvik to SFR 1 a measurement of gamma emitting nuclides is performed. Dominating nuclides are Co-60 and Cs-137. The amount of these nuclides are between 0 and $9.5 \cdot 10^8$ Bq for Co-60 and between 0 and $3.5 \cdot 10^8$ Bq for Cs-137. Other nuclides than gamma emitting ones must be calculated according to Appendix B.

The dose rate limit for this package is 2 mSv/h on the surface, but usually the dose rate is between 0 and 1 mSv/h. Contamination on surface should not exceed 40 kBq/m^2 for gamma and beta emitting nuclides and 4 kBq/m^2 for alpha emitting nuclides on the package. Usually there is no surface contamination.

4.1.4 Waste production

The waste type has been in production since 1980 and is still in production. The annual production is estimated to 15, see Appendix A.

4.2 Reference waste type description

4.2.1 Waste package and material

Table 4-2 shows the content of different materials and the surface areas of steel in the packaging in reference drum from Studsvik.

The assumed composition of the waste in a reference waste package of S.13 is given in Table 4-3. The surface area on the aluminium fraction in the waste are calculated as the area for spheres with a diameter of 4 mm, and a density of 2,700 kg/m^3.

Void and waste volume is specified in Table 4-4.

Table 4-2. Estimated typical packaging composition and surface area and thickness of components in waste type S.13.

Component	Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Inner steel drum	Carbon steel	10	2.7	1
Outer steel drum	Carbon steel	20	4.4	1
	Concrete	240*		

* A review of materials in waste type S.13 was made after the values in SAFE project calculations was chosen. The review has led to revised value for concrete. The value used in the calculations for concrete are 29 kg cement/package. Content in above listed value is 48 kg cement (0.20·240 = 48 kg).

Table 4-3. Estimated reference waste composition in one drum.

Material	Weight (kg)	Surface area (m^2)
Ashes	63.5	
Aluminium	6.5	3.6

Table 4-4. Volumes in one typical waste packages of S.13 drums of ashes.

	Volume one package (m^3)
Waste	0.107
Void	0.006

4.2.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 4-5a and b.

A reference waste package of the type S.13 has a surface dose rate of 2 mSv/h and no surface contamination.

Table 4-5a. Radionuclide composition of a reference waste package of the type S.13 from Studsvik.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	1.3E+07	Pu-239	1.7E–06
Cl-36	8.0E+00	Pu-240	1.7E–05
Ag-108m	3.4E+03	U-232	8.5E+01
Ba-133	1.3E+02	U-234	2.5E+01
Be-10	8.0E-03	U-235	3.4E+01

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
C-14	4.0E+04	U-236	3.4E+01
Fe-55	1.3E+07	U-238	4.2E+04
H-3	1.3E+03	Np-237	8.5E+06
Ho-166m	5.3E+01	Pu-238	2.5E+02
Mo-93	1.3E+01	Pu-241	1.7E+05
Nb-93m	1.3E+04	Pu-242	8.5E+02
Nb-94	1.3E+02	Am-241	2.5E+03
Ni-59	1.3E+04	Am-242m	1.7E+03
Ni-63	1.1E+06	Am-243	4.2E+04
Sb-125	1.5E+06	Cm-243	2.5E+01
Zr-93	1.3E+01	Cm-244	6.8E+00
		Cm-245	5.9E–05
Correlated to Cs-137		Cm-246	4.2E+04
Cs-137	4.8E+06		
Cd-113m	2.9E+03		
Cs-134	2.5E+06		
Cs-135	4.8E+01		
Eu-152	8.3E+05		
Eu-154	7.4E+05		
Eu-155	3.3E+05		
I-129	1.4E+01		
Pd-107	4.8E+00		
Pm-147	4.3E+06		
Ru-106	4.6E+05		
Se-79	1.9E+01		
Sm-151	1.4E+04		
Sn-126	2.4E+00		
Sr-90	2.8E+04		
Tc-99	5.6E+04		

Table 4-6b. Radionuclide composition of a reference waste package of the type S.13 from Svafo.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	3.0E+07	Pu-239	3.8E–06
Cl-36	1.8E+01	Pu-240	3.8E–05
Ag-108m	1.8E+03	U-232	1.9E+02
Ba-133	3.0E+02	U-234	5.6E+01
Be-10	1.8E–02	U-235	7.5E+01
C-14	8.9E+04	U-236	7.5E+01
Fe-55	3.0E+07	U-238	9.4E+04
H-3	3.0E+03	Np-237	1.9E+07
Ho-166m	1.2E+02	Pu-238	5.6E+02
Mo-93	3.0E+01	Pu-241	3.8E+05
Nb-93m	3.0E+04	Pu-242	1.9E+03
Nb-94	3.0E+02	Am-241	5.6E+03
Ni-59	3.0E+04	Am-242m	3.8E+03

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Ni-63	2.4E+06	Am-243	9.3E+04
Sb-125	3.0E+06	Cm-243	5.6E+01
Zr-93	3.0E+01	Cm-244	1.5E+01
		Cm-245	1.3E–04
Correlated to Cs-137		Cm-246	9.4E+04
Cs-137	9.3E+06		
Cd-113m	5.6E+03		
Cs-134	4.4E+06		
Cs-135	9.3E+01		
Eu-152	5.1E+04		
Eu-154	1.1E+06		
Eu-155	6.5E+05		
I-129	2.8E+01		
Pd-107	9.3E+00		
Pm-147	8.4E+06		
Ru-106	7.6E+04		
Se-79	3.7E+01		
Sm-151	2.8E+04		
Sn-126	4.7E+00		
Sr-90	6.2E+04		
Tc-99	1.3E+05		

4.3 Uncertainties

- General uncertainties are described in Section 1.3.
- Waste volume in individual packages depends on size of fragments of the residual ashes.

5 F.99:2

5.1 Waste type description

5.1.1 Waste package

General

The F.99:2 waste type consist of 10 m^3 steel boxes containing steam separators from Forsmark nuclear power plant made mainly from steel. The material is classified as low level waste. All data comes from /Modin 1997, Triumf 2007/.

Following restrictions are always applicable on this waste:

- The general restrictions listed in Section 1.1.

Packaging

The steel box has the dimensions 3.3×2.3×1.3 m. The weight is 1,900 kg and the total surface area of package is 23 m^2. The wall of the package is 4 mm thick. The total weight of the package including waste is about 5,000 kg.

Treatment and conditioning

Three steam separators are put in each steel box. These are then transported to the transport package, an ATB 3T, which can take three steel boxes.

5.1.2 Materials – chemical composition

The amounts of materials are given in Table 5-1. The waste is well defined and consists of steam separators from Forsmark nuclear power plant. The weight of the steam separators in one steel box is normally 5,000 kg.

Table 5-1. Amounts of different material in waste type F.99:2 (per package).

Material	Weight (kg)	Area (m^2)	Weight (%)
Iron/steel	4,900	183	100

5.1.3 Radionuclide inventory

Before the waste is transported from Forsmark to SFR 1 measurement of gamma emitting nuclides was performed. Dominating nuclide is Co-60. Other nuclides than gamma emitting ones must be calculated according to Appendix B.

Dose rate limit for this package is 10 mSv/h.

The surface contamination should not exceed 40 kBq/m^2. The waste packages are usually free of contamination.

5.1.4 Waste production

The waste type has been in production since 2000. No further production is estimated, see Appendix A.

5.2 Reference waste type description

5.2.1 Waste package and material

Table 5-2 shows the content of different materials and the steel surface areas of the packaging in the reference package.

The assumed composition of the waste in the reference waste packages of F.99:2 is given in Table 5-3.

The estimated volumes of the waste, the packaging materials and the void volume inside the package for a reference package are given in Table 5-4. The waste volume is estimated from the weights of the waste components in Table 5-3 and approximate densities of the different material. Average void is estimated to 5% in a package.

Void and waste volume is specified in Table 5-4.

Table 5-2. Estimated reference packaging composition, surface area and thickness of components in F.99:2.

Component	Material	Weight (kg)	Surface area (m²)	Thickness (mm)
Steel box	Steel	1,900	23	4

Table 5-3. Estimated reference waste composition and surface area of components in one F.99:2.

Material	Weight (kg)	Surface area (m²)
Iron/steel	3,000	159

Table 5-4. Volumes in one reference waste packages of F.99:2.

	Volume one package (m³)
Waste	9.5
Void	0.5

5.2.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 5-5.

A reference waste package of the type F.99:2 has a maximum surface dose rate of 10 mSv/h and no surface contamination.

Table 5-5. Radionuclide composition of a reference waste package of the type F.99:2.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	2.8E+11	Pu-239	1.7E+06
Cl-36	1.7E+05	Pu-240	3.5E+06
Ag-108m	1.7E+07	U-232	1.6E+02
Ba-133	2.8E+06	U-234	5.2E+03

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Be-10	1.7E+02	U-235	1.1E+02
C-14	2.0E+05	U-236	1.6E+03
Fe-55	1.7E+10	U-238	2.1E+03
H-3	2.8E+07	Np-237	3.2E+04
Ho-166m	1.1E+06	Pu-238	6.5E+06
Mo-93	2.8E+05	Pu-241	5.2E+08
Nb-93m	2.8E+08	Pu-242	1.6E+04
Nb-94	2.8E+06	Am-241	5.2E+06
Ni-59	2.8E+08	Am-242m	5.2E+04
Ni-63	5.1E+10	Am-243	1.6E+05
Sb-125	2.8E+10	Cm-243	1.1E+05
Zr-93	2.8E+05	Cm-244	1.1E+06
		Cm-245	1.6E+03
Correlated to Cs-137		Cm-246	4.2E+02
Cs-137	–*		
Cd-113m	–*		
Cs-134	–*		
Cs-135	–*		
Eu-152	–*		
Eu-154	–*		
Eu-155	–*		
I-129	–*		
Pd-107	–*		
Pm-147	–*		
Ru-106	–*		
Se-79	–*		
Sm-151	–*		
Sn-126	–*		
Sr-90	–*		
Tc-99	–*		

* Too few measurements to give an estimate.

5.3 Uncertainties

- General uncertainties are described in Section 1.3.

6 O.99:1

6.1 Waste type description

6.1.1 Waste package

General

The O.99:1 type consists of 40 concrete moulds from Oskarshamn. They were originally the same type as O.01:9 and O.02:9, but due to fractures in the moulds they needed special treatment. The waste contains ion exchange resins and the moulds have been put in specially designed cortén boxes due to the fractures. The material is classified as low level waste. All data comes from /Ingemansson 2001, Triumf 2007/.

The following restrictions are always applicable:

- The general restrictions listed in Section 1.1.

Packaging

The fractured concrete moulds have been put in specially designed cortén boxes. These boxes are made from 5 mm thick cortén sheet metal with the measurements of 1.49 m length, 1.49 m width and 1.6 m height. The weight of an empty cortén box is approximately 960 kg. The maximum allowed weight is 7,500 kg.

Treatment and conditioning

The concrete moulds have been put in cortén boxes. When the mould has been put into the box it is provided with a lid. When it is to be disposed of in SFR the boxes will be opened and the moulds grouted to minimise the void volume.

6.1.2 Materials – chemical composition

The amounts of different materials in O.99:1 are given in Table 6-1.

The waste is fairly well defined and consists mainly of ion-exchange resins. The weight of the raw waste material is approximated to 150 kg waste per package, see Table 6-1.

Table 6-1. Amounts of different material in waste type O.99:1 (per package).

Material	Weight (kg)	Area (m^2)	Weight (%)
Cellulose	8.3		0.6
Iron/steel	1,221	38	89.8
Ion exchange resins	130		9.6

6.1.3 Radionuclide inventory

Before the waste is transported from Oskarshamn to SFR 1 a measurement of gamma emitting nuclides is performed. Dominating nuclides are Co-60 and Cs-137. The amounts of these nuclides are approximately $1.6 \cdot 10^{10}$ Bq for Co-60 and $2.7 \cdot 10^{10}$ Bq for Cs-137. Other nuclides than gamma emitting ones must be calculated according to Appendix B.

Dose rate limit for this package is 2 mSv/h but normally it is around 0.2 mSv/h on the surface. The surface contamination should not exceed 40 kBq/m^2 for gamma and beta and 4 kBq/m^2 for alpha emitting nuclides. The waste packages are usually free of contamination.

6.1.4 Waste production

The original concrete moulds were made in Oskarshamn in the beginning of 1970. They were reconditioned in 1979. No further production is foreseen, see Appendix A.

6.2 Reference waste type description

6.2.1 Waste package and material

Table 6-2 shows the content of different materials and the steel surface areas of the packaging in the reference packages.

The assumed composition of the waste in the reference waste packages of O.99:1 is given in Table 6-3.

The estimated volumes of the waste, the packaging materials and the void volume inside the package for a reference package are given in Table 6-4. The waste volume is estimated from the weights of the waste components in Table 6-3 and approximate densities of the different material. Average void is estimated to 5% in a package.

Void and waste volume is specified in Table 6-4.

Table 6-2. Estimated reference packaging composition, surface area and thickness of components in O.99:1.

Component	Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Steel box	Steel	960	38	5

Table 6-3. Estimated reference waste composition in one O.99:1.

Material	Weight (kg)
Cellulose	8.3
Iron/steel	261
Ion exchange resin	130

Table 6-4. Volumes in one reference waste packages of O.99:1.

	Volume one package (m^3)
Waste	3.37
Void	0.18

6.2.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 6-5.

A reference waste package of the type O.99:1 has a maximum surface dose rate of 2 mSv/h and no surface contamination.

Table 6-5. Radionuclide composition of a reference waste package of the type O.99:1.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	1.6E+10	Pu-239	2.9E+06
Cl-36	2.8E+05	Pu-240	5.9E+06
Ag-108m	9.8E+05	U-232	8.8E+01
Ba-133	1.6E+05	U-234	2.9E+03
Be-10	9.8E+00	U-235	5.9E+01
C-14	1.6E+07	U-236	8.8E+02
Fe-55	1.6E+10	U-238	1.2E+03
H-3	1.6E+06	Np-237	1.2E+03
Ho-166m	6.5E+05	Pu-238	7.5E+06
Mo-93	8.2E+05	Pu-241	2.9E+08
Nb-93m	1.6E+07	Pu-242	8.8E+03
Nb-94	1.6E+05	Am-241	7.5E+06
Ni-59	1.6E+07	Am-242m	2.9E+04
Ni-63	3.3E+09	Am-243	8.8E+04
Sb-125	1.6E+09	Cm-243	5.9E+04
Zr-93	1.6E+05	Cm-244	8.8E+06
		Cm-245	8.8E+02
Correlated to Cs-137		Cm-246	2.3E+02
Cs-137	2.8E+10		
Cd-113m	1.7E+07		
Cs-134	2.8E+10		
Cs-135	1.4E+05		
Eu-152	1.9E+06		
Eu-154	2.8E+09		
Eu-155	1.9E+09		
I-129	8.3E+03		
Pd-107	2.8E+04		
Pm-147	2.5E+10		
Ru-106	1.4E+08		
Se-79	1.1E+05		
Sm-151	8.3E+07		
Sn-126	1.4E+04		
Sr-90	2.8E+09		
Tc-99	1.4E+08		

6.3 Uncertainties

- General uncertainties are described in Section 1.3.

7 R.99:1

7.1 Waste type description

7.1.1 Waste package

General

The R.99:1 waste is an old reactor tank lid from Ringhals 2 nuclear power plant. The material is classified as low level waste. All data comes from /Ekenved 1997, Triumf 2007/.

The following restrictions are always applicable:

- The general restrictions listed in Section 1.1.

Packaging

The lid is being fitted with hauling equipment but not packed any further. The waste weighs about 65,000 kg, has a diameter of 4.7 m and a height of 3.2 m with the protective cap.

Treatment and conditioning

The lid is being decontaminated on the outside and after that painted as an extra precautionary measure. The lid is then being provided with a protective cap made from steel with a thickness of 6 mm.

7.1.2 Materials – chemical composition

The amount of material in R.99:1 is given in Table 7-1.

Table 7-1. Amounts of different material in waste type R.99:1 (per package).

Material	Weight (kg)	Area (m^2)	Weight (%)
Iron/steel	65,000	150	100

7.1.3 Radionuclide inventory

Before the waste is transported from Ringhals to SFR 1 a measurement of gamma emitting nuclides is being performed. Before the waste is transported from Ringhals to SFR 1 a measurement of gamma emitting nuclides is performed. Dominating nuclides are Co-60 and Co-58 and Ni-63. Other nuclides than gamma emitting ones must be calculated according to Appendix B.

Dose rate limit for this package is 10 mSv/h upon placement in SFR 1. The surface contamination should not exceed 40 kBq/m^2 for gamma and beta and 4 kBq/m^2 for alpha emitting nuclides. The waste packages are usually free of contamination.

7.1.4 Waste production

The waste type was in production 1996. No further production is foreseen, see Appendix A.

7.2 Reference waste type description

7.2.1 Waste package and material

The waste type dopes not have a waste package per see. The reactor tank lid is grouted with concrete to fill in the void.

7.2.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 7-2.

Table 7-2. Radionuclide composition of a reference waste package of the type R.99:1.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	2.8E+11	Pu-239	1.0E+07
Cl-36	1.7E+05	Pu-240	2.1E+07
Ag-108m	1.7E+07	U-232	6.2E+02
Ba-133	2.8E+06	U-234	2.1E+04
Be-10	1.7E+02	U-235	4.2E+02
C-14	8.4E+08	U-236	6.2E+03
Fe-55	2.8E+11	U-238	8.3E+03
H-3	2.8E+07	Np-237	8.3E+03
Ho-166m	1.1E+06	Pu-238	8.3E+07
Mo-93	2.8E+05	Pu-241	2.1E+09
Nb-93m	2.8E+08	Pu-242	6.2E+04
Nb-94	2.8E+06	Am-241	2.1E+07
Ni-59	2.8E+08	Am-242m	2.1E+05
Ni-63	2.2E+10	Am-243	6.2E+05
Sb-125	2.8E+10	Cm-243	4.2E+05
Zr-93	2.8E+05	Cm-244	6.2E+07
		Cm-245	6.2E+03
Correlated to Cs-137		Cm-246	1.7E+03
Cs-137	–*		
Cd-113m	–*		
Cs-134	–*		
Cs-135	–*		
Eu-152	–*		
Eu-154	–*		
Eu-155	–*		
I-129	–*		
Pd-107	–*		
Pm-147	–*		
Ru-106	–*		
Se-79	–*		
Sm-151	–*		
Sn-126	–*		
Sr-90	–*		
Tc-99	–*		

* Too few measurements to give an estimate.

7.3 Uncertainties

- General uncertainties are described in Section 1.3.

8 Other waste types

The waste types R.01, R.10 and R.23 are put both in BTF and BMA. For further information about them, see Appendix E.

Since SFR 1 still is in operation there is always the possibility that new waste types are needed. It is not possible to define these types per se

9 References

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Appendix E

Waste types in BMA

1 General

1.1 Waste in BMA

In SFR, in the BMA – rock cavern for intermediate level waste – short-lived wastes from the nuclear power plants and the Studsvik research site are disposed off. The BMA cavern is approximately 160 m long, 19.5 m wide and has a height of 16.5 m. Inside the cavern a concrete construction has been raised. The construction consists of 13 big and 2 smaller compartments. The big ones is 10 m · 15 m · 8 m and the small ones 4 m · 7.2 m · 4.8 m. The waste, moulds and drums, are placed in the compartments by remote controlled equipment. When a compartment is filled a lid is built upon the waste. There is also a possibility to back-fill the void between the waste packages in a compartment. The technical barriers are basically the concrete structure that will minimise flow through the waste and lead the ground-water flow around the construction.

The waste that is intended in BMA comes from many different waste streams but the most important one is ion-exchange resins from the nuclear power plants. Other waste like metal components of different origin and contaminated ordinary garbage is also disposed off in BMA.

The dose rates allowed on packages are maximum 100 mSv/h. The amount of radionuclides are fairly low, BMA has been designed to handle approximately 6% of the radionuclides in SFR 1. Dominating nuclide are Co-60 and Cs-137.

Other criteria for acceptance are:

- Construction, geometry, dimensions and weight must be suited to the handling system in SFR and the transportation system in general and in BLA in particular. The strength of the packages must be sufficient to withstand normal and abnormal handling.
- Each package must have an individual identity and the radionuclide content of gamma emitting nuclides shall be known.
- Surface contamination should not exceed 40 kBq/m^2 for gamma and beta emitting nuclides and 4 kBq/m^2 for alpha emitting nuclides.
- The internal dose rates and integrated dose must not affect the barrier properties in the repository.
- The chemical and physical properties and the structure of the waste should be known.
- Chemical aggressive or explosive material, pressurised gas and free liquid are not allowed.
- Gas production from the waste must not be to big.
- The waste shall have enough resistance towards corrosion that the integrity of the package is kept until sealing of the repository.
- Burnable waste must not be subject to self-ignition.
- Complexing agents should if possible be avoided.

All waste in BMA is normally packed in concrete or steel moulds or steel drums.

1.2 Definitions

Surface area is defined as the area subject to anaerobic corrosion and gas production after sealing of the repository.

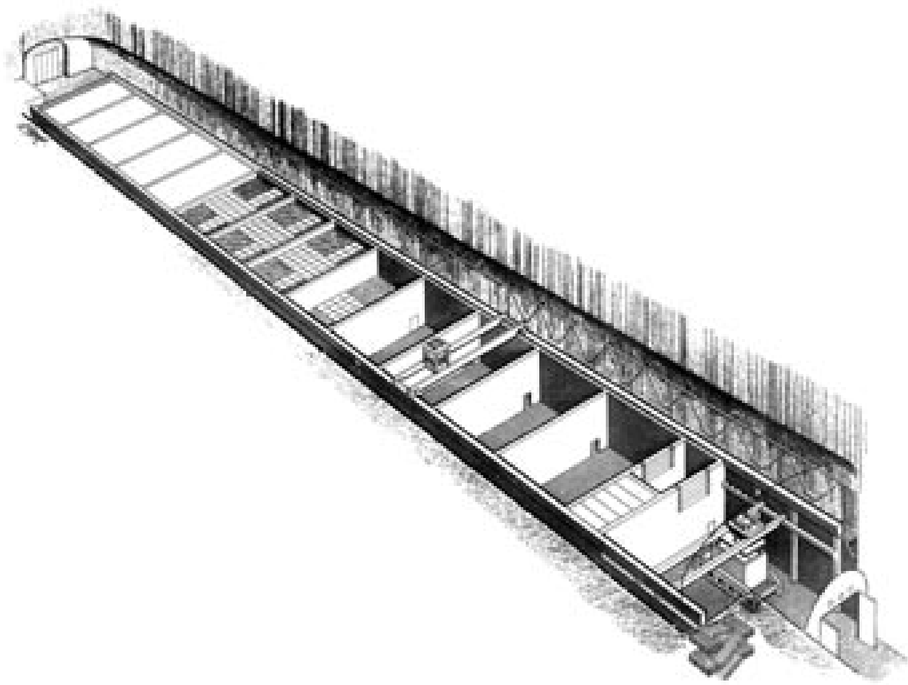

Figure 1-1. BMA.

1.3 Uncertainties

- The presented waste types describes the types that has been, is in or can be foreseen to be produced in Sweden. New waste types may be present in the future.
- Almost all data is based on literature values, smaller changes may not been properly documented.
- All future production is based on a relative simple prognosis model. This model, described in Appendix A, was chosen since the uncertainties about processes etc are large.
- No regard has been taken to improvements in processes etc.
- A reference package has been chosen based on best estimates. Actual packages may differ considerably from this reference package, the uncertainty for the whole population of packages is judged to be smaller.
- Big components as heat exchangers etc could in the future be of interest to dispose off in BMA. See Chapter 19 in this appendix.
- For the radionuclide inventory for each waste package the amount presented is either based on the amount received from correlation factors or actual measurements when applicable. When there is a difference between different sub types of waste packages within one waste type the inventory presented in this appendix is the most conservative one.
- Void presented in this report is based on best estimates, the estimates can be quite crude.
- Tc-99 is correlated against both Co-60 and Cs-137 when quantifying the inventories, according to Appendix B. In the tables "Radionuclide composition of a reference waste package" however Tc-99 is presented as only correlated towards Cs-137.

2 B.05

2.1 Waste type description

2.1.1 Waste package

General

The B.05 waste type consists of a standard 200-litre drum containing ion-exchange resins in a bitumen matrix. The raw waste material consists of ion-exchange resins and sludges generated at the Barsebäck nuclear power plant. The material is classified as intermediate level waste. All data comes from /Berntsson 1991, Johansson 2000, Triumf 2007/.

The physical properties and chemical conditions are well known. The following restrictions are always applicable:

- The general restrictions listed in Section 1.1.
- Since the package contains burnable substances the restrictions regarding self-ignition etc are of special importance.

This waste type is more or less identical to the B.06-type except the radionuclide content.

Packaging

The steel drum is made of stainless steel with diameter 0.595 m and 0.882 m in height. The thickness of the material is 1.2 mm. The drum weighs approximately 23 kg and the surface area is approximately 3 m^2. To facilitate the handling in SFR the drums are placed four by four on a steel plate with the dimension 1.2×1.2 m and the thickness of 0.004 m. The weight of the steel plate is 49 kg with surface area of 2.9 m^2. The variations between packages are minimal. No major changes in design have been made since the start of production of this package except that since 1985 the drums are made of stainless steel. The weight in total of a package including waste and is approximately 200 kg. The maximum allowed weight of a waste package is 500 kg.

Treatment and conditioning

The ion exchange resin containing metal hydroxides and other substances are mixed with bitumen in a custom-made facility in the Barsebäck NPP. The flow in the bitumenisation process is measured with electrical conductivity. A small amount of Na_2SO_4 is mixed with the product to give correct measurements of the conductivity. Emulgator is added in order to make the product more homogene. After filling, a steel lid is placed upon the drum. The drums are placed on the steel plate when they are sent to SFR.

The void varies but is estimated to 0.03 m^3 in a drum (not including expansion box).

2.1.2 Materials – chemical composition

The amounts of different materials in a B.05 drum are given in Table 2-1. The waste matrix contains metal hydroxides from the clean-up system for the reactor and evaporation residues from back flush water from the fuel storage pools. The weight of the raw waste material is normally 50 kg ion exchange resin per package but varies, see Table 2-1.

The matrix is made of bitumen. The specific brand of bitumen has changed during the many years of production of this waste type but it is always of the distilled sort. The brands used are: Mexphalate until 1982, Nynäs industrial bitumen IB 45 between 1983–1995 and Nynäs industrial bitumen IB 55 used from 1995. The amounts of added chemicals are:

Na_2SO_4	0.2–0.5 kg/drum
Emulgator	0.4–1.1 kg/drum

Table 2-1. Amounts of different material in waste type B.05 (per package).

Material	Weight (kg)	Weight (%)
Ion-exchange resin	30–55	16–43
Bitumen	129–155	57–84

2.1.3 Radionuclide inventory

Before the waste is transported from Barsebäck NPP to SFR 1 a measurement of gamma emitting nuclides is performed. Dominating nuclides are Co-60 and Cs-137. The amounts of these nuclides are between 0 and $5.1 \cdot 10^{11}$ Bq for Co-60 and between 0 and $1.1 \cdot 10^{11}$ Bq for Cs-137. Other nuclides than gamma emitting ones must be calculated according to Appendix B.

The dose rate limit for this package is 100 mSv/h but normally it is between 0–10 mSv/h on 1 m. The surface contamination should not exceed 40 kBq/m^2 for gamma and beta and 4 kBq/m^2 for alpha emitting nuclides. The waste packages are usually free of contamination.

2.1.4 Waste production

The waste type has been in production since 1985. No further production is foreseen, see Appendix A.

2.2 Reference waste type description

2.2.1 Waste package and material

Table 2-2 shows the content of different materials and the surface areas of the packaging in B.05 package from Barsebäck. Since the steel plate and expansion box is shared between four drums ¼ of the plate and box is included in a package of B.05.

The assumed composition of the waste in a reference waste package of B.05 is given in Table 2-3.

Void and waste volume is specified in Table 2-4.

Average void is estimated to 0.03 m^3 in a steel drum.

Table 2-2. Estimated reference packaging composition and surface area and thickness of components in each steel drum and one ¼ of a steel plate of B.05.

Component	Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Steel drum	Stainless steel	23	3	1.2
Steel plate	Carbon steel	12.25	0.7	4
Expansion box	Carbon steel	5	0.2	6

Table 2-3. Estimated reference waste composition in one steel drum of B.05.

Material	Weight (kg)
Ion-exchange resin	50
Bitumen	150
Chemicals(Na-sulphate)	0.5

Table 2-4. Volumes in one packages of steel drums including ¼ steel plate per drum of B.05.

	Volume one package (m^3)
Waste	0.2
Void	0.03

2.2.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 2-5.

A reference waste package of the type B.05 has a surface dose rate of 100 mSv/h and no surface contamination.

Table 2-5. Radionuclide composition of a reference waste package of the type B.05.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	9.8E+09	Pu-239	6.2E+03
Cl-36	5.9E+03	Pu-240	1.3E+04
Ag-108m	5.9E+05	U-232	5.6E–01
Ba-133	9.8E+04	U-234	1.9E+01
Be-10	5.9E+00	U-235	6.0E+02
C-14	2.9E+07	U-236	5.2E+02
Fe-55	9.8E+09	U-238	4.4E+02
H-3	9.8E+05	Np-237	3.1E+03
Ho-166m	3.9E+04	Pu-238	2.0E+04
Mo-93	9.8E+03	Pu-241	1.9E+06
Nb-93m	9.8E+06	Pu-242	5.6E+01
Nb-94	9.8E+04	Am-241	1.4E+04
Ni-59	9.8E+06	Am-242m	1.9E+02
Ni-63	7.8E+08	Am-243	5.6E+02
Sb-125	9.5E+08	Cm-243	3.7E+02
Zr-93	9.8E+03	Cm-244	1.1E+04
		Cm-245	5.6E+00
Correlated to Cs-137		Cm-246	1.5E+00
Cs-137	4.7E+09		
Cd-113m	2.8E+06		
Cs-134	1.1E+08		
Cs-135	4.7E+04		
Eu-152	3.3E+05		
Eu-154	4.7E+08		
Eu-155	3.3E+08		
I-129	1.4E+04		
Pd-107	4.7E+03		
Pm-147	4.3E+09		
Ru-106	2.4E+07		
Se-79	1.9E+04		

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Sm-151	1.4E+07		
Sn-126	2.4E+03		
Sr-90	4.7E+08		
Tc-99	4.3E+06		

2.3 Uncertainties

- General uncertainties are described in Section 1.3.

3 F.05

3.1 Waste type description

3.1.1 Waste package

General

The F.05 waste type consists of a standard 200-litre drum containing ion-exchange resins in a bitumen matrix. The raw waste material consists of ion-exchange resins generated at the Forsmark nuclear power plant. The material is classified as intermediate level waste. All data comes from /Lindberg and Malmkvist 1991ab, Johansson 2000, Triumf 2007/.

The physical properties and chemical conditions are well known. The following restrictions are always applicable:

- The general restrictions listed in Section 1.1.
- Since the package contains burnable substances the restrictions regarding self-ignition etc are of special importance.

Packaging

The steel drum is made of stainless steel with diameter 0.595 m and 0.882 m in height. The wall thickness is 1.2 mm. The drum weighs approximately 25 kg and surface area is approximately 2.5 m^2. To facilitate the handling in SFR the drums are placed four by four on a steel plate with the dimension 1.2×1.2 m and the thickness of 4 mm. The weight of the steel plate is 66.5 kg with surface area of 3.9 m^2. The variations between packages are minimal. No major changes in design have been made since the start of production of this package.

The weight in total of a package including waste and is approximately 250 kg but varies between 115 and 350 kg. The maximum allowed weight of a waste package is 500 kg.

Treatment and conditioning

The ion-exchange resins containing metal hydroxides and other substances are mixed with bitumen in a custom-made facility in the Forsmark NPP. After filling, the drums are topped up with a small amount of bitumen and then a steel lid is placed upon the drum. The drums are placed on the steel plate when the drums are sent to SFR.

The void varies but is estimated to 0.03 m^3 in a drum.

3.1.2 Materials – chemical composition

The amounts of different materials in a F.05 drum are given in Table 3-1. The waste contains metal hydroxides from condensate cleaning up system (system 332), system drainage (342) and small amounts from fuel pool cleaning system (324). Some trace amounts of complexing agents from cleaning products could

follow this waste stream. The weight of the raw waste material is normally 130 kg ion-exchange resin per package but varies, see Table 3-1.

The matrix is made of bitumen. The brand used is Nynäs industrial bitumen according to /Lindberg and Malmkvist 1991ab/.

No interval for different material is specified in the literature.

Table 3-1. Amounts of different material in waste type F.05 (per package).

Material	Weight (kg)	Weight (%)
Ion-exchange resin	130	58
Bitumen	95	42

3.1.3 Radionuclide inventory

Before the waste is transported from Forsmark NPP to SFR 1 a measurement of gamma emitting nuclides is performed. Dominating nuclides are Co-60 and Cs-137. The amounts of these nuclides are between 0 and $3 \cdot 10^{10}$ Bq for Co-60 and between 0 and $2 \cdot 10^{9}$ Bq for Cs-137. Other nuclides than gamma emitting ones must be calculated according to Appendix B.

The dose rate limit for this package is 100 mSv/h but normally it is between 0–2 mSv/h on 1 metre. The surface contamination should not exceed 40 kBq/m^2 for gamma and beta and 4 kBq/m^2 for alpha emitting nuclides. The waste packages are usually free of contamination.

3.1.4 Waste production

The waste type has been in production from 1982 to 1990. No further production is foreseen, see Appendix A.

3.2 Reference waste type description

3.2.1 Waste package and material

Table 3-2 shows the content of different materials and the surface areas of the packaging in F.05 package from Forsmark. Since the steel plate is shared between four drums ¼ of the plate and box is included in a package of F.05.

The assumed composition of the waste in a reference waste package of F.05 is given in Table 3-3.

Void and waste volume is specified in Table 3-4.

Average void is estimated to 0.02 m^3 in a steel drum.

Table 3-2. Estimated reference packaging composition and surface area and thickness of components in each steel drum and one ¼ of a steel plate of F.05.

Component	Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Steel drum	Stainless steel	25	2.5	1.25
Steel plate	Stainless steel	16.6	0.98	4

Table 3-3. Estimated reference waste composition in one steel drum of F.05.

Material	Weight (kg)
Ion-exchange resin	130
Bitumen	95

Table 3-4. Volumes in one waste packages of steel drums including ¼ steel plate per drum of F.05.

	Volume one package (m^3)
Waste	0.195
Void	0.02

3.2.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 3-5.

A reference waste package of the type F.05 has a surface dose rate of 30 mSv/h and no surface contamination.

Table 3-5. Radionuclide composition of a reference waste package of the type F.05.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	5.7E+09	Pu-239	9.3E+03
Cl-36	3.4E+03	Pu-240	1.9E+04
Ag-108m	3.4E+05	U-232	8.4E–01
Ba-133	5.7E+04	U-234	2.8E+01
Be-10	3.4E+00	U-235	1.1E+02
C-14	1.7E+07	U-236	8.4E+00
Fe-55	5.7E+09	U-238	1.2E+02
H-3	5.7E+05	Np-237	9.8E+02
Ho-166m	2.3E+04	Pu-238	5.4E+04
Mo-93	5.7E+03	Pu-241	2.8E+06
Nb-93m	5.7E+06	Pu-242	8.4E+01
Nb-94	5.7E+04	Am-241	1.5E+04
Ni-59	5.7E+06	Am-242m	2.8E+02
Ni-63	4.6E+08	Am-243	8.4E+02
Sb-125	5.7E+08	Cm-243	2.8E+02
Zr-93	5.7E+03	Cm-244	5.1E+04
		Cm-245	8.4E+00
Correlated to Cs-137		Cm-246	2.2E+00
Cs-137	2.0E+07		
Cd-113m	1.2E+04		
Cs-134	1.5E+07		
Cs-135	2.0E+02		
Eu-152	1.4E+03		
Eu-154	2.0E+06		
Eu-155	1.4E+06		
I-129	6.0E+01		
Pd-107	2.0E+01		
Pm-147	1.8E+07		
Ru-106	1.0E+05		

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Se-79	8.0E+01		
Sm-151	6.0E+04		
Sn-126	1.0E+01		
Sr-90	2.0E+06		
Tc-99	1.8E+04		

3.3 Uncertainties

- General uncertainties are described in Section 1.3.

4 F.15

4.1 Waste type description

4.1.1 Waste package

General

The F.15 waste type consists of 1.73 m^3 steel moulds containing ion-exchange resins in a concrete matrix. The raw waste material consists of ion-exchange resins, filter aids and evaporator residues generated at the Forsmark nuclear power plant. The material is classified as intermediate level waste. All data comes from /Lindberg and Malmkvist 1991c, Johansson 2000, Triumf 2007/.

Following restrictions are always applicable on this waste:

- The general restrictions listed in Section 1.1.

Packaging

The mould is a cubic box made of steel with the dimensions 1.2×1.2×1.2 m. The wall is 5 mm thick, the bottom is 6 mm thick. The mould weighs approximately 425 kg and the surface area is approximately 20 m^2. The variations between moulds are minimal. No major change in design has been made since the start of production of this package.

The weight in total of a package including waste and is approximately 2,400 kg but differs between 2,300–3,100 kg. The maximum allowed weight of a waste package is 5,000 kg.

Treatment and conditioning

The waste, used ion-exchange resin, is heated and then mixed with water and cement additives. The slurry is then pumped into the mould and cement is added. When the cement is added the stirrer is started and the waste matrix is mixed. The matrix is then allowed to harden. A steel lid is placed upon the mould.

4.1.2 Materials – chemical composition

The amounts of different materials in a F.15 mould are given in Table 4-1. The waste is well defined and consists of ion-exchange resins, evaporator residues and filter-aid with radio-nuclides.

The waste includes powder resins and filter-aids from cleaning of condensate (system 332), system drainage (system 342/1), cleaning of floor drainage (system 342/2) and bead resins from, system drainage (system 342/1) and evaporator residues from system 342/5. Some resins from other systems can be present if the radionuclide content is low enough.

The weight of the raw waste material is normally 536 kg of waste per package but varies, see Table 4-1.

The matrix is made of cement, standard Portland type. Silix GP and Sika AER are used as additives.

Table 4-1. Amounts of different material in waste type F.15 (per package)

Material	Weight (kg)	Weight (%)
Ion-exchange resin	355–550	24.8–41.7
Evaporator residues	0–200	0–15.1
Cement*	770–875	50.7–71.1

* Including water 0.4 kg/kg cement.

4.1.3 Radionuclide inventory

Before the waste is transported from Forsmark NPP to SFR 1 a measurement of gamma emitting nuclides is performed. Dominating nuclides are Co-60 and Cs-137. The amounts of these nuclides are between 0 and $5 \cdot 10^9$ Bq for Co-60 and between 0 and $5 \cdot 10^9$ Bq for Cs-137. Other nuclides than gamma emitting ones must be calculated according to Appendix B.

The dose rate limit for this package is 100 mSv/h on the surface. Normally the dose rate at 1 metre is 0–0.1 mSv/h. The surface contamination should not exceed 40 kBq/m^2 for gamma and beta and 4 kBq/m^2 for alpha emitting nuclides on the package. The waste packages are usually free of contamination.

4.1.4 Waste production

The waste type was in production between 1981 and 1988. No more production is foreseen, see Appendix A.

4.2 Reference waste type description

4.2.1 Waste package and material

Table 4-2 shows the content of different materials and the steel surface areas of the packaging in a reference package.

The assumed composition of the waste in a reference waste package of F.15 is given in Table 4-3.

The estimated volumes of the waste, the packaging materials and the void volume inside the package for a reference package are given in Table 4-4. The waste volume is estimated from the weights of the waste components in Table 4-3 and approximate densities of the different material. Average void is estimated to 10% in a package.

Void and waste volume is specified in Table 4-4.

Table 4-2. Estimated reference packaging composition and surface area and thickness of components in concrete mould of F.15.

Component	Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Steel box	Stainless steel	400	17	5
Stirrer	Stainless steel	25	3	2

Table 4-3. Estimated reference waste composition and surface area and thickness of components in one of F.15.

Material	Weight (kg)
Ion-exchange resin	375
Evaporator concentrates	161
Cement*	805

* Including 0.4 kg water/kg cement.

Table 4-4. Volumes in one packages of F.15.

	Volume in one (m^3)
Waste	1.53
Void	0.17

4.2.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 4-5. The transuranic elements have been calculated in a different way and it is not possible to calculate a reference waste type.

A reference waste package of the type F.15 has a surface dose rate of 30 mSv/h and no surface contamination.

Table 4-5. Radionuclide composition of a reference waste package of the type F.15.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	3.4E+09	Pu-239	5.6E+03
Cl-36	2.1E+03	Pu-240	1.1E+04
Ag-108m	2.1E+05	U-232	5.0E–01
Ba-133	3.4E+04	U-234	1.7E+01
Be-10	2.1E+00	U-235	6.6E+01
C-14	1.0E+07	U-236	5.0E+00
Fe-55	3.4E+09	U-238	7.2E+01
H-3	3.4E+05	Np-237	5.9E+02
Ho-166m	1.4E+04	Pu-238	3.2E+04
Mo-93	3.4E+03	Pu-241	1.7E+06
Nb-93m	3.4E+06	Pu-242	5.0E+01
Nb-94	3.4E+04	Am-241	9.2E+03
Ni-59	3.4E+06	Am-242m	1.7E+02
Ni-63	2.8E+08	Am-243	5.0E+02
Sb-125	3.4E+08	Cm-243	1.7E+02
Zr-93	3.4E+03	Cm-244	3.1E+04
		Cm-245	5.0E+00
Correlated to Cs-137		Cm-246	1.3E+00
Cs-137	2.2E+09		
Cd-113m	1.3E+06		
Cs-134	5.5E+08		

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Cs-135	2.2E+04		
Eu-152	1.6E+05		
Eu-154	2.2E+08		
Eu-155	1.6E+08		
I-129	6.7E+03		
Pd-107	2.2E+03		
Pm-147	2.0E+09		
Ru-106	1.1E+07		
Se-79	8.9E+03		
Sm-151	6.7E+06		
Sn-126	1.1E+03		
Sr-90	2.2E+08		
Tc-99	2.0E+06		

4.3 Uncertainties

- General uncertainties are described in Section 1.3.
- Evaporator concentrate could also contain fragments of metals, plastics, oil, paint and different salts.

5 F.17

5.1 Waste type description

5.1.1 Waste package

General

The F.17 waste type consists of 1.73 m^3 steel moulds containing ion-exchange resins in a bitumen matrix. The raw waste material consists of ion-exchange resins, filter aids and evaporator concentrates generated at the Forsmark nuclear power plant. The material is classified as intermediate level waste. All data comes from /Lindberg and Malmkvist 1999, Johansson 2000, Triumf 2007/.

Following restrictions are always applicable on this waste:

- The general restrictions listed in Section 1.1.
- Since the package contains burnable substances the restrictions regarding self-ignition etc are of special importance.

Packaging

The mould is a cubic box made of steel with the dimensions 1.2×1.2×1.2 m. The wall is 5 mm thick, the bottom is 6 mm thick. The mould weighs approximately 400 kg and the surface area is approximately 10 m^2. The variations between moulds are minimal. No major change in design has been made since the start of production of this package.

The weight in total of a package including waste and is approximately 2,080 kg but differs between 1,900–2,250 kg. The maximum allowed weight of a waste package is 5,000 kg.

Treatment and conditioning

The ion-exchange resins are dried in 150°C in a conical dryer and are then homogenised with bitumen in a conical mixer. The mixed waste is then poured in a steel mould. Each batch is maximum 500 litres. The mix is then allowed to cool for at least 17 h before a thin layer (~ 1 cm) of pure bitumen is poured upon the waste matrix. The waste package is thereafter lidded with a steel lid.

The void varies but is estimated to 10% in a package.

5.1.2 Materials – chemical composition

The amounts of different materials in a F.17 mould are given in Table 5-1. The waste is well defined and consists of ion-exchange resins, filter aids and evaporator concentrates with radionuclides from system 342. The weight of the raw waste material is normally 770 kg waste per package but varies, see Table 5-1. Until 1992 the waste type F.17 contained filter aids based on cellulose, each package until this year is estimated to have an average of 3.6 kg cellulose per package. In total there is 195 packages with this amount of cellulose /Carlsson 2001/.

The matrix is made of bitumen. The specific brand of bitumen has changed during the years of production of this waste type but it is always the distilled kind. The brands used are Nynäs industrial bitumen IB 45 or IB 55.

Table 5-1. Amounts of different material in waste type F.17 (per package).

Material	Weight (kg)	Weight (%)
Ion-exchange resin	0–1,100	0–65
Evaporator concentrate	0–1,100	0–65
Bitumen	600–680	35–40

5.1.3 Radionuclide inventory

Before the waste is transported from Forsmark NPP to SFR 1 a measurement of gamma emitting nuclides is performed. Dominating nuclides are Co-60 and Cs-137. The amounts of these nuclides are between 0 and $5 \cdot 10^{11}$ Bq for Co-60 and between 0 and $2 \cdot 10^{11}$ Bq for Cs-137. Other nuclides than gamma emitting ones must be calculated according to Appendix B.

The dose rate limit for this package is 100 mSv/h. Normal dose rate at 1 m is between 0.1–4.5 mSv/h. The surface contamination should not exceed 40 kBq/m^2 for gamma and beta and 4 kBq/m^2 for alpha emitting nuclides on the package. The waste packages are usually free of contamination.

5.1.4 Waste production

The waste type has been in production since 1989 and is still in production. Annual production is estimated to 30 packages per year, see Appendix A.

5.2 Reference waste type description

5.2.1 Waste package and material

Table 5-2 shows the content of different materials and the steel surface areas of the packaging in a reference package. The assumed composition of the waste in a reference waste package of F.17 is given in Table 5-3.

Average void is estimated 10% in a steel mould. Void and waste volume is specified in Table 5-4.

Table 5-2. Estimated reference packaging composition and surface area and thickness of components in steel mould of F.17.

Component	Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Steel mould	Stainless steel	400	10	5

Table 5-3. Estimated reference waste composition in one of F.17.

Material	Weight (kg)
Ion-exchange resin	650
Evaporator concentrates	120
Bitumen	820

Table 5-4. Volumes in one reference waste package of F.17.

	Volume in one (m^3)
Waste	1.615
Void	0.085

5.2.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 5-5.

A reference waste package of the type F.17 has a surface dose rate of 100 mSv/h and no surface contamination.

Table 5-5. Radionuclide composition of a reference waste package of the type F.17.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	6.7E+10	Pu-239	1.1E+05
Cl-36	4.0E+04	Pu-240	2.3E+05
Ag-108m	4.0E+06	U-232	1.0E+01
Ba-133	6.7E+05	U-234	3.4E+02
Be-10	4.0E+01	U-235	1.3E+03
C-14	2.0E+08	U-236	1.0E+02
Fe-55	6.7E+10	U-238	1.5E+03
H-3	6.7E+06	Np-237	1.2E+04
Ho-166m	2.7E+05	Pu-238	6.5E+05
Mo-93	6.7E+04	Pu-241	3.4E+07
Nb-93m	6.7E+07	Pu-242	1.0E+03
Nb-94	6.7E+05	Am-241	1.9E+05
Ni-59	6.7E+07	Am-242m	3.4E+03
Ni-63	5.3E+09	Am-243	1.0E+04
Sb-125	6.6E+09	Cm-243	3.4E+03
Zr-93	6.7E+04	Cm-244	6.2E+05
		Cm-245	1.0E+02
Correlated to Cs-137		Cm-246	2.7E+01
Cs-137	8.2E+09		
Cd-113m	4.9E+06		
Cs-134	2.9E+09		
Cs-135	8.2E+04		
Eu-152	5.7E+05		
Eu-154	8.2E+08		
Eu-155	5.7E+08		
I-129	2.5E+04		
Pd-107	8.2E+03		
Pm-147	7.4E+09		
Ru-106	4.1E+07		

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Se-79	3.3E+04		
Sm-151	2.5E+07		
Sn-126	4.1E+03		
Sr-90	8.2E+08		
Tc-99	7.4E+06		

5.3 Uncertainties

- General uncertainties are described in Section 1.3.
- Evaporator concentrate could also contain fragments of metals, plastics, oil, paint and different salts.
- Data on cellulose in filter aids until 1992 is probably overestimated.

6 F.23

6.1 Waste type description

6.1.1 Waste package

General

The F.23 waste type consists of 1.73 m^3 concrete or steel moulds with scrap metal and refuse in a concrete matrix. The raw waste material consists of mainly iron/steel, cellulose and other organics generated at the Forsmark nuclear power plant. The material is classified as intermediate level waste. All data comes from /Forsmark 2000, Johansson 2000, Triumf 2007/.

Following restrictions are always applicable on this waste:

- The general restrictions listed in Section 1.1.

Packaging

The steel mould has the dimension 1.2×1.2×1.2 m. The weight is 445 kg and the total metal surface area of package is 14.5 m^2. The wall of the package is 5 mm thick, the bottom is 6 mm thick. The mould also has two lids and some reinforcements. The variations between moulds are minimal. Total weight of package including waste is approximately 2,100 kg but differs between 1,470–4,270 kg.

The concrete mould is a cubic box made of reinforced concrete with the dimensions 1.2×1.2×1.2 m. The walls are 10 cm thick. The reinforcement is made of 12 mm steel bars with a total surface of 10 m^2. The mould weighs approximately 1,600 kg. The variations between moulds are minimal. Total weight of package including waste is approximately 2,400 kg but differs between 2,200–4,100 kg.

In some special cases a concrete mould with 25 cm thick walls can be used.

The maximum allowed weight of a waste package is 5,000 kg.

Treatment and conditioning

Waste from the power plant is sorted in compactable and non-compactable. Non-compactable is put directly in a mould and the mould is then filled with concrete. Compactable waste is put in the mould and compacted. A ‘middle-lid’ is placed in the mould to prevent re-expansion of the waste. This operation is repeated until the mould is full. Then the mould is filled with concrete.

Pouring concrete on top of the package makes a lid.

6.1.2 Materials – chemical composition

The amounts of different materials in a F.23 mould are given in Table 6-1. The waste is fairly well defined and consists mainly of scrap metal and refuses as

filters, wood, cloth, plastics and cables. The weight of the raw waste material is normally 770 kg waste per package but varies, see Table 6-1a and b.

The matrix is made of concrete. The brand used is of the standard Portland.

Table 6-1a. Amounts of different material in waste type F.23 steel mould (per package).

Material	Weight (kg)*	Area (m^2)	Weight (%)
Sludges	18		0.8
Cellulose	150		7.0
Other organic material	450		21.1
Iron/steel	150	7.6	7.0
Aluminium	5	0.5	0.2
Concrete**	1,356		63.7

* Variation of amounts of material is not available in referred documents.

** Including 0.13 kg water/kg concrete.

Table 6-1b. Amounts of different material in waste type F.23 10-cm concrete mould (per package).

Material	Weight (kg)*	Area (m^2)	Weight (%)
Sludges	0		0
Cellulose	29		3.6
Other organic material	186		23.0
Iron/steel	30	1.5	3.7
Aluminium	0	0	0
Concrete**	565		69.8

* Variation of amounts of material is not available in referred documents.

** Including 0.13 kg water/kg concrete.

6.1.3 Radionuclide inventory

Before the waste is transported from Forsmark to SFR 1 a measurement of gamma emitting nuclides is performed. Dominating nuclides are Co-60 and Cs-137. The amounts of these nuclides are between 0 and $9\cdot10^{10}$ Bq for Co-60 and between 0 and $3\cdot10^{10}$ Bq for Cs-137. Other nuclides than gamma emitting ones must be calculated according to Appendix B.

Dose rate limit for this package is 100 mSv/h but normally it is between 0.01–8 mSv/h on 1 m. The surface contamination should not exceed 40 kBq/m^2 for gamma and beta and 4 kBq/m^2 for alpha emitting nuclides. The waste packages are usually free of contamination.

6.1.4 Waste production

The waste type has been in production since 1986 and is still in production. Annual production is estimated to 8 packages per year, see Appendix A.

6.2 Reference waste type description

6.2.1 Waste package and material

The steel mould is used as reference waste package.

Table 6-2a and b shows the content of different materials and the steel surface areas of the packaging in the reference packages.

The assumed composition of the waste in the reference waste packages of F.23 is given in Table 6-3a and b.

Table 6-2a. Estimated reference packaging composition, surface area and thickness of components in steel mould of F.23.

Component	Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Steel box	Steel	400	10	5
Lid	Steel	48	5.8	3
Reinforcement	Steel	14	0.5	14

Table 6-2b. Estimated reference packaging composition, surface area and thickness of components in concrete mould of F.23.

Component	Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Concrete box	Concrete	1,340		10
Lid	Steel	13	1.8	2
Reinforcement	Steel	261	10	12

Table 6-3a. Estimated reference waste composition and surface area and thickness of components in one steel mould F.23.

Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Sludge	18		
Cellulose	150		
Other organic material	450		
Iron/steel	150	7.8	5
Aluminium	5	0.7	5
Concrete*	1,356		

* Including 0.13 kg water/kg concrete.

Table 6-3b. Estimated reference waste composition and surface area and thickness of components in one concrete mould F.23.

Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Sludge	0		
Cellulose	29		
Other organic material	186		
Iron/steel	30	1.5	5
Aluminium	0	0	5
Concrete*	565		

* Including 0.13 kg water/kg concrete.

The estimated volumes of the waste, the packaging materials and the void volume inside the package for a reference package are given in Table 6-4. The waste volume is estimated from the weights of the waste components in Table 6-3a and b and approximate densities of the different material. Average void is estimated to 5% in a package. Void and waste volume is specified in Table 6-4.

Table 6-4. Volumes in one package of F.23.

	Volume in one (m^3)
Waste	1.615
Void	0.085

6.2.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 6-5a and b.

Table 6-5a. Radionuclide composition of a reference waste package of the type F.23 concrete mould.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	1.5E+10	Pu-239	3.9E+03
Ag-108m	9.0E+05	Pu-240	7.7E+03
Ba-133	1.5E+05	U-232	3.5E–01
Be-10	9.0E+00	U-234	1.2E+01
C-14	4.5E+07	U-235	7.8E+01
Fe-55	1.5E+10	U-236	3.5E+00
H-3	1.5E+06	U-238	6.0E+01
Ho-166m	6.0E+04	Np-237	3.8E+02
Mo-93	1.5E+04	Pu-238	1.8E+04
Nb-93m	1.5E+07	Pu-241	1.2E+06
Nb-94	1.5E+05	Pu-242	3.5E+01
Ni-59	1.5E+07	Am-241	5.3E+03
Ni-63	1.2E+09	Am-242m	1.2E+02
Sb-125	1.5E+09	Am-243	3.5E+02
Zr-93	1.5E+04	Cm-243	5.2E–02
		Cm-244	6.6E+03
Correlated to Cs-137		Cm-245	3.5E+00
Cs-137	–*	Cm-246	9.3E-01
Cd-113m	–*		
Cs-134	–*		
Cs-135	–*		
Eu-152	–*		
Eu-154	–*		
Eu-155	–*		
I-129	–*		
Pd-107	–*		
Pm-147	–*		
Ru-106	–*		
Se-79	–*		
Sm-151	–*		
Sn-126	–*		
Sr-90	–*		
Tc-99	–*		

* Too few measurements to give an estimate.

Table 6-5b. Radionuclide composition of a reference waste package of the type F.23 steel mould.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	1.3E+10	Pu-239	3.2E+03
Cl-36	8.0E+03	Pu-240	6.4E+03
Ag-108m	8.0E+05	U-232	2.9E–01
Ba-133	1.3E+05	U-234	9.7E+00
Be-10	8.0E+00	U-235	6.6E+01
C-14	4.0E+07	U-236	2.9E+00
Fe-55	1.3E+10	U-238	5.0E+01
H-3	1.3E+06	Np-237	3.2E+02
Ho-166m	5.3E+04	Pu-238	1.5E+04
Mo-93	1.3E+04	Pu-241	9.7E+05
Nb-93m	1.3E+07	Pu-242	2.9E+01
Nb-94	1.3E+05	Am-241	4.7E+06
Ni-59	1.3E+07	Am-242m	9.7E+01
Ni-63	1.1E+09	Am-243	2.9E+02
Sb-125	1.3E+09	Cm-243	4.4E–02
Zr-93	1.3E+04	Cm-244	5.6E+03
		Cm-245	2.9E+00
Correlated to Cs-137		Cm-246	7.8E–01
Cs-137	2.2E+08		
Cd-113m	1.3E+05		
Cs-134	1.5E+08		
Cs-135	2.2E+03		
Eu-152	1.5E+04		
Eu-154	2.2E+07		
Eu-155	1.5E+07		
I-129	6.6E+02		
Pd-107	2.2E+02		
Pm-147	2.0E+08		
Ru-106	1.1E+06		
Se-79	8.8E+02		
Sm-151	6.6E+05		
Sn-126	1.1E+02		
Sr-90	2.2E+07		
Tc-99	2.0E+05		

A reference waste package of the type F.23 has a surface dose rate of 100 mSv/h and no surface contamination.

6.3 Uncertainties

- General uncertainties are described in Section 1.3.
- All future production of F.23 is assumed to be in steel moulds. Some packages may be of the concrete kind.

7 O.01

7.1 Waste type description

7.1.1 Waste package

General

The O.01 waste type consists of 1.73 m^3 concrete moulds containing ion-exchange resins and inert filter-aid in a concrete matrix. The raw waste material consists of ion-exchange resins and filter aids generated at the Oskarshamn nuclear power plant. The material is classified as intermediate level waste /Ingemansson 1999, Johansson 2000, Triumf 2007/.

Following restrictions are always applicable on this waste:

- The general restrictions listed in Section 1.1.

Packaging

The mould is a cubic box made of reinforced concrete with the 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 is made of 12 mm steel bars with a total surface of 10 m^2. The 10-cm mould weighs approximately 1,600 kg and the 25-cm one weighs about 3,200 kg. A stirrer made of carbon steel is included in the waste package. It weighs 16 kg and has a estimated surface of 1 m^2. To avoid cracking of the mould, due to expansion of the concrete matrix, a lining of compactable material (polyethene) is put inside of the mould, the lining has a thickness of 20 mm and a total weight of 10 kg. Lining is only placed in the 10-cm mould. The variations between moulds are minimal.

Some changes in the design have been made since the start of production of this package. 1975–78 the reinforcement was gradually improved and in 1981 a new design of reinforcement was introduced. In 1979 the moulds was fitted with a expansion cassette of polyurethane and wood, this was used until 1988 when the polyethene cassette was introduced. In 1986 a new brand of concrete to make the lid was introduced, instead of Sabema A, the Betokem EXM-4 was used. Today no chemical cement additives is used, earlier Silix GP was used.

The weight of a package including waste is approximately 3,200 kg but differs between 3,300–3,600 kg. The maximum allowed weight of a waste package is 5,000 kg.

Treatment and conditioning

The waste, used ion-exchange resin, is pumped into the mould and cement is added. If needed, water is added. When the cement is added the stirrer is started and the waste matrix is mixed for 30 minutes. The matrix is then allowed to harden for two days. Pouring concrete on top of the matrix then makes a lid. The lid is at least 10 cm thick. The concrete is allowed 24 hours to harden before the waste is transported to intermediate storage.

The waste, used ion-exchange resin, is pumped into the mould and cement is added. If needed, water is added. When the cement is added the stirrer is started and the waste matrix is mixed for 30 minutes. The matrix is then allowed to harden for two days. Pouring concrete on top of the matrix then makes a lid. The concrete is allowed 24 hours to harden before the waste is transported to intermediate storage. The lid is at least 10 cm thick.

7.1.2 Materials – chemical composition

The amounts of different materials in a O.01 mould are given in Table 7-1. The waste is well defined and consists of ion exchange resins and filter-aid with radionuclides from systems for cleaning reactor water (system 331), treatment of liquid waste (system 342), decontamination (system 347) and cleaning of fuel ponds (system 324). Systems 331, 347 and 342 uses bead resin, system 342 uses powder resin. The weight of the raw waste material is normally 130 kg ion exchange resin per package but varies, see Table 7-1.

The matrix is made of cement. The specific brand of cement has changed during the years of production of this waste type. The brands used are:
Sabema A-bruk 1970–1986,
LH cement 1970–1981,
Massiv cement 1981–1986,
Anläggningscement 1986–1999,
Höghållfasthetscement from 1999–
Betokem EXM used from 1986–.

Table 7-1. Amounts of different material in waste type O.01 (per package).

Material	Weight (kg)	Weight (%)
Ion-exchange resin	90–150	5.7–10.0
Cement*	1,350–1,500	90.0–94.3

* Including water 0.4 kg/kg cement.

7.1.3 Radionuclide inventory

Before the waste is transported from Oskarshamn NPP to SFR 1 a measurement of gamma emitting nuclides is performed. Dominating nuclides are Co-60 and Cs-137. The amounts of these nuclides are between 0 and $6 \cdot 10^{11}$ Bq for Co-60 and between 0 and $9 \cdot 10^{11}$ Bq for Cs-137. Other nuclides than gamma emitting ones must be calculated according to Appendix B.

Dose rate limit for this package is 30 mSv/h but normally it is between 0.01–11 mSv/h on 1 m. The surface contamination should not exceed 40 kBq/m^2 for gamma and beta and 4 kBq/m^2 for alpha emitting nuclides. The waste packages are usually free of contamination.

7.1.4 Waste production

The waste type has been in production since 1970. No further production is foreseen, see Appendix A.

7.2 Reference waste type description

7.2.1 Waste package and material

Table 7-2 shows the content of different materials and the steel surface areas of the packaging in a reference package.

The assumed composition of the waste in a reference waste package of O.01 is given in Table 7-3.

The estimated volumes of the waste, the packaging materials and the void volume inside the package for a reference package are given in Table 7-4. The waste volume is estimated from the weights of the waste components in Table 7-3 and approximate densities of the different material. Average void is estimated to 15% in a package.

Void and waste volume is specified in Table 7-4.

Table 7-2. Estimated reference packaging composition and surface area and thickness of components in concrete mould of O.01.

Component	Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Concrete mould	Concrete	1,340		100
Reinforcement	Steel	261	10	12
Steel lid	Steel	13	1.8	2
Stirrer	Steel	16	1	5
Other organics	Polyethene	10		

Table 7-3. Estimated reference waste composition and surface area and thickness of components in one of O.01.

Material	Weight (kg)
Ion-exchange resin	130
Cement*	1,540

* Including water 0.4 kg/kg cement.

Table 7-5. Volumes in one waste package of O.01.

	Volume in one (m^3)
Waste	0.85
Void	0.15

7.2.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 7-5.

A reference waste package of the type O.01 has a maximum surface dose rate of 100 mSv/h and no surface contamination.

Table 7-5. Radionuclide composition of a reference waste package of the type O.01.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	9.2E+10	Pu-239	3.8E+05
Cl-36	5.5E+04	Pu-240	7.7E+05
Ag-108m	5.5E+06	U-232	3.5E+01
Ba-133	9.2E+05	U-234	1.2E+03
Be-10	5.5E+01	U-235	1.9E+01
C-14	2.8E+08	U-236	8.0E+02
Fe-55	9.2E+10	U-238	2.6E+03
H-3	9.2E+06	Np-237	2.8E+03
Ho-166m	3.7E+05	Pu-238	1.9E+06
Mo-93	9.2E+04	Pu-241	1.2E+08
Nb-93m	9.2E+07	Pu-242	3.5E+03
Nb-94	9.2E+05	Am-241	5.5E+05

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Ni-59	9.2E+07	Am-242m	1.2E+04
Ni-63	7.3E+09	Am-243	3.4E+04
Sb-125	1.0E+10	Cm-243	1.6E+03
Zr-93	9.2E+04	Cm-244	9.0E+05
		Cm-245	3.5E+02
Correlated to Cs-137		Cm-246	9.2E+01
Cs-137	2.1E+10		
Cd-113m	1.2E+07		
Cs-134	1.7E+10		
Cs-135	2.1E+05		
Eu-152	1.5E+06		
Eu-154	2.1E+09		
Eu-155	1.5E+09		
I-129	6.2E+04		
Pd-107	2.1E+04		
Pm-147	1.9E+10		
Ru-106	1.0E+08		
Se-79	8.3E+04		
Sm-151	6.2E+07		
Sn-126	1.0E+04		
Sr-90	2.1E+09		
Tc-99	1.9E+07		

7.3 Uncertainties

- No production of O.01 is planned but production is possible.
- General uncertainties are described in Section 1.3.

8 C.01

8.1 Waste type description

8.1.1 Waste package

General

The C.01 waste type consists of 1.73 m^3 concrete moulds containing ion-exchange resins and inert filter-aid in a concrete matrix. The raw waste material consists of ion-exchange resins and filter aids generated at the Clab facility in Oskarshamn. The material is classified as intermediate level waste /Ingemansson 1999, Johansson 2000, Triumf 2007/.

The C.01 waste type is identical with O.01 with some small exceptions. The type is in everyday use called O.01.

Following restrictions are always applicable on this waste:

- The general restrictions listed in Section 1.1.

Packaging

The mould is a cubic box made of reinforced concrete with the 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 is made of 12 mm steel bars with a total surface of 10 m^2. The 10-cm mould weighs approximately 1,600 kg and the 25-cm one weighs about 3,200 kg. A stirrer made of carbon steel is included in the waste package. It weighs 16 kg and has a estimated surface of 1 m^2. To avoid cracking of the mould, due to expansion of the concrete matrix, a lining of compactible material (polyethene) is put inside of the mould, the lining has a thickness of 20 mm and a total weight of 10 kg. Lining is only placed in the 10-cm mould. The variations between moulds are minimal.

Some changes in the design have been made since the start of production of this package. 1975–78 the reinforcement was gradually improved and in 1981 a new design of reinforcement was introduced. In 1979 the moulds was fitted with a expansion cassette of polyurethane and wood, this was used until 1988 when the polyethene cassette was introduced. In 1986 a new brand of concrete to make the lid was introduced, instead of Sabema A, the Betokem EXM-4 was used. Today no chemical cement additives is used, earlier Silix GP was used.

The weight of a package including waste is approximately 3,200 kg but differs between 3,300–3,600 kg. The maximum allowed weight of a waste package is 5,000 kg.

Treatment and conditioning

The waste, used ion-exchange resin, is pumped into the mould and cement is added. If needed, water is added. When the cement is added the stirrer is started and the waste matrix is mixed for 30 minutes. The matrix is then allowed to harden for two days. Pouring concrete on top of the matrix then makes a lid. The lid is at least 10 cm thick. The concrete is allowed 24 hours to harden before the waste is transported to intermediate storage. Treatment and conditioning

The waste, used ion-exchange resin, is pumped into the mould and cement is added. If needed, water is added. When the cement is added the stirrer is started and the waste matrix is mixed for 30 minutes. The matrix is then allowed to harden for two days. Pouring concrete on top of the matrix then makes a lid. The concrete is allowed 24 hours to harden before the waste is transported to intermediate storage. The lid is at least 10 cm thick.

8.1.2 Materials – chemical composition

The amounts of different materials in a C.01 mould are given in Table 8-1. The waste is well defined and consists of ion exchange resins and filter-aid with radionuclides from systems for cleaning reactor water (system 331), treatment of liquid waste (system 342), decontamination (system 347) and cleaning of fuel ponds (system 324). Systems 331, 347 and 342 uses bead resin, system 342 uses powder resin. The weight of the raw waste material is normally 130 kg ion exchange resin per package but varies, see Table 8-1.

The matrix is made of cement. The specific brand of cement has changed during the years of production of this waste type. The brands used are:
Sabema A-bruk –1986,
Massiv cement –1986,
Anläggningscement 1986–1999,
Höghållfasthetscement from 1999–
Betokem EXM used from 1986–.

Table 8-1. Amounts of different material in waste type C.01 (per package).

Material	Weight (kg)	Weight (%)
Ion-exchange resin	90–150	5.7–10.0
Cement*	1,350–1,500	90.0–94.3

* Including water 0.4 kg/kg cement.

8.1.3 Radionuclide inventory

Before the waste is transported from Oskarshamn NPP/Clab to SFR 1 a measurement of gamma emitting nuclides is performed. Dominating nuclides are Co-60 and Cs-137. The amounts of these nuclides are between 0 and $6 \cdot 10^{11}$ Bq for Co-60 and between 0 and $9 \cdot 10^{11}$ Bq for Cs-137. Other nuclides than gamma emitting ones must be calculated according to Appendix B.

Dose rate limit for this package is 30 mSv/h but normally it is between 0.01–30 mSv/h on the surface. The surface contamination should not exceed 40 kBq/m^2 for gamma and beta and 4 kBq/m^2 for alpha emitting nuclides. The waste packages are usually free of contamination.

8.1.4 Waste production

The waste type has been in production since 1970 and can still produced, but no further production is foreseen, see Appendix A.

8.2 Reference waste type description

8.2.1 Waste package and material

Table 8-2 shows the content of different materials and the steel surface areas of the packaging in a reference package.

The assumed composition of the waste in a reference waste package of C.01 is given in Table 8-3.

The estimated volumes of the waste, the packaging materials and the void volume inside the package for a reference package are given in Table 8-4. The waste volume is estimated from the weights of the waste components in Table 8-3 and approximate densities of the different material. Average void is estimated to 15% in a package.

Void and waste volume is specified in Table 8-4.

Table 8-2. Estimated reference packaging composition and surface area and thickness of components in concrete mould of C.01.

Component	Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Concrete mould	Concrete	1,340		100
Reinforcement	Steel	261	10	12
Steel lid	Steel	13	1.8	2
Stirrer	Steel	16	1	5
Other organics	Polyethene	10		

Table 8-3. Estimated reference waste composition and surface area and thickness of components in one of C.01.

Material	Weight (kg)
Ion-exchange resin	130
Cement*	1,540

* Including water 0.4 kg/kg cement.

Table 8-4. Volumes in one reference waste package of C.01.

	Volume in one (m^3)
Waste	0.85
Void	0.15

8.2.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 8-5.

A reference waste package of the type C.01 has a maximum surface dose rate of 100 mSv/h and no surface contamination.

Table 8-5. Radionuclide composition of a reference waste package of the type C.01.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	5.8E+10	Pu-239	7.6E+04
Ag-108m	3.4E+06	Pu-240	1.2E+05
Ba-133	5.8E+05	U-232	6.8E+00
Be-10	3.5E+01	U-234	2.3E+02
C-14	1.8E+08	U-235	4.5E+00
Fe-55	5.8E+10	U-236	6.8E+01
H-3	5.8E+06	U-238	9.1E+01
Ho-166m	2.3E+05	Np-237	9.1E+01
Mo-93	5.8E+04	Pu-238	9.3E+05
Nb-93m	5.8E+07	Pu-241	2.3E+07
Nb-94	5.8E+05	Pu-242	6.8E+02
Ni-59	5.8E+07	Am-241	2.6E+05

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Ni-63	4.7E+09	Am-242m	2.3E+03
Sb-125	5.8E+09	Am-243	3.6E+04
Zr-93	5.8E+04	Cm-243	1.5E+04
		Cm-244	5.7E+05
Correlated to Cs-137		Cm-245	6.8E+01
Cs-137	–*	Cm-246	1.8E+01
Cd-113m	–*		
Cs-134	–*		
Cs-135	–*		
Eu-152	–*		
Eu-154	–*		
Eu-155	–*		
I-129	–*		
Pd-107	–*		
Pm-147	–*		
Ru-106	–*		
Se-79	–*		
Sm-151	–*		
Sn-126	–*		
Sr-90	–*		
Tc-99	–*		

* Too few measurements to give an estimate.

8.3 Uncertainties

- No production of C.01 is planned but production is possible.
- General uncertainties are described in Section 1.3.

9 O.23

9.1 Waste type description

9.1.1 Waste package

General

The O.23 waste type consists of 1.73 m^3 concrete moulds with scrap metal and refuse in a concrete matrix. The raw waste material consists of mainly iron/steel, cellulose and other organics generated at the Oskarshamn nuclear power plant. The material is classified as intermediate level waste. All data comes from /Ingemansson 2000, Johansson 2000, Triumf 2007/.

Following restrictions are always applicable on this waste:

- The general restrictions listed in Section 1.1.

Packaging

The concrete mould is a cubic box made of reinforced concrete with the dimensions 1.2×1.2×1.2 m. The walls are 10 cm thick. The reinforcement is made of 12 mm steel bars with a total surface of 10 m^2. The 10-cm mould weighs approximately 1,600 kg. The variations between moulds are minimal. Total weight of package including waste is approximately 2,400 kg but differs between 1,900–4,000 kg.

In some special cases a concrete mould with 25 cm or 35 cm thick walls can be used.

The maximum allowed weight of a waste package is 5,000 kg.

Treatment and conditioning

Waste from the power plant is sorted in compactable and non-compactable. Non-compactable is put directly in a mould and the mould is then filled with concrete. Compactable waste is put in the mould and compacted. A 'middle-lid' is placed in the mould to prevent re-expansion of the waste. This operation is repeated until the mould is full. Then the mould is filled with concrete, which is allowed to harden for two days. Pouring concrete on top of the package makes a lid.

9.1.2 Materials – chemical composition

The amounts of different materials in a O.23 mould are given in Table 9-1. The waste is fairly well defined and consists mainly of scrap metal and refuses as filters, wood, cloth, plastics and cables. The weight of the raw waste material is normally 780 kg waste per package but varies, see Table 9-1.

The matrix is made of concrete. The brand used is Betokem EXM 4PA.

Table 9-1. Amounts of different material in waste type O.23 10-cm concrete mould (per package).

Material	Weight (kg)*	Area (m^2)	Weight (%)
Sludges	52.5		6.2
Cellulose	30		3.5
Other organic material	66.5		7.9
Iron/steel	112	5.7	13.2
Other inorganic material	17.5		2.1
Aluminium	3.5	0.5	0.4
Concrete**	565		66.7

* Variation of amounts of material is not available in referred documents.

** Including 0.13 kg water/kg concrete.

9.1.3 Radionuclide inventory

Before the waste is transported from Oskarshamn to SFR 1 a measurement of gamma emitting nuclides is performed. Dominating nuclides are Co-60 and Cs-137. The amounts of these nuclides are between 0 and $1 \cdot 10^{11}$ Bq for Co-60 and between 0 and $2 \cdot 10^{11}$ Bq for Cs-137. Other nuclides than gamma emitting ones must be calculated according to Appendix B.

Dose rate limit for this package is 30 mSv/h but normally it is between 0–2 mSv/h on 1 m. The surface contamination should not exceed 40 kBq/m^2 for gamma and beta and 4 kBq/m^2 for alpha emitting nuclides. The waste packages are usually free of contamination.

9.1.4 Waste production

The waste type has been in production since 1981 and is still in production. Annual production is estimated to 7 packages per year, see Appendix A.

9.2 Reference waste type description

9.2.1 Waste package and material

Table 9-2 shows the content of different materials and the steel surface areas of the packaging in a reference package.

The assumed composition of the waste in a reference waste package of O.23 is given in Table 9-3.

The estimated volumes of the waste, the packaging materials and the void volume inside the package for a reference package are given in Table 9-4. The waste volume is estimated from the weights of the waste components in Table 9-3 and approximate densities of the different material. Average void is estimated to 5% in a package.

Void and waste volume is specified in Table 9-4.

Table 9-2. Estimated reference packaging composition, surface area and thickness of components in steel mould of O.23.

Component	Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Concrete box	Concrete	1,340		10
Lid	Steel	13	1.8	2
Reinforcement	Steel	261	10	12

Table 9-3. Estimated reference waste composition and surface area and thickness of components in one O.23.

Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Sludge	52.5		
Cellulose	30		
Other organic material	66.5		
Iron/steel	112	5.7	5
Other inorganics	17.5		
Aluminium	3.5	0.5	5
Concrete*	565		

* Including 0.13 kg water/kg concrete.

Table 9-4. Volumes in one reference waste package of O.23.

	Volume in one (m^3)
Waste	0.95
Void	0.05

9.2.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 9-5. Inventory based on data for B.12, F.12 and R.12.

A reference waste package of the type O.23 has a surface dose rate of 30 mSv/h and no surface contamination.

Table 9-5. Radionuclide composition of a reference waste package of the type O.23.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	1.1E+10	Pu-239	2.4E+04
Cl-36	6.6E+03	Pu-240	4.9E+04
Ag-108m	6.6E+05	U-232	2.1E+00
Ba-133	1.1E+05	U-234	7.3E+01
Be-10	6.6E+00	U-235	1.5E+00
C-14	3.3E+07	U-236	2.2E+01
Fe-55	1.1E+10	U-238	2.9E+01
H-3	1.1E+06	Np-237	2.9E+01
Ho-166m	4.4E+04	Pu-238	1.4E+05
Mo-93	1.1E+04	Pu-241	7.3E+06
Nb-93m	1.1E+07	Pu-242	2.2E+02
Nb-94	1.1E+05	Am-241	3.9E+04
Ni-59	1.1E+07	Am-242m	7.3E+02
Ni-63	8.8E+08	Am-243	2.2E+02
Sb-125	1.1E+09	Cm-243	1.5E+03
Zr-93	1.1E+04	Cm-244	8.4E+04
		Cm-245	2.2E+01

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Cs-137		Cm-246	5.8E+00
Cs-137	1.0E+09		
Cd-113m	6.0E+05		
Cs-134	1.0E+09		
Cs-135	1.0E+04		
Eu-152	7.0E+04		
Eu-154	1.0E+08		
Eu-155	7.0E+07		
I-129	3.0E+03		
Pd-107	1.0E+03		
Pm-147	9.0E+08		
Ru-106	5.0E+06		
Se-79	4.0E+03		
Sm-151	3.0E+06		
Sn-126	5.0E+02		
Sr-90	1.0E+08		
Tc-99	9.0E+05		

9.3 Uncertainties

- General uncertainties are described in Section 1.3.

10 C.23

10.1 Waste type description

10.1.1 Waste package

General

The C.23 waste type consists of 1.73 m^3 concrete moulds with scrap metal and refuse in a concrete matrix. The raw waste material consists of mainly iron/steel, cellulose and other organics generated at the Oskarshamn nuclear power plant. The material is classified as intermediate level waste /Ingemansson 2000, Johansson 2000, Triumf 2007/.

The C.23 waste type is identical with O.23 with some small exceptions. The type is in everyday use called O.23.

Following restrictions are always applicable on this waste:

- The general restrictions listed in Section 1.1.

Packaging

The concrete mould is a cubic box made of reinforced concrete with the dimensions 1.2×1.2×1.2 m. The walls are 10 cm thick. The reinforcement is made of 12 mm steel bars with a total surface of 10 m^2. The 10-cm mould weighs approximately 1,600 kg. The variations between moulds are minimal. Total weight of package including waste is approximately 2,400 kg but differs between 1,900–4,000 kg.

In some special cases a concrete mould with 25 cm or 35 cm thick walls can be used

The maximum allowed weight of a waste package is 5,000 kg.

Treatment and conditioning

Waste from the power plant is sorted in compactable and non-compactable. Non-compactable is put directly in a mould and the mould is then filled with concrete. Compactable waste is put in the mould and compacted. A 'middle-lid' is placed in the mould to prevent re-expansion of the waste. This operation is repeated until the mould is full. Then the mould is filled with concrete, which is allowed to harden for two days. Pouring concrete on top of the package makes a lid.

10.1.2 Materials – chemical composition

The amounts of different materials in a C.23 mould are given in Table 10-1. The waste is fairly well defined and consists mainly of scrap metal and refuses as filters, wood, cloth, plastics and cables. The weight of the raw waste material is normally 780 kg waste per package but varies, see Table 10-1.

The matrix is made of concrete. The brand used is Betokem EXM 4PA.

Table 10-1. Amounts of different material in waste type C.23 10-cm concrete mould (per package).

Material	Weight (kg)*	Area (m^2)	Weight (%)
Sludges	52.5		6.2
Cellulose	30		3.5
Other organic material	66.5		7.9
Iron/steel	112	5.7	13.2
Other inorganic material	17.5		2.1
Aluminium	3.5	0.5	0.4
Concrete**	565		66.7

* Variation of amounts of material is not available in referred documents.

** Including 0.13 kg water/kg concrete.

10.1.3 Radionuclide inventory

Before the waste is transported from Oskarshamn to SFR 1 a measurement of gamma emitting nuclides is performed. Dominating nuclides are Co-60 and Cs-137. The amounts of these nuclides are between 0 and $1 \cdot 10^{11}$ Bq for Co-60 and between 0 and $2 \cdot 10^{11}$ Bq for Cs-137. Other nuclides than gamma emitting ones must be calculated according to Appendix B.

Dose rate limit for this package is 30 mSv/h but normally it is between 0–2 mSv/h on 1 metre. The surface contamination should not exceed 40 kBq/m^2 for gamma and beta and 4 kBq/m^2 for alpha emitting nuclides. The waste packages are usually free of contamination.

10.1.4 Waste production

The waste type has been in production since 1985 and is still in production. Annual production is estimated to 2 packages per year until 2015 when the estimated production is increased to 4 packages per year, see Appendix A.

10.2 Reference waste type description

10.2.1 Waste package and material

Table 10-2 shows the content of different materials and the steel surface areas of the packaging in a reference package.

The assumed composition of the waste in a reference waste package of C.23 is given in Table 10-3.

The estimated volumes of the waste, the packaging materials and the void volume inside the package for a reference package are given in Table 10-4. The waste volume is estimated from the weights of the waste components in Table 10-3 and approximate densities of the different material. Average void is estimated to 5% in a package.

Void and waste volume is specified in Table 10-4.

Table 10-2. Estimated reference packaging composition, surface area and thickness of components in steel mould of C.23.

Component	Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Concrete box	Concrete	1,340		10
Lid	Steel	13	1.8	2
Reinforcement	Steel	261	10	12

Table 10-3. Estimated reference waste composition and surface area and thickness of components in one C.23.

Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Sludge	52.5		
Cellulose	30		
Other organic material	66.5		
Iron/steel	112	5.7	5
Other inorganics	17.5		
Aluminium	3.5	0.5	5
Concrete*	565		

* Including 0.13 kg water/kg concrete.

Table 10-4. Volumes in one reference waste package of C.23.

	Volume per package (m^3)
Waste	0.95
Void	0.05
Waste package	1.73

10.2.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 10-5.

A reference waste package of the type C.23 has a surface dose rate of 30 mSv/h and no surface contamination.

Table 10-5. Radionuclide composition of a reference waste package of the type C.23.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	1.2E+10	Pu-239	1.3E+04
Ag-108m	7.0E+05	Pu-240	2.6E+04
Ba-133	1.2E+05	U-232	1.2E+00
Be-10	7.0E+00	U-234	3.9E+01
C-14	3.5E+07	U-235	7.8E–01
Fe-55	1.2E+10	U-236	1.2E+01
H-3	1.2E+06	U-238	1.6E+01
Ho-166m	4.7E+04	Np-237	1.6E+01
Mo-93	1.2E+04	Pu-238	1.3E+05
Nb-93m	1.2E+07	Pu-241	3.9E+06
Nb-94	1.2E+05	Pu-242	1.2E+02
Ni-59	1.2E+07	Am-241	2.3E+04
Ni-63	9.4E+08	Am-242m	3.9E+02
Sb-125	1.2E+09	Am-243	1.2E+04
Zr-93	1.2E+04	Cm-243	7.8E+02
		Cm-244	1.2E+05

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Cs-137		Cm-245	1.2E+01
Cs-137	–*	Cm-246	3.1E+00
Cd-113m	–*		
Cs-134	–*		
Cs-135	–*		
Eu-152	–*		
Eu-154	–*		
Eu-155	–*		
I-129	–*		
Pd-107	–*		
Pm-147	–*		
Ru-106	–*		
Se-79	–*		
Sm-151	–*		
Sn-126	–*		
Sr-90	–*		
Tc-99	–*		

* Too few measurements to give an estimate.

10.3 Uncertainties

- General uncertainties are described in Section 1.3.

11 R.01

11.1 Waste type description

11.1.1 Waste package

General

The R.01 waste type consists of 1.73 m^3 concrete moulds containing ion-exchange resins and inert filter-aid in a concrete matrix. The raw waste material consists of ion-exchange resins and filter aids generated at the Ringhals nuclear power plant. The material is classified as intermediate level waste /Eriksson 1990, Johansson 2000, Triumf 2007/.

Following restrictions are always applicable on this waste:

- The general restrictions listed in Section 1.1.

Packaging

The mould is a cubic box made of reinforced concrete with the 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 is made of steel bars with a total surface of 7.9 m^2 and a weight of 160 kg in a 10-cm mould and 11.8 m^2 and 250 kg in a 25-cm mould. The 10-cm mould weighs approximately 1,600 kg and the 25-cm weighs about 3,100 kg. A stirrer made of carbon steel is included in the waste package. It weighs 16 kg and has an estimated surface of 1 m^2. To avoid cracking of the mould, due to expansion of the concrete matrix, a lining with compressible material (polyethene or polystyrene) is put inside the walls of the mould. The lining has a thickness of 20 mm and a total weight of 10 kg in a 10-cm mould and 5 mm and approximately 1 kg in a 25-cm mould. The variations between moulds are minimal.

Some changes in the design have been made since the start of production of this package. Before 1976 no lining was used. Some changes have been made regarding reinforcement and steel details in the mould. For the lid Betokem EXM-4 was used, nowadays Fiberbruk VF50 is used. Today no chemical cement additives are used but earlier Silix GP and Sika AER were used.

The weight of a package including waste and is approximately 3,400 kg but differs between 3,000–3,700 kg. The maximum allowed weight of a waste package is 4,000 kg.

Treatment and conditioning

The waste, used ion-exchange resin, is pumped into the mould and cement is added. If needed, water is added. When the cement is added the stirrer is started and the waste matrix is mixed for 60 minutes. The matrix is then allowed to harden for two days. Pouring concrete on top of the matrix then makes a lid. The concrete is allowed 24 hours to harden before the waste is transported to interim storage. The lid is at least 10 cm thick.

11.1.2 Materials – chemical composition

The amounts of different materials in a R.01 mould are given in Table 11-1. The waste is well defined and consists of ion exchange resins and filter-aid with radionuclides.

The resins are:

- Bead resin from PWR: cleaning of the primary reactor circuit (system 334), the fuel ponds (system 324), the waste water (system 342), water from the secondary side of the steam generators (system 417/337) and from drainage water.

- Bead resin from BWR: systems for cleaning reactor water (system 331), treatment of liquid waste (system 342), analytical ion-exchangers (system 336) and cleaning of fuel ponds (system 324).
- Powder resins from BWR: cleaning of condensate (system 332), fuel ponds (system 324) and treatment of liquid waste (system 324). The resins from liquid waste treatment contain filter-aids.

Resins from the PWR-reactors contain boric acid (H_3BO_3) up to 90 g/kg of resin. Some resins contain lithium. Resins from system 417/337 can contain ammonia and hydrazine.

The weight of the raw waste material is normally 130 kg of ion exchange resin per package but varies, see Table 11-1.

The matrix is made of cement. The specific brand of cement has changed during the years of production of this waste type. The brand used nowadays is Skövde standardcement. In earlier days Limhamn LH-cement was used.

Table 11-1. Amounts of different material in waste type R.01 (per package).

Material	Weight (kg)	Weight (%)
Ion-exchange resin	100–150	5.9–12.0
Cement*	1,100–1,600	88.0–94.1

* Including water 0.4 kg/kg cement.

11.1.3 Radionuclide inventory

Before the waste is transported from Ringhals NPP to SFR 1 a measurement of gamma emitting nuclides is performed. Dominating nuclides are Co-60 and Cs-137. The amounts of these nuclides are between 0 and $4 \cdot 10^{12}$ Bq for Co-60 and between 0 and $1 \cdot 10^{11}$ Bq for Cs-137. Other nuclides than gamma emitting ones must be calculated according to Appendix B.

The dose rate limit for this package is 30 mSv/h on the surface. Normal dose rate at 1 m is 4 mSv/h but varies between 0–20 mSv/h. The surface contamination should not exceed 40 kBq/m^2 for gamma and beta and 4 kBq/m^2 for alpha emitting nuclides on the package.

11.1.4 Waste production

The waste type has been in production since 1984. No further production is foreseen, see Appendix A.

11.2 Reference waste type description

11.2.1 Waste package and material

Table 11-2 shows the content of different materials and the steel surface areas of the packaging in a reference package.

The assumed composition of the waste in a reference waste package of R.01 is given in Table 11-3.

The estimated volumes of the waste, the packaging materials and the void volume inside the package for a reference package are given in Table 11-4. The waste volume is estimated from the weights of the waste components in Table 11-3 and approximate densities of the different material. Average void is estimated to 15% in a package.

Void and waste volume is specified in Table 11-4.

Table 11-2. Estimated reference packaging composition and surface area and thickness of components in concrete mould of R.01.

Component	Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Concrete box	Concrete	1,340		100
Reinforcement	Steel	261	10	12
Steel lid	Steel	13	1.8	2
Stirrer	Steel	16	1	5
Other organics	Polyethene	10		

Table 11-3. Estimated reference waste composition and surface area and thickness of components in one of R.01.

Material	Weight (kg)
Ion-exchange resin	130
Cement*	1,400

* Including water 0.4 kg/kg cement.

Table 11-5. Volumes in one reference waste package of R.01.

	Volume per package (m^3)
Waste	0.85
Void	0.15

11.2.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 11-5.

A reference waste package of the type R.01 has a surface dose rate of 30 mSv/h and no surface contamination.

Table 11-5. Radionuclide composition of a reference waste package of the type R.01.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	1.0E+11	Pu-239	1.0E+06
Cl-36	6.1E+04	Pu-240	2.0E+06
Ag-108m	6.1E+06	U-232	9.0E+01
Ba-133	1.0E+06	U-234	3.0E+03
Be-10	6.1E+01	U-235	6.0E+01
C-14	3.0E+08	U-236	9.0E+02
Fe-55	1.0E+11	U-238	1.2E+03
H-3	1.0E+07	Np-237	1.2E+03
Ho-166m	4.0E+05	Pu-238	5.0E+06
Mo-93	1.0E+05	Pu-241	3.0E+08
Nb-93m	1.0E+08	Pu-242	9.0E+03

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Nb-94	1.0E+06	Am-241	1.7E+06
Ni-59	1.0E+08	Am-242m	3.0E+04
Ni-63	8.1E+09	Am-243	9.0E+04
Sb-125	9.7E+09	Cm-243	6.0E+04
Zr-93	1.0E+05	Cm-244	2.0E+06
		Cm-245	9.0E+02
Correlated to Cs-137		Cm-246	2.4E+02
Cs-137	1.2E+10		
Cd-113m	7.2E+06		
Cs-134	7.4E+09		
Cs-135	1.2E+05		
Eu-152	8.3E+05		
Eu-154	1.2E+09		
Eu-155	8.3E+08		
I-129	3.6E+04		
Pd-107	1.2E+04		
Pm-147	1.1E+10		
Ru-106	2.7E+08		
Se-79	4.8E+04		
Sm-151	3.6E+07		
Sn-126	6.0E+03		
Sr-90	1.2E+09		
Tc-99	1.1E+07		

11.3 Uncertainties

- General uncertainties are described in Section 1.3.

12 R.10

12.1 Waste type description

12.1.1 Waste package

General

The R.10 waste type consists of 1.73 m^3 concrete moulds containing sludges in a concrete matrix. The raw waste material consists of ion-exchange resins and filter aids generated at the Ringhals nuclear power plant. The material is classified as intermediate level waste /Ahlqvist 1995, Johansson 2000, Triumf 2007/.

Following restrictions are always applicable on this waste:

- The general restrictions listed in Section 1.1.

Packaging

The mould is a cubic box made of reinforced concrete with the 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 is made of steel bars with a total surface of 7.9 m^2 and a weight of 160 kg in a 10-cm mould and 11.8 m^2 and 250 kg in a 25-cm mould. The 10-cm mould weighs approximately 1,600 kg and the 25-cm weighs about 3,100 kg. A stirrer made of carbon steel is included in the waste package. It weighs 16 kg and has an estimated surface of 1 m^2. To avoid cracking of the mould, due to expansion of the concrete matrix, a lining with compressible material (polyethene or polystyrene) is put inside the walls of the mould. The lining has a thickness of 20 mm and a total weight of 10 kg in a 10-cm mould and 5 mm and approximately 1 kg in a 25-cm mould. The variations between moulds are minimal.

The weight of a package including waste and is approximately 4,000 kg but differs between 2,700–4,500 kg. The maximum allowed weight of a waste package is 4,000 kg. some packages weighs more than 4,000 kg, but it is possible to handle packages up to 5,000 kg.

Treatment and conditioning

The waste is pumped into the mould and cement is added. If needed, water is added. When the cement is added the stirrer is started and the waste matrix is mixed for 60 minutes. The matrix is then allowed to harden for two days. Pouring concrete on top of the matrix then makes a lid. The concrete is allowed 24 hours to harden before the waste is transported to interim storage. The lid is at least 10 cm thick.

12.1.2 Materials – chemical composition

The amounts of different materials in a R.10 mould are given in Table 12-1. The waste is quite well defined and consists of sludges from different waste streams in Ringhals.

Table 12-1. Amounts of different material in waste type R.10 (per package)

Material	Weight (kg)	Weight (%)
Sludge	56–128	5–12
Cement*	980–1,120	88–95

* Including water 0.4 kg/kg cement.

12.1.3 Radionuclide inventory

Before the waste is transported from Ringhals NPP to SFR 1 a measurement of gamma emitting nuclides is performed. Dominating nuclides are Co-60 and Cs-137. The amounts of these nuclides are between 0 and $4 \cdot 10^{12}$ Bq for Co-60 and between 0 and $1 \cdot 10^{11}$ Bq for Cs-137. Other nuclides than gamma emitting ones must be calculated according to Appendix B.

The dose rate limit for this package is 100 mSv/h on the surface. Normal dose rate at 1 m varies between 0–9 mSv/h. The surface contamination should not exceed 40 kBq/m^2 for gamma and beta and 4 kBq/m^2 for alpha emitting nuclides on the package.

12.1.4 Waste production

Annual production is estimated to 2 per year, see Appendix A.

12.2 Reference waste type description

12.2.1 Waste package and material

Table 12-2 shows the content of different materials and the steel surface areas of the packaging in a reference package.

The assumed composition of the waste in a reference waste package of R.10 is given in Table 12-3.

The estimated volumes of the waste, the packaging materials and the void volume inside the package for a reference package are given in Table 12-4. The waste volume is estimated from the weights of the waste components in Table 12-3 and approximate densities of the different material. Average void is estimated to 10% in a package.

Void and waste volume is specified in Table 12-4.

Table 12-2. Estimated reference packaging composition and surface area and thickness of components in concrete mould of R.10.

Component	Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Concrete box	Concrete	1,340		100
Reinforcement	Steel	261	10	12
Steel lid	Steel	13	1.8	2
Stirrer	Steel	16	1	5
Other organics	Polyethene	10		

Table 12-3. Estimated reference waste composition and surface area and thickness of components in one of R.10.

Material	Weight (kg)
sludge	115
Cement*	1,120

* Including water 0.4 kg/kg cement.

Table 12-4. Volumes in one reference waste package of R.10.

	Volume per package (m^3)
Waste	0.90
Void	0.10

12.2.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 12-5.

A reference waste package of the type R.10 has a surface dose rate of 30 mSv/h and no surface contamination.

Table 12-5. Radionuclide composition of a reference waste package of the type R.10.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	3.1E+10	Pu-239	3.0E+05
Cl-36	1.8E+04	Pu-240	6.1E+05
Ag-108m	1.8E+06	U-232	2.7E+01
Ba-133	3.1E+05	U-234	9.1E+02
Be-10	1.8E+01	U-235	1.8E+01
C-14	9.2E+07	U-236	2.7E+02
Fe-55	3.1E+10	U-238	3.6E+02
H-3	3.1E+06	Np-237	3.6E+02
Ho-166m	1.2E+05	Pu-238	1.5E+06
Mo-93	3.1E+04	Pu-241	9.1E+07
Nb-93m	3.1E+07	Pu-242	2.7E+03
Nb-94	3.1E+05	Am-241	5.4E+05
Ni-59	3.1E+07	Am-242m	9.1E+03
Ni-63	2.5E+09	Am-243	2.7E+04
Sb-125	3.1E+09	Cm-243	1.8E+04
Zr-93	3.1E+04	Cm-244	5.9E+05
		Cm-245	2.7E+02
Correlated to Cs-137		Cm-246	7.3E+01
Cs-137	3.0E+08		
Cd-113m	1.8E+05		
Cs-134	2.5E+08		
Cs-135	3.0E+03		
Eu-152	2.1E+04		
Eu-154	3.0E+07		
Eu-155	2.1E+07		
I-129	9.0E+02		
Pd-107	3.0E+02		
Pm-147	2.7E+08		
Ru-106	1.5E+06		
Se-79	1.2E+03		
Sm-151	9.0E+05		
Sn-126	1.5E+02		
Sr-90	3.0E+07		
Tc-99	2.7E+05		

12.3 Uncertainties

- General uncertainties are described in Section 1.3.

13 R.15

13.1 Waste type description

13.1.1 Waste package

General

The R.15 waste type consists of 1.73 m^3 steel moulds containing ion-exchange resins and inert filter-aid in a concrete matrix. The raw waste material consists of ion-exchange resins and filter aids generated at the Ringhals nuclear power plant. The material is classified as intermediate level waste. All data comes from /Ahlqvist 1998, Johansson 2000, Triumf 2007/.

Following restrictions are always applicable on this waste:

- The general restrictions listed in Section 1.1.

Packaging

The mould is a cubic box made of steel with the dimensions 1.2×1.2×1.2 m. The wall is 5 mm thick, the bottom is 6 mm thick. The mould weighs approximately 425 kg and the surface area is approximately 20 m^2. The variations between moulds are minimal. No major change in design has been made since the start of production of this package.

The weight in total of a package including waste and is approximately 3,000 kg but differs between 2,600–3,300 kg. The maximum allowed weight of a waste package is 5,000 kg.

Treatment and conditioning

The waste, used ion-exchange resin, is pumped into the mould and cement is added. If needed, water is added. When the cement is added the stirrer is started and the waste matrix is mixed for 60 minutes. The matrix is then allowed to harden for two days. Pouring concrete on top of the matrix then makes a lid. The concrete is allowed 24 hours to harden before the waste is transported to interim storage. The lid is at least 10 cm thick.

13.1.2 Materials – chemical composition

The amounts of different materials in a R.15 mould are given in Table 13-1. The waste is well defined and consists of ion-exchange resins and filter-aid with radionuclides.

The resins are:

- Bead resin from PWR: cleaning of the primary reactor circuit (system 334), the fuel ponds (system 324), the waste water (system 342), water from the secondary side of the steam generators (system 417/337) and from drainage water.
- Bead resin from BWR: systems for cleaning reactor water (system 331), treatment of liquid waste (system 342), analytical ion-exchangers (system 336) and cleaning of fuel ponds (system 324).
- Powder resins from BWR: cleaning of condensate (system 332), fuel ponds (system 324) and treatment of liquid waste (system 324).
  The resins from liquid waste treatment contain filter-aids.

Resins from the PWR-reactors contain boric acid (H_3BO_3) up to 90 g/kg of resin. Some resins contain lithium. Resins from system 417/337 can contain ammonia and hydrazine.

The weight of the raw waste material is normally 130 kg of ion exchange resin per package but varies, see Table 8-1.

The matrix is made of cement. The brand used is Skövde standardcement. Nowadays also Degerham standardcement can be used. Silix GP and Sika AER was used as additives until 1997 (and a few exceptions during 1998), from 1997 to 1998 Hydrafix was used and from 1997 a special additive is used (brand name is kept on commercial secrecy by demand from Ringhals NPP).

Table 13-1. Amounts of different material in waste type R.15 (per package).

Material	Weight (kg)	Weight (%)
Ion-exchange resin	10–500	0.4–26.3
Cement*	1,400–2,800	99.6–73.7

* Including water 0.4 kg/kg cement.

13.1.3 Radionuclide inventory

Before the waste is transported from Ringhals NPP to SFR 1 a measurement of gamma emitting nuclides is performed. Dominating nuclides are Co-60 and Cs-137. The amounts of these nuclides are between 0 and $3 \cdot 10^{11}$ Bq for Co-60 and between 0 and $7 \cdot 10^{11}$ Bq for Cs-137. Other nuclides than gamma emitting ones must be calculated according to Appendix B.

Dose rate limit for this package is 100 mSv/h but normally it is between 0.1–10 mSv/h on 1 metre. The surface contamination should not exceed 40 kBq/m^2 for gamma and beta and 4 kBq/m^2 for alpha emitting nuclides. The waste packages are usually free of contamination.

13.1.4 Waste production

The waste type has been in production since 1991 and is still in production. Annual production is estimated to 1 packages per year, see Appendix A.

13.2 Reference waste type description

13.2.1 Waste package and material

Table 13-2 shows the content of different materials and the steel surface areas of the packaging in a reference package.

The assumed composition of the waste in a reference waste package of R.15 is given in Table 13-3.

The estimated volumes of the waste, the packaging materials and the void volume inside the package for a reference package are given in Table 13-4. The waste volume is estimated from the weights of the waste components in Table 13-3 and approximate densities of the different material. Average void is estimated to 5% in a package.

Void and waste volume is specified in Table 13-4.

Table 13-2. Estimated reference packaging composition and surface area and thickness of components in concrete mould of R.15.

Component	Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Steel box	Stainless steel	400	17	5
Stirrer	Stainless steel	25	3	2

Table 13-3. Estimated reference waste composition and surface area and thickness of components in one of R.15.

Material	Weight (kg)
Ion-exchange resin	250
Cement*	2,100

* Including water 0.4 kg/kg cement.

Table 13-4. Volumes in one reference waste package of R.15.

	Volume per package (m^3)
Waste	1.615
Void	0.085

13.2.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 13-5.

A reference waste package of the type R.15 has a surface dose rate of 100 mSv/h and no surface contamination.

Table 13-5. Radionuclide composition of a reference waste package of the type R.15.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	8.7E+10	Pu-239	8.7E+05
Cl-36	5.2E+04	Pu-240	1.7E+06
Ag-108m	5.2E+06	U-232	7.8E+01
Ba-133	8.7E+05	U-234	2.6E+03
Be-10	5.2E+01	U-235	5.2E+01
C-14	2.6E+08	U-236	7.8E+02
Fe-55	8.7E+10	U-238	1.0E+03
H-3	8.7E+06	Np-237	1.0E+03
Ho-166m	3.5E+05	Pu-238	4.4E+06
Mo-93	8.7E+04	Pu-241	2.6E+08
Nb-93m	8.7E+07	Pu-242	7.8E+03
Nb-94	8.7E+05	Am-241	1.5E+06
Ni-59	8.7E+07	Am-242m	2.6E+04
Ni-63	7.0E+09	Am-243	7.8E+04
Sb-125	8.7E+09	Cm-243	5.2E+04
Zr-93	8.7E+04	Cm-244	1.7E+06
		Cm-245	7.8E+02
Correlated to Cs-137		Cm-246	2.1E+02
Cs-137	1.5E+10		
Cd-113m	8.8E+06		
Cs-134	1.0E+10		
Cs-135	1.5E+05		
Eu-152	1.0E+06		

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Eu-154	1.5E+09		
Eu-155	1.03E+09		
I-129	4.4E+04		
Pd-107	1.5E+04		
Pm-147	1.3E+10		
Ru-106	7.4E+07		
Se-79	5.9E+04		
Sm-151	4.4E+07		
Sn-126	7.4E+03		
Sr-90	1.5E+09		
Tc-99	1.3E+07		

13.3 Uncertainties

- General uncertainties are described in Section 1.3.

14 R.23

14.1 Waste type description

14.1.1 Waste package

General

The R.23 waste type consists of 1.73 m^3 concrete or steel moulds with scrap metal and refuse in a concrete matrix. The raw waste material consists of mainly iron/steel, cellulose and other organics generated at the Ringhals nuclear power plant. The material is classified as intermediate level waste. All data comes from /Ahlqvist 1993, Johansson 2000, Triumf 2007/.

Following restrictions are always applicable on this waste:

- The general restrictions listed in Section 1.1.

Packaging

The steel mould has the dimension 1.2×1.2×1.2 m. The weight is 661 kg including lid and the total metal surface area of package is 23 m^2. The wall of the package is 5 mm thick, the bottom is 6 mm thick. The mould also has two lids and some reinforcements. The variations between moulds are minimal. Total weight of package including waste is approximately 2,700 kg but differs between 2,100–4,900 kg.

The concrete mould is a cubic box made of reinforced concrete with the dimensions 1.2×1.2×1.2 m. The walls are 10 cm thick. The reinforcement is made of 12 mm steel bars with a total surface of 10 m^2. The mould weighs approximately 1,600 kg. The variations between moulds are minimal. Total weight of package including waste is approximately 2,400 kg but differs between 2,200–4,100 kg.

In some special cases a concrete mould with 25 cm thick walls can be used.

The maximum allowed weight of a waste package is 5,000 kg.

Treatment and conditioning

Waste from the power plant is sorted in compactible and non-compactible. Non-compactible is put directly in a mould and the mould is then filled with concrete. Compactible waste is put in the mould and compacted. A 'middle-lid' is placed in the mould to prevent re-expansion of the waste. This operation is repeated until the mould is full. Then the mould is filled with concrete.

Pouring concrete on top of the package makes a lid.

14.1.2 Materials – chemical composition

The amounts of different materials in a R.23 mould are given in Table 14-1. The waste is fairly well defined and consists mainly of scrap metal and refuses as filters, wood, cloth, plastics and cables. The weight of the raw waste material is normally 770 kg waste per package but varies, see Table 14-1a and b.

The matrix is made of concrete. The brand used is of the standard Portland.

Table 14-1a. Amounts of different material in waste type R.23 steel mould (per package).

Material	Weight (kg)*	Area (m^2)	Weight (%)
Cellulose	44		2.7
Other organic material	100		6.2
Iron/steel	100	5.1	6.2
Aluminium	4	0.6	0.2
Concrete**	1,356		84.5

* Variation of amounts of material is not available in referred documents.

** Including 0.13 kg water/kg concrete.

Table 14-1b. Amounts of different material in waste type R.23 10-cm concrete mould (per package).

Material	Weight (kg)*	Area (m^2)	Weight (%)
Cellulose	11		1.8
Other organic material	25		4.0
Iron/steel	25	1.3	4.0
Aluminium	1	0.1	0.2
Concrete**	565		90

* Variation of amounts of material is not available in referred documents.

** Including 0.13 kg water/kg concrete.

14.1.3 Radionuclide inventory

Before the waste is transported from Ringhals to SFR 1 a measurement of gamma emitting nuclides is performed. Dominating nuclides are Co-60 and Cs-137. The amounts of these nuclides are between 0 and $6 \cdot 10^{11}$ Bq for Co-60 and between 0 and $7 \cdot 10^{10}$ Bq for Cs-137. Other nuclides than gamma emitting ones must be calculated according to Appendix B.

Dose rate limit for this package is 100 mSv/h but normally it is between 0.01–10 mSv/h on 1 metre. The surface contamination should not exceed 40 kBq/m^2 for gamma and beta and 4 kBq/m^2 for alpha emitting nuclides. The waste packages are usually free of contamination.

14.1.4 Waste production

The waste type has been in production since 1960 and is still in production. Annual production is estimated to 5 packages per year, see Appendix A.

14.2 Reference waste type description

14.2.1 Waste package and material

Table 14-2a and b shows the content of different materials and the steel surface areas of the packaging in the reference packages.

The assumed composition of the waste in the reference waste packages of R.23 is given in Table 14-3a and b.

The estimated volumes of the waste, the packaging materials and the void volume inside the package for a reference package are given in Table 14-4. The waste volume is estimated from the weights of the waste components in Table 14-3a and b and approximate densities of the different material. Average void is estimated to 5% in a package. Void and waste volume is specified in Table 14-4.

Table 14-2a. Estimated reference packaging composition, surface area and thickness of components in steel mould of R.23.

Component	Material	Weight (kg)	Surface area (m²)	Thickness (mm)
Steel box	Steel	478	14.4	5
Lid	Steel	183	8.6	6

Table 14-2b. Estimated reference packaging composition, surface area and thickness of components in concrete mould of R.23.

Component	Material	Weight (kg)	Surface area (m²)	Thickness (mm)
Concrete box	Concrete	1,340		10
Lid	Steel	13	1.8	2
Reinforcement	Steel	261	10	12

Table 14-3a. Estimated reference waste composition and surface area and thickness of components in one steel mould R.23.

Material	Weight (kg)	Surface area (m²)	Thickness (mm)
Cellulose	44		2.7
Other organic material	100		6.2
Iron/steel	100	5.1	6.2
Aluminium	4	0.6	0.2
Concrete*	1,356		84.5

* Including 0.13 kg water/kg concrete.

Table 14-3b. Estimated reference waste composition and surface area and thickness of components in one concrete mould R.23.

Material	Weight (kg)	Surface area (m²)	Thickness (mm)
Cellulose	11		1.8
Other organic material	25		4.0
Iron/steel	25	1.3	4.0
Aluminium	1	0.1	0.2
Concrete*	565		90

Table 14-4. Volumes in one reference waste package of R.23.

	Volume per package (m³)
Waste	1.615
Void	0.085

14.2.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 14-5.

Table 14-5a. Radionuclide composition of a reference waste package of the type R.23 in concrete moulds.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	1.6E+10	Pu-239	7.4E+04
Cl-36	9.9E+03	Pu-240	1.5E+05
Ag-108m	9.9E+05	U-232	6.6E+00
Ba-133	1.6E+05	U-234	2.2E+02
Be-10	9.9E+00	U-235	4.4E+00
C-14	4.9E+07	U-236	6.6E+01
Fe-55	1.6E+10	U-238	8.8E+01
H-3	1.6E+06	Np-237	8.8E+01
Ho-166m	6.6E+04	Pu-238	5.8E+05
Mo-93	1.6E+04	Pu-241	2.2E+07
Nb-93m	1.6E+07	Pu-242	6.6E+02
Nb-94	1.6E+05	Am-241	5.5E+05
Ni-59	1.6E+07	Am-242m	2.2E+03
Ni-63	1.3E+09	Am-243	6.6E+03
Sb-125	1.7E+09	Cm-243	4.4E+03
Zr-93	1.6E+04	Cm-244	1.5E+06
		Cm-245	6.6E+01
Correlated to Cs-137		Cm-246	1.8E+01
Cs-137	6.1E+08		
Cd-113m	3.6E+05		
Cs-134	3.7E+08		
Cs-135	6.1E+03		
Eu-152	4.2E+04		
Eu-154	6.1E+07		
Eu-155	4.2E+07		
I-129	1.8E+03		
Pd-107	6.1E+02		
Pm-147	5.5E+08		
Ru-106	2.4E+08		
Se-79	2.4E+03		
Sm-151	1.8E+06		
Sn-126	3.0E+02		
Sr-90	6.1E+07		
Tc-99	5.5E+05		

Table 14-5b. Radionuclide composition of a reference waste package of the type R.23 in steel moulds.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	4.0E+10	Pu-239	1.8E+04
Cl-36	2.4E+04	Pu-240	3.4E+05
Ag-108m	2.4E+06	U-232	1.7E+01
Ba-133	4.0E+05	U-234	5.5E+02
Be-10	2.4E+01	U-235	1.1E+01

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
C-14	1.2E+08	U-236	1.7E+02
Fe-55	4.0E+10	U-238	2.2E+02
H-3	4.0E+06	Np-237	2.2E+02
Ho-166m	1.6E+05	Pu-238	1.4E+06
Mo-93	4.0E+04	Pu-241	5.5E+07
Nb-93m	4.0E+07	Pu-242	1.7E+03
Nb-94	4.0E+05	Am-241	1.4E+06
Ni-59	4.0E+07	Am-242m	5.5E+03
Ni-63	3.2E+09	Am-243	1.7E+04
Sb-125	4.0E+09	Cm-243	1.1E+04
Zr-93	4.0E+04	Cm-244	3.8E+06
		Cm-245	1.7E+02
Correlated to Cs-137		Cm-246	4.4E+01
Cs-137	3.0E+08		
Cd-113m	1.8E+05		
Cs-134	3.0E+08		
Cs-135	3.0E+03		
Eu-152	2.1E+04		
Eu-154	3.0E+07		
Eu-155	2.1E+07		
I-129	9.1E+02		
Pd-107	3.0E+02		
Pm-147	2.7E+08		
Ru-106	1.5E+06		
Se-79	1.2E+03		
Sm-151	9.1E+05		
Sn-126	1.5E+02		
Sr-90	3.0E+07		
Tc-99	2.7E+05		

A reference waste package of the type R.23 has a surface dose rate of 100 mSv/h and no surface contamination.

14.3 Uncertainties

- General uncertainties are described in Section 1.3.
- All future production of R.23 is assumed to be in steel moulds. Some packages may be of the concrete kind.

15 R.29

15.1 Waste type description

15.1.1 Waste package

General

The R.29 waste type consist of 1.73 m^3 concrete moulds with evaporator concentrate in a concrete matrix. It is generated at Ringhals nuclear power plant. The material is classified as intermediate level waste. All data comes from /Einarsson 2006, Triumf 2007/.

Following restrictions are always applicable on this waste:

- The general restrictions listed in Section 1.1.

Packaging

The concrete mould has the dimension 1.2×1.2×1.2 m. The weight is 1,600 kg including lid. The walls of the package are 10 cm thick. The mould weighs approximately 1,600 kg. The variations between moulds are minimal. Total weight of package including waste is approximately 4,000 kg.

The maximum allowed weight of a waste package is 5,000 kg.

Treatment and conditioning

The waste is first processed in an evaporator, put through a process that oxidises the organic matter, then it is dried further to a substance with a dry matter content of 10–20%. Then the waste is put into the concrete mould and later filled with concrete.

15.1.2 Materials – chemical composition

The amounts of different materials in a R.29 mould are given in Table 15-1. The weight of the raw waste material is normally 700 kg waste per package but varies between 350 and 1,000 kg.

The matrix is made of concrete.

Table 15-1. Amounts of different material in waste type R.29 (per package).

Material	Weight (kg)*	Area (m^2)	Weight (%)
Cement	3,200		79
Evaporator concentrate	700		17
Iron/steel	160	7.9	4

15.1.3 Radionuclide inventory

Before the waste is transported from Ringhals to SFR 1 a measurement of gamma emitting nuclides is performed. Dominating nuclide is Co-60. Other nuclides than gamma emitting ones must be calculated according to Appendix B.

Dose rate limit for this package is 100 mSv/h. The surface contamination should not exceed 40 kBq/m^2 for gamma and beta and 4 kBq/m^2 for alpha emitting nuclides.

15.1.4 Waste production

The waste type is estimated to start production in 2008. Annual production is estimated to 30 packages per year, see Appendix A.

15.2 Reference waste type description

15.2.1 Waste package and material

Annual production in estimated to 30 waste packages. Table 15-2 shows the content of different materials and the steel surface areas of the packaging in the reference packages.

The assumed composition of the waste in the reference waste packages of R.29 is given in Table 15-3.

The estimated volumes of the waste, the packaging materials and the void volume inside the package for a reference package are given in Table 15-4. The waste volume is estimated from the weights of the waste components in Table 15-3 and approximate densities of the different material. Average void is estimated to 10% in a package.

Table 15-2. Estimated reference packaging composition, surface area and thickness of components in R.29.

Component	Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Concrete mould	Concrete	1,600		100
Reinforcement	Steel	160	7.9	

Table 15-3. Estimated reference waste composition in one R.29.

Material	Weight (kg)
Evaporator concentrate	700
Concrete*	1,600

* Including 0.13 kg water/kg concrete.

Table 15-4. Volumes in one reference waste packages of R.29.

	Volume per package (m^3)
Waste	1.53
Void	0.17

15.2.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 15-5.

A reference waste package of the type R.29 has a surface dose rate of 100 mSv/h.

Table 15-5. Radionuclide composition of a reference waste package of the type R.29.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	2.2E+08	Pu-239	2.0E+03
Cl-36	1.3E+02	Pu-240	4.0E+03
Ag-108m	1.3E+04	U-232	1.8E–01
Ba-133	2.2E+03	U-234	6.0E+00
Be-10	1.3E–01	U-235	1.2E–01
C-14	6.6E+05	U-236	1.8E+00
Fe-55	2.2E+08	U-238	2.4E+00
H-3	2.2E+04	Np-237	2.4E+00
Ho-166m	8.8E+02	Pu-238	2.4E+04
Mo-93	2.2E+02	Pu-241	6.0E+05
Nb-93m	2.2E+05	Pu-242	1.8E+01
Nb-94	2.2E+03	Am-241	6.0E+03
Ni-59	2.2E+05	Am-242m	6.0E+01
Ni-63	1.8E+07	Am-243	1.8E+02
Sb-125	2.2E+07	Cm-243	1.2E+02
Zr-93	2.2E+02	Cm-244	1.8E+04
		Cm-245	1.8E+00
Correlated to Cs-137		Cm-246	4.8E–01
Cs-137	2.7E+07		
Cd-113m	1.6E+04		
Cs-134	2.7E+07		
Cs-135	2.7E+02		
Eu-152	1.9E+03		
Eu-154	2.7E+06		
Eu-155	1.9E+06		
I-129	8.1E+01		
Pd-107	2.7E+01		
Pm-147	2.4E+07		
Ru-106	1.4E+05		
Se-79	1.1E+02		
Sm-151	8.1E+04		
Sn-126	1.4E+01		
Sr-90	2.7E+06		
Tc-99	2.4E+04		

15.3 Uncertainties

- General uncertainties are described in Section 1.3.

16 S.09

16.1 Waste type description

16.1.1 Waste package

General

The S.09 waste type consists of a standard 200-litre drum containing sludges in a cement matrix. The raw waste material consists sludges generated at the Studsvik research centre. The material is classified as intermediate level waste. All data comes from /Chyssler 2000, Johansson 2000, Studsvik 2000/.

The physical properties and chemical conditions are well known. The following restrictions are always applicable:

- The general restrictions listed in Section 1.1.
- The content of complexing agents should be well known.

Packaging

The steel drum is made of stainless steel with diameter 0.57 m and 0.84 m in height. The wall thickness is 1.5 mm. The drum weighs approximately 60 kg and surface area is approximately 5.1 m^2. To facilitate the handling in SFR the drums are placed four by four on a steel plate with the dimension 1.2 m ·1.2 m and the thickness of 4 mm. The weight of the steel plate is 66 kg with surface area of 4.0 m^2. The variations between packages are minimal. No major changes in design have been made since the start of production of this package.

The weight in total of a package including waste and is approximately 350 kg. The maximum allowed weight of a waste package is 500 kg.

Treatment and conditioning

The sludges containing radionuclides are mixed with concrete in a custom-made facility at the Studsvik site. Cement is added in the drum and then sludge and water is pumped into the drum.

When the waste is added the stirrer is started and the waste matrix is mixed. The matrix is then allowed to harden for one day. After measurements a steel lid is placed on top of the drum. The drums are placed on the steel plate when the drums are sent to SFR.

The void varies but is estimated to 0.01 m^3 in a drum.

16.1.2 Materials – chemical composition

The average amounts of different materials in an ‘typical package’ are given in Table 16-1. Waste matrix contains sludge (with dry weight 3–5%) from cleaning up system for Studsvik radioactive waste water treatment. Main compounds in the sludge is ferric hydroxide, cupper ferric cyanide, this waste could also contain some complexing agents originating form the laundry.

Most important from laundry is the organic chelating compounds EDTA and gluconic acid. The weight of the raw waste material is normally 5 kg sludge per package, see Table 16-1. The matrix is made of concrete. No cement additives are used.

Table 16-1. Amounts of different material in waste type S.09 (per package).

Material	Weight (kg)*	Weight (%)
Sludge	5	1.8
Concrete	280	98.2

* No data concerning variations is available.

16.1.3 Radionuclide inventory

Before the waste is solidified a measurement of gamma emitting nuclides is performed. Dominating nuclides are Co-60 and Cs-137. The amounts of these nuclides are between 0 and $2 \cdot 10^{10}$ Bq for Co-60 and between 0 and $4 \cdot 10^{9}$ Bq for Cs-137. Other nuclides than gamma emitting ones must be calculated according to Appendix B.

The dose rate limit for this package is 50 mSv/h. No estimation on variation of dose rates are available in referred documents. The surface contamination should not exceed 40 kBq/m^2 for gamma and beta and 4 kBq/m^2 for alpha emitting nuclides. The waste packages are usually free of contamination.

16.1.4 Waste production

Annual production is estimated to 20 packages, see Appendix A.

16.2 Reference waste type description

16.2.1 Waste package and material

Table 16-2 shows the content of different materials and the surface areas of the packaging in S.09 package from Studsvik. Since the steel plate and expansion box is shared between four drums ¼ of the plate and box is included in a package of S.09.

The assumed composition of the waste in a reference waste package of S.09 is given in Table 16-3.

Void and waste volume is specified in Table 16-4. Average void is estimated to 0.01 m^3 in a steel drum.

Table 16-2. Estimated reference packaging composition and surface area and thickness of components in each steel drum and one ¼ of a steel plate and expansion box of S.09.

Component	Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Steel drum	steel	60	5.1	1.5
Steel plate	steel	16.6	1.0	4

Table 16-3. Estimated reference waste composition in one steel drum of S.09.

Material	Weight (kg)
Sludge	5
Cement*	280

* Including water 0.4 kg/kg cement.

Table 16-4. Volumes in one reference waste package of steel drums including ¼ steel plate and expansion box per drum of S.09.

	Volume per package (m^3)
Waste	0.2
Void	0.01

16.2.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 16-5.

A reference waste package of the type S.09 has a surface dose rate of 50 mSv/h and no surface contamination.

Table 16-5. Radionuclide composition of a reference waste package of the type S.09.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	1.8E+08	Pu-239	1.1E+04
Cl-36	6.0E+04	Pu-240	2.2E+04
Ag-108m	1.1E+04	U-232	9.7E–01
Ba-133	1.8E+03	U-234	3.2E+01
Be-10	1.1E–01	U-235	6.5E–01
C-14	1.8E+05	U-236	9.7E+00
Fe-55	1.8E+08	U-238	1.3E+01
H-3	1.8E+04	Np-237	1.3E+01
Ho-166m	7.3E+03	Pu-238	1.3E+05
Mo-93	9.1E+03	Pu-241	3.2E+06
Nb-93m	1.8E+05	Pu-242	9.7E+01
Nb-94	1.8E+03	Am-241	3.2E+04
Ni-59	1.8E+05	Am-242m	3.2E+02
Ni-63	3.6E+07	Am-243	9.7E+02
Sb-125	1.8E+07	Cm-243	6.5E+02
Zr-93	1.8E+03	Cm-244	9.7E+04
		Cm-245	9.7E+00
Correlated to Cs-137		Cm-246	2.6E+00
Cs-137	6.0E+09		
Cd-113m	3.6E+06		
Cs-134	6.0E+09		
Cs-135	3.0E+04		
Eu-152	4.2E+05		
Eu-154	6.0E+08		
Eu-155	4.2E+08		
I-129	1.8E+03		
Pd-107	6.0E+03		
Pm-147	5.4E+09		
Ru-106	3.0E+07		

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Se-79	2.4E+04		
Sm-151	1.8E+07		
Sn-126	3.0E+03		
Sr-90	6.0E+08		
Tc-99	3.0E+07		

16.3 Uncertainties

- General uncertainties are described in Section 1.3.

17 S.23

17.1 Waste type description

17.1.1 Waste package

General

The S.23 waste type consists of 1.73 m^3 concrete moulds with scrap metal and refuse in a concrete matrix. The raw waste material consists of mainly iron/steel, cellulose and other organics generated at the Studsvik research site. The material is classified as intermediate level waste. All data comes from /Chyssler 1997, Johansson 2000, Triumf 2007/.

Following restrictions are always applicable on this waste:

- The general restrictions listed in Section 1.1.

Packaging

The concrete mould is a cubic box made of reinforced concrete with the dimensions 1.2×1.2×1.2 m. The walls are 10 cm thick. The reinforcement is made of 12 mm steel bars with a total surface of 10 m^2. The 10-cm mould weighs approximately 1,600 kg. The variations between moulds are minimal. Total weight of package including waste is approximately 3,500 kg but differs between 2,900–4,900 kg.

The maximum allowed weight of a waste package is 5,000 kg.

Treatment and conditioning

Waste from the power plant is sorted in compactable and non-compactable. Non-compactable is put directly in a mould and the mould is then filled with concrete. Compactable waste is put in the mould and compacted. A 'middle-lid' is placed in the mould to prevent re-expansion of the waste. This operation is repeated until the mould is full. Then the mould is filled with concrete, which is allowed to harden. Pouring concrete on top of the package makes a lid.

17.1.2 Materials – chemical composition

The amounts of different materials in a S.23 mould are given in Table 17-1. The waste is fairly well defined and consists mainly of scrap metal and refuses as filters, wood, cloth, plastics and cables. The weight of the raw waste material is normally 780 kg waste per package but varies, see Table 17-1.

The matrix is made of concrete. The brand used is Betokem EXM 4PA.

Table 17-1. Amounts of different material in waste type S.23 10-cm concrete mould (per package).

Material	Weight (kg)*	Area (m^2)	Weight (%)
Sludge	3.8		0.5
Cellulose	29		3.4
Iron/steel	113	5.8	13.2
Aluminium	3.8	0.6	0.5
Other inorganic material	139		16.3
Concrete**	565		66.2

* Variation of amounts of material is not available in referred documents.

** Including 0.13 kg water/kg concrete.

17.1.3 Radionuclide inventory

Before the waste is transported from Studsvik to SFR 1 a measurement of gamma emitting nuclides is performed. Dominating nuclides are Co-60 and Cs-137. Other nuclides than gamma emitting ones must be calculated according to Appendix B.

Dose rate limit for this package is 100 mSv/h, information on variations are not available. The surface contamination should not exceed 40 kBq/m^2 for gamma and beta and 4 kBq/m^2 for alpha emitting nuclides. The waste packages are usually free of contamination.

17.1.4 Waste production

Annual production is estimated to 10 packages, see Appendix A.

17.2 Reference waste type description

17.2.1 Waste package and material

Table 17-2 shows the content of different materials and the steel surface areas of the packaging in a reference package.

The assumed composition of the waste in a reference waste package of S.23 is given in Table 17-3.

The estimated volumes of the waste, the packaging materials and the void volume inside the package for a reference package are given in Table 17-4. The waste volume is estimated from the weights of the waste components in Table 17-3 and approximate densities of the different material. Average void is estimated to 5% in a package. Void and waste volume is specified in Table 17-4.

Table 17-2. Estimated reference packaging composition, surface area and thickness of components in steel mould of S.23.

Component	Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Concrete box	Concrete	1,340		10
Lid	Steel	13	1.8	2
Reinforcement	Steel	261	10	12

Table 17-3. Estimated reference waste composition and surface area and thickness of components in one S.23.

Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Sludge	3.8		0.5
Cellulose	29		3.4
Iron/steel	113	5.8	13.2
Aluminium	3.8	0.6	0.5
Other inorganic material	139		16.3
Concrete*	565		66.2

* Including 0.13 kg water/kg concrete.

Table 17-4. Volumes in one reference waste package of S.23.

	Volume per package (m^3)
Waste	0.95
Void	0.05

17.2.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 17-5. Inventory based on data for B.12, F.12 and R.12.

A reference waste package of the type S.23 has a surface dose rate of 100 mSv/h and no surface contamination.

Table 17-5. Radionuclide composition of a reference waste package of the type S.23.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	1.7E+10	Pu-239	1.5E+06
Cl-36	9.4E+03	Pu-240	3.1E+06
Ag-108m	1.0E+06	U-232	1.4E+02
Ba-133	1.7E+05	U-234	4.6E+03
Be-10	1.0E+01	U-235	9.2E+01
C-14	1.7E+07	U-236	1.4E+03
Fe-55	1.7E+10	U-238	1.8E+03
H-3	1.7E+06	Np-237	1.8E+03
Ho-166m	6.8E+05	Pu-238	1.8E+07
Mo-93	8.5E+05	Pu-241	4.6E+08
Nb-93m	1.7E+07	Pu-242	1.4E+04
Nb-94	1.7E+05	Am-241	4.6E+06
Ni-59	1.7E+07	Am-242m	4.6E+04
Ni-63	3.4E+09	Am-243	1.4E+05
Sb-125	1.7E+09	Cm-243	9.2E+04
Zr-93	1.7E+05	Cm-244	1.4E+07
		Cm-245	1.4E+03
Correlated to Cs-137		Cm-246	3.7E+02
Cs-137	9.4E+08		
Cd-113m	5.6E+05		
Cs-134	9.4E+08		
Cs-135	4.7E+03		
Eu-152	6.6E+04		
Eu-154	9.4E+07		
Eu-155	6.6E+07		
I-129	2.8E+02		
Pd-107	9.4E+02		
Pm-147	8.5E+08		

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Ru-106	4.7E+06		
Se-79	3.8E+03		
Sm-151	2.8E+06		
Sn-126	4.7E+02		
Sr-90	9.4E+07		
Tc-99	4.7E+06		

17.3 Uncertainties

- General uncertainties are described in Section 1.3.

18 F.99:1

18.1 Waste type description

18.1.1 Waste package

General

The F.99:1 waste type consists of 1.73 m^3 steel moulds containing steel drums with a rest slurry from oxidation tests conducted at Forsmark nuclear power plant. The material is classified as intermediate level waste. All data comes from /MAAS 6/94, Triumf 2007/.

Following restrictions are always applicable on this waste:

- The general restrictions listed in Section 1.1.

Packaging

The steel mould has the dimension 1.2×1.2×1.2 m. The weight is 661 kg including lid and the total metal surface area of package is 23 m^2. The wall of the package is 5 mm thick, the bottom is 6 mm thick.

The steel drums are made of stainless steel with diameter 0.595 m and 0.882 m in height. The thickness of the material is 1.2 mm. The drum weighs approximately 23 kg and the surface area is approximately 3 m^2.

The maximum allowed weight of a waste package is 5,000 kg.

Treatment and conditioning

The slurry is mixed with cement in a custom made facility in the Forsmark NPP: After filling, the drums are topped with a small amount of cement. The void varies but is estimated to 0.02–0.03 m^3 in a drum.

18.1.2 Materials – chemical composition

The amounts of different materials in a F.99:1 mould are given in Table 18-1.

Table 18-1. Amounts of different material in waste type F.99:1 (per package).

Material	Weight (kg)	Area (m^2)
Other organic material	35	
Iron/steel	225	11

18.1.3 Radionuclide inventory

Before the waste is transported from Forsmark to SFR 1 a measurement of gamma emitting nuclides is performed. Dominating nuclide is Co-60. Other nuclides than gamma emitting ones must be calculated according to Appendix B.

Dose rate limit for this package is 100 mSv/h. The surface contamination should not exceed 40 kBq/m^2 for gamma and beta and 4 kBq/m^2 for alpha emitting nuclides. The waste packages are usually free of contamination.

18.1.4 Waste production

The waste type has been in production since 1987. No further production is foreseen, see Appendix A.

18.2 Reference waste type description

18.2.1 Waste package and material

Table 18-2 shows the content of different materials and the steel surface areas of the packaging in the reference packages.

The estimated volumes of the waste, the packaging materials and the void volume inside the package for a reference package are given in Table 18-3. Average void is estimated to 10% in a package.

Table 18-2. Estimated reference packaging composition, surface area and thickness of components in F.99:1.

Component	Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Steel mould	Stainless steel	400	10	5

Table 18-3. Volumes in one reference waste packages of F.99:1.

	Volume per package (m^3)
Waste	1.615
Void	0.085

18.2.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 18-4.

A reference waste package of the type F.99:1 has a maximum surface dose rate of 100 mSv/h and no surface contamination.

Table 18-4. Radionuclide composition of a reference waste package of the type F.99:1.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	7.0E+08	Pu-239	1.1E+03
Cl-36	4.2E+02	Pu-240	2.3E+03
Ag-108m	4.2E+04	U-232	1.0E–01
Ba-133	7.0E+03	U-234	3.4E+00
Be-10	4.2E–01	U-235	1.3E+01
C-14	2.1E+06	U-236	1.0E+00
Fe-55	7.0E+08	U-238	1.5E+01
H-3	7.0E+04	Np-237	1.2E+02
Ho-166m	2.8E+03	Pu-238	6.6E+03
Mo-93	7.0E+02	Pu-241	3.4E+05

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Nb-93m	7.0E+05	Pu-242	1.0E+01
Nb-94	7.0E+03	Am-241	1.9E+03
Ni-59	7.0E+05	Am-242m	3.4E+01
Ni-63	5.6E+07	Am-243	1.0E+02
Sb-125	7.0E+07	Cm-243	3.5E+01
Zr-93	7.0E+02	Cm-244	6.2E+03
		Cm-245	1.0E+00
Correlated to Cs-137		Cm-246	2.7E–01
Cs-137	1.6E+08		
Cd-113m	9.9E+04		
Cs-134	1.0E+07		
Cs-135	1.6E+03		
Eu-152	1.2E+04		
Eu-154	1.6E+07		
Eu-155	1.2E+07		
I-129	4.9E+02		
Pd-107	1.6E+02		
Pm-147	1.5E+08		
Ru-106	8.2E+05		
Se-79	6.6E+02		
Sm-151	4.9E+05		
Sn-126	8.2E+01		
Sr-90	1.6E+07		
Tc-99	1.5E+05		

18.3 Uncertainties

- General uncertainties are described in Section 1.3.

19 Other waste types

Since SFR 1 still is in operation there is always the possibility that new waste types are needed. It is not possible to define these types per se.

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Appendix F

Waste types in the Silo

1 General

1.1 Waste in the Silo

In SFR, in the Silo – Silo for intermediate level waste – short-lived wastes from the nuclear power plants and the Studsvik research site are disposed off. The Silo consists of a cylindrical concrete construction with shafts of different sizes for waste packages. The biggest shafts are 2.5 m by 2.5 m. The Silo is operated with remote controlled equipment for waste handling. The Silo is approximately 50 m high and has a diameter of approximately 30 m. The Silo has been designed to handle 80–90% of the activity in SFR 1 and hence has the most efficient barriers in the repository. The Silo is placed in a low flow area of the rock, there is approximately 1 m of bentonite clay filling between the rock and the concrete construction and finally the void between the construction and the waste packages are back-filled with concrete. All this will ensure that after closure of the facility the transport of radionuclides will be very small.

The waste that is intended in the Silo comes from many different waste streams but the most important one is ion exchange resins from the nuclear power plants. Other waste like metal components of different origin is also disposed off in Silo. The amount of organic material is kept to a minimum.

Dose rates allowed on a package is maximum 500 mSv/h on the surface. The dominating nuclides are Co-60 and Cs-137. Most of the transuranic elements in SFR 1 are following the waste streams to the Silo.

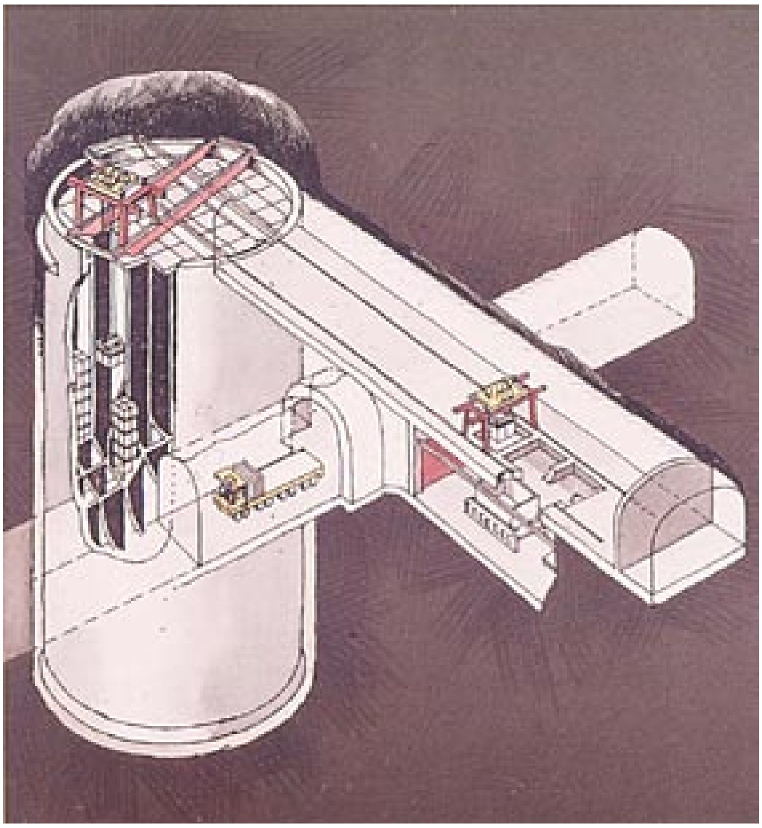

Figure 1-1. *The Silo and some adjacent tunnels.*

Other criteria for acceptance are:

- Construction, geometry, dimensions and weight must be suited to the handling system in SFR and the transportation system in general and in
  the Silo in particular. The strength of the packages must be sufficient to withstand normal and abnormal handling.
- Each package must have an individual identity and the radionuclide content of gamma emitting nuclides shall be known.
- Surface contamination should not exceed 40 kBq/m^2 for gamma and beta emitting nuclides and 4 kBq/m^2 for alpha emitting nuclides.
- The internal dose rates and integrated dose must not affect the barrier properties in the repository.
- The chemical and physical properties and the structure of the waste should be known.
- Chemical aggressive or explosive material, pressurised gas and free liquid are not allowed.
- Gas production from the waste must not be too big.
- The waste shall have enough resistance towards corrosion that the integrity of the package is kept until sealing of the repository.
- Burnable waste must not be subject to self-ignition.
- Complexing agents should be avoided.

All waste in the Silo is normally packed in concrete or steel moulds, other packages that are used are steel drums.

1.2 Definitions

- Surface area is defined as the area subject to anaerobic corrosion and gas production after sealing of the repository.

1.3 Uncertainties

- The presented waste types describe the types that have been, are in or can be foreseen to be produced in Sweden. New waste types may be present in the future.
- Almost all data is based on literature values, smaller changes may not been properly documented.
- All future production is based on a relative simple prognosis model. This model, described in Appendix A, was chosen since the uncertainties about processes etc are large.
- No regard has been taken to improvements in processes etc.
- A reference package has been chosen. Actual packages may differ considerably from this reference package, the uncertainty for the whole population of packages is judged to be smaller.
- For the radionuclide inventory for each waste package the amount presented is either based on the amount received from correlation factors or actual measurements when applicable. When there is a difference between different sub types of waste packages within one waste type the inventory presented in this appendix is the most conservative one.
- Void presented in this report is based on best estimates; the estimates can be quite crude.
- Tc-99 is correlated against both Co-60 and Cs-137 when quantifying the inventories, according to Appendix B. In the tables "Radionuclide composition of a reference waste package" however Tc-99 is presented as only correlated towards Cs-137.

2 B.04

2.1 Waste type description

2.1.1 Waste package

General

The B.04 waste type consists of a standard 200-litre drum containing ion exchange resins in a cement matrix. The raw waste material consists of ion exchange resins generated at the Barsebäck nuclear power plant. The material is classified as intermediate level waste. All data comes from /Jönsson 2006, Johansson 2000, Triumf 2007/.

The physical properties and chemical conditions are well known. The following restrictions are always applicable:

- The general restrictions listed in Section 1.1.
- Since the package contains burnable substances the restrictions regarding self-ignition etc are of special importance.

Packaging

The steel drum is made of stainless steel with diameter 0.590 m and 0.880 m in height. The thickness of the material is 1.2 mm. The drum weighs approximately 21 kg and the surface area is approximately 3 m^2. The weight in total of a package including waste is approximately 375 kg. Maximum allowed weight of a waste package is 500 kg.

Treatment and conditioning

The ion exchange resin is mixed with cement in a custom made facility in the Barsebäck NPP. After filling, a steel lid is placed on the drum. The void is approximately 10%.

2.1.2 Materials – chemical composition

The amounts of different materials in a B.04 are given in Table 2-1.

The waste is well defined and consists of ion exchange resins with radionuclides from cleaning systems (system 331, 352, 482 and 324). The systems contains both bead and power resins. The weight of the raw waste material is normally 33 kg ion exchange resin per package but varies.

Table 2-1. Amounts of different material in waste type B.04 (per package).

Material	Weight (kg)	Weight (%)
Ion-exchange resin	33	9
Iron/steel	40	11
Cement	302	80

2.1.3 Radionuclide inventory

Before the waste is transported from Barsebäck NPP to SFR 1 a measurement of gamma emitting nuclides is performed. Dominating nuclides are Co-60 and Cs-137. Other nuclides than gamma emitting ones must be calculated according to Appendix B.

The dose rate limit for this package is 500 mSv/h. The surface contamination should not exceed 40 kBq/m^2 for gamma and beta and 4 kBq/m^2 for alpha emitting nuclides. The waste packages are usually free of contamination.

2.1.4 Waste production

No further production is foreseen, see Appendix A.

2.2 Reference waste type description

2.2.1 Waste package and material

Table 2-2 shows the content of different materials and the surface areas of the packaging in B.04 package from Barsebäck.

The assumed composition of the waste in a reference waste package of B.04 is given in Table 2-3.

Void and waste volume is specified in Table 2-4.

Average void is estimated to 10% in a steel drum.

Table 2-2. Estimated reference packaging composition and surface area and thickness of components in each steel drum of B.04.

Component	Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Steel drum	Stainless steel	21	3	1.2
Stirrer	Stainless steel	10		

Table 2-3. Estimated reference waste composition in one steel drum of B.04.

Material	Weight (kg)
Ion-exchange resin	33

Table 2-4. Volumes in one reference waste packages of B.04.

	Volume one package (m^3)
Waste	0.19
Void	0.02

2.5.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 2-5.

A reference waste package of the type B.04 has a surface dose rate of 100 mSv/h and no surface contamination.

Table 2-5. Radionuclide composition of a reference waste package of the type B.04.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	2.0E+10	Pu-239	1.7E+04
Cl-36	4.1E+04	Pu-240	1.7E+04
Ag-108m	1.2E+06	U-232	5.0E–01
Ba-133	2.0E+05	U-234	1.7E+01
Be-10	1.2E+01	U-235	3.3E–01
C-14	2.0E+07	U-236	5.0E+00
Fe-55	2.0E+10	U-238	6.6E+00
H-3	2.0E+06	Np-237	6.6E+00
Ho-166m	7.9E+05	Pu-238	6.6E+04
Mo-93	9.9E+05	Pu-241	1.7E+06
Nb-93m	2.0E+07	Pu-242	5.0E+01
Nb-94	2.0E+05	Am-241	1.7E+04
Ni-59	2.0E+07	Am-242m	1.7E+02
Ni-63	3.9E+09	Am-243	5.0E+02
Sb-125	2.0E+09	Cm-243	3.3E+02
Zr-93	2.0E+05	Cm-244	5.0E+04
		Cm-245	5.0E+00
Correlated to Cs-137		Cm-246	1.3E+00
Cs-137	4.1E+09		
Cd-113m	2.5E+05		
Cs-134	4.1E+09		
Cs-135	2.1E+04		
Eu-152	2.9E+05		
Eu-154	4.1E+08		
Eu-155	2.9E+08		
I-129	1.2E+03		
Pd-107	4.1E+03		
Pm-147	3.7E+09		
Ru-106	2.1E+07		
Se-79	1.7E+04		
Sm-151	1.2E+07		
Sn-126	2.1E+03		
Sr-90	4.1E+08		
Tc-99	2.1E+07		

2.3 Uncertainties

- General uncertainties are described in Section 1.3.

3 B.06

3.1 Waste type description

3.1.1 Waste package

General

The B.06 waste type consists of a standard 200-litre drum containing ion exchange resins in a bitumen matrix. The raw waste material consists of ion exchange resins and sludges generated at the Barsebäck nuclear power plant. The material is classified as intermediate level waste. All data comes from /Roth 1996, Johansson 2000, Triumf 2007/.

The physical properties and chemical conditions are well known. The following restrictions are always applicable:

- The general restrictions listed in Section 1.1.
- Since the package contains burnable substances the restrictions regarding self-ignition etc are of special importance.

Packaging

The steel drum is made of stainless steel with diameter 0.595 m and 0.882 m in height. The thickness of the material is 1.2 mm. The drum weighs approximately 23 kg and the surface area is approximately 3 m^2. To facilitate the handling in SFR the drums are placed four by four on a steel plate with the dimension 1.2×1.2 m and the thickness of 0.004 m. Between the four drums a expansion box is placed to provide expansion volume in case of swelling of the bitumen matrix. The box is made of steel and has a weight of 20 kg, the walls are 6 mm thick and the whole box has a surface area of 0.8 m^2. The weight of the steel plate is 49 kg with surface area of 2.9 m^2. The variations between packages are minimal. No major changes in design have been made since the start of production of this package except that since 1985 the drums are made of stainless steel. The weight in total of a package including waste and is approximately 200 kg. The maximum allowed weight of a waste package is 500 kg.

Treatment and conditioning

The ion exchange resin containing metal hydroxides and other substances are mixed with bitumen in a custom made facility in the Barsebäck NPP. The flow in the bitumenisation process is measured with electrical conductivity. A small amount of Na_2SO_4 is mixed with the product to give correct measurements of the conductivity. Emulgator is added in order to make the product more homogene. After filling, a steel lid is placed upon the drum. The drums are placed on the steel plate when they are sent to SFR.

The void varies but is estimated to 0.03 m^3 in a drum (not including expansion box).

3.1.2 Materials – chemical composition

The amounts of different materials in a B.06 drum are given in Table 3-1. The waste matrix contains metal hydroxides from the clean-up system for the reactor and evaporation residues from back flush water from the fuel storage pools. The weight of the raw waste material is normally 50 kg ion exchange resin per package but the weight varies, see Table 3-1.

The matrix is made of bitumen. The specific brand of bitumen has changed during the many years of production of this waste type but it is always of the distilled sort. The brands used are Mexphalate until 1982, Nynäs industrial bitumen IB 45 between 1983–1995 and Nynäs industrial bitumen IB 55 used from 1995. The amounts of added chemicals are:

Na_2SO_4	0.2–0.5 kg/drum
Emulgator	0.4–1.1 kg/drum

Table 3-1. Amounts of different material in waste type B.06 (per package).

Material	Weight (kg)	Weight (%)
Ion-exchange resin	30–55	16–43
Bitumen	129–155	57–84

3.1.3 Radionuclide inventory

Before the waste is transported from Barsebäck NPP to SFR 1 a measurement of gamma emitting nuclides is performed. Dominating nuclides are Co-60 and Cs-137. The amounts of these nuclides are between 0 and $1.1 \cdot 10^{11}$ Bq for Co-60 and between 0 and $2.7 \cdot 10^{11}$ Bq for Cs-137. Other nuclides than gamma emitting ones must be calculated according to Appendix B.

The dose rate limit for this package is 500 mSv/h. Normal dose rate at 1 m is between 0–10 mSv/h. The surface contamination should not exceed 40 kBq/m^2 for gamma and beta and 4 kBq/m^2 for alpha emitting nuclides. The waste packages are usually free of contamination.

3.1.4 Waste production

The waste type was in production from 1985. No further production is foreseen, see Appendix A.

3.2 Reference waste type description

3.2.1 Waste package and material

Table 3-2 shows the content of different materials and the surface areas of the packaging in B.06 package from Barsebäck. Since the steel plate and expansion box is shared between four drums ¼ of the plate and box is included in a package of B.06.

The assumed composition of the waste in a reference waste package of B.06 is given in Table 3-3.

Void and waste volume is specified in Table 3-4.

Average void is estimated to 0.03 m^3 in a steel drum.

Table 3-2. Estimated reference packaging composition and surface area and thickness of components in each steel drum and one ¼ of a steel plate and expansion box of B.06.

Component	Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Steel drum	Stainless steel	23	3	1.2
Steel plate	Carbon steel	12.25	0.7	4
Expansion box	Carbon steel	5	0.2	6

Table 3-3. Estimated reference waste composition in one steel drum of B.06.

Material	Weight (kg)
Ion-exchange resin	50
Bitumen	150
Chemicals(Na-sulphate)	0.5

Table 3-4. Volumes in one reference waste packages of steel drums including ¼ steel plate and expansion box per drum of B.06.

	Volume one package (m^3)
Waste	0.2
Void	0.03

3.2.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 3-5.

A reference waste package of the type B.06 has a surface dose rate of 100 mSv/h and no surface contamination.

Table 3-5. Radionuclide composition of a reference waste package of the type B.06.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	1.9E+10	Pu-239	9.9E+03
Cl-36	1.2E+04	Pu-240	2.0E+04
Ag-108m	1.2E+06	U-232	8.9E–01
Ba-133	1.9E+05	U-234	3.0E+01
Be-10	1.2E+01	U-235	9.6E+02
C-14	5.8E+07	U-236	8.3E+02
Fe-55	1.9E+10	U-238	7.1E+02
H-3	1.9E+06	Np-237	4.9E+03
Ho-166m	7.7E+04	Pu-238	3.1E+04
Mo-93	1.9E+04	Pu-241	3.0E+06
Nb-93m	1.9E+07	Pu-242	8.9E+01
Nb-94	1.9E+05	Am-241	2.3E+04
Ni-59	1.9E+07	Am-242m	3.0E+02
Ni-63	1.5E+09	Am-243	8.9E+02
Sb-125	1.3E+09	Cm-243	6.0E+02
Zr-93	1.9E+04	Cm-244	1.8E+04
		Cm-245	8.9E+00
Correlated to Cs-137		Cm-246	2.4E+00
Cs-137	3.8E+09		
Cd-113m	2.3E+06		
Cs-134	3.3E+08		
Cs-135	3.8E+04		
Eu-152	2.7E+05		

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Eu-154	3.8E+08		
Eu-155	2.7E+08		
I-129	1.2E+04		
Pd-107	3.8E+03		
Pm-147	3.4E+09		
Ru-106	1.9E+07		
Se-79	1.5E+04		
Sm-151	1.2E+07		
Sn-126	1.9E+03		
Sr-90	3.8E+08		
Tc-99	3.4E+06		

3.3 Uncertainties

- General uncertainties are described in Section 1.3.
- The production of type B.06 was closed down in the end of 2000.

4 F.18

4.1 Waste type description

4.1.1 Waste package

General

The F.18 waste type consists of 1.73 m^3 steel moulds containing ion-exchange resins in a bitumen matrix. The raw waste material consists of ion exchange resins generated at the Forsmark nuclear power plant. The material is classified as intermediate level waste. All data comes from /Jansson 1995, Johansson 2000, Triumf 2007/.

Following restrictions are always applicable on this waste:

- The general restrictions listed in Section 1.1.
- Since the package contains burnable substances the restrictions regarding self-ignition etc are of special importance.

Packaging

The mould is a cubic box made of steel with the dimensions 1.2×1.2×1.2 m. The wall is 5 mm thick, the bottom is 6 mm thick. The mould weighs approximately 400 kg and the surface area is approximately 10 m^2. The variations between moulds are minimal. No major change in design has been made since the start of production of this package.

The weight in total of a package including waste and is approximately 1,960 kg but differs between 1,860–2,260 kg. The maximum allowed weight of a waste package is 5,000 kg.

Treatment and conditioning

The ion-exchange resins are dried in 150°C in a conical dryer and are then homogenised with bitumen in a conical mixer. The mixed waste is then poured in a steel mould. Each batch is maximum 500 litres. The mix is then allowed to cool for at least 17 h before a thin layer (~1 cm) of pure bitumen is poured upon the waste matrix. The waste package is thereafter lidded with a steel lid.

The void varies but is estimated to 10% in a package.

4.1.2 Materials – chemical composition

The amounts of different materials in a F.18 mould are given in Table 4-1. The waste is well defined and consists of ion exchange resins with radionuclides from system 324 and 331. The weight of the raw waste material is normally 600 kg ion exchange resin per package but varies, see Table 4-1.

The matrix is made of bitumen. The specific brand of bitumen has changed during the years of production of this waste type but it is always the distilled sort. The brands used are IB 45 Nynäs industrial bitumen used until 1994 and IB 55 Nynäs industrial bitumen from 1995.

No variation of material is identified in references.

Table 4-1. Amounts of different material in waste type F.18 (per package).

Material	Weight (kg)	Weight (%)
Ion-exchange resin	600	38
Bitumen	960	62

4.1.3 Radionuclide inventory

Before the waste is transported from Forsmark NPP to SFR 1 a measurement of gamma emitting nuclides is performed. Dominating nuclides are Co-60 and Cs-137. The amounts of these nuclides are between 0 and $2.5 \cdot 10^{12}$ Bq for Co-60 and between 0 and $1.9 \cdot 10^{11}$ Bq for Cs-137. Other nuclides than gamma emitting ones must be calculated according to Appendix B.

The dose rate limit for this package is 500 mSv/h on the surface. Normal dose rate at 1 m is approximately 30 mSv/h; normal interval for dose rate at 1 m is between 5–80 mSv/h. The surface contamination should not exceed 40 kBq/m^2 for gamma and beta and 4 kBq/m^2 for alpha emitting nuclides on the package. The waste packages are usually free of contamination.

4.1.4 Waste production

The waste type has been in production since 1991 and is still in production. The annual production is estimated to20 packages per year, see Appendix A.

4.2 Reference waste type description

4.2.1 Waste package and material

Table 4-2 shows the content of different materials and the steel surface areas of the packaging in a reference package.

The assumed composition of the waste in a reference waste package of F.18 is given in Table 4-3.

Average void is estimated 10% in a steel mould. Void and waste volume is specified in Table 4-4.

Table 4-2. Estimated reference packaging composition and surface area and thickness of components in steel mould of F.18.

Component	Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Steel mould	steel	400	10	5

Table 4-3. Estimated reference waste composition and surface area and thickness of components in one of F.18.

Material	Weight (kg)
Ion-exchange resin	600
Bitumen	960

Table 4-4. Volumes in one waste packages of F.18.

	Volume one package (m^3)
Waste	1.53
Void	0.17

4.2.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 4-5.

A reference waste package of the type F.18 has a surface dose rate of 500 mSv/h and no surface contamination.

Table 4-5. Radionuclide composition of a reference waste package of the type F.18.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	7.6E+11	Pu-239	1.2E+06
Cl-36	4.6E+05	Pu-240	2.5E+06
Ag-108m	4.6E+07	U-232	1.1E+02
Ba-133	7.6E+06	U-234	3.8E+03
Be-10	4.6E+02	U-235	1.5E+04
C-14	2.3E+09	U-236	1.1E+03
Fe-55	7.6E+11	U-238	1.6E+04
H-3	7.6E+07	Np-237	1.3E+05
Ho-166m	3.0E+06	Pu-238	7.3E+06
Mo-93	7.6E+05	Pu-241	3.8E+08
Nb-93m	7.6E+08	Pu-242	1.1E+04
Nb-94	7.6E+06	Am-241	2.1E+06
Ni-59	7.6E+08	Am-242m	3.8E+04
Ni-63	6.1E+10	Am-243	1.1E+05
Sb-125	7.4E+10	Cm-243	3.8E+04
Zr-93	7.6E+05	Cm-244	6.9E+06
		Cm-245	1.1E+03
Correlated to Cs-137		Cm-246	3.1E+02
Cs-137	7.1E+10		
Cd-113m	4.3E+07		
Cs-134	2.6E+10		
Cs-135	7.1E+05		
Eu-152	5.0E+06		
Eu-154	7.1E+09		
Eu-155	5.0E+09		
I-129	2.1E+05		
Pd-107	7.1E+04		
Pm-147	6.4E+10		

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Ru-106	3.6E+08		
Se-79	2.9E+05		
Sm-151	2.1E+08		
Sn-126	3.6E+04		
Sr-90	7.1E+09		
Tc-99	6.4E+07		

4.3 Uncertainties

- General uncertainties are described in Section 1.3.

5 O.02

5.1 Waste type description

5.1.1 Waste package

General

The O.02 waste type consists of 1.73 m^3 concrete moulds containing ion-exchange resins and inert filter-aid in a concrete matrix. The raw waste material consists of ion exchange resins and filter aids generated at the Oskarshamn nuclear power plant. The material is classified as intermediate level waste. All data comes from /Ingemansson 1999, Johansson 2000, Triumf 2007/.

Following restrictions are always applicable on this waste:

- The general restrictions listed in Section 1.1.

Packaging

The mould is a cubic box made of reinforced concrete with the 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 is made of 12 mm steel bars with a total surface of 10 m^2. The 10-cm mould weighs approximately 1,600 kg and the 25-cm one weighs about 3,200 kg. A stirrer made of carbon steel is included in the waste package. It weighs 16 kg and has an estimated surface of 1 m^2. To avoid cracking of the mould, due to expansion of the concrete matrix, a lining of compactable material (polyethene) is put inside of the mould, the lining has a thickness of 20 mm and a total weight of 10 kg. Lining is only placed in the 10-cm mould. The variations between moulds are minimal.

Some changes in the design have been made since the start of production of this package. 1975–78 the reinforcement was gradually improved and in 1981 a new design of reinforcement was introduced. In 1979 the moulds was fitted with a expansion cassette of polyurethane and wood, this was used until 1988 when the polyethene cassette was introduced. In 1986 a new brand of concrete to make the lid was introduced, instead of Sabema A, the Betokem EXM-4 was used. Today no chemical cement additives are used, earlier Silix GP was used.

The weight of a package including waste is approximately 3,200 kg but differs between 3,300–3,600 kg. The maximum allowed weight of a waste package is 5,000 kg.

Treatment and conditioning

The waste, used ion-exchange resins, is pumped into the mould and cement is added. If needed, water is added. When the cement is added the stirrer is started and the waste matrix is mixed for 30 minutes. The matrix is then allowed to harden for two days. Pouring concrete on top of the matrix then makes a lid. The lid is at least 10 cm thick. The concrete is allowed 24 hours to harden before the waste is transported to intermediate storage. The waste, used ion-exchange resins, is pumped into the mould and cement is added. If needed, water is added. When the cement is added the stirrer is started and the waste matrix is mixed for 30 minutes. The matrix is then allowed to harden for two days. Pouring concrete on top of the matrix then makes a lid. The concrete is allowed 24 hours to harden before the waste is transported to intermediate storage. The lid is at least 10 cm thick.

5.1.2 Materials – chemical composition

The amounts of different materials in a O.02 mould are given in Table 5-1. The waste is well defined and consists of ion exchange resins and filter-aid with radionuclides from systems for cleaning reactor water (system 331), treatment of liquid waste (system 342), decontamination (system 347) and cleaning of fuel ponds (system 324). Systems 331, 347 and 342 uses bead resin, system 342 uses powder resin. The weight of the raw waste material is normally 130 kg ion exchange resin per package but varies, see Table 5-1.

The matrix is made of cement. The specific brand of cement has changed during the years of production of this waste type. The brands used are:
Sabema A-bruk 1970–1986,
LH cement 1970–1981,
Massiv cement 1981–1986,
Anläggningscement 1986–1999,
Höghållfasthetscement from 1999–,
Betokem EXM used from 1986–.

Table 5-1. Amounts of different material in waste type O.02 (per package).

Material	Weight (kg)	Weight (%)
Ion-exchange resin	90–150	5.7–10.0
Cement*	1,350–1,500	90.0–94.3

* Including water 0.4 kg/kg cement.

5.1.3 Radionuclide inventory

Before the waste is transported from Oskarshamn NPP to SFR 1 a measurement of gamma emitting nuclides is performed. Dominating nuclides are Co-60 and Cs-137. The amounts of these nuclides are between 0 and $9.4 \cdot 10^{11}$ Bq for Co-60 and between 0 and $7.9 \cdot 10^{12}$ Bq for Cs-137. Other nuclides than gamma emitting ones must be calculated according to Appendix B.

The dose rate limit for this package is 30 mSv/h on the surface. Normal dose rate at 1 m is 2 mSv/h but varies between 0–10 mSv/h. The contamination on surface should not exceed 40 kBq/m^2 for gamma and beta and 4 kBq/m^2 for alpha emitting nuclides on the package. The waste packages are usually free of contamination.

5.1.4 Waste production

The waste type has been in production since 1970 and is still in production. The annual production is estimated to 40 packages per year, see Appendix A.

5.2 Reference waste type description

5.2.1 Waste package and material

Table 5-2 shows the content of different materials and the steel surface areas of the packaging in a reference package.

The assumed composition of the waste in a reference waste package of O.02 is given in Table 5-3.

The estimated volumes of the waste, the packaging materials and the void volume inside the package for a reference package are given in Table 5-4. The waste volume is estimated from the weights of the waste components in Table 5-3 and approximate densities of the different material. Average void is estimated to 15% in a package. Void and waste volume is specified in Table 5-4.

Table 5-2. Estimated reference packaging composition and surface area and thickness of components in concrete mould of O.02.

Component	Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Concrete mould	Concrete	1,340		100
Reinforcement	Steel	261	10	12
Steel lid	Steel	13	1.8	2
Stirrer	Steel	16	1	5
Other organics	Polyethene	10		

Table 5-3. Estimated reference waste composition and surface area and thickness of components in one of O.02.

Material	Weight (kg)
Ion-exchange resin	130
Cement*	1,540

* Including water 0.4 kg/kg cement.

Table 5-4. Volumes in one waste packages of O.02.

	Volume one package (m^3)
Waste	0.85
Void	0.15

5.2.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 5-5.

A reference waste package of the type O.02 has a maximum surface dose rate of 300 mSv/h and no surface contamination.

Table 5-5. Radionuclide composition of a reference waste package of the type O.02.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	1.1E+11	Pu-239	4.4E+05
Cl-36	6.3E+04	Pu-240	8.7E+05
Ag-108m	6.3E+06	U-232	3.9E+01
Ba-133	1.1E+06	U-234	1.3E+03
Be-10	6.3E+01	U-235	2.2E+01
C-14	3.2E+08	U-236	9.1E+02
Fe-55	1.1E+11	U-238	2.9E+03
H-3	1.1E+07	Np-237	3.2E+03
Ho-166m	4.2E+05	Pu-238	2.2E+06
Mo-93	1.1E+05	Pu-241	1.3E+08
Nb-93m	1.1E+08	Pu-242	3.9E+03
Nb-94	1.1E+06	Am-241	6.3E+05

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Ni-59	1.1E+08	Am-242m	1.3E+04
Ni-63	8.4E+09	Am-243	3.9E+04
Sb-125	1.1E+10	Cm-243	1.8E+03
Zr-93	1.1E+05	Cm-244	1.0E+06
		Cm-245	3.9E+02
Correlated to Cs-137		Cm-246	1.1E+02
Cs-137	1.6E+10		
Cd-113m	9.9E+06		
Cs-134	1.2E+10		
Cs-135	1.6E+05		
Eu-152	1.2E+06		
Eu-154	1.6E+09		
Eu-155	1.2E+09		
I-129	4.9E+04		
Pd-107	1.6E+04		
Pm-147	1.5E+10		
Ru-106	8.2E+07		
Se-79	6.6E+04		
Sm-151	4.9E+07		
Sn-126	8.2E+03		
Sr-90	1.6E+09		
Tc-99	1.5E+07		

5.3 Uncertainties

- General uncertainties are described in Section 1.3.

6 C.02

6.1 Waste type description

6.1.1 Waste package

General

The C.02 waste type consists of 1.73 m^3 concrete moulds containing ion-exchange resins and inert filter-aid in a concrete matrix. The raw waste material consists of ion exchange resins and filter aids generated at the Clab facility in Oskarshamn. The material is classified as intermediate level waste. All data comes from /Ingemansson 1999, Johansson 2000, Triumf 2007/.

Basically the C.02 waste type is identical with O.02 with some small exceptions. The type is in everyday use called O.02.

Following restrictions are always applicable on this waste:

- The general restrictions listed in Section 1.1.

Packaging

The mould is a cubic box made of reinforced concrete with the 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 is made of 12 mm steel bars with a total surface of 10 m^2. The 10-cm mould weighs approximately 1,600 kg and the 25-cm one weighs about 3,200 kg. A stirrer made of carbon steel is included in the waste package. It weighs 16 kg and has a estimated surface of 1 m^2. To avoid cracking of the mould, due to expansion of the concrete matrix, a lining of compactable material (polyethene) is put inside of the mould, the lining has a thickness of 20 mm and a total weight of 10 kg. Lining is only placed in the 10-cm mould. The variations between moulds are minimal.

Some changes in the design have been made since the start of production of this package. In 1986 a new brand of concrete to make the lid was introduced, instead of Sabema A, the Betokem EXM-4 was used. Today no chemical cement additives is used, earlier Silix GP was used.

The weight of a package including waste is approximately 3,200 kg but differs between 3,300–3,600 kg. The maximum allowed weight of a waste package is 5,000 kg.

Treatment and conditioning

The waste, used ion-exchange resin, is pumped into the mould and cement is added. If needed, water is added. When the cement is added the stirrer is started and the waste matrix is mixed for 30 minutes. The matrix is then allowed to harden for two days. Pouring concrete on top of the matrix then makes a lid. The lid is at least 10 cm thick. The concrete is allowed 24 hours to harden before the waste is transported to intermediate storage. Treatment and conditioning

The waste, used ion-exchange resin, is pumped into the mould and cement is added. If needed, water is added. When the cement is added the stirrer is started and the waste matrix is mixed for 30 minutes. The matrix is then allowed to harden for two days. Pouring concrete on top of the matrix then makes a lid. The concrete is allowed 24 hours to harden before the waste is transported to intermediate storage. The lid is at least 10 cm thick.

6.1.2 Materials – chemical composition

The amounts of different materials in a C.02 mould are given in Table 6-1. The waste is well defined and consists of ion exchange resins and filter-aid with radionuclides from systems for cooling and cleaning of fuel ponds (system 313, 324), treatment of process water (system 371) and treatment of drainage (system 372). The systems contains both bead and power resins. The weight of the raw waste material is normally 130 kg ion exchange resin per package but varies, see Table 6-1.

The matrix is made of cement. The specific brand of cement has changed during the years of production of this waste type. The brands used are:
Sabema A-bruk 1985–1986,
Massiv cement 1985–1986,
Anläggningscement 1986–1999,
Höghållfasthetscement from 1999–,
Betokem EXM used from 1986–.

Table 6-1. Amounts of different material in waste type C.02 (per package).

Material	Weight (kg)	Weight (%)
Ion-exchange resin	90–150	5.7–10.0
Cement*	1,350–1,500	90.0–94.3

* Including water 0.4 kg/kg cement.

6.1.3 Radionuclide inventory

Before the waste is transported from Oskarshamn NPP/Clab to SFR 1 a measurement of gamma emitting nuclides is performed. Dominating nuclides are Co-60 and Cs-137. The amounts of these nuclides are between 0 and $6.38 \cdot 10^{11}$ Bq for Co-60 and between 0 and $8.8 \cdot 10^{11}$ Bq for Cs-137. Other nuclides than gamma emitting ones must be calculated according to Appendix B.

The dose rate limit for this package is 30 mSv/h on the surface. Normal dose rate at 1 m is 2 mSv/h but varies between 0–10 mSv/h. The contamination on surface should not exceed 40 kBq/m^2 for gamma and beta and 4 kBq/m^2 for alpha emitting nuclides on the package. The waste packages are usually free of contamination.

6.1.4 Waste production

The waste type has been in production since 1985 and is still in production. The annual production is estimated to 15 packages per year until 2015 and then 25 annually onwards, see Appendix A.

6.2 Reference waste type description

6.2.1 Waste package and material

Table 6-2 shows the content of different materials and the steel surface areas of the packaging in a reference package.

The assumed composition of the waste in a reference waste package of C.02 is given in Table 6-3.

The estimated volumes of the waste, the packaging materials and the void volume inside the package for a reference package are given in Table 6-4. The waste volume is estimated from the weights of the waste components in Table 6-3 and approximate densities of the different material. Average void is estimated to 15% in a package.

Void and waste volume is specified in Table 6-4.

Table 6-2. Estimated reference packaging composition and surface area and thickness of components in concrete mould of C.02.

Component	Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Concrete mould	Concrete	1,340		100
Reinforcement	Steel	261	10	12
Steel lid	Steel	13	1.8	2
Stirrer	Steel	16	1	5
Other organics	Polyethene	10		

Table 6-3. Estimated reference waste composition and surface area and thickness of components in one of C.02.

Material	Weight (kg)
Ion-exchange resin	130
Cement*	1,540

* Including water 0.4 kg/kg cement.

Table 6-4. Volumes in one waste packages of C.02.

	Volume one package (m^3)
Waste	0.85
Void	0.15

6.2.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 6-5.

A reference waste package of the type C.02 has a maximum surface dose rate of 300 mSv/h and no surface contamination.

Table 6-5. Radionuclide composition of a reference waste package of the type C.02.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	8.4E+10	Pu-239	1.1E+05
Cl-36	5.1E+04	Pu-240	2.2E+05
Ag-108m	5.1E+06	U-232	9.9E+00
Ba-133	8.4E+05	U-234	3.3E+02
Be-10	5.1E+01	U-235	6.6E+00
C-14	2.5E+08	U-236	9.9E+01
Fe-55	8.4E+10	U-238	1.3E+02
H-3	8.4E+06	Np-237	1.3E+02
Ho-166m	3.4E+05	Pu-238	1.4E+06
Mo-93	8.4E+04	Pu-241	3.3E+07
Nb-93m	8.4E+07	Pu-242	9.9E+02
Nb-94	8.4E+05	Am-241	3.8E+05

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Ni-59	8.4E+07	Am-242m	3.3E+03
Ni-63	6.8E+09	Am-243	5.3E+04
Sb-125	9.8E+09	Cm-243	2.2E+04
Zr-93	8.4E+04	Cm-244	8.3E+05
		Cm-245	9.9E+01
Correlated to Cs-137		Cm-246	2.6E+01
Cs-137	2.7E+08		
Cd-113m	1.6E+05		
Cs-134	7.0E+08		
Cs-135	2.7E+03		
Eu-152	1.9E+04		
Eu-154	2.7E+07		
Eu-155	1.9E+07		
I-129	8.1E+02		
Pd-107	2.7E+02		
Pm-147	2.4E+08		
Ru-106	1.4E+06		
Se-79	1.1E+03		
Sm-151	8.1E+05		
Sn-126	1.4E+02		
Sr-90	2.7E+07		
Tc-99	2.4E+05		

6.3 Uncertainties

- General uncertainties are described in Section 1.3.
- The allowed dose rate will probably be raised in the future.

7 C.24

7.1 Waste type description

7.1.1 Waste package

General

The C.24 waste type consists of 1.73 m^3 concrete moulds with scrap metal in a concrete matrix. A small volume of garbage is allowed. The raw waste material consists of mainly iron/steel. Cellulose and other organic material may be present but is kept to a minimum. All waste is generated at the Clab facility in Oskarshamn. The material is classified as intermediate level waste. All data comes from /Ingemansson 2000, Johansson 2000/.

The type is in everyday use called O.24.

Following restrictions are always applicable on this waste:

- The general restrictions listed in Section 1.1.

Packaging

The mould is a cubic box made of reinforced concrete with the dimensions 1.2×1.2×1.2 m. The walls are 10 cm thick. The reinforcement is made of 12 mm steel bars with a total surface of 10 m^2. The 10-cm mould weighs approximately 1,600 kg. The variations between moulds are minimal.

Inside the concrete mould, modified steel drums could be used as an inner package for the waste. A special drum basket is used inside the concrete mould to centre the drum inside the mould. Approximately 10% of the concrete moulds are foreseen to have this inner drum.

The weight of a package including waste is approximately 2,600 kg. No variation of weight is documented. The maximum allowed weight of a waste package is 5,000 kg.

Treatment and conditioning

The waste is transported to the 'treatment hot cell' in Clab where the waste is packed in the moulds or in the inner drums mentioned above. When the mould is filled the package is filled with concrete. The waste package is then allowed two days to harden. Pouring concrete on top of the package makes a lid.

7.1.2 Materials – chemical composition

The amounts of different materials in a C.24 mould are given in Table 7-1. The waste is fairly well defined and consists mainly of scrap metal e.g. valves, filters, etc. A small fraction consists of garbage as e.g. air filters or plastics used in the hot cell. The weight of the raw waste material is normally 640 kg waste per package but can vary, see Table 7-1.

The matrix is made of concrete. The brand used is Betokem EXM 4.

Table 7-1. Amounts of different material in waste type C.24 (per package).

Material	Weight (kg)***	Area (m^2)	Weight (%)
Cellulose	35		5.4
Iron/steel**	17	0.9	2.7
Other inorganics	24		3.7
Concrete*	565		88.1

* Including 0.13 kg water/kg concrete.

** Up to 10% of the packages could have a content of special boxes of steel with a weight of 420 kg and an area of 2.3 m^2 per each.

*** No variation of material is available in references.

7.1.3 Radionuclide inventory

Before the waste is transported from Oskarshamn NPP/Clab to SFR 1 a measurement of gamma emitting nuclides is performed. Dominating nuclides are Co-60 and Cs-137. Variations of nuclides are not available. Other nuclides than gamma emitting ones must be calculated according to Appendix B.

Dose rate limit for this package is 30 mSv/h. The surface contamination should not exceed 40 kBq/m^2 for gamma and beta and 4 kBq/m^2 for alpha emitting nuclides. The waste packages are usually free of contamination.

7.1.4 Waste production

The waste type has been in production since 1985 and is still in production. The annual production is 2 packages per year until 2015 and then 4 packages per year annually, see Appendix A.

7.2 Reference waste type description

7.2.1 Waste package and material

Table 7-2 shows the content of different materials and the steel surface areas of the packaging in a reference package.

The assumed composition of the waste in a reference waste package of C.24 is given in Table 7-3.

10% of the packages is calculated to have steel packages inside the concrete mould. These packages are averaged over all packages.

The estimated volumes of the waste, the packaging materials and the void volume inside the package for a reference package are given in Table 7-4. The waste volume is estimated from the weights of the waste components in Table 7-3 and approximate densities of the different material. Average void is estimated to 1% in a package.

Void and waste volume is specified in Table 7-4.

Table 7-2. Estimated reference packaging composition, surface area and thickness of components in concrete mould of C.24.

Component	Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Concrete mould	Concrete	1,340		100
Reinforcement	Steel	261	10	12
Steel packaging	Steel	420	2.3	5

Table 7-3. Estimated reference waste composition and surface area and thickness of components in one of C.24.

Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Cellulose	35		
Iron/steel	17	0.9	5
Other inorganics	24		
Concrete*	565		

* Including 0.13 kg water/kg concrete.

Table 7-4. Volumes in one packages of C.24.

	Volume one package (m^3)
Waste	0.99
Void	0.01

7.2.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 7-5.

A reference waste package of the type C.24 has a maximum surface dose rate of 300 mSv/h and no surface contamination.

Table 7-5. Radionuclide composition of a reference waste package of the type C.24.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	1.5E+11	Pu-239	1.1E+04
Cl-36	6.5E+05	Pu-240	2.2E+04
Ag-108m	9.0E+06	U-232	9.8E–01
Ba-133	1.5E+06	U-234	3.3E+01
Be-10	9.0E+01	U-235	6.6E–01
C-14	1.5E+08	U-236	9.8E+00
Fe-55	1.5E+11	U-238	1.3E+01
H-3	1.5E+07	Np-237	1.3E+01
Ho-166m	6.0E+05	Pu-238	1.3E+05
Mo-93	7.5E+05	Pu-241	3.3E+06
Nb-93m	1.5E+08	Pu-242	9.8E+01
Nb-94	1.5E+06	Am-241	3.3E+04
Ni-59	1.5E+08	Am-242m	3.3E+02
Ni-63	3.0E+10	Am-243	9.8E+02
Sb-125	1.5E+10	Cm-243	6.6E+02
Zr-93	1.5E+05	Cm-244	9.8E+04
		Cm-245	9.8E+00
Correlated to Cs-137		Cm-246	2.6E+00
Cs-137	6.5E+10		
Cd-113m	3.9E+07		
Cs-134	6.5E+10		

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Cs-135	3.3E+05		
Eu-152	4.6E+06		
Eu-154	6.5E+09		
Eu-155	4.6E+09		
I-129	2.0E+04		
Pd-107	6.5E+04		
Pm-147	5.9E+10		
Ru-106	3.3E+08		
Se-79	2.6E+05		
Sm-151	2.0E+08		
Sn-126	3.3E+04		
Sr-90	6.5E+09		
Tc-99	3.3E+08		

7.3 Uncertainties

- Production of C.24 is estimated to optimise available volume in Silo, probably the total production of C.24 will be lower than foreseen above.

8 R.02

8.1 Waste type description

8.1.1 Waste package

General

The R.02 waste type consists of 1.73 m^3 concrete moulds containing ion-exchange resins and inert filter-aid in a concrete matrix. The raw waste material consists of ion exchange resins and filter aids generated at the Ringhals nuclear power plant. The material is classified as intermediate level waste. All data comes from /Eriksson 1990, Johansson 2000, Triumf 2007/.

Following restrictions are always applicable on this waste:

- The general restrictions listed in Section 1.1.

Packaging

The mould is a cubic box made of reinforced concrete with the 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 is made of steel bars with a total surface of 7.9 m^2 and a weight of 160 kg in a 10-cm mould and 11.8 m^2 and 250 kg in a 25-cm mould. The 10-cm mould weighs approximately 1,600 kg and the 25-cm weighs about 3,100 kg. A stirrer made of carbon steel is included in the waste package. It weighs 16 kg and has an estimated surface of 1 m^2. To avoid cracking of the mould, due to expansion of the concrete matrix, a lining with compressible material (polyethene or polystyrene) is put inside the walls of the mould. The lining has a thickness of 20 mm and a total weight of 10 kg in a 10-cm mould and 5 mm and approximately 1 kg in a 25-cm mould. The variations between moulds are minimal.

Some changes in the design have been made since the start of production of this package. Before 1976 no lining was used. Some changes have been made regarding reinforcement and steel details in the mould. For the lid Betokem EXM-4 was used, nowadays Fiberbruk VF50 is used. Today no chemical cement additives are used but earlier Silix GP and Sika AER were used.

The weight of a package including waste and is approximately 3,400 kg but differs between 3,000–3,700 kg. The maximum allowed weight of a waste package is 4,000 kg.

Treatment and conditioning

The waste, used ion-exchange resins, is pumped into the mould and cement is added. If needed, water is added. When the cement is added the stirrer is started and the waste matrix is mixed for 60 minutes. The matrix is then allowed to harden for two days. Pouring concrete on top of the matrix then makes a lid. The concrete is allowed 24 hours to harden before the waste is transported to interim storage. The lid is at least 10 cm thick.

8.1.2 Materials – chemical composition

The amounts of different materials in a R.02 mould are given in Table 8-1. The waste is well defined and consists of ion exchange resins and filter-aid with radionuclides.

The resins are:

- Bead resin from PWR: cleaning of the primary reactor circuit (system 334), the fuel ponds (system 324), the waste water (system 342), water from the secondary side of the steam generators (system 417/337) and from drainage water.

- Bead resin from BWR: systems for cleaning reactor water (system 331), treatment of liquid waste (system 342), analytical ion-exchangers (system 336) and cleaning of fuel ponds (system 324).
- Powder resins from BWR: cleaning of condensate (system 332), fuel ponds (system 324) and treatment of liquid waste (system 324). The resins from liquid waste treatment contain filter-aids.

Resins from the PWR-reactors contain boric acid (H_3BO_3) up to 90 g/kg of resin. Some resins contain lithium. Resins from system 417/337 can contain ammonia and hydrazine.

The weight of the raw waste material is normally 130 kg of ion exchange resin per package but the weight varies, see Table 8-1.

The matrix is made of cement. The specific brand of cement has changed during the years of production of this waste type. The brand used nowadays is Skövde standardcement. In earlier days Limhamn LH-cement was used.

Table 8-1. Amounts of different material in waste type R.02 (per package).

Material	Weight (kg)	Weight (%)
Ion-exchange resin	100–150	5.9–12.0
Cement*	1,100–1,600	88.0–94.1

* Including water 0.4 kg/kg cement.

8.1.3 Radionuclide inventory

Before the waste is transported from Ringhals NPP to SFR 1 a measurement of gamma emitting nuclides is performed. Dominating nuclides are Co-60 and Cs-137. The amounts of these nuclides are between 0 and $1.4 \cdot 10^{12}$ Bq for Co-60 and between 0 and $7.7 \cdot 10^{11}$ Bq for Cs-137. Other nuclides than gamma emitting ones must be calculated according to Appendix B.

The dose rate limit for this package is 500 mSv/h on the surface. Normal dose rate at 1 m is 4 mSv/h but varies between 0–14 mSv/h. The surface contamination should not exceed 40 kBq/m^2 for gamma and beta and 4 kBq/m^2 for alpha emitting nuclides on the package.

8.1.4 Waste production

The waste type has been in production since 1984 and is still in production even though in a small scale. Type R.16 is used for this kind of waste nowadays, see Appendix A.

8.2 Reference waste type description

8.2.1 Waste package and material

Table 8-2 shows the content of different materials and the steel surface areas of the packaging in a reference package.

The assumed composition of the waste in a reference waste package of R.02 is given in Table 8-3.

The estimated volumes of the waste, the packaging materials and the void volume inside the package for a reference package are given in Table 8-4. The waste volume is estimated from the weights of the waste components in Table 8-3 and approximate densities of the different material. Average void is estimated to 15% in a package.

Void and waste volume is specified in Table 8-4.

Table 8-2. Estimated reference packaging composition and surface area and thickness of components in concrete mould of R.02.

Component	Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Concrete box	Concrete	1,340		100
Reinforcement	Steel	261	10	12
Steel lid	Steel	13	1.8	2
Stirrer	Steel	16	1	5
Other organics	Polyethene	10		

Table 8-3. Estimated reference waste composition and surface area and thickness of components in one of R.02.

Material	Weight (kg)
Ion-exchange resin	130
Cement*	1,400

* Including water 0.4 kg/kg cement.

Table 8-4. Volumes in one waste packages of R.02.

	Volume one package (m^3)
Waste	0.85
Void	0.15

8.2.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 8-5.

A reference waste package of the type R.02 has a maximum surface dose rate of 500 mSv/h and no surface contamination.

Table 8-5. Radionuclide composition of a reference waste package of the type R.02.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	5.0E+10	Pu-239	5.0E+05
Cl-36	3.0E+04	Pu-240	1.0E+06
Ag-108m	3.0E+06	U-232	4.5E+01
Ba-133	5.0E+05	U-234	1.5E+03
Be-10	3.0E+01	U-235	3.0E+01
C-14	1.5E+08	U-236	4.5E+02
Fe-55	5.0E+10	U-238	6.1E+02
H-3	5.0E+06	Np-237	6.1E+02
Ho-166m	2.0E+05	Pu-238	2.5E+06
Mo-93	5.0E+04	Pu-241	1.5E+08
Nb-93m	5.0E+07	Pu-242	4.5E+03
Nb-94	5.0E+05	Am-241	8.9E+05

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Ni-59	5.0E+07	Am-242m	1.5E+04
Ni-63	4.0E+09	Am-243	4.5E+04
Sb-125	5.0E+09	Cm-243	3.0E+04
Zr-93	5.0E+04	Cm-244	9.8E+05
		Cm-245	4.5E+02
Correlated to Cs-137		Cm-246	1.2E+02
Cs-137	5.1E+10		
Cd-113m	3.1E+07		
Cs-134	3.2E+10		
Cs-135	5.1E+05		
Eu-152	3.6E+06		
Eu-154	5.1E+09		
Eu-155	3.6E+09		
I-129	1.5E+05		
Pd-107	5.1E+04		
Pm-147	4.6E+10		
Ru-106	2.5E+08		
Se-79	2.0E+05		
Sm-151	1.5E+08		
Sn-126	2.5E+04		
Sr-90	5.1E+09		
Tc-99	4.6E+07		

8.3 Uncertainties

- General uncertainties are described in Section 1.3.

9 R.16

9.1 Waste type description

9.1.1 Waste package

General

The R.16 waste type consists of 1.73 m^3 steel moulds containing ion-exchange resins and inert filter-aid in a concrete matrix. The raw waste material consists of ion exchange resins and filter aids generated at the Ringhals nuclear power plant. The material is classified as intermediate level waste. Data comes from /Ahlqvist 1998, Johansson 2000, Triumf 2007/.

Following restrictions are always applicable on this waste:

- The general restrictions listed in Section 1.1.

Packaging

The mould is a cubic box made of steel with the dimensions 1.2×1.2×1.2 m. The wall is 5 mm thick, the bottom is 6 mm thick. The mould weighs approximately 425 kg and the surface area is approximately 20 m^2. The variations between moulds are minimal. No major change in design has been made since the start of production of this package.

The weight in total of a package including waste and is approximately 3,200 kg but differs between 2,700–3,500 kg. The maximum allowed weight of a waste package is 5,000 kg.

Treatment and conditioning

The waste, used ion-exchange resin, is pumped into the mould and cement is added. If needed, water is added. When the cement is added the stirrer is started and the waste matrix is mixed for 60 minutes. The matrix is then allowed to harden for two days. Pouring concrete on top of the matrix then makes a lid. The concrete is allowed 24 hours to harden before the waste is transported to interim storage. The lid is at least 10 cm thick.

9.1.2 Materials – chemical composition

The amounts of different materials in a R.16 mould are given in Table 9-1. The waste is well defined and consists of ion exchange resins and filter-aid with radionuclides.

The resins are:

- Bead resin from PWR: cleaning of the primary reactor circuit (system 334), the fuel ponds (system 324), the waste water (system 342), water from the secondary side of the steam generators (system 417/337) and from drainage water.
- Bead resin from BWR: systems for cleaning reactor water (system 331), treatment of liquid waste (system 342), analytical ion-exchangers (system 336) and cleaning of fuel ponds (system 324).
- Powder resins from BWR: cleaning of condensate (system 332), fuel ponds (system 324) and treatment of liquid waste (system 324).
  The resins from liquid waste treatment contain filter-aids.

Resins from the PWR-reactors contain boric acid (H_3BO_3) up to 90 g/kg of resin. Some resins contain lithium. Resins from system 417/337 can contain ammonia and hydrazine.

The weight of the raw waste material is normally 130 kg of ion exchange resin per package but it varies, see Table 9-1.

The matrix is made of cement. The brand used is Skövde standardcement. Nowadays also Degerham standardcement can be used. Silix GP and Sika AER was used as additives until 1997 (and a few exceptions during 1998), from 1997 to 1998 Hydrafix was used and from 1997 a special additive is used (brand name is kept on commercial secrecy by demand from Ringhals NPP).

Table 9-1. Amounts of different material in waste type R.16 (per package).

Material	Weight (kg)	Weight (%)
Ion-exchange resin	10–500	0.4–26.3
Cement*	1,400–2,800	99.6–73.7

* Including water 0.4 kg/kg cement.

9.1.3 Radionuclide inventory

Before the waste is transported from Ringhals NPP to SFR 1 a measurement of gamma emitting nuclides is performed. Dominating nuclides are Co-60 and Cs-137. The amounts of these nuclides are between 0 and $9.0\cdot10^{11}$ Bq for Co-60 and between 0 and $2.0\cdot10^{11}$ Bq for Cs-137. Other nuclides than gamma emitting ones must be calculated according to Appendix B.

The dose rate limit for this package is 500 mSv/h on the surface. Normal dose rate at 1 m is 9 mSv/h but varies between 0.1–35 mSv/h. The surface contamination should not exceed 40 kBq/m^2 for gamma and beta and 4 kBq/m^2 for alpha emitting nuclides on the package. The waste packages are usually free of contamination.

9.1.4 Waste production

The waste type has been in production since 1991 and is still in production. The annual production is estimated to 70 packages per year, see Appendix A.

9.2 Reference waste type description

9.2.1 Waste package and material

Table 9-2 shows the content of different materials and the steel surface areas of the packaging in a reference package.

The assumed composition of the waste in a reference waste package of R.16 is given in Table 9-3.

The estimated volumes of the waste, the packaging materials and the void volume inside the package for a reference package are given in Table 9-4. The waste volume is estimated from the weights of the waste components in Table 9-3 and approximate densities of the different material. Average void is estimated to 5% in a package.

Void and waste volume is specified in Table 9-4.

Table 9-2. Estimated reference packaging composition and surface area and thickness of components in concrete mould of R.16.

Component	Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Steel box	Stainless steel	400	17	5
Stirrer	Stainless steel	25	3	2

Table 9-3. Estimated reference waste composition and surface area and thickness of components in one of R.16.

Material	Weight (kg)
Ion-exchange resin	250
Cement*	2,100

* Including water 0.4 kg/kg cement.

Table 9-4. Volumes in one waste packages of R.16.

	Volume one package (m^3)
Waste	1.615
Void	0.085

9.2.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 9-5.

A reference waste package of the type R.16 has a maximum surface dose rate of 500 mSv/h and no surface contamination.

Table 9-5. Radionuclide composition of a reference waste package of the type R.16.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	1.9E+11	Pu-239	1.8E+06
Cl-36	1.2E+05	Pu-240	3.5E+06
Ag-108m	1.2E+07	U-232	1.7E+02
Ba-133	1.9E+06	U-234	5.7E+03
Be-10	1.2E+02	U-235	1.1E+02
C-14	5.8E+08	U-236	1.7E+03
Fe-55	1.9E+11	U-238	2.3E+03
H-3	1.9E+07	Np-237	2.3E+03
Ho-166m	7.7E+05	Pu-238	9.6E+06
Mo-93	1.9E+05	Pu-241	5.7E+08
Nb-93m	1.9E+08	Pu-242	1.7E+04
Nb-94	1.9E+06	Am-241	3.4E+06
Ni-59	1.9E+08	Am-242m	5.7E+04
Ni-63	1.6E+10	Am-243	1.7E+05
Sb-125	1.9E+10	Cm-243	1.1E+05
Zr-93	1.9E+05	Cm-244	3.7E+06
		Cm-245	1.7E+03
Correlated to Cs-137		Cm-246	4.6E+02
Cs-137	2.9E+10		
Cd-113m	1.8E+07		
Cs-134	2.0E+10		
Cs-135	2.9E+05		
Eu-152	2.1E+06		

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Eu-154	2.9E+09		
Eu-155	2.1E+09		
I-129	8.8E+04		
Pd-107	2.9E+04		
Pm-147	2.6E+10		
Ru-106	1.5E+08		
Se-79	1.2E+05		
Sm-151	8.8E+07		
Sn-126	1.5E+04		
Sr-90	2.9E+09		
Tc-99	2.6E+07		

9.3 Uncertainties

- Production of R.16 is estimated to optimise available volume in Silo, probably the total production of R.16 will be lower than foreseen above.

10 S.04

10.1 Waste type description

10.1.1 Waste package

General

The S.04 waste type consists of a standard 200-litre drum containing ion exchange resins in a cement matrix. The raw waste material consists of ion exchange resins generated at the Studsvik research centre. The material is classified as intermediate level waste. All data comes from /Öberg 1997, Johansson 2000, Studsvik 2000/.

The physical properties and chemical conditions are well known. The following restrictions are always applicable:

- The general restrictions listed in Section 1.1.

Packaging

The steel drum is made of stainless steel with diameter 0.57 m and 0.84 m in height. The wall thickness is 1.5 mm. The drum weighs approximately 60 kg and surface area is approximately 5.1 m^2. To facilitate the handling in SFR the drums are placed four by four on a steel plate with the dimension 1.2 m ·1.2 m and the thickness of 4 mm. The weight of the steel plate is 66 kg with surface area of 4.0 m^2. The variations between packages are minimal. No major changes in design have been made since the start of production of this package.

The weight in total of a package including waste and is approximately 350 kg. The maximum allowed weight of a waste package is 500 kg.

Treatment and conditioning

The ion exchange resin containing metal hydroxides and other substances is mixed with concrete in a custom-made facility at the Studsvik site. Cement is added in the drum and then the waste, used ion-exchange resin, and water is pumped into the drum.

When the waste is added the stirrer is started and the waste matrix is mixed. The matrix is then allowed to harden for one day. After measurements a steel lid is placed on top of the drum. The drums are placed on the steel plate when the drums are sent to SFR.

The void varies but is estimated to 0.01 m^3 in a drum.

10.1.2 Materials – chemical composition

The amounts of different materials in a S.04 drum are given in Table 10-1. The waste contains ion-exchange resins (bead-type) with metal hydroxides from the fuel ponds and primary circuits in the Studsvik R2-reactor. The weight of the raw waste material is normally 65 kg ion-exchange resin per package but varies, see Table 10-1.

The matrix is made of concrete. No cement additives are used.

Table 10-1. Amounts of different material in waste type S.04 (per package).

Material	Weight (kg)	Weight (%)
Ion-exchange resin	20	25
Concrete	60	75

10.1.3 Radionuclide inventory

Before the waste is solidified a measurement of gamma emitting nuclides is performed. Dominating nuclides are Co-60 and Cs-137. The amounts of these nuclides are between 0 and $4.6 \cdot 10^8$ Bq for Co-60 and between 0 and $1.2 \cdot 10^9$ Bq for Cs-137. Other nuclides than gamma emitting ones must be calculated according to Appendix B.

The dose rate limit for this package is 50 mSv/h. No estimation on variation of dose rates are available in referred documents. The surface contamination should not exceed 40 kBq/m^2 for gamma and beta and 4 kBq/m^2 for alpha emitting nuclides. The waste packages are usually free of contamination.

10.1.4 Waste production

The annual production is estimated to 4, see Appendix A.

10.2 Reference waste type description

10.2.1 Waste package and material

Table 10-2 shows the content of different materials and the surface areas of the packaging in S.04 package from Studsvik. Since the steel plate and expansion box is shared between four drums ¼ of the plate and box is included in a package of S.04.

The assumed composition of the waste in a reference waste package of S.04 is given in Table 9-3.

Void and waste volume is specified in Table 10-4.

Table 10-2. Estimated reference packaging composition and surface area and thickness of components in each steel drum and one ¼ of a steel plate and expansion box of S.04.

Component	Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Steel drum	steel	60	5.1	1.5
Steel plate	steel	16.6	1.0	4

Table 10-3. Estimated reference waste composition in one steel drum of S.04.

Material	Weight (kg)
Ion-exchange resin	65
Cement*	238

* Including water 0.4 kg/kg cement.

Table 10-4. Volumes in one waste packages of steel drums including ¼ steel plate and expansion box per drum of S.04.

	Volume one package (m^3)
Waste	0.2
Void	0.01

10.2.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 10-5.

A reference waste package of the type S.04 has a surface dose rate of 50 mSv/h and no surface contamination.

Table 10-5. Radionuclide composition of a reference waste package of the type S.04.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	5.3E+07	Pu-239	7.7E+02
Cl-36	3.2E+01	Pu-240	1.6E+03
Ag-108m	3.2E+03	U-232	7.0E–02
Ba-133	5.3E+02	U-234	2.3E+00
Be-10	3.2E–02	U-235	4.6E–02
C-14	1.6E+05	U-236	7.0E–01
Fe-55	5.3E+07	U-238	9.3E–01
H-3	5.3E+03	Np-237	9.3E–01
Ho-166m	2.1E+02	Pu-238	1.5E+06
Mo-93	5.3E+01	Pu-241	2.3E+05
Nb-93m	5.3E+04	Pu-242	7.0E+00
Nb-94	5.3E+02	Am-241	2.3E+03
Ni-59	5.3E+04	Am-242m	2.3E+01
Ni-63	4.2E+06	Am-243	7.0E+01
Sb-125	5.3E+06	Cm-243	4.6E+01
Zr-93	5.3E+01	Cm-244	1.6E+06
		Cm-245	7.0E–01
Correlated to Cs-137		Cm-246	1.9E–01
Cs-137	2.0E+07		
Cd-113m	1.2E+04		
Cs-134	8.2E+06		
Cs-135	2.0E+02		
Eu-152	2.4E+06		
Eu-154	8.3E+05		
Eu-155	1.4E+06		
I-129	6.0E+01		
Pd-107	2.0E+01		
Pm-147	1.8E+07		
Ru-106	3.2E+06		
Se-79	8.0E+01		
Sm-151	6.0E+04		
Sn-126	1.0E+01		
Sr-90	2.0E+06		
Tc-99	1.8E+04		

10.3 Uncertainties

- General uncertainties are described in Section 1.3.

11 S.11

11.1 Waste type description

11.1.1 Waste package

General

The S.11 waste type consists of 1.73 m^3 steel moulds containing ion-exchange resins and sludges in a cement matrix. The raw waste material consists of ion-exchange resins and sludges generated at the Studsvik site or at the closed down Ågesta Reactor. The material is classified as intermediate level waste. All data comes from /Rolandsson 1998, Johansson 2000/.

Following restrictions are always applicable on this waste:

- The general restrictions listed in Section 1.1.

Packaging

The mould is a cubic box made of steel with the dimensions 1.2×1.2×1.2 m. The wall is 5 mm thick, the bottom is 6 mm thick. The mould weighs approximately 425 kg including stirrer and the surface area is approximately 20 m^2. The variations between moulds are minimal.

The weight in total of a package including waste and is approximately 4,000 kg. The maximum allowed weight of a waste package is 5,000 kg.

Treatment and conditioning

The waste is pumped into the mould and cement is added. If needed, water is added. When the cement is added the stirrer is started and the waste matrix is mixed. The matrix is then allowed to harden. Pouring concrete on top of the matrix then makes a lid. The concrete is allowed 24 hours to harden before the waste is transported to interim storage. The lid is at least 10 cm thick.

11.1.2 Materials – chemical composition

The amounts of different materials in a S.11 mould are given in Table 11-1. The waste has been thoroughly analysed and is fairly well defined. It consists of ion exchange resins and sludges containing radionuclides. No harmful amounts of complexing agents are present in the waste matrix that could improve mobility of the radionuclides.

The weight of the raw waste material is normally 667 kg of ion exchange resin and 333 kg sludge per package but varies, some parameters have an estimated variation see Table 11-1.

The matrix is made of cement. No additives are used in this waste.

Table 11-1. Amounts of different material in waste type S.11 (per package).

Material	Weight (kg)	Weight (%)
Ion-exchange resin	667	21–26.2
Sludge	333	10.7–13.1
Cellulose	3	0–0.1
Cement*	1,540–2,100	60.6–67.7

* Including water 0.4 kg/kg cement.

11.1.3 Radionuclide inventory

Before the waste is transported from Studsvik to SFR 1 a measurement of gamma emitting nuclides is performed. Dominating nuclides are Co-60 and Cs-137, no data is available since no production of waste packages has been done yet. Other nuclides than gamma emitting ones must be calculated according to Appendix B.

The dose rate limit for this package is 5 mSv/h on the surface. No variation on dose rate is available since no production of waste packages has been done yet. The surface contamination should not exceed 40 kBq/m^2 for gamma and beta and 4 kBq/m^2 for alpha emitting nuclides on the package. Usually the packages are free of contamination.

11.1.4 Waste production

The waste type has not yet been put in production. The amount of waste is limited to about 50 m^3 stored in a Silo in at the Studsvik site.

11.2 Reference waste type description

11.2.1 Waste package and material

Table 11-2 shows the content of different materials and the steel surface areas of the packaging in a reference package.

The assumed composition of the waste in a reference waste package of S.11 is given in Table 11-3.

The estimated volumes of the waste, the packaging materials and the void volume inside the package for a reference package are given in Table 11-4. The waste volume is estimated from the weights of the waste components in Table 11-3 and approximate densities of the different material. Average void is estimated to 5% in a package.

Void and waste volume is specified in Table 11-4.

Table 11-2. Estimated reference packaging composition and surface area and thickness of components in concrete mould of S.11.

Component	Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Steel box	Stainless steel	400	17	5
Stirrer	Stainless steel	25	3	2

Table 11-3. Estimated reference waste composition and surface area and thickness of components in one of S.11.

Material	Weight (kg)
Ion-exchange resin	667
Sludge	333
Cellulose	3
Cement*	1,820

* Including water 0.4 kg/kg cement.

Table 11-4. Volumes in one packages of S.11.

	Volume one package (m^3)
Waste	1.615
Void	0.085

11.2.2 Radionuclide inventory

The radionuclide composition of a reference waste package is calculated from the amount of Co-60 and Cs-137 in the waste according to Appendix B. The reference inventory in one package at the time of production is presented in Table 11-5.

A reference waste package of the type S.11 has a surface dose rate of 5 mSv/h and no surface contamination.

Table 11-5. Radionuclide composition of a reference waste package of the type S.11.

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Correlated to Co-60		Correlated to Pu	
Co-60	3.0E+08	Pu-239	1.3E+07
Cl-36	1.8E+02	Pu-240	2.6E+07
Ag-108m	4.7E+05	U-232	1.2E+03
Ba-133	3.3E+05	U-234	3.9E+04
Be-10	1.8E–01	U-235	7.7E+02
C-14	9.1E+05	U-236	1.2E+04
Fe-55	3.0E+08	U-238	1.6E+04
H-3	3.0E+04	Np-237	1.6E+04
Ho-166m	1.2E+03	Pu-238	7.3E+07
Mo-93	3.0E+02	Pu-241	3.9E+09
Nb-93m	3.0E+05	Pu-242	1.2E+05
Nb-94	3.0E+03	Am-241	3.4E+07
Ni-59	3.0E+05	Am-242m	3.9E+05
Ni-63	2.4E+07	Am-243	1.2E+06
Sb-125	3.0E+07	Cm-243	7.7E+05
Zr-93	3.0E+02	Cm-244	1.1E+07
		Cm-245	1.2E+04
Correlated to Cs-137		Cm-246	3.1E+03
Cs-137	7.2E+09		
Cd-113m	4.3E+06		
Cs-134	2.0E+09		
Cs-135	7.2E+04		
Eu-152	5.1E+05		
Eu-154	6.1E+06		
Eu-155	4.1E+08		
I-129	2.2E+04		
Pd-107	7.2E+03		
Pm-147	6.5E+09		
Ru-106	3.6E+07		

Nuclide	Activity (Bq)	Nuclide	Activity (Bq)
Se-79	2.9E+04		
Sm-151	2.2E+07		
Sn-126	3.6E+03		
Sr-90	7.2E+08		
Tc-99	6.5E+06		

11.3 Uncertainties

- General uncertainties are described in Section 1.3.

12 S.24

12.1 Waste type description

12.1.1 Waste package

General

The S.24 waste type consists of 1.73 m^3 concrete moulds with scrap metal in a concrete matrix. Waste material consists of iron/steel, aluminium and other inorganic material generated at Studsvik nuclear research centre. This package is supposed to contain smoke detectors with the dominating radionuclide Am-241. The material is classified as intermediate level waste. All data comes from /Chyssler 1997, Johansson 2000/. The packages are not produced yet.

The physical properties and chemical conditions can vary a lot, but the following restrictions are always applicable on this waste:

- The general restrictions listed in Section 1.1.

Packaging

The mould is a cubic box made of reinforced concrete with the dimensions 1.2×1.2×1.2 m. The walls are 10 cm thick. The reinforcement is made of 12 mm steel bars with a total surface of 10 m^2. The 10-cm mould weighs approximately 1,600 kg. The variations between moulds are minimal.

The maximum allowed weight of a waste package is 5,000 kg.

Treatment and conditioning

Waste, mostly smoke detectors are placed in the mould, which is then filled with concrete. Pouring concrete on top of the package makes a lid.

12.1.2 Materials – chemical composition

The amounts of different materials in a S.24 mould are given in Table 12-1. The waste is fairly well defined and consists mainly of smoke detectors. Some other scrap metal and other inorganic waste could be present. No variation of material is estimated.

The matrix is made of concrete.

Table 12-1. Amounts of different material in waste type S.24 (per package).

Material	Weight (kg)	Area (m^2)	Weight (%)
Iron/steel	113	5.8	13.8
Aluminium	3.8	0.6	0.5
Other inorganic material	139		16.9
Concrete*	565		68.8

* Including 0.13 kg water/kg concrete.

12.1.3 Radionuclide inventory

The dominating nuclide is Am-241, the amount in total is estimated to $5.0 \cdot 10^{11}$ Bq. Other nuclides may be present but amounts will probably be very small.

The surface dose rate limit for this package is 500 mSv/h but the actual dose rate will probably be a lot less. The contamination on surface should not exceed 40 kBq/m^2 for gamma and beta and 4 kBq/m^2 for alpha emitting nuclides on the package. Based on experience of other waste type no contamination is expected.

12.1.4 Waste production

The annual production is estimated to 3, see Appendix A.

12.2 Reference waste type description

12.2.1 Waste package and material

The number of packages that will be produced is calculated to 2 packages during the operation of Studsvik.

Table 12-2 shows the content of different materials and the steel surface areas of the packaging in a reference package.

The assumed composition of the waste in a reference waste package of S.24 is given in Table 12-3.

The estimated volumes of the waste and the packaging materials and the void volume inside the package for a typical package of concrete box are given in Table 12-4. The waste volume is estimated from the weights of the waste components in Table 13-3 and approximate densities of the different material. Average void is estimated to 5% in a package.

Void and waste volume is specified in Table 12-4.

Table 12-2. Estimated reference packaging composition, surface area and thickness of components in concrete mould of S.24.

Component	Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Concrete box	Concrete	1,340		10
Reinforcement	Steel	261	10	12
Lid	Steel	13	1.8	2

Table 12-3. Estimated reference waste composition and surface area and thickness of components in one of S.24.

Material	Weight (kg)	Surface area (m^2)	Thickness (mm)
Iron/steel	113	5.8	
Aluminium	3.8	0.6	5
Other inorganic material	139		5
Concrete*	565		

* Including 0.13 kg water/kg concrete.

Table 12-4. Volumes in one reference waste packages of S.24.

	Volume one package (m^3)
Waste	0.95
Void	0.05

12.2.2 Radionuclide inventory

The reference inventory in one package at the time of production has not been calculated since it is assumed that the activity in the waste comes solely from the smoke detectors, i.e. only Am-241, the amount in total is $5.0 \cdot 10^{11}$ Bq.

A reference waste package of the type S.24 has a surface dose rate of 500 mSv/h and no surface contamination.

12.3 Uncertainties

- General uncertainties are described in Section 1.3.
- This type has not been produced yet, there are large uncertainties regarding all data.

13 Other waste types

Since SFR 1 still is in operation there is always the possibility that new waste types are needed. It is not possible to define these types per se. Since the Silo is not well suited for odd geometries, no other waste types have been assumed.

13.1 Uncertainties

Up to this date no kind of odd wastes have been disposed off in the Silo, in the future odd waste types could be disposed off.

14 References

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