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Long-lived intermediate level waste from Swedish nuclear power plants

Reference Inventory

Björn Herschend
Svensk Kärnbränslehantering AB

April 2014

Svensk Kärnbränslehantering AB
Swedish Nuclear Fuel
and Waste Management Co

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

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The original report, dated April 2014, was found to contain editorial errors which have been corrected in this updated version.

Summary

In order to accommodate all radioactive waste from the Swedish nuclear power plants within the Swedish system for managing radioactive waste, a repository for long-lived low and intermediate level waste (SFL) is planned for by SKB. Compared to the amounts of short-lived operational and decommissioning waste, the total amount of long-lived waste from the nuclear power plants is relatively small, ~4,600 tonnes. However, the total activity is several orders of magnitude higher, in particular for long-lived nuclides. The waste is mostly in the form of scrap metal and most of the activity originates from neutron irradiation of the construction material of reactor components close to the core.

The radionuclide inventory is mainly based on calculations of activity content where the specific operational histories and material compositions for the core components have been accounted for. Initially, the activity is dominated by short-lived nuclides such as Fe-55 and Co-60, but at closure of SFL and onwards the activity is dominated by Ni-63 and later Ni-59. The total activity at the point of closure is 2.1×10^{17} Bq.

Actinides and fission products are also present in the waste, but with much lower activities. This activity originates from fissile material in neutron detectors and surface contamination due to fuel leakage.

For the volume estimate, it has been assumed that waste will be disposed of in steel tanks. The same type of tank is currently used for interim storage of segmented reactor internals already taken out of the reactors as part of maintenance. It is further assumed that the waste will be stabilized with grout prior to disposal in SFL. For the PWRs, the waste includes the reactor pressure vessels and here the option of disposing the each RPV as a whole (with or without the reactor internals intact) has also been considered. Steel is the predominant material in the waste although alloys such as Zircalloy and Inconel can be found in small amounts. Control rods from BWR also contribute with boron carbide and hafnium.

Sammanfattning

För att kunna omhänderta allt radioaktivt avfall från de svenska kärnkraftverken planerar SKB för ett slutförvar för långlivat låg- och medelaktivt avfall (SFL) inom det svenska systemet för hantering av radioaktivt avfall. Jämfört med mängden kortlivat drift- och rivningsavfall, är den totala mängden långlivat avfall från kärnkraftverken förhållandevis liten, ~4 600 ton. Den totala aktiviteten är dock flera storleksordningar högre, särskilt för långlivade nuklider. Avfallet består till största delen av skrot och det mesta av aktiviteten har sitt ursprung i neutronbestrålning av konstruktionsmaterialet i de reaktorkomponenter som befinner sig närmast härden.

Radionuklidinventariet är huvudsakligen baserat på beräkningar av aktivitetsinnehåll där de härdnära komponenternas specifika drifthistorik och materialsammansättning beaktats. Initialt domineras aktiviteten av kortlivade nuklider som t ex Fe-55 och Co-60, men från och med tidpunkten för förslutning av SFL domineras aktiviteten av Ni-63 och senare Ni-59. Totalaktiviteten vid förslutning uppgår till $2{,}1 \times 10^{17}$ Bq.

I avfallet förekommer även aktinider och fissionsprodukter, men med väsentligt lägre aktivitet. Denna aktivitet härrör från fissilt material i neutrondetektorer samt ytkontamination till följd av bränsleläckage.

Vid uppskattning av avfallsvolym har det antagits att avfallet deponeras i ståltankar. Samma typ av tankar används idag för mellanlagring av segmenterade interndelar som tagits ur drift. Vidare antas att avfallet kringgjuts med betong innan deponering. Avfallet från PWR innefattar reaktortankarna, och för dessa har även möjligheten att deponera tankarna hela (med eller utan interndelar) beaktats.

Stål är det dominerande materialet även om andra legeringar så som Zircalloy och Inconel även finns i avfallet. Styrstavar från BWR bidrar även med borkarbid och hafnium.

Contents

1 Introduction

In order to enable safe disposal of long-lived low and intermediate level waste a repository, SFL, will be built with planned start of operations in 2045. Investigations (SKB 1999a, b) of the long term safety of earlier SFL-concepts, have been performed with a waste inventory from 1998 (SKBdoc 1416968). This inventory presents, not only the waste, but also a concept design for the repository as well as future waste packaging for different waste types.

The waste in the reference inventory from 1998 includes neutron irradiated reactor internals from the nuclear power plants (NPP), legacy-, operational- and decommissioning waste from the Studsvik site and operational and decommissioning waste from the central interim storage (Clab) and the encapsulation plant for spent nuclear fuel (Clink). However, the remit of the SFL repository has changed significantly since 1998:

- The planned operational time for the NPPs have been prolonged.
- The repository for short-lived low and intermediate level waste (LLW and ILW, respectively) is planned to remain in operation until end of 2075, and will accommodate all the LLW and ILW from Clab/Clink, which is assumed to be short-lived.
- Investigations into legacy waste have revealed larger waste volumes than previously expected.

In light of these changes, it was decided to update the reference inventory. The aim was to update the inventory with current knowledge about existing waste as well as taking into account the recently finalized decommissioning studies of the NPPs (SKBdoc 1403739, Lindow 2012, Anunti et al. 2013, Larsson et al. 2013, Hansson et al. 2013).

For the BWR core components, the reference inventory from 1998 was largely based on activity calculations for the model reactor BWR 3000 (similar to the F3 and O3 rectors). Activity in PWR components was estimated from calculations for the R2 reactor. With the recent decommissioning studies and reported data on existing waste a more detailed and reactor specific description of the waste can be achieved, in terms of material content and activity.

However, even though investigations into the legacy waste have given a better view of the actual waste volumes, it has also become clear that our current knowledge about the waste content is still very limited. For this reason, the present report is limited to an updated reference inventory for long-lived ILW from the Swedish NPPs since more investigation of the legacy waste will be required to produce a complete reference inventory for the future SFL.

This reference inventory is based on the long-lived waste described in the decommissioning studies of the NPPs as well as reported inventories of long-lived waste in interim storage at the NPPs and Clab. The inventory includes twelve nuclear reactors from the four large scale NPPs as well as the pressurized heavy-water reactor Ågesta. Table 1-1 lists the reactors with start and end of operation as well as the assumed thermal power used for estimating the radionuclide inventory. In general, this corresponds to the highest expected power level, and in the earlier years of operation the power levels were significantly lower. This has introduced some conservatism in the calculated inventory.

The components contributing to the long-lived waste are described in Chapter 2. The methodology for calculating the radionuclide inventory is explained in Chapter 3. In Chapter 4, the general assumptions constituting the reference case for estimating the total waste inventory are presented and the resulting reference inventory is presented in Chapter 5. An analysis of the uncertainties in volume, material and activity is given in Chapter 6.

Table 1-1. Swedish nuclear reactors included in this inventory with dates for start and end of operation and the assumed thermal power used to estimate the radionuclide inventory. For NPPs in operation, the end date is set to correspond to the expected operational time as per 2012.

Reactor	Type	Thermal power/MW	Start	End
B1	BWR	1,800	Jan. 1976	Nov. 1999
B2	BWR	1,800	Jan. 1977	May 2005
F1	BWR	3,253	Dec. 1980	Dec. 2040
F2	BWR	3,253	Jul. 1981	Jul. 2041
F3	BWR	3,300[1]	Aug. 1985	Aug. 2045
R1	BWR	2,600	Jan. 1976	Jan. 2026
R2	PWR	2,652	May 1975	May 2025
R3	PWR	3,174	Sep. 1981	Sep. 2041
R4	PWR	3,292	Nov. 1983	Nov. 2043
O1	BWR	1,375	Feb. 1972	Feb. 2032
O2	BWR	1,800[1]	Aug. 1974	Aug. 2034
O3	BWR	3,900	Mar. 1985	Mar. 2045
Å	PHWR	65	Jul. 1963	Jun. 1974

[1]Planned increase to 3,750 MWt (F3) and 2,300 MWt (O2) have not been accounted for.

2 Description of neutron induced components from the Swedish nuclear power plants

Long-lived waste from the NPPs typically consists of neutron irradiated components from inside and close to (0.5–1.0 m) the reactor core. For most of these components, only a specific part resides close enough to the core to generate significant induced activity. For the remaining parts, surface contamination is the only relevant source of activity, and these parts can be separated from the induced material and disposed of in SFR. This has been taken into account when estimating the waste volume from each component. However, since details are not known about how components will be segmented to separate long-lived from short-lived waste, the total activity calculated for each component has been attributed to the parts contributing to the long-lived waste.

In the following, core components from the different reactors are presented. Unless otherwise specified, reported weights correspond to the assumed activated part of each component, used in the calculation of neutron induced activity. Specified weights may be significantly lower than the weights of entire components depending on how the components have been modeled in the calculation of neutron induced activity. Differences between reactors for the same component type do not always represent a difference in actual component weight.

Some components have been replaced as part of reactor maintenance and in some cases these components have already been segmented. The sum of reported segment weights do not always match the weights reported here and the reasons for this discrepancy are both uncertainties in actual component weights and that the actual separation of activated and non-activated material has been done differently from the assumptions made for the inventory calculations. Furthermore, old and new components may differ in weight, and this has only been accounted for when different weights have been explicitly reported.

2.1 BWR components

2.1.1 Core shroud

The core shroud (Figure 2-1) is an internal tank surrounding the core inside the reactor vessel. It is open at the top and bottom and the main function of the core shroud is to separate the downward flowing cooling water from the water flowing upwards through the core. The core shroud also acts as a support for the shroud head, core grid, steam separator and steam dryer as well as material test tubes mounted on the core shroud walls.

The core shroud is made of austenitic stainless steel and consists of a cylindrical part with an upper and lower flange. At the bottom of the cylinder there is a frame structure shaped to accommodate the core plate. The lower flange stands on a support frame, mounted to the bottom part of the reactor vessel. The frame is assumed to have only surface contamination and has therefore been excluded from the inventory (waste amounts as well as activity).

Weight- and material specification for core shrouds from the different reactors are given in Table 2-1. Core shrouds have been replaced in F1, F2 and O1.

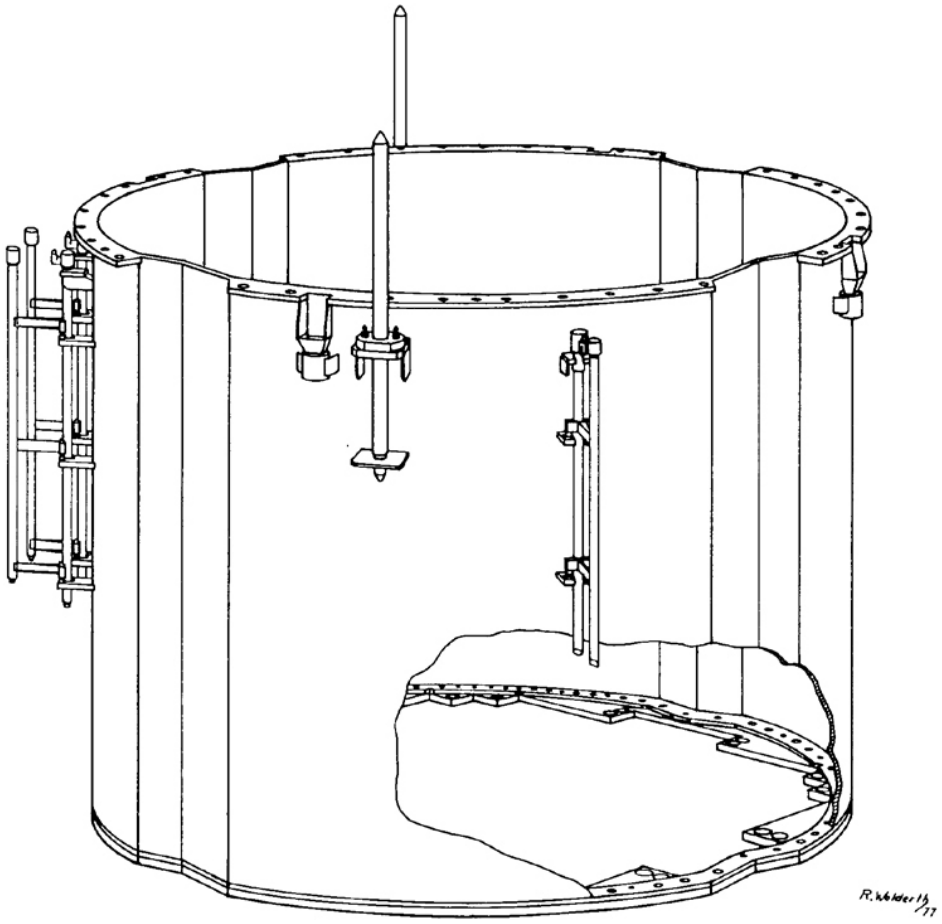

Figure 2-1. Schematic figure of the core shroud showing the cylindrical jacket and the bottom frame shaped to accommodate the core plates (SKBdoc 1416968).

Table 2-1. Weight, material specification and number of core shrouds in the waste to SFL. The reported weights correspond to the assumed activated part for each component.

Reactor	Weight/kg	Material	Number of components
B1/B2	24,400	SIS2333	2
F1/F2	24,000	AISI 316L	4
F3	32,000	SIS2352	1
O1	26,600	SIS2353	2
O2	11,000[1]	SIS2333	1
O3	32,000	SIS2352	1
R1	12,300	SIS2333	1

[1]The total component weight is the same as the reported weight for B1/B2 (24,400 kg).

2.1.2 Core grid

The core grid (Figure 2-2) is mounted on top of the core shroud and the main function is to keep the upper part of the fuel assemblies and the detector guide tubes in the correct position. The core grid consists of a frame and a grid. The grid consists of square holes with room for 4 fuel assemblies in each. The core grid is made of austenitic stainless steel.

Weight- and material specification for core grids from the different reactors are given in Table 2-2. Core grids have been replaced in B1, B2, F1, F2, O1, O2 and R1.

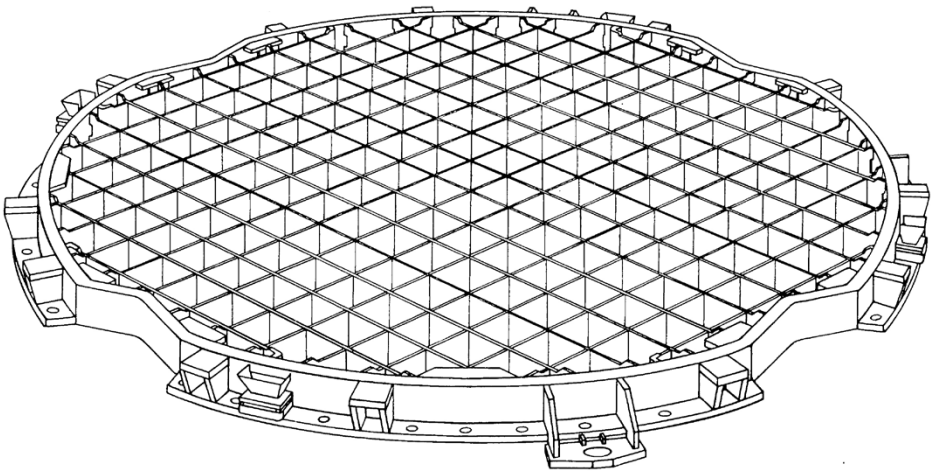

Figure 2-2. *Schematic picture of the core grid (SKBdoc 1416968).*

Table 2-2. Weight, material specification and number of core grids in the waste to SFL. The reported weights correspond to the assumed activated part for each component.

Reactor	Weight/kg	Material	Number of components
B1/B2	2,500	SIS2353	4
F1/F2	8,900	AISI 316L	4
F3	6,000	SIS2352	1
O1	4,100	SIS2353	2
O2	2,500	SIS2353	2
O3	6,000	SIS2352	1
R1	7,000	SIS2353	2

2.1.3 Core shroud head

The core shroud head is lying directly on top of the core grid. The shroud head directs the steam/water flow from the core to the steam separators and separates this flow from the circulation water coming back from the steam separator and steam dryer.

A picture of the shroud head is given in Figure 2-3. It is made of austenitic stainless steel and consists of a cupola, a flange, and connection pieces for the steam separators, an upper connection ring for the steam dryer with support legs and supplying tubes for the core spray (for the reactors with the core spray still in operation). The cupola, flange and the connection pieces will contain significant induced activity.

Weight- and material specification for shroud heads from the different reactors are given in Table 2-3. The shroud heads have been replaced at least once in all reactors.

Table 2-3. Weight, material specification and number of core shroud heads in the waste to SFL. The reported weights correspond to the assumed activated part for each component.

Reactor	Weight/kg	Material	Number of components
B1/B2	16,500	SIS2352	4
F1/F2	19,500	SIS2333	4
F3	31,350	AISI 316L	2
O1	11,000	AISI 316L	2
O2	14,160	AISI 316L	3
O3	31,350	AISI 316L	2
R1	25,300	AISI 304L	2

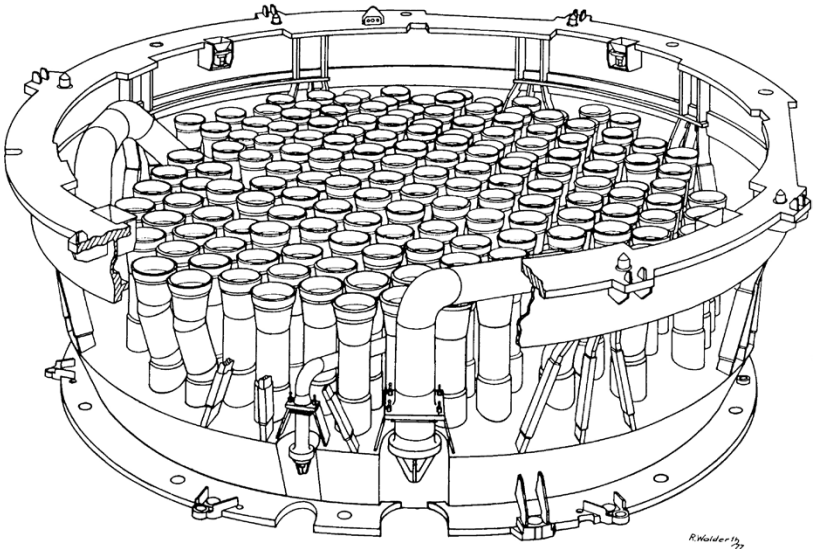

Figure 2-3. Schematic picture of a core shroud head (SKBdoc 1416968).

2.1.4 Control rod guide tubes

BWR control rods enter the core through the control rod guide tube (Figure 2-4). The bottom of the tube is mounted at the bottom of the reactor vessel and the upper part, including the core plate, is supported by the bottom of the core shroud. Each core plate supports four fuel boxes. The control rod guide tube also directs a portion of the cooling water flow to provide cooling for the control rods and the space between the fuel boxes.

The uppermost part of the tube together with the core plate at the top will contain significant neutron induced activity and will need to be disposed of as long-lived waste. Each guide tube will contribute with ~75 kg of long-lived waste except for the B1 and B2 reactors where the expected weight is 45 kg per guide tube. The tube and the upper plate are made of stainless steel (SIS2333).

The control rod guide tubes remain intact in all the BWRs and therefore only waste from decommissioning is expected from this component category. The number of guide tubes in each reactor is the same as the reported number of control rods in Table 2-5.

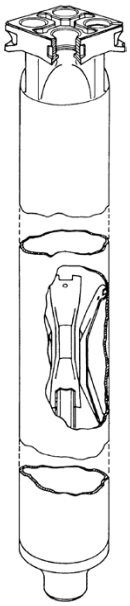

Figure 2-4. Schematic picture of a control road guide tube showing the upper part with the core plate expected to become long-lived waste (SKBdoc 1416968).

2.1.5 Core spray

The core spray system is a safety system, which allows cooling of the core by sprinkling water on the core from the top in the event of a loss of coolant accident. In some reactors, e.g. O2, the system has the additional function of making and maintaining the core subcritical by sprinkling the core with boron solution.

The core spray consists of a support bolted to the core shroud head, and a core spray (Figure 2-5). The parts are mainly constructed from austenitic stainless steel. The core spray consists of two equal parts, each including a main tube with two different dimensions and a fork-shaped distributor with parallel distributor pipes and spray nozzles. The two distributor forks are inter-lapping, and independent, providing redundancy.

Waste from the core sprays also include the upper part of the pipes providing the cooling water and boron solution to the core spray. These pipes enter through the bottom of the core shroud, and therefore the upper part will contain neutron induced radioactivity.

Weight- and material specification for core sprays from the different reactors are given in Table 2-4. The core sprays of B1, B2 and O2 have been replaced and in the F1, F2, F3 and O3 reactors the core sprays are no longer in operation and have been permanently removed.

Table 2-4. Weight, material specification and number of core sprays in the waste to SFL. The reported weights correspond to the assumed activated part for each component.

Reactor	Weight/kg	Material	Number of components
B1/B2 (old)	2,900[1]	SIS2333	2
B1/B2	3,130[2]	SIS2353	2
F1/F2	2,840	SIS2333	2
F3	9,600	SIS2333	1
O1	9,700[2]	AISI 316L	1
O2	3,200[2]	SIS2353	3
O3	9,600	SIS2333	1
R1	5,840[2]	AISI 316L	1

[1]Including upper part of 323-pipes.
[2]Including upper part of 323- and 351-pipes.

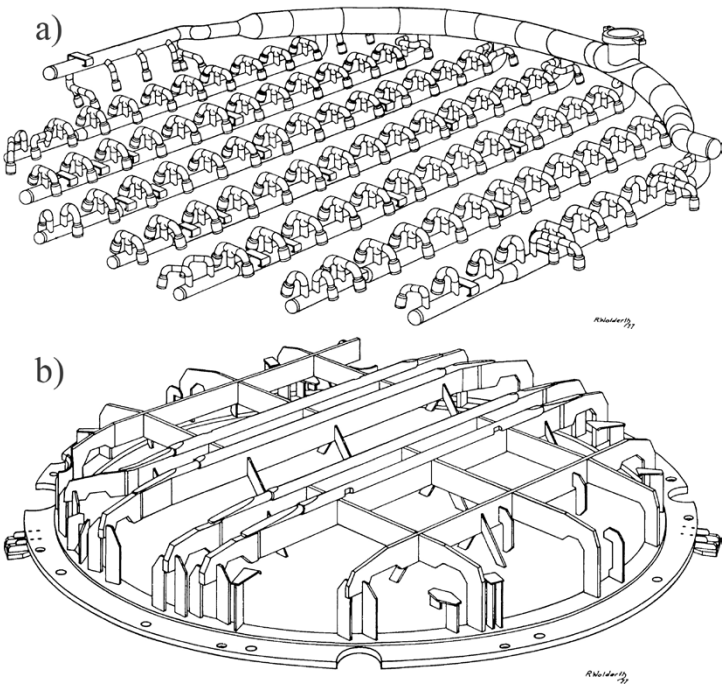

Figure 2-5. *Schematic picture of a core spray (a) and support frame (b) (SKBdoc 1416968).*

2.1.6 Control rods

BWR control rods are cruciform permanent absorbers used for power regulation. The control rods are constructed from four absorber plates, ~4 m in length, mounted together perpendicularly to each other to form a shape with the transection of a cross. The absorber plates are mounted on a steel rod at the lower end and have a welded handle at the upper end (Figure 2-6).

In the inventory, two types of control rods are modeled. Control rods for power regulation and control rods used during shutdown. The first type has a short operational time (4–6 years) and is subject to intense neutron irradiation, whereas the latter type is used only during start-up and shutdown, and have much longer operational time, ~20 years.

The information on control rods taken out of operation is stored in the database for interim storage in Clab (DRAAK version 1.07p). However, the operational history is not generally documented in sufficient detail to allow for radionuclide inventories to be calculated individually for each control rod. Hence, all control rods in interim storage are assigned activity corresponding to a number-weighted average for all control rod types in each reactor, calculated for the maximum operational time of each control rod type. This is a reasonable assumption since control rods are often used in different positions prior to being decommissioned. The same assumption is used for assigning activity to control rods from future operation and decommissioning.

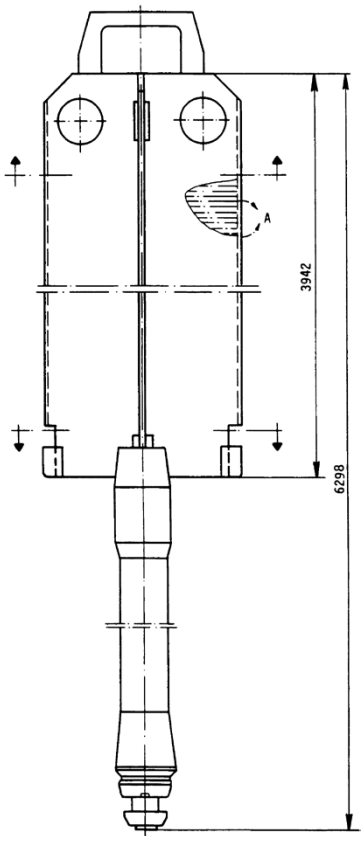

Figure 2-6. *Schematic picture of a BWR control rod including absorbers, handle and steel rod (SKBdoc 1416968).*

The absorbing materials include boron carbide and in most cases hafnium. The remaining weight of the control rod is assumed to be made of stainless steel. The expected waste mass from the control rods differ depending on the information source:

- 132 kg is used in the decommissioning inventory, and therefore this weight is used for the final set of control rods from reactors that remain in operation.
- For control rods in the interim storage Clab, and for control rods stored at the NPPs, the reported weights have been used. These weights are somewhat lower than for the total control rod because the shafts of the rods are removed prior to transport to Clab.
- For the forecast on control rods from future operation of the NPPs, the average weight of the presently stored rods in Clab has been used (calculated individually for each reactor).

For control rods originating from the same reactor, the hafnium and boron carbide amounts are assumed to be the same and the difference in weight is taken into account by varying the amount of steel. Table 2-5 shows the number of control rods in each reactor as well as the nominal material composition. The number of control rods expected in the SFL waste is given in Table 2-6. The number of decommissioned control rods has been taken from the DRAAK database as well as reported data on interim storage of control rods from the NPPs. The forecast for the future number of decommissioned BWR control rods has been calculated using the assumption that the full set of control rods have to be replaced with a 15-year interval except the last set, which will be kept in operation for up to 20 years.

Table 2-5. Material specification and number of control rods in use in each reactor. The reported weights correspond to the contribution from each material to the assumed activated part of a single control rod.

Reactor	Steel/kg	Hafnium/kg	Boron carbide/kg	Number of control rods in operation
B1/B2	73.2	3.2	10.2	109
F1/F2	73.2	3.2	10.2	161
F3	74	3.2	10.2	169
O1	100.5	3.1	10.1	112
O2	81.7	3.1	10.1	109
O3	74	3.2	10.2	169
R1	74.1	0	10.5	157

Table 2-6. Number of control rods in the SFL waste.

Reactor	Number of control rods		
	Existing (interim storage)	Forecast (operation)	Forecast (decom.)
B1	218	0	0
B2	228	0	0
F1	72	322	161
F2	117	322	161
F3	307	338	169
O1	63	112	112
O2	126	218	109
O3	297	338	169
R1	153	157	157
Total	**1,581**	**1,807**	**1,038**

2.1.7 Detectors and guide tubes

For monitoring neutron flux, several detectors are used in the BWRs. These detectors are located in thin steel guide tubes placed vertically inside the core. Both the detectors and the guide tubes are subject to intense neutron irradiation, which causes buildup of induced activity. Furthermore, the detectors themselves contain small amounts of fissile material, which also contribute to the activity in the form of actinides and fission products.

Most of the waste comes from local power range monitors (LPRM), which have fixed positions during reactor operation. There are additional detectors such as source range-, intermediate range-, short- and intermediate range- and wide range monitors (SRM, IRM, SIRM and WRNM, respectively), which are temporarily inserted into guide tubes at lower power levels. The LPRM detectors therefore receive the most intense neutron flux and have the shortest operational time. Table 2-7 lists the number of neutron detectors for the different reactors. In addition to the above mentioned detectors, there are also travelling in core probes (TIP) that are movable inside the LPRM guide tubes.

The detector type is not always specified for existing waste containing detector material. Therefore, the activity is generally estimated from a number-weighted average for all detector types (except TIP), calculated for the maximum operational time of each detector type. This assumption is used both for existing waste and for the forecast on waste from future operation and decommissioning.

Since the existing waste from detectors has often been reported as bulk weight of segmented detectors including guide tubes, and occasionally mixed with other scrap metal, it is difficult to estimate the exact number of detectors in the inventory. Instead the contribution to the inventory from detectors has been calculated with the following assumptions:

- Volumes for existing waste from decommissioned neutron detectors have been estimated from reported data from the NPP as well as data from the database for interim storage of long-lived waste in Clab (DRAAK version 1.07p).
- A forecast for the expected volumes of waste from neutron detectors have been calculated from the assumption that the full set of PRM-detectors have to be replaced with a 10 year interval.

2.1.8 Boron plates

Temporary absorbers in the form of boron steel plates have been used in the first cores of B1, O1, O2 and R1. Most of the absorbers are stored in Clab, except the absorbers from O1, which are still stored at the NPP. Approximately 1,000 absorber plates are kept in interim storage.

Material and surface data has been taken from the reference inventory from 1998 (SKBdoc 1416968). No activity data has been reported for this waste. Instead, activity has been calculated with the following assumptions:

- Time of operation has been set to one year.
- Neutron flux levels have been assumed to be the same as for the LPRM tubes.
- The material composition for a single absorber has been assumed to be 0.7 kg SIS2353 and 19.3 kg boron steel (similar to SIS2353, but with 0.32 % boron).
- Surface contamination has been calculated in the same manner as for stainless steel with the specified area of 1.39 m^2 (corresponding to the total area reported in the reference inventory from 1998 (SKBdoc 1416968) for a single absorber).

Table 2-7. The type and number of neutron detectors in operation in each reactor.

Reactor	Number of LPRM detectors per reactor	Number of guide tubes for SRM, IRM etc.	Number of TIP-detectors
B1/B2	24	4	2
F1/F2	37	8	4
F3	37	8	4
O1	23	4	2
O2	24	4	2
O3	37	8	4
R1	36	6	2

2.1.9 Fuel assembly components

In the existing long-lived waste, components from the fuel assemblies are present. These include fuel boxes, spacers, springs and handles. However, this type of waste is not expected in the forecast for waste from future operation and decommissioning.

Both in terms of waste volume and in terms of activity this waste category is dominated by irradiated fuel boxes (Figure 2-7). Table 2-8 shows the material composition of the fuel boxes used for calculating neutron induced activity. However, the actual weights reported for fuel boxes in the waste are closer to 30 kg. In general, it is assumed that the specified weight is divided between steel and Zircalloy in the same proportions as the weights specified in Table 2-8. For compacted boxes from Oskarshamn, it is assumed that the unit weight is 30 kg, containing only the Zircalloy part. No updated model exists for calculating the radionuclide inventory of the fuel boxes from O1. Therefore, induced activity is assumed to be the same as for fuel boxes from O2.

For fuel boxes from Oskarshamn and Forsmark, no surface contamination levels have been determined, since surface contamination on Zircalloy is not covered by the model (Section 3.3). For the fuel boxes from Barsebäck, it is obvious from the reported total activities that surface contamination has been accounted for, but since separation of induced activity and surface contamination has not been presented for this waste category the activity is treated as induced activity in the present report.

Table 2-8. Assumed material composition of fuel boxes in the SFL waste. The reported weights correspond to the contribution from each material to the waste mass of a single fuel box.

Reactor	Steel/kg	Zircalloy/kg	Total/kg
B1/B2	10.6	31.5	42.1
F1/F2	7.8	34.6	42.4
F3	10.6	31.5	42.1
O1	–	30.0	30.0

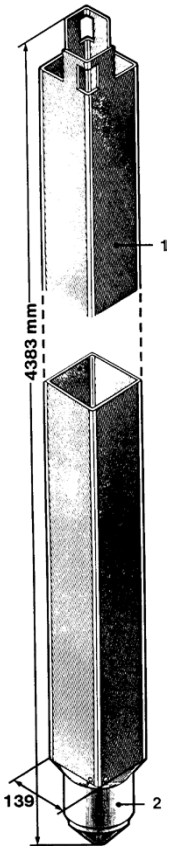

Figure 2-7. Schematic picture of a fuel box (SKBdoc 1416968).

2.1.10 Miscellaneous scrap

In the present inventory there are small amounts of (~10.5 tonnes in total) scrap metal from reactor internals, but the exact origin (component) is unknown. This waste has been accounted for in terms of weight and has been assumed to be SIS2333 stainless steel. No activity has been attributed to this waste since all the activity originating from the core region is accounted for in other waste types.

In the waste there are also twelve start-up neutron sources from R1. No information about the material or radionuclide contents of this waste has been reported, but the waste is in the form of metal tubes (similar to PRM-tubes) and the neutron source is Cf-252. This waste type should have been generated at all the BWRs. By extrapolating the reported number of neutron sources from R1 to all BWRs (scaling with core size), it can be estimated that there should be ~90 neutron sources in the waste to SFL, weighing less than one tonne in total.

2.2 PWR components

In terms of component weights and material, the three PWRs R2, R3 and R4 are considered identical. Table 2-9 summarizes the weight and material composition of the PWR components in the long-lived waste. Waste amounts will be lower if the components are segmented due to the fact that parts with insignificant induced activity can be removed an treated as short-lived waste.

2.2.1 Reactor pressure vessel

The reactor pressure vessel in a Ringhals PWR is a steel cylinder with an inner diameter of 4 m, with half spheres at the top and bottom resulting in a total height of 13 m. The walls are 20 cm thick. The vessel is made of carbon steel with an internal stainless steel cladding. Due to the compact design of the PWR RPVs the amount of induced long-lived activity in the core region will require disposal in SFL.

The RPVs from R2, R3 and R4 can be assumed to be identical in terms of waste volumes and material. The entire vessel weighs 327.5 tonnes, out of which 245 tonnes has been considered long-lived waste in the decommissioning studies (Hansson et al. 2013). Hence, these weights have also been used in this inventory.

2.2.2 Burnable absorbers

In the inventory from 1998 (SKBdoc 1416968), so called burnable absorbers or burnable poisons from PWR were included. In the first core of a PWR reactor, burnable absorbers in the form of absorber rod bundles are used. They are made from stainless steel tubes covered with boron glass and are positioned in the control rod guide tubes. At present, these absorber rods are stored together with the spent fuel. Like the control rods from PWR, they have been excluded from the present inventory since they are likely to be disposed of with the spent fuel.

2.2.3 PWR internals

PWR internals consist of the core baffle, core barrel, thermal panels, upper internals assembly and the lower core plate. All components are made of ASIS 304 stainless steel.

Table 2-10 lists the parts of the reactor internals considered in the activity calculations (making up the 94 tonnes reported as long-lived waste from internals in Table 2-9). The whole set of internals are assumed to remain intact for the entire operational life time of the reactor. Instrumental guide tubes (so called flux thimbles) are not specifically accounted for in the calculations, but the dismantled tubes in interim storage at the NPP are included in the waste volume. The activity of these tubes is assumed to be accounted for in the total activity for the reactor internals.

Table 2-9. Weight and material specifications for components from PWR in the SFL waste. The reported weights correspond to the expected waste mass from each component.

Component	Total weight/kg	Weight of activated part/kg	Material
RPV	327,500	245,000	AISI 304 and carbon steel
Internals	135,500	94,000	AISI 304
Biological shield	588,000	370,318	Concrete and carbon steel (reinforcement)

Table 2-10. Weight of PWR internals considered in the calculations of activity.

Component	Weight/kg
Core baffle	14,000
Core barrel and thermal panels	35,000
Lower core plate	25,000
Upper internals assembly	2,000

2.2.4 Biological shield

The biological shield is a reinforced concrete structure surrounding the RPV to protect the surroundings from gamma- and neutron radiation from the reactor core. The shield is made of reinforced concrete with a total weight of ~588 tonnes, 88 tonnes being steel reinforcement bars.

The biological shield of the PWRs has a total C-14 activity of more than 10^{10} Bq, and has therefore been excluded from the volume of short-lived decommissioning waste. The C-14 activity is concentrated to the innermost concrete.

When estimating neutron induced activity in the concrete, only ~370 tonnes of the total concrete weight is considered. The remaining concrete is assumed to be non-activated. For this reason it can be assumed that at least the reinforcement and the outermost 130 tonnes of concrete could be removed and treated as short-lived waste. However, since the entire system has been excluded from the short-lived waste fraction, the entire biological shield must be accounted for in this inventory.

2.3 Components from Ågesta

The Ågesta reactor is a pressurized heavy-water reactor. It went critical for the first time in 1963 and was permanently shut down in 1974. The RPV (Figure 2-8) is designed with an ellipsoidal bottom and a flat vessel head including refueling and control rod stand pipes. Inside the RPV there are two (inner and outer) radial thermal shields and an upper thermal shield immediately below the vessel head. At the bottom of the vessel there is a flow distributor and a filler body as well as inlet- and outlet nozzles for the heavy water.

The waste consists of the reactor pressure vessel including the head, thermal shields inside the reactor and control rods. Table 2-11 shows the different components, materials and weights.

Table 2-11. Weight and material specifications for components from the Ågesta PHWR in the SFL waste. The reported weights correspond to the expected waste mass.

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

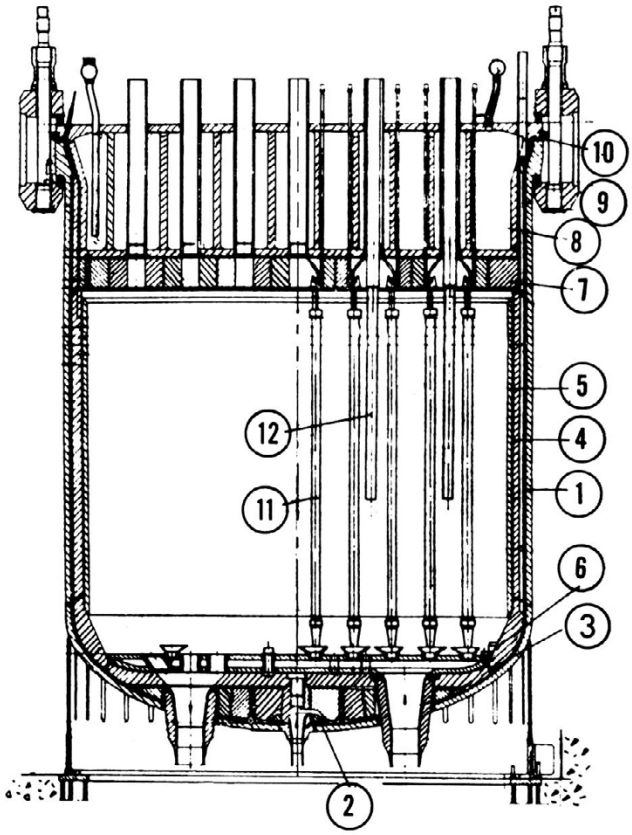

Figure 2-8. *Schematic picture of the Ågesta RPV (McHugh 1964, Figure 7, p 13). 1. RPV. 2. Filler ring. 3. Filler body. 4. Outer thermal shield. 5. Inner thermal shield. 6. Distributor. 7. Upper thermal shield. 8. RPV head. 9. Flange. 10. Seal ring. 11. Fuel assembly. 12. Control rod.*

3 Method for estimating radionuclide inventory in reactor internals

The radionuclide inventory for the reactor internals is based on calculations rather than direct measurements. The methods used are the same as those used for the recent decommissioning studies for the Swedish nuclear power plants (SKBdoc 1374191), which have been developed by Studsvik ALARA Engineering. In the following, the models used for calculating the radionuclide inventory are described.

3.1 Sources of radioactivity

The radioactivity in reactor internals originates mainly from direct neutron irradiation and subsequent transmutation of stable nuclides into radionuclides. To a lesser extent, surface contamination in the form of activated corrosion products and dissolved fuel also contribute to the overall activity. Hence, three different models have been used to estimate the radionuclide inventory.

3.2. Neutron induced activity

The change in nuclide composition in a material can be described by the Bateman equation

$$\frac{dN_i}{dt} = -\sum_{j\neq i} N_i\left[\lambda_{ij} + \int \sigma_{ij}(E)\phi(E)dE\right] + \sum_{j\neq i} N_j\left[\lambda_{ji} + \int \sigma_{ji}(E)\phi(E)dE\right] + S_i \qquad 3\text{-}1$$

Where, N_i is the number of atoms of species i. λ_{ij} is the decay constant for the decay of i into j. σ_{ij} is the cross section for the reaction of i into j at the energy E due to the neutron flux ϕ. S_i is production of species i due to fission. The first term is the sum of all decay paths and burnup of species i, the second is the sum of all reactions producing species i due to neutron capture or decay of heavier nuclides. For neutron induced activity only decay and neutron induced reactions are applicable.

The calculations of neutron induced activity have been based on neutron transport calculations. Taking into account the specific geometry, material- and operational data for the different reactors, the neutron flux has been calculated using Monte-Carlo simulations of neutron transport. The reaction cross sections have mostly been taken from the European activation file (Forrest et al. 2007), in the case of Barsebäck and Ågesta the internal cross section library of FISPACT (Forrest 2002) has been used.

For the Barsebäck and Ågesta reactors the Bateman equation has been solved numerically using the FISPACT code, explicitly taking account of the change in material composition due to neutron irradiation and decay. However for the other reactors, the Bateman equation has been solved analytically using a somewhat simpler three group approach, ignoring the change in abundance of stable nuclides. In the following, the three group approach is described.

3.2.1 Activation rate

The activation rate due to neutron irradiation can be written as

$$R = \sum_{j\neq i} N_j \int \sigma_{ji}(E)\phi(E)dE = \sum_{j\neq i} \int \Sigma_{ji}(E)\phi(E)dE \qquad 3\text{-}2$$

where N_j is the number of atoms of nuclide j, Σ_{ji} is the macroscopic cross section for the reaction of nuclide j to form nuclide i and φ is the neutron flux. In the three group model, the activation rate is divided into three terms

$$R = R_{th} + R_{epi} + R_{fiss} \qquad 3\text{-}3$$

for reactions due to neutrons in the thermal, epithermal and fission (fast) energy groups. The upper boundary for thermal neutrons is set to 0.5 eV. Below this energy the neutrons are assumed to be in thermal equilibrium with the surrounding medium and follow the Maxwell speed distribution:

$$f(v)=\sqrt{\left(\frac{m}{2\pi kT}\right)^3}\,4\pi v^2\cdot e^{-\frac{mv^2}{2kT}} \qquad 3\text{-}4$$

where m is the neutron mass, k is the Boltzmann constant, v is the neutron speed and T is the temperature.

The most probable speed at a given reference temperature is given by

$$\frac{df(v)}{dv}=0\rightarrow v_T=\sqrt{\frac{2kT}{m}} \qquad 3\text{-}5$$

And the average speed is

$$\langle v\rangle=\int v\cdot f(v)dv=\sqrt{\frac{8kT}{\pi\cdot m}}=\frac{2}{\sqrt{\pi}}v_T \qquad 3\text{-}6$$

The neutron flux can be written as

$$\phi(v)=v\cdot n(v)=v\cdot n\cdot f(v)=\frac{4\cdot n\cdot v^3}{v_T^3\sqrt{\pi}}\cdot e^{-\frac{v^2}{v_T^2}} \qquad 3\text{-}7$$

where $n(v)$ is the neutron density distribution as a function of speed and n is the total neutron density. The macroscopic cross section in the thermal region can be assumed to be inversely proportional to the square root of the neutron energy. Hence, the cross section can be determined at some reference temperature and extrapolated over the thermal energy range.

$$\Sigma_a(v)=\Sigma_0\sqrt{\frac{E_0}{E}}=\Sigma_0\frac{v_0}{v} \qquad 3\text{-}8$$

$$v_0=\sqrt{\frac{2kT_0}{m}} \qquad 3\text{-}9$$

The thermal activation rate can then be written as

$$R_{th}=\int\Sigma_a(v)\phi(v)dE=\int\Sigma_a(v)\phi(v)\cdot vdv \qquad 3\text{-}10$$

$$R_{th}=\int\frac{\Sigma_0 v_0}{v}n(v)\cdot vdv=\frac{4\Sigma_0 v_0 n}{v_T^3\sqrt{\pi}}\int v^2\cdot e^{\frac{v^2}{v_T^2}}dv=\frac{4\Sigma_0 v_0 n}{v_T^3\sqrt{\pi}}\left[\frac{v_T^3}{4}\sqrt{\pi}\right]=\Sigma_0 v_0 n \qquad 3\text{-}11$$

The overall thermal neutron flux can be written as

$$\Phi_{th}=n\cdot\langle v\rangle=n\frac{2}{\sqrt{\pi}}v_T=n\cdot v_0\sqrt{\frac{4T_0}{\pi\cdot T}} \qquad 3\text{-}12$$

With Φ from the neutron transport simulations, the reaction rate can be calculated according to

$$R_{th}=\Phi_{th}\sqrt{\frac{\pi\cdot T_0}{4T}}\Sigma_0 \qquad 3\text{-}13$$

Above about 0.1 eV (E_1), the epithermal range starts where the neutron flux is assumed to be inversely proportional to the energy. Hence, the neutron flux can be written as

$$\Phi(E)=\begin{cases}\Phi_{th}(E)+\dfrac{\Phi_{epi}}{E}, & \text{for } E>E_1\\ \Phi_{th}(E), & \text{for } E<E_1\end{cases} \qquad 3\text{-}14$$

The macroscopic cross section can be assumed to be linear up to 0.5 eV (E_2), but above this energy resonance contributions must be taken into account (Σ_{res}).

$$\Sigma_a(E) = \begin{cases} \Sigma_0\sqrt{\frac{E_0}{E}} + \Sigma_{res}(E), \text{ for } E > E_2 \\ \Sigma_0\sqrt{\frac{E_0}{E}}, \text{ for } E < E_2 \end{cases} \quad 3\text{-}15$$

Therefore the activation rate including epithermal neutrons can be written as

$$R = \int_0^{E_2} \Sigma_0\sqrt{\frac{E_0}{E}} \cdot \Phi_{th}(E)dE + \int_{E_1}^{E_2} \Sigma_0\sqrt{\frac{E_0}{E}} \cdot \frac{\Phi_{epi}}{E}dE + \int_{E_2}^{E_3} \Sigma_a(E) \cdot \frac{\Phi_{epi}}{E}dE \quad 3\text{-}16$$

The first integral is just the thermal activation rate, and the second integral can be solved analytically. The third integral can be rewritten using the resonance integral (RI) according to

$$N \cdot RI = \int_{E_2}^{E_3} \frac{\Sigma_a(E)}{E}dE \quad 3\text{-}17$$

The activation rate in the thermal and epithermal range (up to E_3= 0.1 MeV) can then be rewritten.

$$R = \Phi_{th}\sqrt{\frac{\pi \cdot T_0}{4T}}\Sigma_0 + \Phi_{epi}\left[2\Sigma_0\left(\sqrt{\frac{E_0}{E_1}} - \sqrt{\frac{E_0}{E_2}}\right) + N \cdot RI\right] \quad 3\text{-}18$$

Finally, the activation rate in the fast neutron range can be calculated from the empirically determined fission spectrum (N_{fiss}) and the fission spectrum weighted cross section (Σ_{fiss}).

$$\Phi_{fiss}(E') = \Phi_{fiss} \cdot N_{fiss}(E') \quad 3\text{-}19$$

$$N_{fiss}(E') \approx 0{,}453 \cdot e^{-E'/0{,}965} \cdot \sinh\left(\sqrt{2{,}29E'}\right) \quad 3\text{-}20$$

$$R_{fiss} = \int_{E_3}^{\infty} \Sigma_a(E)\Phi_{fiss}N_{fiss}(E')dE' = \Sigma_a^{fiss}\Phi_{fiss} \quad 3\text{-}21$$

The total reaction rate can now be calculated using overall neutron flux in the thermal, epithermal and fission energy range from Monte Carlo simulations and tabulated cross sections and resonance integrals.

$$R = \Phi_{th}\sqrt{\frac{\pi \cdot T_0}{4T}}\Sigma_0 + \Phi_{epi}\left[2\Sigma_0\left(\sqrt{\frac{E_0}{E_1}} - \sqrt{\frac{E_0}{E_2}}\right) + N_j RI\right] + \Sigma_a^{fiss}\Phi_{fiss} \quad 3\text{-}22$$

Neutron transport calculations have been performed using the MCNP (X-5 Monte Carlo Team 2003) Monte-Carlo particle transport code. The reactor internals have been divided into sub-components with given material compositions (see Appendix B). For each sub-component, the neutron flux has been tallied to give the thermal, epithermal and fast neutron flux levels.

3.2.2 Activity buildup and burnup

By only including decay, burnup and radionuclide production from neutron capture by stable nuclides, the Bateman equation can be simplified and solved.

$$\frac{dN_i}{dt} = -N_i(\lambda_i + R_i') + \sum_j R_{ji} \quad 3\text{-}23$$

$$a_i = \lambda_i N_i = \frac{\lambda_i}{(\lambda_i + R_i')}\sum_j R_{ji}\left(1 - e^{-(\lambda_i + R_i')t}\right) \quad 3\text{-}24$$

where a_i is the activity of nuclide i and R_i' is the burnup rate. The burnup rate is calculated in the same manner as the activation rate, but here only thermal and epithermal neutrons are considered relevant.

3.3 Surface contamination

3.3.1 Activated corrosion products BWR

Surface contamination due to activated corrosion products has been calculated using a mass balance for metal ions in the process system. The model depends on the water chemistry, the process water system, and the reactor operation and therefore, the model differs between BWR and PWR. The model parameters such as adsorption/desorption rates, corrosion rates etc have been determined empirically. Furthermore, parameter values differ between the reactors depending on the water chemistry applied in each case.

The total influx of dissolved corrosion products is given by

$$F_{i,tot} = F_{i,FW} + \sum_{mtrl} F_{i,mtrl} \qquad 3\text{-}25$$

Where $F_{i,FW}$ is the feed water influx of species i and $F_{i,mtrl}$ is the influx from different construction materials. The contribution from the feed water is given by the feed water concentration ($C_{i,FW}$) and feed water flow at full effect ($f_{,FW}$) scaled with the capacity factor (η_p)

$$F_{i,FW} = C_{i,FW} \cdot f_{FW} \cdot \eta_P \qquad 3\text{-}26$$

The contribution from corrosion of the different construction materials is given by

$$F_{i,mtrl} = x_{i,mtrl} \cdot k_{mtrl} \cdot A_{mtrl} \qquad 3\text{-}27$$

where x is the mass fraction of i in the material, k is the corrosion rate and A is the surface area. The reactor water concentration of species i is then determined from the total influx and the loss of ions (G) due to the reactor water clean-up unit and buildup of fuel crud.

$$C_{i,RW} = \frac{F_{i,tot}}{G_{i,331} + G_{i,FC}} \qquad 3\text{-}28$$

The clean-up efficiency of the reactor water clean-up is assumed to be 100%. Hence the removal rate is equal to the flow through the 331 system.

$$G_{i,331} = f_{331} \qquad 3\text{-}29$$

The buildup of fuel crud is determined by the deposition rate (d) for the specific species. These model parameters are changed to match the water chemistry specified for each reactor. Some elements are assumed to deposit independently of other elements.

$$G_{i,FC} = d_{i,FC} \cdot f_{FW} \cdot \eta_P \qquad 3\text{-}30$$

Some elements crystallize as iron spinels which makes the rate of crud buildup dependent on the relative influx of iron.

$$G_{i,FC} = \frac{d_{i,FC}}{\delta_i}\left(1 - \exp\left(-\frac{F_{Fe,tot}}{2\sum_j F_{j,tot}}\right)\right) \cdot f_{FW} \cdot \eta_P \qquad 3\text{-}31$$

where δ_i is a parameter that depends on the propensity for spinel formation compared to Ni (i.e. $\delta_{Ni} = 1$). Finally, Ni is assumed to deposit both as spinel and as NiO.

$$G_{Ni,FC} = \left[d_{Ni,FC}\left(1 - \exp\left(-\frac{F_{Fe,tot}}{2\sum_j F_{j,tot}}\right)\right) + d_{NiO}\right] \cdot f_{FW} \cdot \eta_P \qquad 3\text{-}32$$

The amount (M) of species i that accumulate in the fuel crud is calculated from

$$M_{i,FC} = \frac{G_{i,FC} \cdot C_{i,RW} \cdot t_{FC}}{e_{i,FC} + 1} \qquad 3\text{-}33$$

where t_{FC} is the average time for fuel in the core and $e_{i,FC}$ is a time-exponent for buildup of i as a function of operational time for the fuel. The nuclide inventory of the fuel crud will depend on the buildup rate mother species in the crud. An expression for the activation cannot be found analytically. The following empirical expression has been used:

$$a_{j,FC} \approx \frac{p_{i,j,FC} \cdot M_{i,FC}}{\lambda_j + \alpha_{j,FC} + \dfrac{e_{i,FC} + 1}{t_{FC}}} \qquad 3\text{-}34$$

where p is the activation rate, α is the burnup rate and λ is the decay constant. Release of activated species from the fuel crud is also described with an empirically based expression. The release is assumed to follow two pathways; species independent release of particulate matter (r_1) and diffusion controlled dissolution of the crud (r_2).

$$f_{j,FC} \approx \left[r_1 + r_{j,2} \cdot \sqrt{\frac{\lambda_j + 1/t_{FC}}{\lambda_{Co60} + 1/t_{FC}}} \cdot \sqrt{\frac{100}{\sum_i M_{i,FC}}} \right] a_{j,FC} \qquad 3\text{-}35$$

Release of radionuclides from corrosion of reactor internals is based on the three-group model for neutron induced activity described in the previous section. However, rather than applying specific operational time for each component, an average specific activity is calculated from the following expression:

$$\bar{a}_{j,mtrl} = a(\infty)_{j,mtrl} - \frac{a(t_{mtrl})_{j,mtrl}}{t_{mtrl} \cdot (\lambda_j + \alpha_{j,mtrl})} \qquad 3\text{-}36$$

where the t_{mtrl} is the operational time for the material and a is the activity as a function of operational time. Release of radionuclides is calculated analogous to the release of stable species due to corrosion

$$f_{j,mtrl} = \bar{a}_{j,mtrl} \cdot k_{mtrl} \cdot A_{mtrl} \qquad 3\text{-}37$$

The total flow of radionuclides to the reactor water is given by

$$f_{j,tot} = f_{j,FC} + \sum_{mtrl} f_{j,mtrl} \qquad 3\text{-}38$$

and the corresponding concentration is

$$c_{j,RW} = \frac{f_{j,tot}}{g_{j,FC} + g_{j,SS} + g_{j,331} + \lambda_j \cdot M_{RW}} \qquad 3\text{-}39$$

where g is the rate of removal of radionuclides from the reactor water. The clean-up efficiency and deposition rate to the fuel crud for active species are assumed to be the same as for the corresponding stable nuclide (G). The deposition rate to the surfaces of reactor internals is assumed to be similar to the deposition on the fuel, but also includes assumption of diffusion into the surface oxides, where decay has to be taken into account:

$$g_{j,SS} \approx d_{SS} \cdot \left[1 - \exp\left(- \frac{g_{j,FC}}{G_{Fe,FC}} \sqrt{\frac{\lambda_j + 1/t_{end}}{\lambda_{Co60} + 1/t_{end}}} \right) \right] \qquad 3\text{-}40$$

Finally the nuclide inventory from surface contamination of activated corrosion products can be calculated from

$$a_{j,SS} = \left(\frac{g_{j,SS} \cdot c_{j,RW}}{\lambda_j \cdot A_{SS}} + \frac{g_{m,SS} \cdot c_{m,RW}}{\lambda_m \cdot A_{SS}} \right) \cdot (1 - \exp(-\lambda_j \cdot t_{end})) \qquad 3\text{-}41$$

where t_{end} is the total operational time for the reactor, and m refers to possible mother nuclides to j.

3.3.2 Activated corrosion products PWR

To account for the difference in process chemistry between the BWR and PWR, a different compartment model is used for the PWR. The total influx of corrosion products is defined as

$$F_{i,tot} = F_{j,X} + \sum_{mtrl} F_{i,mtrl} \qquad 3\text{-}42$$

Here, there is no feed water contribution, but an extra term is included to allow for influx from unknown/unspecified sources $F_{i,X}$,. This term is applied for Ag, Sb and Zn, to rationalize the observed buildup of related activity (Ag-110m, Sb-122, Sb-124 and Zn-65).

For PWR, the materials considered also differ, most notably by the influx of corrosion products from the large Inconel surface of the steam generator tubes. The influx of corrosion products is calculated in the same way as for BWR, but the corrosion rates differ to account for the difference in water chemistry.

The overall removal rate (d) of corrosion products from the reactor water is defined as the sum of deposition rates for the core and other surfaces in contact with the reactor water as well as the removal due to the CVCS clean-up system (flow (f) and efficiency (η)).

$$d_{i,tot} = d_{i,core} + \sum_{mtrl} d_{i,mtrl} + f_{CVCS} + \eta_{i,CVCS} \qquad 3\text{-}43$$

From this expression the amount of species i deposited on the fuel is calculated.

$$M_{i,core} = F_{i,tot} \cdot \frac{d_{i,core}\left(1 - \exp\left(-r_{i,core} \cdot \left(1 - \frac{d_{i,core}}{d_{i,tot}}\right) t_{core}\right)\right)}{d_{i,core} \cdot r_{i,core} \cdot \left(1 - \frac{d_{i,core}}{d_{i,tot}}\right)} \qquad 3\text{-}44$$

where r is the desorption rate and t is the assumed residence time for the corrosion products deposited on the core. The activation of corrosion products deposited in the core is treated similarly to the BWR model

$$a_{j,core} \approx \frac{p_{i,j,FC} \cdot M_{i,core}}{\lambda_j + \alpha_{j,FC} + r_{j,core}} \qquad 3\text{-}45$$

The influx of activated corrosion products from the core is given by

$$f_{j,core} = a_{j,core} \cdot r_{j,core} \qquad 3\text{-}46$$

The activation of core materials is performed identically to the BWR model, using the same expression for average specific activity and release of active nuclides due to corrosion. The total influx of activate materials is then given by

$$f_{j,tot} = f_{j,core} + \sum_{mtrl} f_{j,mtrl} \qquad 3\text{-}47$$

The concentration of activated species in the reactor water under normal operation ($c_{i,op}$) is then calculated according to

$$c_{j,op} = \frac{f_{j,tot}}{d_{j,core} + \sum_{mtrl}\left(\frac{d_{j,mtrl}}{1 + \frac{r_{j,mtrl}}{\lambda_j}}\right) + f_{CVCS} \cdot \eta_{j,CVCS} + \lambda_j \cdot M_{RW}} \qquad 3\text{-}48$$

Finally, the buildup of activity on the system surfaces is given by

$$a_{j,mtrl} = \frac{d_{j,mtrl} \cdot c_{j,op}}{\left(\lambda_j + r_{j,mtrl}\right) \cdot A_{mtrl}} \qquad 3\text{-}49$$

3.3.3 Contamination from fuel leakage

Fuel leakage causes incorporation of actinides and insoluble fission products in the oxide layers on system surfaces in contact with the reactor water. The total accumulated amount of activity from fuel release has been calculated from the fuel leakage history of each reactor multiplied with the core inventory for the current power level and fuel burnup. This makes the estimated inventory conservative because final burnup has increased since the start of operation for the reactors.

Experience from the decontamination of B2 indicates that approximately 10% of the accumulated activity due to fuel leakage remains in the surface oxide layers after final shutdown. The following expression has been used for contamination of fission products and actinides, assuming linear release of fuel:

$$a_{j,mtrl} = \frac{0.1 \cdot u_{diss} \cdot x_{mtrl}}{A_{mtrl}} \left(\frac{a_{j,fuel}}{\lambda_j t_{tot}} (1 - \exp(-\lambda_j t_{tot})) + \sum_m \eta_{m,j} \frac{a_{m,fuel}}{\lambda_m t_{tot}} (1 - (\lambda_j t_{tot} + 1) \cdot \exp(-\lambda_j t_{tot})) \right) \quad 3\text{-}50$$

u_{diss} is the total mass of dissolved/leaked fuel, $a_{j,fuel}$ is the core inventory activity of species j, t_{tot} is the total operational time and m is the mother nuclide giving rise to nuclide i with a yield of $\eta_{m,j}$. x_{mtrl} is a distribution coefficient for the material surface. In the BWR case only stainless steel is considered and therefore $x_{SS} = 1$. For PWR, the Inconel surface of the steam generators must also be taken into account. The distribution coefficient is then calculated as the relative Co-60 activity in the crud:

$$x_{mtrl} = \left(\frac{a_{Co60,mtrl}}{\sum_{mtrl} a_{Co60,mtrl}} \right) \quad 3\text{-}51$$

3.4 Radionuclide inventory based on dose-rate measurements

For some of the waste categories in the existing operational waste from Forsmark, the reported inventory is based on dose rate measurements in combination with calculated induced activity according to the methodology described in Chapter 3. The waste is typically of mixed origin, making it difficult to determine the exact contribution of different components. The calculations have been performed according to the following procedure:

- Dose rates have been measured at 5 and 15 cm in water for the entire waste container or for all individual components in the waste giving the total dose rate of the container.
- The measured dose rates have been assumed to originate from a monoenergetic gamma source with an energy of 1.25 MeV (i.e. the average gamma energy of Co-60).
- The source is assumed to be a homogenous body corresponding to the waste volume.
- Using the program MicroShield (Grove Software 2007), the activity has been calculated to fit the measured dose rates.
- The calculated activity has been assumed to correspond to the Co-60 activity.
- A nuclide vector with nuclide specific activities relative to Co-60 has been calculated using the model for induced activity described in Section 3.2.
- Using the estimated Co-60 and the nuclide vector, nuclide specific activity has been assigned to each waste package.

The procedure outlined above can be said to improve the activity estimate by scaling the calculated activity to fit actual measured dose rates. However, the procedure does not take into account surface contamination from activated corrosion products or fuel leakage. Thus, no actinides or fission products are assigned to this waste. For waste including material from neutron detectors and guide tubes, the possible presence of fissile material from the detector units, is also neglected. However, this only concerns a small fraction of the total waste amount, and the lack of actinide activities for this waste does not significantly affect the total inventory from the Forsmark NPP.

3.5 Radionuclide inventory from B1, B2, O1 and Ågesta

The method described above has been used consistently for all reactors except B1, B2, O1 and Ågesta. For all reactors except O1, updated models for activity calculations have been implemented when the decommissioning studies were updated. In the decommissioning studies, activity data from O1 safety assessment has been complemented with data for O2. All activity data is taken from the decommissioning inventory where all core components are assumed to remain in operation for the entire operational lifetime of the reactor. Hence, replaced core components from O1 only contribute with material and waste amounts in the present inventory since the activity is implicitly included in the decommissioning inventory.

For, B1, B2 and Ågesta the inventory has been based on the most recently updated neutron transport calculations. Rather than applying the three-group model described above, the inventory has been calculated using the numerical solution to the Bateman equation implemented in FISPACT (Forrest 2002). This approach takes more activation pathways into account and also provides higher resolution of the neutron flux and cross sections.

Furthermore, in the case of Ågesta, the use of heavy water as a primary coolant gives rise to tritium contamination. This has been taken into account using an empirically determined contamination factor for all steel surfaces.

3.6 Radioactive decay

Radioactive decay has been calculated according to the half-lives listed in Table 3-1. The ingrowth of radio nuclides due to decay of heavier nuclides has been accounted for in for the decay chains listed in Table 3-2.

Table 3-1. Half-lives for radioactive decay.

Nuclide	$t_{½}$/years	Nuclide	$t_{½}$/years
H-3	1.23E+01	Cs-137	3.00E+01
Be-10	1.60E+06	Ba-133	1.05E+01
C-14	5.70E+03	Pm-147	2.62E+00
Cl-36	3.01E+05	Sm-151	9.00E+01
Ca-41	1.03E+05	Eu-152	1.35E+01
Fe-55	2.73E+00	Eu-154	8.59E+00
Co-60	5.27E+00	Eu-155	4.75E+00
Ni-59	7.60E+04	Ho-166m	1.20E+03
Ni-63	1.01E+02	U-232	6.98E+01
Se-79	1.10E+06	U-235	7.04E+08
Sr-90	2.88E+01	U-236	2.37E+07
Zr-93	1.53E+06	Np-237	2.14E+06
Nb-93m	1.61E+01	Pu-238	8.77E+01
Nb-94	2.00E+04	Pu-239	2.41E+04
Mo-93	4.00E+03	Pu-240	6.56E+03
Tc-99	2.14E+05	Pu-241	1.43E+01
Ru-106	1.02E+00	Pu-242	3.74E+05
Ag-108m	4.18E+02	Am-241	4.33E+02
Pd-107	6.50E+06	Am-242m	1.41E+02
Cd-113m	1.41E+01	Am-243	7.37E+03
Sn-126	2.30E+05	Cm-243	3.00E+01
Sb-125	2.76E+00	Cm-244	1.80E+01
I-129	1.61E+07	Cm-245	8.50E+03
Cs-134	2.07E+00	Cm-246	4.73E+03
Cs-135	2.30E+06		

Table 3-2. Decay chains accounted for in the inventory.

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

4 Reference case

In order to estimate the inventory of long-lived waste from the NPPs, a reference case has to be assumed concerning future waste amounts and packaging prior to disposal. The following general assumptions have been made in order to estimate the reference inventory (component specific assumptions are described in Chapter 2):

- Volumes for existing waste from replaced reactor internals have been taken from reported values for each NPP.
- In the forecast on future operational waste, only Oskarshamn has reported volumes for planned replacements of reactor internals. At the other NPPs, the current set of internals is expected to last the full operational life time of the reactors.
- Existing waste has been assumed to have the same material specifications as decommissioning waste from similar components.
- The metallic waste is assumed to be packaged in steel tanks and stabilized with grout (see Section 4.1). The biological shields from the Ringhals PWRs are accounted for by their entire volume, but without packaging.
- In the forecast for waste volumes from decommissioning of the NPPs, only the parts of the reactor internals with significant induced activity have been taken into account. Here, the calculated amounts from the decommissioning studies have been used for all NPPs except Ågesta. In the decommissioning study of Ågesta, steel tanks have not been included due to the low expected dose rates of the waste. Hence, in the present inventory, waste amounts and volume have been calculated from the reported component weights expected to contribute to the long-lived waste.
- Secondary waste from decommissioning of reactor internals have been included only as reported from the NPPs. Hence, it can be expected that additional waste will be produced depending on how the current set of reactor internals and control rods will be handled prior to packaging.
- The disposal volume has been calculated from the reported number of tanks for the waste from Barsebäck as well as for the existing waste in Forsmark and Oskarshamn. For the remaining waste, general packing densities of 0.677 tonne/m^3 for core components and 1.1 tonne/m^3 for segmented RPVs have been assumed.
- For core components from future operation and decommissioning it is assumed that the waste can be mixed.
- For the PWR RPVs three different scenarios have been evaluated:
  1. Segmentation of RPVs (base case).
  2. Segmentation of the PWR internals, but disposal of the RPVs intact.
  3. Disposal of intact RPVs including internals.

4.1 Waste packaging

Final packaging of waste in SFL will be decided when the design and safety requirements of the repository has been finalized. However, in order to estimate the required repository volume as well as the material contribution from packaging and grout, a waste package has to be assumed. In the following, the assumed packaging is described.

Reactor internals are assumed to be disposed of in steel tanks of the same dimensions, as the tanks used today for interim storage. The tank has a 1.3 × 3.3 m base and a height of 2.3 m. Waste is loaded into a steel cassette (typically under water), which is then placed inside the tank. The cassette with the waste is loaded into the tank and dried under vacuum. Prior to disposal it is assumed that the waste will be stabilized with grout.

The tanks used today differ in wall thickness depending on the radiation shielding requirements of the components stored in them. In the present inventory, it is assumed that the waste already stored in steel tanks (from Oskarshamn and Forsmark) will be disposed of in the existing tanks. The amount of package material and grout is calculated from the reported tank types and waste amounts. The same analysis has been done for the waste from Barsebäck where a detailed packing plan has been presented. From this packing plan, and the packing of existing waste, an average packing density (0.677 tonne/m^3) and an average tank wall thickness (96 mm) have been calculated.

For the waste from future operation and decommissioning of Forsmark, Oskarshamn, Ringhals and Ågesta, volume and waste packaging material has been calculated using the average packing density reported above and steel tanks with 100 mm wall thickness. This tank choice is somewhat conservative. However, the waste from Barsebäck has undergone radioactive decay for several years and the waste from the other reactors will likely require more shielding.

The PWR RPVs have significantly lower specific activity compared to the core components. Hence for the case of segmented RPVs, 50 mm tanks are assumed. Furthermore, the thick-walled steel should allow for a higher packing density than for the core components. Therefore, in the volume estimates, 1.1 tonne/m^3 is assumed for the RPVs.

Table 4-1 lists properties for the steel tanks (SKBdoc 1262717), relevant to the calculation of volume and material in SFL. Void is calculated as 25% of the inner volume, which is consistent with the estimate of void in short-lived waste for containers with scrap and refuse (SKB 2013). The volume of grout is calculated as the difference between the inner volume and the combined volume of the waste, cassette and void.

Table 4-1. Data for steel tanks. The T-50 and T-100 tanks are used as reference waste package for core components and RPV segments in the inventory. Waste volume refers to part of the inner volume available for the waste. The T-150 and T-200 tanks are included since they are currently used for storage of waste.

Property	T-50	T-100	T-150	T-200	Unit
Outer dimensions	3.3 × 1.3 × 2.3	3.3 × 1.3 × 2.3	3.3 × 1.3 × 2.3	3.3 × 1.3 × 2.3	m
Wall thickness	0.05	0.1	0.15	0.2	m
Packing density	1,100	677	–	–	kg/m^3
Volumes					
Inner volume	8.5	7.4	6.6	5.6	m^3
Waste volume	7	6	4.9	4	m^3
Repository volume	9.9	9.9	9.9	9.9	m^3
Carbon steel (tank and cassette)	1.7	2.8	4.0	5.0	m^3
Concrete (grout)	5.0	4.6	–	–	m^3
Void	2.1	1.9	1.9	1.4	m^3
Weights					
Weight of cassette	2,800	3,600	5,500	5,800	kg
Weight of tank	10,300	18,500	25,600	33,200	kg
Carbon steel (tank an cassette)	13,100	22,100	31,100	39,000	kg
Concrete (grout)	12,094	10,981	–	–	kg
Total	32,894	37,141			
Surface area (including cassette)					
Carbon steel	85	89	102	98	m^2

[1]Assuming standard packing densities: 0.677 tonne/m^3 for core components and 1.1 tonne/m^3 for segmented RPVs.

4.2 Surface area

The surface area of the metallic waste has been estimated using an approach similar to the one used to estimate surface contamination. For each component or waste item the surface area has been calculated as the surface area of sheet metal of the same weight and density as the component, but with an assumed thickness. Table 4-2 shows the values of metal thickness and density assumed for different materials and components.

For control rods and detector guide tubes, specific surface areas have been reported in the inventories for the decommissioning studies. For the SFL inventory, the surface area in the waste has been estimated as twice the reported surface area used in calculation of surface contamination, since segmentation will expose the inner surface not normally exposed to the reactor water.

For hafnium, indium, cadmium and silver the surface area has been calculated as a fraction of the control rod surface area in proportion to each materials contribution to the total weight.

Table 4-2. Assumed densities and material thickness used for estimating material surface areas in the waste.

Component type	Material	Density m^3/kg	Material thickness/mm
Core support structure	Steel	7,860	10
Core spray	Steel	7,860	5
Spacers	Inconel	8,140	2.5
Fuel boxes	Zircalloy	6,550	2.5
Swarf	Steel	7,860	0.5
Scrap metal (unknown origin)	Steel	7,860	5

5 Inventory

Reactor internals close to the core are exposed to intense neutron radiation, which result in significant buildup of long-lived radionuclides in the construction material of these components. These components along with neutron detectors, BWR control rods and reactor pressure vessels (RPVs) from PWR constitute most of the long-lived intermediate level waste from the NPPs. There are also small volumes of waste such as irradiated parts of the fuel assembly (fuel boxes, spacers etc.), start-up neutron sources and burnable absorbers, which have also been included in the SFL inventory. Control rods from PWR are expected to be disposed of together with the spent fuel and are therefore not taken into account here, except for control rods from the Ågesta reactor. A more comprehensive description of the components constituting the waste is given in Chapter 2.

In the following, the inventory of long-lived waste from the NPPs is summarized in terms of waste volume, material content and radionuclide inventory.

5.1 Volume

When large core components are replaced during reactor maintenance, the replaced components are segmented under water using different cutting techniques (e.g. abrasive water jet, mechanical cutting, electrical discharge machining etc.). The segments are placed in steel cassettes and taken out of the water using a radiation shielding hood. The cassette is placed in the steel tank and dried under vacuum (SKBdoc 1262717).

In this report, it is assumed that metallic long-lived waste from the NPPs will be handled in the same manner. Even though it still remains to be seen if this is a viable approach for BWR control rods, which are not as straight forwards to segment (for example, due to potential release of activated boron carbide during segmentation) as the reactor internals. Furthermore smaller components such as segmented neutron detectors, secondary waste and smaller pieces removed from the larger components (e.g. metal samples, bolt and nuts) will likely require some form of primary packaging before being placed into steel tanks. However, it is beyond the scope of the present report to solve these technical challenges, and therefore all the metallic waste is assumed to be handled and packed in the same way.

In addition to the core components, the biological shields from the Ringhals PWRs have been excluded from the inventory of short-lived decommissioning waste due to the relatively high C-14 content (in excess of 10^{10} Bq). Hence, it must be accounted for in the SFL-inventory. In the future, the choice may be taken to separate the most activated parts of the biological shield to reduce the volume of long-lived waste, and it may turn out that separating the reinforcement from the concrete will be required. However, for the present analysis the whole biological shield will be included in the SFL-inventory, but the volume is included without packaging since it is unrealistic to assume that the entire biological shield will be segmented and disposed of in steel tanks.

The total volume of long-lived waste from the Swedish NPPs has been calculated to ~6,800 m^3 (for scenario 1 above). Table 5-1 lists the waste volume for different types of waste and Table 5-2 to Table 5-6 show the amount of waste for each NPP. From Table 5-1 it is clear that the disposal volume can be reduced with 1,150 m^3 or 140 m^3 by allowing disposal of whole PWR RPVs with or without internals, respectively.

Secondary waste in the form of swarf has been reported for the segmentation of core components and reactor internals from Forsmark and Oskarshamn. A forecast on similar waste from the future segmentation of reactor internals from Barsebäck has also been reported. This waste is accounted for in the volume of waste from core components.

Forsmark and Oskarshamn have also reported values of actual *secondary waste* in the form of filters, abrasives and decommissioned segmentation equipment (reported as Secondary waste in Table 5-3 and Table 5-4). This waste is stored in 1.2 × 1.2 × 1.2 m moulds, and the volume estimates for this waste category have been based on these dimensions.

Table 5-1. Weight and volume for long-lived waste from the NPPs for the three scenarios listed in Chapter 4.

Scenario:	1. Segmented PRW RPVs and internals		2. Intact PRW RPVs, but segmented internals		3. Intact PRW RPVs and internals	
Waste type	**Weight/ tonnes**	**Volume/ m^3**	**Weight/ tonnes**	**Volume/ m^3**	**Weight/ tonnes**	**Volume/ m^3**
Core components BWR (existing)[1]	386	1,048	386	1,048	386	1,048
Core components BWR (forecast)	83	203	83	203	83	203
Core components BWR (decom.)	588	1,392	588	1,392	588	1,392
Secondary waste	22	116	22	116	22	116
Control rods BWR (existing)	168	417	168	417	168	417
Control rods BWR (forecast)	323	791	323	791	323	791
Control rods (Ågesta)	2	5	2	5	2	5
RPV PWR	735	950	983	862	983	862
RPV PWR (Ågesta)	140	188	140	135	140	135
Biological shield PWR	1,764	659	1,764	659	1,764	659
Reactor internals PWR	282	692	282	692	405	0
Reactor internals (Ågesta)	125	312	125	312	125	0
Sum (existing)	569	1,540	569	1,540	569	1,540
Sum (forecast)	414	1,036	414	1,036	414	1,036
Sum (decom.)	3,636	4,199	3,883	4,057	4,006	3,052
Total	**4,619**	**6,774**	**4,866**	**6,632**	**4,989**	**5,627**

[1]This waste fraction also includes a small amount of existing waste from PWR (3.2 tonnes)

Table 5-2. Weight and volume for long-lived waste from Barsebäck.

Barsebäck	Weight/kg	Volume/m^3
Core components (existing)	55,382	113
Core components (decom.)	110,090	224
Control rods (existing)	45,890	119
Total	**211,362**	**455**

Table 5-3. Weight and volume for long-lived waste from Forsmark.

Waste type	Weight/kg	Volume/m^3
Core components (existing)	152,596	396
Core components (forecast)	5,828	14
Core components (decom.)	218,005	534
Control rods (existing)	51,994	127
Control rods (forecast)	102,026	250
Control rods (decom.)	66,873	164
Secondary waste	966[1]	7
Total	**598,288**	**1,492**

[1]Waste mass calculated from the reported number of secondary waste packages from Forsmark multiplied with the average waste mass for secondary waste from Oskarshamn.

Table 5-4. Weight and volume for long-lived waste from Oskarshamn.

Waste type	Weight/kg	Volume/m^3
Core components (existing)	143,909	455
Core components (forecast)	77,219	188
Core components (decom.)	198,150	483
Control rods (existing)	48,554	118
Control rods (forecast)	66,916	163
Control rods (decom.)	51,480	126
Secondary waste	13,501	67
Secondary waste (forecast)	7,728	41
Total	**607,457**	**1,643**

Table 5-5. Weight and volume for long-lived waste from Ringhals.

Ringhals	Segmented RPVs and internals from PWRs		Intact RPVs and internals from PWRs	
	Weight/kg	Volume/m^3	Weight/kg	Volume/m^3
Core components (existing)	34,198[1]	84	34,198[1]	84
Core components (forecast)	133	0	133	0
Core components BWR (decom.)	61,595	151	61,595	151
Reactor internals PWR	282,000	692	405,000	0
RPV PWR	735,000	950	982,500	862
Biological shield PWR	1,764,000	659	1,764,000	659
Control rods (existing)	21,513	53	21,513	53
Control rods (forecast)	15,395	38	15,395	38
Control rods (decom.)	20,724	51	20,724	51
Total	**2,934,559**	**2,678**	**3,305,059**	**1,896**

[1]This waste fraction also includes a small amount of existing waste from PWR (3.2 tonnes)

Table 5-6. Weight and volume for long-lived waste from Ågesta.

Ågesta	Weight/kg	Segmented RPV Volume/m^3	Whole RPV Volume/m^3
Core components	125,000	312	0
RPV	140,200	188	135
Control rods	1,811	5	10
Total	**267,011**	**505**	**145**

The amount and type of secondary waste generated in the future will strongly depend on the choice of segmentation technique, and lessons learned from the segmentation at Oskarshamn and Forsmark may very well serve to reduce future secondary waste volumes. Hence, no reliable forecast on future waste volumes of this type can be given at present, but all activity in such waste is accounted for in the inventory from the core components.

5.2 Material

The materials inventory has been calculated from the reported and calculated waste amounts and the specification of construction materials for each component. Material specifications have been taken from the radionuclide inventory calculations for the decommissioning studies (SKBdoc 1403739, Lindow 2012, Anunti et al. 2013, Larsson et al. 2013, Hansson et al. 2013). Detailed material compositions for each construction material are given in Appendix B.

Table 5-7 summarizes the material content for the different NPPs as well as for the total waste mass. Most of the material is stainless steel. It should be pointed out that steel grades differ between components. Boron carbide and hafnium originate from the BWR-control rods. Indium, cadmium and silver originate from the Ågesta reactor control rods. Small amounts of Zircalloy and Inconel in the waste stems from parts of the BWR fuel assemblies (e.g. fuel boxes and spacers).

Apart from the materials specified in Table 5-7, the neutron detectors contain a small amount of fissile material. However, from a volume point of view this amount is insignificant.

Using this approach the total surface area of the different metals in the waste can be calculated. Table 5-8 shows the total surface area for the different waste materials and Table 5-9 shows the material contribution from the waste packaging calculated according to the number of packages and the data given in Table 4-1.

Table 5-7. Materials amounts in SFL-waste from the Swedish NPPs. Packaging and grout material is not included.

Material	Material/tonnes						
	Barsebäck	Forsmark	Oskarshamn	Ringhals	Ågesta	Total	w/w%
Stainless steel	200	570	541	439	266	2,015	**44%**
Carbon steel	0	0	0	984	0	984	**21%**
Boron steel	6	0	8	6	0	20	**0.4%**
Boron carbide	5	20	16	5	0	46	**1%**
Hafnium	1	6	5	0	0	13	**0.3%**
Inconel	0	0.3	0	0	0	0.3	**0.01%**
Zircalloy	0	0	17	0	0	18	**0.4%**
Silver	0	0	0	0	1	1	**0.02%**
Indium	0	0	0	0	0.2	0.2	**0.004%**
Cadmium	0	0	0	0	0.1	0.1	**0.001%**
Concrete	0	0	0	1,500	0	1,500	**32%**
Misc. (filters etc.)	0	1	21	0	0	22	**0.5%**
Total	**211**	**598**	**607**	**2,935**	**267**	**4,619**	**100%**

Table 5-8. Surface areas of different waste materials expected to be exposed to corrosion in the repository.

Material	Surface area/m^2
Stainless steel	67,960
Carbon steel	1,313
Boron steel	7,472
Boron carbide	N/A
Hafnium	946
Inconel	18
Zircalloy	1,797
Silver	7
Indium	1
Cadmium	0.4
Concrete	N/A
Misc. (filters etc.)	N/A

Table 5-9. Material in waste packaging and grout.

Carbon steel/tonnes	12,285
Carbon steel /m^2	53,679
Concrete/tonnes	5,415
Void/m^3	1,157
Number of packages	606

5.3 Total radionuclide inventory

The radionuclide inventory of the long-lived waste from the NPPs is dominated by induced radioactivity due to neutron irradiation of the components close to the reactor core. Surface contamination is ~0.1 % of the total activity and consists of activated corrosion products and fuel material deposited on the component surfaces. Fission products and actinides originates from fissile material in the neutron detectors (induced activity) and from fuel leakage (surface contamination).

The total activity of the NPP inventory is estimated to 2.1 10^{17} Bq by the end of 2075. Table 5-10 and Table 5-11 show the nuclide specific activity levels for induced radioactivity and surface contamination, respectively. Figure 5-1 shows the total activity as well as the dominating nuclide specific activities as a function of time. Initially the activity is dominated by the short lived activity from Co-60. For the first 1,000 years after closure, Ni-63 is the dominating nuclide followed by Ni-59 in the 1,000 –10^6 year time frame.

The total activity is higher than the activity reported for the corresponding waste in the reference inventory from 1998 (SKBdoc 1416968), which is reasonable given the extended time of operation for the NPPs. The difference is also within the expected error margin. This modest difference also holds for many of the nuclides arising from neutron capture in the construction material (e.g. Co-60, Ni-63, Ni-59, Cl-36 and C-14). The larger differences for induced activity are in the range of one order of magnitude.

For actinides and some fission products the new inventory is several orders of magnitude higher. However, surface contamination in the reference inventory from 1998 was calculated from correlation factors, not taking the fuel leakage history of the reactors into account. Furthermore, fissile material in the detectors was not accounted for. In the reference inventory from 1998, significant surface contamination of Cs-137 was assumed, which is why the Cs-137 activity is higher in the reference inventory from 1998. In the present report, the only source of Cs-137 is fissile material in the detectors. Cs and other aqueous soluble fission products are not assumed to be present in the surface oxides of the core components.

Table 5-10. Neutron induced radionuclide inventory in long-lived waste from the NPPs at 2075.

Nuclide	Activity/Bq	Nuclide	Activity/Bq
H-3	8.31E+14	Cs-137	1.88E+11
Be-10	6.41E+04	Ba-133	4.95E+09
C-14	3.99E+14	Pm-147	1.87E+07
Cl-36	1.18E+11	Sm-151	4.43E+11
Ca-41	3.28E+11	Eu-152	3.37E+12
Fe-55	2.62E+14	Eu-154	6.02E+10
Co-60	5.53E+15	Eu-155	3.38E+09
Ni-59	1.91E+15	Ho-166m	4.21E+09
Ni-63	1.99E+17	U-232	9.30E+06
Se-79	8.03E+10	U-235	6.69E+02
Sr-90	1.73E+11	U-236	2.67E+06
Zr-93	6.14E+11	Np-237	3.76E+06
Nb-93m	8.55E+14	Pu-238	1.77E+10
Nb-94	1.54E+13	Pu-239	8.68E+09
Mo-93	2.38E+13	Pu-240	6.79E+07
Tc-99	3.55E+12	Pu-241	1.17E+09
Ru-106	1.43E+01	Pu-242	1.11E+09
Ag-108m	2.13E+12	Am-241	8.75E+08
Pd-107	1.19E+05	Am-242m	1.93E+06
Cd-113m	1.47E+09	Am-243	7.00E+06
Sn-126	1.52E+06	Cm-243	1.05E+07
Sb-125	1.68E+10	Cm-244	1.05E+08
I-129	1.98E+05	Cm-245	7.20E+07
Cs-134	2.27E+07	Cm-246	6.18E+04
Cs-135	2.34E+06	**Total**	**2.09E+17**

Table 5-11. Radionuclide inventory in surface contamination of long-lived waste from the NPPs at 2075.

Nuclide	Activity/Bq	Nuclide	Activity/Bq
H-3	0.00E+00	Cs-137	0.00E+00
Be-10	0.00E+00	Ba-133	0.00E+00
C-14	0.00E+00	Pm-147	1.91E+07
Cl-36	0.00E+00	Sm-151	8.08E+09
Ca-41	0.00E+00	Eu-152	9.03E+06
Fe-55	4.74E+10	Eu-154	1.44E+09
Co-60	2.49E+12	Eu-155	3.30E+07
Ni-59	3.04E+12	Ho-166m	3.35E+05
Ni-63	2.79E+14	U-232	1.61E+05
Se-79	1.01E+09	U-235	4.82E+02
Sr-90	4.67E+11	U-236	9.94E+06
Zr-93	7.54E+10	Np-237	1.24E+07
Nb-93m	2.00E+13	Pu-238	6.48E+10
Nb-94	1.93E+11	Pu-239	1.11E+10
Mo-93	7.19E+09	Pu-240	1.47E+10
Tc-99	5.01E+09	Pu-241	1.53E+11
Ru-106	0.00E+00	Pu-242	1.10E+10
Ag-108m	6.12E+11	Am-241	5.25E+10
Pd-107	4.05E+07	Am-242m	3.77E+08
Cd-113m	0.00E+00	Am-243	1.08E+09
Sn-126	1.92E+07	Cm-243	1.37E+08
Sb-125	4.64E+08	Cm-244	1.25E+10
I-129	1.69E+08	Cm-245	6.28E+08
Cs-134	0.00E+00	Cm-246	6.34E+06
Cs-135	0.00E+00	**Total**	**3.07E+14**

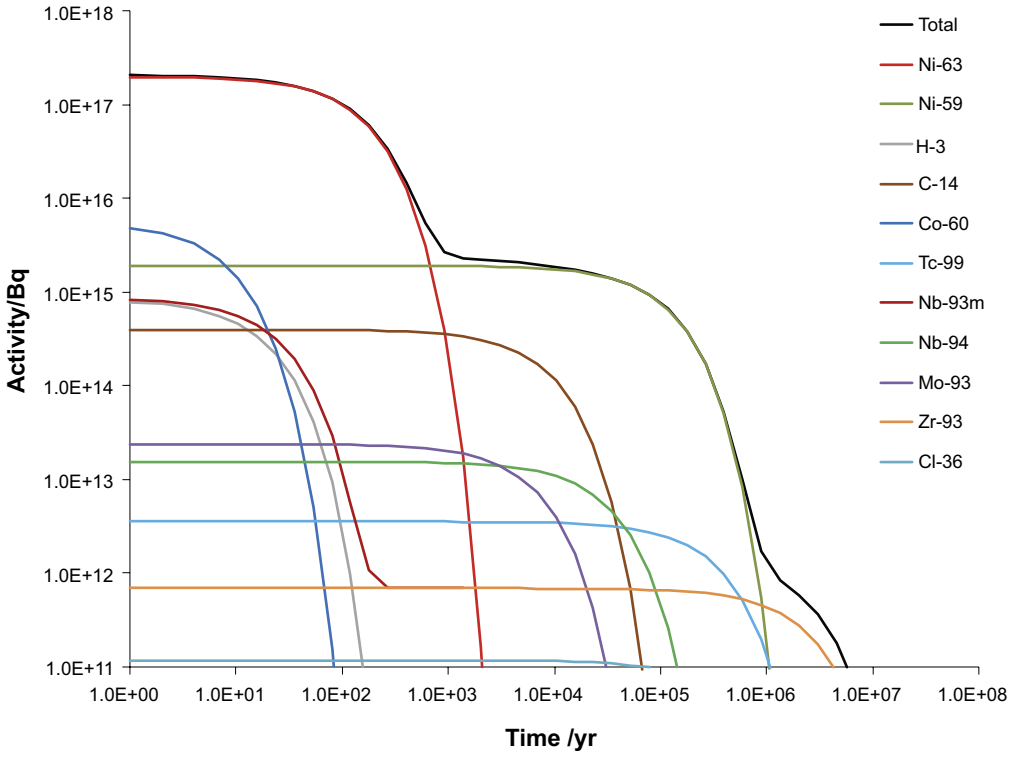

Figure 5-1. *Total radionuclide inventory in SFL as a function of time after closure of SFL 2075. Total activity as well as nuclide specific activity is shown for the dominating nuclides and Cl-36.*

5.3.1 Radionuclide inventory per producer

In the following (Table 5-12 to Table 5-21), neutron induced activity and activity in surface contamination is reported for each NPP.

Table 5-12. Neutron induced radionuclide inventory in long-lived waste from Barsebäck at 2075.

Nuclide	Activity/Bq	Nuclide	Activity/Bq
H-3	1.00E+13	Cs-137	1.62E+10
Be-10	4.62E+03	Ba-133	1.94E+02
C-14	6.22E+12	Pm-147	1.81E+02
Cl-36	2.16E+09	Sm-151	7.98E+07
Ca-41	0.00E+00	Eu-152	1.33E+04
Fe-55	1.12E+10	Eu-154	8.39E+06
Co-60	1.64E+12	Eu-155	4.17E+04
Ni-59	4.27E+13	Ho-166m	6.89E+01
Ni-63	3.29E+15	U-232	1.99E+06
Se-79	2.47E+04	U-235	2.87E+00
Sr-90	1.48E+10	U-236	3.87E+05
Zr-93	2.77E+09	Np-237	5.28E+05
Nb-93m	1.85E+12	Pu-238	2.34E+09
Nb-94	1.42E+11	Pu-239	4.63E+07
Mo-93	1.10E+12	Pu-240	3.25E+07
Tc-99	1.29E+11	Pu-241	3.17E+08
Ru-106	0.00E+00	Pu-242	1.67E+05
Ag-108m	5.83E+01	Am-241	4.88E+08
Pd-107	1.62E+04	Am-242m	3.08E+05
Cd-113m	6.53E+04	Am-243	1.80E+06
Sn-126	2.09E+05	Cm-243	9.63E+06
Sb-125	4.66E+05	Cm-244	8.72E+06
I-129	2.68E+04	Cm-245	3.08E+04
Cs-134	1.51E+00	Cm-246	8.81E+03
Cs-135	3.18E+05	**Total**	**3.35E+15**

Table 5-13. Radionuclide inventory in surface contamination of long-lived waste from Barsebäck at 2075.

Nuclide	Activity/Bq	Nuclide	Activity/Bq
H-3	0.00E+00	Cs-137	0.00E+00
Be-10	0.00E+00	Ba-133	0.00E+00
C-14	0.00E+00	Pm-147	6.23E+00
Cl-36	0.00E+00	Sm-151	7.82E+06
Ca-41	0.00E+00	Eu-152	4.45E+03
Fe-55	1.61E+06	Eu-154	2.77E+05
Co-60	3.18E+09	Eu-155	7.99E+02
Ni-59	1.40E+11	Ho-166m	1.43E+02
Ni-63	1.13E+13	U-232	2.56E+02
Se-79	0.00E+00	U-235	1.20E+00
Sr-90	6.19E+08	U-236	1.78E+04
Zr-93	5.99E+08	Np-237	1.85E+04
Nb-93m	2.40E+11	Pu-238	7.82E+07
Nb-94	5.32E+09	Pu-239	1.77E+07
Mo-93	7.11E+07	Pu-240	3.06E+07
Tc-99	1.19E+07	Pu-241	9.87E+07
Ru-106	0.00E+00	Pu-242	1.25E+05
Ag-108m	4.22E+09	Am-241	9.59E+07
Pd-107	0.00E+00	Am-242m	3.24E+05
Cd-113m	0.00E+00	Am-243	1.38E+06
Sn-126	2.32E+04	Cm-243	1.29E+05
Sb-125	8.32E+04	Cm-244	7.52E+06
I-129	0.00E+00	Cm-245	1.52E+04
Cs-134	0.00E+00	Cm-246	7.12E+03
Cs-135	0.00E+00	**Total**	**1.17E+13**

Table 5-14. Neutron induced radionuclide inventory in long-lived waste from Forsmark at 2075.

Nuclide	Activity/Bq	Nuclide	Activity/Bq
H-3	4.17E+14	Cs-137	5.85E+10
Be-10	0.00E+00	Ba-133	2.32E+03
C-14	1.44E+14	Pm-147	6.24E+06
Cl-36	3.16E+10	Sm-151	2.01E+08
Ca-41	0.00E+00	Eu-152	1.01E+05
Fe-55	8.62E+13	Eu-154	1.64E+08
Co-60	6.47E+14	Eu-155	9.63E+06
Ni-59	5.78E+14	Ho-166m	1.50E+02
Ni-63	5.73E+16	U-232	2.60E+06
Se-79	5.07E+04	U-235	9.38E+01
Sr-90	5.45E+10	U-236	6.58E+05
Zr-93	3.07E+10	Np-237	1.05E+06
Nb-93m	7.19E+13	Pu-238	4.54E+09
Nb-94	9.56E+11	Pu-239	2.01E+09
Mo-93	9.55E+12	Pu-240	1.21E+07
Tc-99	1.48E+12	Pu-241	2.98E+08
Ru-106	9.01E+00	Pu-242	1.11E+09
Ag-108m	9.62E+01	Am-241	1.28E+08
Pd-107	3.48E+04	Am-242m	1.34E+06
Cd-113m	2.56E+04	Am-243	1.53E+06
Sn-126	4.42E+05	Cm-243	3.69E+05
Sb-125	2.29E+09	Cm-244	3.24E+07
I-129	5.77E+04	Cm-245	7.18E+07
Cs-134	1.13E+06	Cm-246	2.13E+04
Cs-135	6.83E+05	**Total**	**5.92E+16**

Table 5-15. Radionuclide inventory in surface contamination of long-lived waste from Forsmark at 2075.

Nuclide	Activity/Bq	Nuclide	Activity/Bq
H-3	0.00E+00	Cs-137	0.00E+00
Be-10	0.00E+00	Ba-133	0.00E+00
C-14	0.00E+00	Pm-147	9.45E+06
Cl-36	0.00E+00	Sm-151	2.95E+09
Ca-41	0.00E+00	Eu-152	3.75E+06
Fe-55	1.11E+10	Eu-154	5.71E+08
Co-60	8.57E+11	Eu-155	1.44E+07
Ni-59	7.03E+11	Ho-166m	7.68E+04
Ni-63	6.67E+13	U-232	5.21E+04
Se-79	0.00E+00	U-235	1.47E+02
Sr-90	1.74E+11	U-236	3.53E+06
Zr-93	3.64E+08	Np-237	4.39E+06
Nb-93m	4.89E+12	Pu-238	2.20E+10
Nb-94	4.19E+10	Pu-239	4.47E+09
Mo-93	1.06E+09	Pu-240	5.04E+09
Tc-99	3.44E+08	Pu-241	6.12E+10
Ru-106	0.00E+00	Pu-242	1.10E+10
Ag-108m	1.10E+10	Am-241	1.93E+10
Pd-107	0.00E+00	Am-242m	1.76E+08
Cd-113m	0.00E+00	Am-243	3.89E+08
Sn-126	6.89E+06	Cm-243	4.94E+07
Sb-125	8.63E+07	Cm-244	4.40E+09
I-129	0.00E+00	Cm-245	6.15E+08
Cs-134	0.00E+00	Cm-246	1.91E+06
Cs-135	0.00E+00	**Total**	**7.35E+13**

Table 5-16. Neutron induced radionuclide inventory in long-lived waste from Oskarshamn at 2075.

Nuclide	Activity/Bq	Nuclide	Activity/Bq
H-3	2.65E+14	Cs-137	6.98E+10
Be-10	0.00E+00	Ba-133	1.85E+03
C-14	3.23E+13	Pm-147	3.65E+06
Cl-36	1.30E+10	Sm-151	1.79E+08
Ca-41	0.00E+00	Eu-152	8.44E+04
Fe-55	3.17E+13	Eu-154	1.26E+08
Co-60	2.06E+14	Eu-155	6.22E+06
Ni-59	2.35E+14	Ho-166m	1.34E+02
Ni-63	2.09E+16	U-232	2.32E+06
Se-79	4.55E+04	U-235	5.62E+02
Sr-90	6.40E+10	U-236	7.44E+05
Zr-93	5.80E+11	Np-237	9.96E+05
Nb-93m	2.83E+13	Pu-238	5.23E+09
Nb-94	4.35E+11	Pu-239	6.34E+09
Mo-93	3.92E+12	Pu-240	1.08E+07
Tc-99	6.19E+11	Pu-241	3.66E+08
Ru-106	5.32E+00	Pu-242	1.29E+05
Ag-108m	8.63E+01	Am-241	1.44E+08
Pd-107	3.12E+04	Am-242m	1.36E+05
Cd-113m	2.14E+04	Am-243	1.68E+06
Sn-126	3.96E+05	Cm-243	2.73E+05
Sb-125	1.10E+09	Cm-244	3.94E+07
I-129	5.17E+04	Cm-245	3.72E+04
Cs-134	6.60E+05	Cm-246	1.45E+04
Cs-135	6.12E+05	**Total**	**2.17E+16**

Table 5-17. Radionuclide inventory in surface contamination of long-lived waste from Oskarshamn at 2075.

Nuclide	Activity/Bq	Nuclide	Activity/Bq
H-3	0.00E+00	Cs-137	0.00E+00
Be-10	0.00E+00	Ba-133	0.00E+00
C-14	0.00E+00	Pm-147	7.30E+06
Cl-36	0.00E+00	Sm-151	3.14E+09
Ca-41	0.00E+00	Eu-152	3.07E+06
Fe-55	9.95E+09	Eu-154	5.42E+08
Co-60	8.65E+11	Eu-155	1.33E+07
Ni-59	1.57E+12	Ho-166m	2.02E+05
Ni-63	1.45E+14	U-232	5.67E+04
Se-79	1.01E+09	U-235	2.08E+02
Sr-90	1.77E+11	U-236	3.85E+06
Zr-93	3.06E+09	Np-237	4.62E+06
Nb-93m	5.84E+12	Pu-238	2.30E+10
Nb-94	5.98E+10	Pu-239	4.18E+09
Mo-93	5.96E+08	Pu-240	6.08E+09
Tc-99	3.07E+08	Pu-241	5.66E+10
Ru-106	0.00E+00	Pu-242	3.20E+07
Ag-108m	5.03E+11	Am-241	1.94E+10
Pd-107	4.05E+07	Am-242m	1.25E+08
Cd-113m	0.00E+00	Am-243	3.65E+08
Sn-126	7.30E+06	Cm-243	4.97E+07
Sb-125	1.35E+08	Cm-244	4.23E+09
I-129	1.69E+08	Cm-245	5.35E+06
Cs-134	0.00E+00	Cm-246	1.95E+06
Cs-135	0.00E+00	**Total**	**1.54E+14**

Table 5-18. Neutron induced radionuclide inventory in long-lived waste from Ringhals at 2075.

Nuclide	Activity/Bq	Nuclide	Activity/Bq
H-3	1.38E+14	Cs-137	4.31E+10
Be-10	5.94E+04	Ba-133	4.95E+09
C-14	2.16E+14	Pm-147	8.86E+06
Cl-36	7.11E+10	Sm-151	4.43E+11
Ca-41	3.28E+11	Eu-152	3.37E+12
Fe-55	1.44E+14	Eu-154	5.99E+10
Co-60	4.68E+15	Eu-155	3.36E+09
Ni-59	1.06E+15	Ho-166m	4.21E+09
Ni-63	1.17E+17	U-232	2.39E+06
Se-79	8.03E+10	U-235	1.01E+01
Sr-90	3.94E+10	U-236	8.85E+05
Zr-93	1.22E+08	Np-237	1.18E+06
Nb-93m	7.53E+14	Pu-238	5.55E+09
Nb-94	1.39E+13	Pu-239	2.90E+08
Mo-93	9.25E+12	Pu-240	1.26E+07
Tc-99	1.32E+12	Pu-241	1.92E+08
Ru-106	0.00E+00	Pu-242	1.53E+05
Ag-108m	1.46E+12	Am-241	1.16E+08
Pd-107	3.71E+04	Am-242m	1.50E+05
Cd-113m	3.60E+07	Am-243	1.99E+06
Sn-126	4.71E+05	Cm-243	2.27E+05
Sb-125	1.34E+10	Cm-244	2.50E+07
I-129	6.16E+04	Cm-245	4.42E+04
Cs-134	2.10E+07	Cm-246	1.72E+04
Cs-135	7.29E+05	**Total**	**1.24E+17**

Table 5-19. Radionuclide inventory in surface contamination of long-lived waste from Ringhals at 2075.

Nuclide	Activity/Bq	Nuclide	Activity/Bq
H-3	0.00E+00	Cs-137	0.00E+00
Be-10	0.00E+00	Ba-133	0.00E+00
C-14	0.00E+00	Pm-147	2.31E+06
Cl-36	0.00E+00	Sm-151	1.99E+09
Ca-41	0.00E+00	Eu-152	2.21E+06
Fe-55	2.64E+10	Eu-154	3.26E+08
Co-60	7.69E+11	Eu-155	5.28E+06
Ni-59	6.29E+11	Ho-166m	5.57E+04
Ni-63	5.63E+13	U-232	5.23E+04
Se-79	0.00E+00	U-235	1.23E+02
Sr-90	1.15E+11	U-236	2.54E+06
Zr-93	7.13E+10	Np-237	3.39E+06
Nb-93m	8.98E+12	Pu-238	1.97E+10
Nb-94	8.56E+10	Pu-239	2.43E+09
Mo-93	5.47E+09	Pu-240	3.48E+09
Tc-99	4.35E+09	Pu-241	3.52E+10
Ru-106	0.00E+00	Pu-242	2.10E+07
Ag-108m	9.36E+10	Am-241	1.37E+10
Pd-107	0.00E+00	Am-242m	7.64E+07
Cd-113m	0.00E+00	Am-243	3.29E+08
Sn-126	4.95E+06	Cm-243	3.73E+07
Sb-125	2.43E+08	Cm-244	3.89E+09
I-129	0.00E+00	Cm-245	7.46E+06
Cs-134	0.00E+00	Cm-246	2.47E+06
Cs-135	0.00E+00	**Total**	**6.71E+13**

Table 5-20. Neutron induced radionuclide inventory in long-lived waste from Ågesta at 2075.

Nuclide	Activity/Bq	Nuclide	Activity/Bq
H-3	1.98E+09	Cs-137	0.00E+00
Be-10	0.00E+00	Ba-133	0.00E+00
C-14	6.89E+10	Pm-147	0.00E+00
Cl-36	3.17E+02	Sm-151	0.00E+00
Ca-41	0.00E+00	Eu-152	0.00E+00
Fe-55	8.54E+03	Eu-154	0.00E+00
Co-60	1.34E+08	Eu-155	0.00E+00
Ni-59	4.02E+11	Ho-166m	0.00E+00
Ni-63	2.42E+13	U-232	0.00E+00
Se-79	0.00E+00	U-235	0.00E+00
Sr-90	0.00E+00	U-236	0.00E+00
Zr-93	0.00E+00	Np-237	0.00E+00
Nb-93m	0.00E+00	Pu-238	0.00E+00
Nb-94	3.13E+05	Pu-239	0.00E+00
Mo-93	0.00E+00	Pu-240	0.00E+00
Tc-99	6.94E+08	Pu-241	0.00E+00
Ru-106	0.00E+00	Pu-242	0.00E+00
Ag-108m	6.66E+11	Am-241	0.00E+00
Pd-107	0.00E+00	Am-242m	0.00E+00
Cd-113m	1.43E+09	Am-243	0.00E+00
Sn-126	0.00E+00	Cm-243	0.00E+00
Sb-125	0.00E+00	Cm-244	0.00E+00
I-129	0.00E+00	Cm-245	0.00E+00
Cs-134	0.00E+00	Cm-246	0.00E+00
Cs-135	0.00E+00	**Total**	**2.53E+13**

Table 5-21. Radionuclide inventory in surface contamination of long-lived waste from Ågesta at 2075.

Nuclide	Activity/Bq	Nuclide	Activity/Bq
H-3	0.00E+00	Cs-137	0.00E+00
Be-10	0.00E+00	Ba-133	0.00E+00
C-14	0.00E+00	Pm-147	0.00E+00
Cl-36	0.00E+00	Sm-151	0.00E+00
Ca-41	0.00E+00	Eu-152	0.00E+00
Fe-55	1.32E+00	Eu-154	0.00E+00
Co-60	1.87E+05	Eu-155	0.00E+00
Ni-59	2.08E+08	Ho-166m	0.00E+00
Ni-63	1.27E+10	U-232	0.00E+00
Se-79	0.00E+00	U-235	2.23E+00
Sr-90	5.50E+05	U-236	4.37E+01
Zr-93	0.00E+00	Np-237	1.00E+03
Nb-93m	0.00E+00	Pu-238	2.32E+06
Nb-94	3.97E+07	Pu-239	4.03E+07
Mo-93	0.00E+00	Pu-240	2.66E+07
Tc-99	2.67E+05	Pu-241	1.02E+07
Ru-106	0.00E+00	Pu-242	0.00E+00
Ag-108m	2.94E+07	Am-241	5.41E+07
Pd-107	0.00E+00	Am-242m	0.00E+00
Cd-113m	0.00E+00	Am-243	0.00E+00
Sn-126	0.00E+00	Cm-243	0.00E+00
Sb-125	0.00E+00	Cm-244	1.18E+03
I-129	0.00E+00	Cm-245	0.00E+00
Cs-134	0.00E+00	Cm-246	0.00E+00
Cs-135	0.00E+00	**Total**	**1.31E+10**

6 Uncertainties

One of the aims of updating the reference inventory for long-lived low and intermediate level waste was to reduce the uncertainties in the reference inventory from 1998 (SKBdoc 1416968). In several ways this has been achieved. Data for actual waste has replaced forecasts, generalized data for core components have been replaced by reactor-specific data and the calculation of the radionuclide inventory is based on a more consistent approach (see Chapter 3). However, several sources of uncertainty still remain in the present inventory. In the following, these sources are described and quantified.

6.1 Waste amounts and volume

Even though reactor-specific data for component weights are used, there are still uncertainties around the actual waste amounts and all quantities that depend on waste amounts. The greatest uncertainty comes from the forecasted weight and packing densities. For NPPs still in operation, there may very well be a need for replacing additional parts of the core support structure in the future, and except for the reactors in Oskarshamn there are no forecasted waste amounts for such replacements. Additionally, with an average replacement frequency of 15–20 years, the forecast for replacement of BWR control rods may turn out to overestimate the total number of control rods in the waste. Furthermore, there is currently no available technique to segment the control rods for disposal in steel tanks. Hence, there is still significant uncertainty in the actual waste volume for the control rods.

In addition to the above mentioned uncertainties, there are also uncertainties in the actual component weights and the extent to which the waste amounts can be reduced by separating neutron induced material from surface contaminated parts (the latter would be disposed of as short-lived waste). However, these uncertainties are likely to be much smaller than the uncertainties in forecasted waste amounts.

Packing densities and choice of tank wall thickness will impact the estimated repository volume significantly. From the data reported from Barsebäck, Forsmark and Oskarshamn, the packing density varies between 1.7 and 0.12 tonne/m^3 with an average of 0.677 tonne/m^3 and the average wall thickness is 96 mm. Assuming a normal distribution of the average packing density, 0.75 and 0.61 tonne/m^3 are the upper and lower limits, respectively, calculated as a 95% confidence interval.

Figure 6-1 shows the total disposal volume for the waste, assuming different packing densities and tank types for the core components, with volume contributions from segmented PWR RPVs and biological shields kept constant.

6.2 Material

Material specifications have been reported for almost all of the components in the waste, except for the small amounts of miscellaneous scrap reported in Section 2.1.10. For the core components only the main construction materials have been reported, but in many cases small amount of other materials may be present. For example, the core spray of O2 is mainly constructed from stainless steel, but contain parts made of Inconel. However, these uncertainties are insignificant compared to the uncertainties in component weights.

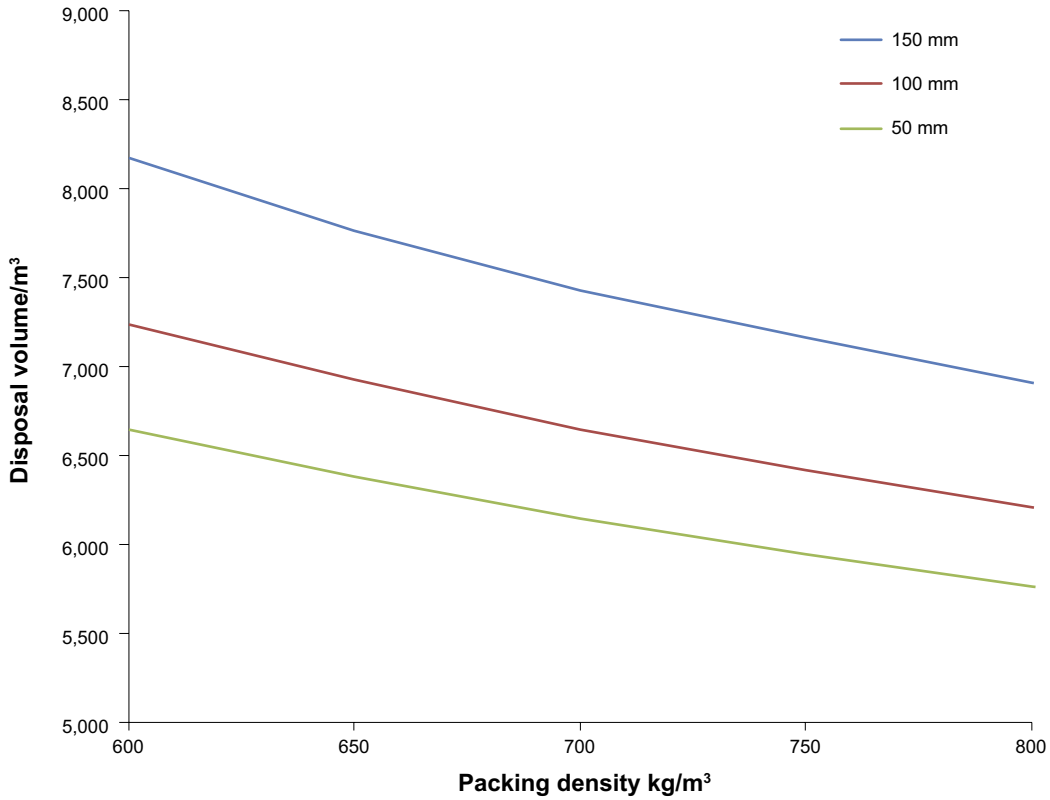

Figure 6-1. *Total disposal volume as a function of packing density and wall thickness for core components keeping the volume contribution from segmented PWR RPVs and PWR biological shields constant.*

6.3 Radionuclide inventory

The uncertainties in waste amounts do not have the same effect on the radionuclide inventory as on the waste volume. Burnup of stable nuclides due to neutron irradiation is not accounted for in the model. Hence for components that are expected to remain in the reactor for the entire operational life time of the reactor, almost all radioactivity is accounted for regardless of how many times the component is replaced. The only error introduced by ignoring replacement of components is that burnup of active nuclides will be overestimated.

Errors in the radionuclide inventory arise from approximations made in the model for the neutron transport calculations and the radionuclide transport model for estimating surface contamination (see Chapter 9). However, uncertainties in the material composition are likely to be the dominating sources of error. In an assessment of the overall uncertainty (SKBdoc 1344093) of the total activity from calculations made for the decommissioning studies, the total activity is estimated to lie within ±50% of the reported value. For the systems constituting long-lived waste from the NPPs, an uncertainty factor of 2 is assumed, which is interpreted as the true value being found between twice and half the reported activity. For nuclides originating from trace impurities in the construction material the error is likely higher. However, for these nuclides the impurity level used in the material specification is conservatively chosen (e.g. 1 ppm Cl in steel (Parry et al. 1997) giving rise to Cl-36).

As mentioned in Section 3.4, the activity in some waste packages from the existing waste from Forsmark have been assessed using measured dose rates in combination with calculated nuclide specific activities. This approach is likely to give more accurate activity estimates for the considered nuclides. However, for nuclides excluded from this assessment (actinides and fission products) the error is much larger than a factor of two for the individual waste packages. However, the overall error for the waste from Forsmark should not be significantly affected by this discrepancy.

The inventory from Barsebäck deviates from the inventory of the other NPPs, in that tritium is reported in neutron irradiated steel components (for the other BWRs the only sources of tritium are neutron irradiated boron and fissile material in neutron detectors). This may be attributed to additional reaction pathways to form tritium, not considered in the three group model described in Section 3.2. This means that the tritium inventory of the steel components from the NPPs may be underestimated by approximately two orders of magnitude. However, the overall tritium inventory is dominated by the BWR control rods, making the possible error for the steel components insignificant.

However, the example of tritium shows that the general estimated uncertainty factor of two only holds under the assumption that all relevant activation pathways have been considered. For example, the formation of Fe-60 in the neutron irradiated material has not been considered here. This activation pathway could turn out to be relevant since, at longer time scales, the Co-60 activity will be in equilibrium with Fe-60.

References

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1344093 ver 1.0	Svenska LWR – Aktivitetsinventarium vid rivning – Utvärdering av onoggrannheter. (In Swedish.)	Studsvik ALARA Engineering, 2012
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1416968 ver 1.0	Low and intermediate level waste in SFL 3-5: Reference inventory	SKB, 1998

Appendix A

Abbreviations

B1, B2	Barsebäck nuclear power plant units 1 and 2
BWR	Boiling water reactor
Clab	Central interim storage for spent nuclear fuel
Clink	Central interim storage and encapsulation plant for spent nuclear fuel
CVCS	Chemical and volume control system
F1, F2, F3	Forsmark nuclear power plant units 1, 2 and 3
ILW	Intermediate level waste
IRM	Intermediate range monitor
LLW	Low level waste
LPRM	Local power range monitors
NPP	Nuclear power plant
O1, O2, O3	Oskarshamn nuclear power plant units 1, 2 and 3
PHWR	Pressurized heavy-water reactor
PWR	Pressurized water reactor
R1, R2, R3, R4	Ringhals nuclear power plant units 1, 2, 3 and 4
RPV	Reactor pressure vessel
SFL	Repository for long-lived waste
SFR	Repository for short-lived radioactive waste
SIRM	Source- and intermediate range monitor
SKB	Swedish nuclear fuel and waste management Co.
SRM	Source range monitor
WRNM	Wide range neutron monitor
Å	Ågesta nuclear power plant

Material compositions

In the calculation of neutron induced activity, the following material compositions have been assumed. For some materials, there are reactor-specific differences in the composition.

Table B-1. Material compositions for different construction materials found in the inventory.

	Material composition w/w%									
Element	**1.4541**	**2103-R3**	**A24L**	**A3MM**	**AISI 302**	**AISI 304**	**AISI 304 (PWR)**	**AISI 304L**	**AISI 316L**	**AISI 316L (O3, F3)**
H										
Li							0.00001			
Be										
B										
C	0.04	0.16	0.029	0.05	0.07	0.04	0.07	0.02	0.018	0.025
N	0.04	0.015	0.068	0.1	0.04	0.04	0.04	0.04	0.08	0.04
O	0.01				0.01	0.01	0.01	0.01	0.01	0.01
F										
Na							0.001			
Mg										
Al	0.002				0.002	0.002	0.002	0.002	0.03	0.002
Si	0.5	0.5	0.58	0.6	0.6	0.6	0.6	0.6	0.5	0.6
P	0.015	0.02	0.029	0.03	0.02	0.02	0.02	0.02	0.015	0.02
S	0.008	0.03	0.024	0.025	0.015	0.015	0.015	0.015	0.001	0.015
Cl	0.0001				0.0001	0.0001	0.0001	0.0001	0.0001	0.0001
K										
Ca							0.002			
Ti	0.5				0.01	0.01	0.01	0.01	0.005	0.01
V	0.001				0.001	0.001	0.001	0.001	0.04	0.001
Cr	18	0.1	17	18.5	18.5	18.5	18.5	18.5	17	17
Mn	15	1.9	1.9	0.8	1.3	1.3	1.3	1.3	1.7	1.3
Fe	55.0	97.2	64.5	70.4	69.1	69.0	68.9	69.1	67.3	66.3
Co	0.02	0.02	0.012	0.02	0.03	0.1	0.1	0.03	0.035	0.03
Ni	10.5	0.1	13	9.5	10	10	10	10	11	12
Cu	0.1				0.1	0.1	0.1	0.1	0.1	0.1
Zn							0.01			
As	0.01				0.01	0.01	0.01	0.01	0.01	0.01
Se							0.004			
Sr										
Nb	0.01				0.01	0.01	0.03	0.01	0.01	0.01
Mo	0.2		2.9		0.2	0.2	0.2	0.2	2.15	2.5
Ag							0.0001			
Cd										
In										
Sn	0.01				0.01	0.01	0.01	0.01	0.01	0.01
Sb	0.001				0.001	0.001	0.001	0.001	0.001	0.001
Cs										
Ba										
Ce							0.01			
Nd										
Sm										
Eu										
Ho										
Ta	0.01				0.01	0.01	0.01	0.01	0.01	0.01
W	0.01				0.01	0.01	0.01	0.01	0.01	0.01
Re										
Au										
Hf										
Zr							0.001			
Th										
U										

Table B-1 (continued).

Element	Material composition w/w%											
	AISI 316L (R1)	Ag	Boron carbide	Concrete	Boron steel	C1070	Cd	Carbon steel	Hf	In	MgO	SIS 2333
H				0.6	0							
Li				0.004	0			0.00003				
Be				0.0004	0							
B			77.874	0.02	0.32							
C	0.02		22		0.025	0.7		0.7				0.04
N	0.04			0.01	0.040	0.01		0.01				0.04
O	0.01		0.1	45.81	0.010	0.05		0.05			39.70	0.01
F				0.05	0							
Na			0.001	1	0							
Mg				1	0						60.30	
Al	0.002			6	0.002	0.025		0.025				0.002
Si	0.6			30	0.60	0.25		0.25				0.6
P	0.02			0.05	0.020	0.01		0.01				0.02
S	0.015			0.2	0.015	0.01		0.01				0.015
Cl	0.0001		0.006	0.003	0.0001	0.0001		0.0001				0.0001
K				1	0							
Ca				10	0							
Ti	0.01		0.005	1	0.010	0.01		0.01				0.01
V	0.001			0.01	0.001	0.001		0.001				0.001
Cr	16.7			0.01	17.1	0.2		0.2				18.5
Mn	0.6		0.001	0.04	1.1	0.8		0.8				1.3
Fe	66.5		0.01	3	64.8	96.6		96.6	0.002			69.1
Co	0.03		0.002	0.002	0.011	0.02		0.02	0.001			0.03
Ni	13			0.003	13.1	0.6		0.6	0.001			10
Cu	0.1		0.001	0.002	0.088	0.1		0.1				0.1
Zn				0.006	0							
As	0.01			0.001	0.01	0.01		0.01				0.01
Se				0.0001	0							
Sr				0.04	0							
Nb	0.01			0.001	0.01	0.01		0.004	0.01			0.01
Mo	2.3			0.001	2.7	0.55		0.55	0.001			0.2
Ag		100		0.00002	0							
Cd				0.00001	0		100					
In					0					100		
Sn	0.01			0.0002	0.010	0.01		0.01				0.01
Sb	0.001			0.0002	0.001	0.0025		0.0025				0.001
Cs				0.0004	0							
Ba				0.1	0							
Ce				0.003	0							
Nd				0.004	0							
Sm				0.004	0							
Eu				0.0001	0							
Ho				0.0001	0							
Ta	0.01			0.0002	0.010	0.001		0.001	0.02			0.01
W	0.01			0.0001	0.010	0.01		0.01				0.01
Re				1.00E-07	0							
Au				1.00E-07	0							
Hf				0.0003	0				99.97			
Zr				0.02	0							
Th				0.0004	0							
U				0.0003	0							

Table B-1 (continued).

Element	Material composition w/w% SIS 2352	SS 2353	X-750	Zircalloy	Total inventory[1]
H					0.2
Li					0.001
Be					0.0001
B					0.8
C	0.025	0.025	0.03	0.015	0.4
N	0.04	0.04	0.01	0.004	0.026
O	0.01	0.01	0.01	0.14	15.0
F					0.02
Na					0.3
Mg					0.3
Al	0.002	0.002	0.7	0.005	2.0
Si	0.6	0.6	0.3	0.01	10.1
P	0.02	0.02	0.005		0.03
S	0.015	0.015	0.005		0.1
Cl	0.0001	0.0001	0.0001		0.001
K					0.3
Ca					3.3
Ti	0.01	0.01	2.5	0.004	0.3
V	0.001	0.001	0.01		0.01
Cr	18.5	17	16	0.1	7.4
Mn	1.3	1.3	0.2	0.003	0.8
Fe	68.1	65.1	7	0.15	52.3
Co	0.03	0.03	0.01	0.0001	0.02
Ni	11	13	72.0	0.05	4.8
Cu	0.1	0.1	0.1	0.003	0.1
Zn					0.003
As	0.01	0.01	0.05		0.01
Se					0.0003
Sr					0.01
Nb	0.01	0.01	0.9		0.01
Mo	0.2	2.7	0.05	0.0005	0.7
Ag					0.02
Cd					0.001
In					0.004
Sn	0.01	0.01	0.01	1.5	0.01
Sb	0.001	0.001	0.005		0.001
Cs					0.0001
Ba					0.03
Ce					0.002
Nd					0.001
Sm					0.001
Eu					3.26E-05
Ho					3.26E-05
Ta	0.01	0.01	0.1		0.004
W	0.01	0.01	0.01	0.005	0.006
Re					3.26E-08
Au					3.26E-08
Hf				0.01	0.3
Zr				98	0.4
Th				0.00004	0.0001
U				0.0003	0.0001

[1] *Total inventory* corresponds to the overall material composition of the waste (excluding packaging and grout), calculated from the mass fraction and composition of each material in the waste.