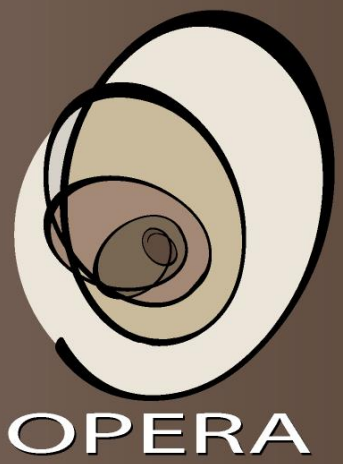

Waste families in OPERA

OPERA-PG-COV023

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Title: Waste families in OPERA
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Summary

After OPLA and CORA, OPERA (*OnderzoeksProgramma Eindberging Radioactief Afval*), begun in 2011, is the third Dutch research programme to have been undertaken on the topic of geological disposal of radioactive waste. After at least 100 years of storage in above-ground facilities, geological disposal is foreseen for the High-Level Waste (HLW) and Low- and Intermediate-Level Waste (LILW) described in this report.

It is important to collect the information necessary for geological disposal when the waste is generated. If not properly collected and documented upon generation, it will be difficult to trace back information about the waste at the time of disposal. The outline of a concept for disposal in clay [Verhoef, 2014a] describes the waste packages, intended emplacement of waste packages, and the underground facility located in a deep Boom Clay (Rupel) formation. To assess the performance of the facility, the potential rate of release of radionuclides from the waste has to be determined. The release of radionuclides from the waste matrix depends on how and where radionuclides are present in the waste package and in what matrix. In this report, the waste intended for disposal is characterised in order to determine the potential release from the packages. The report may also assist in setting criteria for the acceptance of waste and requirements for the accompanying information.

For the characterisation, the waste is grouped into so-called waste families. Families are groups of radioactive waste from the same origin, of similar nature, and having identical or closely related conditioning characteristics, while belonging to the same category of the current waste classification. The grouping into families facilitates calculating the source term, but is necessarily a simplification. Criteria for grouping the waste include the available information on the content or degradation mechanisms and the potential contribution to the source term (e.g. grouping of small volumes of waste).

For each of these families a standardised description is derived. On the one hand, the level of detail in the description matches the requirements of other tasks within OPERA, and on the other hand it reflects the available information. The standardised description includes: the origin of the waste, the number of packages, characteristics of the waste container (dimensions, steel or concrete type), the waste matrix (chemical composition of the waste), the radionuclides per waste container, how these radionuclides are expected to be present in the waste matrix and, if relevant, the expected heat output in 2130.

1. Introduction

1.1. Background

Radioactive substances and ionizing radiation are used and generated in medicine, industry, agriculture, research, education and electricity production. As a consequence, these activities generate radioactive waste. The current policy in the Netherlands is that radioactive waste be collected, treated and stored by COVRA (*Centrale Organisatie Voor Radioactief Afval*), the Central Organisation for Radioactive Waste. After an interim storage period of at least 100 years, radioactive waste is prepared for disposal. There is a world-wide scientific and technical consensus that geological repositories represent a safe disposal option for radioactive waste.
Geological disposal is the emplacement of radioactive waste in deep underground formations. The goal of geological disposal is the underground containment of the radioactive waste and isolation from our living environment in order to avoid the exposure of future generations to ionising radiation originating from the waste. In 2014 the outline of a concept for disposal in clay was revised [Verhoef, 2014a]. The description of the inventory of the waste to be disposed in this concept is classified into Low- and Intermediate-Level Waste (LILW), (Technically Enhanced) Naturally Occurring Radioactive Materials ((TE)NORM), and heat-generating as well as non-heat-generating High-Level Waste (HLW). This waste classification has been used in the Netherlands since 1985 [EA, 2014: p.24].

1.2. Objectives

In this report, the waste is grouped into families. Families are groups of radioactive waste from the same origin, of similar nature, and having identical or closely related conditioning characteristics while belonging to the same category of the current waste classification. For each of these families, a standardised composition is derived, of which the level of detail matches the requirements of other tasks within OPERA and reflects the available information. The decisions taken to derive these standardised compositions have been described in order to allow for the reproduction of the compositions.
The aim of this report is to provide the researchers within OPERA with the necessary information about the waste types, enable them to carry out their research work, and eventually to feed the safety assessment. It is important to collect this information when the waste is generated and processed. If not properly collected and documented upon generation and treatment, it will be difficult to trace back information about the waste at the time of disposal. A second aim is, therefore, to assist COVRA in setting criteria for the acceptance of waste and requirements for the accompanying information.

1.3. Realisation

A workshop was organised with experts of NRG, Brenk Systemplanung and COVRA in order to define a suitable set of waste families. As groundwork for the workshop, COVRA prepared a suggestion — based on the available information — for the families to be considered in OPERA. Criteria for grouping the waste include available information on the content or degradation mechanisms[a] and the potential contribution to the source term (grouping of small volumes of waste).

[a] The degradation mechanisms studied in OPERA do not cover all wastes. Knowledge about degradation of the waste matrix is essential to determining the source term. Therefore, only waste families for which the degradation is investigated within OPERA or in an appropriate alternative, e.g. EU research project Carbon-14 Source Term (CAST), are considered.

The following three questions were used to assess whether a composition in sufficient detail could be made for each family:

- What information is necessary for disposal?
- What (traceable) information is available?
- What assumptions need to be made for OPERA?

GKN has provided additional information of decommissioning waste from nuclear power plants and has reviewed the waste families relevant for nuclear power plants.

1.4.Explanation of contents

The relevant available knowledge regarding the radionuclides in the waste and their origins is presented for each waste family. This report is a follow-up to the reports produced by OPERA concerning waste inventory. It provides descriptions of the waste amounts [Verhoef, 2014a], the radionuclides [Hart, 2014] and the waste matrix [Meeussen, 2014] in order to fill in the missing details. The next chapter describes the choices and assumptions made when determining the characteristics of each waste family that are relevant to disposal in 2130. For clarity, the subsequent headings of the chapters list the waste for disposal as it is classified in the Netherlands. The identified waste families are categorised according to this waste classification.

2. Methodology

An assessment of the safety and feasibility of the geological disposal of waste requires a characterisation of the radioactive waste intended for disposal. To assess potential releases (i.e., the source term), one needs to know what radionuclides are present in the waste matrix and the disposal package. This chapter describes how the waste inventory and characterisation for determining the source term was developed.

2.1.Waste families

For the purposes of characterisation, waste is grouped into so-called waste families. Families are groups of radioactive waste from the same origin which are similar in nature and which have identical or closely related conditioning characteristics while belonging to the same category of the current waste classification. This grouping into families facilitates calculation of the source term, but is necessarily a simplification. Criteria for grouping the waste include the available information on the content or degradation mechanisms and the potential contribution to the source term[b]. Figure 2-1 below depicts the different families considered in this report.
For each of these families a standardised description has been derived. On the one hand, the level of detail in the description matches the requirements of other tasks within OPERA, and on the other hand it reflects the available information. The standardised description includes: the origin of the waste (the generation and processing), the number of packages, characteristics of the waste container (dimensions, steel or concrete type), the waste matrix (chemical composition of the waste), the radionuclides per waste container, how these radionuclides are expected to be present in the waste matrix and, if relevant, the expected heat output in 2130.

2.1.Information sources

The traceability of the standardised descriptions is important for any review or reproduction of the research, both as a starting point for future research and to support public confidence in the outcomes of the work. In the descriptions, therefore, the underlying sources, methodologies, choices and assumptions are documented along with the information supplied. When the source documentation did not clearly describe the data or how it was obtained, where possible the authors were contacted. These contacts proved useful in finding other information sources and/or estimating the reliability of the information (sources).

Every year, the volume and activity of the different types of waste in storage are published on the COVRA website. COVRA maintains an internal database on all Low- and Intermediate-Level Waste (LILW) stored at its facilities. The database contains all information provided by the waste generators to COVRA. The information is primarily collected for transport, storage and processing purposes and includes activity of radionuclides and dose rates. For information about chemical composition, EURAL codes are used. EURAL is the European system for waste classification that was implemented by the (chemical) waste directive [EU, 2001].

[b] This means that small volumes of different wastes have been grouped into one family (e.g. compacted waste), where the impact on the source term was limited.

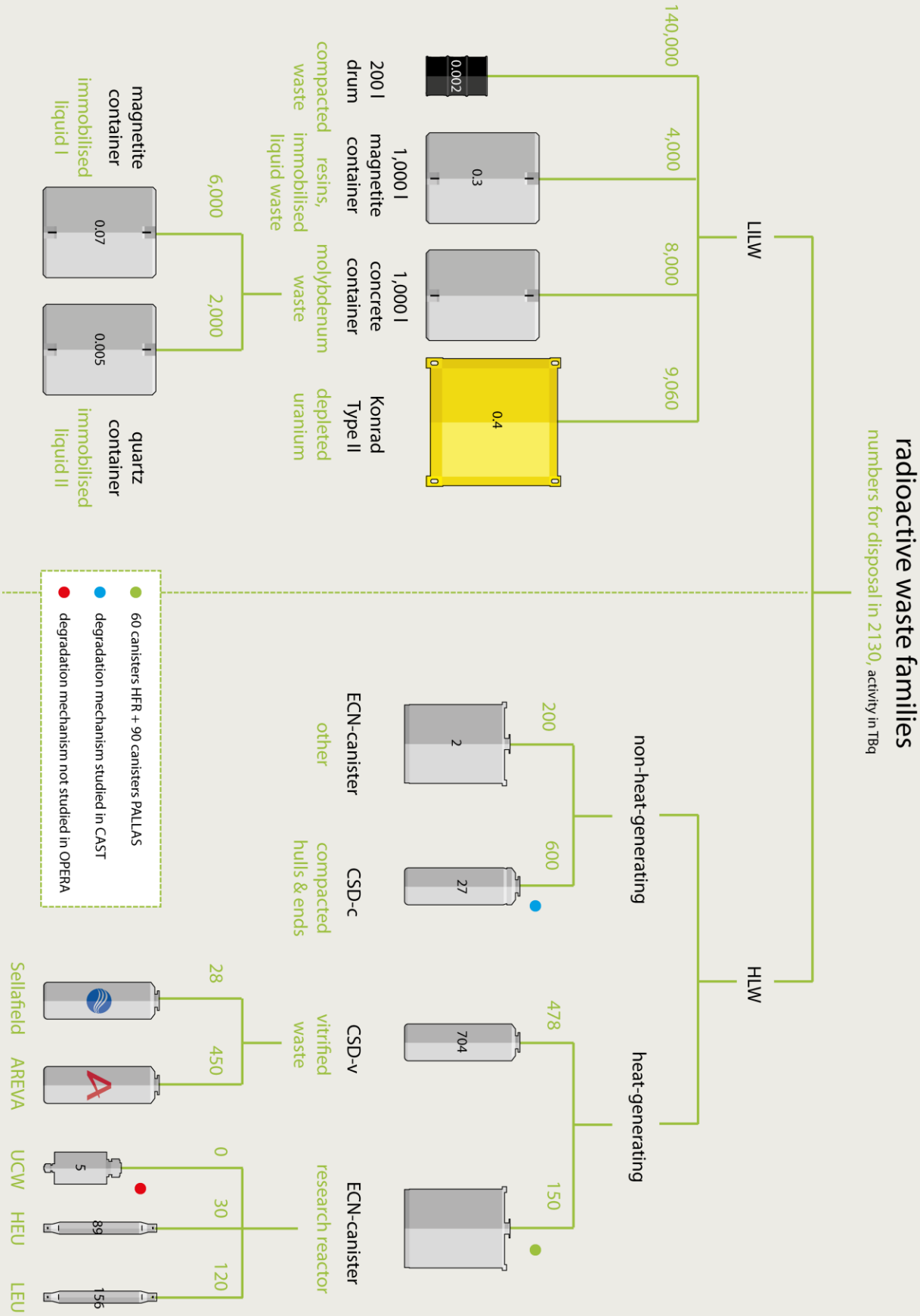

Figure 2-1. Different waste families in the waste inventory. The number of packages intended for disposal and the expected activity for each waste package at disposal in 2130 are included.

In the current research programme, only radionuclides are considered. It is expected that hazardous chemical elements will be taken into account in future research on geological disposal of this radioactive waste in other research programmes. The chemical composition of the waste (i.e., the matrix) is useful in assessing release and degradation mechanisms. The information that can be derived from the EURAL codes is not sufficient to make such assessments.

For a number of waste streams — including HLW, depleted uranium, spent ion exchangers and other liquid waste from nuclear power plants, and waste resulting from producing molybdenum for medical purposes — guaranteed parameters are agreed upon delivery to COVRA. Where available, the guaranteed parameters have been used in the body text of this report, and other and derived values have been listed in Appendices. However, in most cases the separate documents are confidential and not able to be published publicly. The information used for the inventory from these documents is listed in the References. The number of waste transports and amounts of stored HLW waste, including their activity and also distinguished in numbers of (classified) waste containers, are published every year on COVRA's website.

In addition to keeping a waste inventory, COVRA also periodically estimates future waste streams, based among others on extrapolation of the current rates and (long-term) contracts with waste generators. These internal predictions are used to plan new storage capacity on time and to calculate waste fees. The expected future streams also form the basis for the number of waste packages per family in this report.

2.2. Selection of radionuclides

One of the boundary conditions in the Dutch safety strategy is that all radioactive waste be stored above ground for a period for at least 100 years [Verhoef, 2014c:p.6]. The contribution to the activity of the waste is considered negligible at the time of disposal for radionuclides with a half-life:

- shorter than 10 years, due to the long-term storage period; and
- longer than 1×10^{10} years; this half-life is similar to primordial radionuclides (except for uranium isotopes).

To begin with, the (alphabetical) list of radionuclides from the COVRA database for LILW is used in this document; radionuclides with a half-life of longer than 10 years and shorter than 1×10^{10} years have been selected, including:

- radionuclides with daughters that meet the above half-life criteria; and
- so-called key nuclides, radionuclides used to obtain the missing radionuclides (see 2.3).

Radionuclides in the HLW with a half-life of more than 10 years — including their daughters — have been added to this alphabetical list.

2.3. Obtaining missing radionuclides

Usually waste generators and waste producers report radionuclides that can easily be measured using gamma spectrometry, e.g. ^{137}Cs and ^{60}Co, and contribute most to the dose rate of the waste upon delivery to COVRA. These nuclides are relevant for transport, treatment and storage but not necessarily for disposal. A methodology for completing missing radionuclides is to consider ^{137}Cs as a representative for fission products and ^{60}Co for activation products (key-nuclides), as well as using scaling factors (correlations

between the key-nuclides and the other nuclides). Methodologies to derive these scaling factors as well as a number of scaling factors themselves are available for waste from nuclear power plants [IAEA, 2009a].

In a nuclear plant, fission of actinides results in fission products. Most fission products decay during plant operation or storage prior to disposal. In this report, the scaling factor applied is derived from the chain yield distribution of fission products in the isobars provided in the Karlsruher Nuklidkarte [Magill, 2012]. Isobars with a chain yield greater than 0.045% indicated for the thermal fission of ^{235}U have been taken into account. The radionuclides that result from fission and which are relevant for the geological disposal of Dutch waste are shown in Figure 2-2.

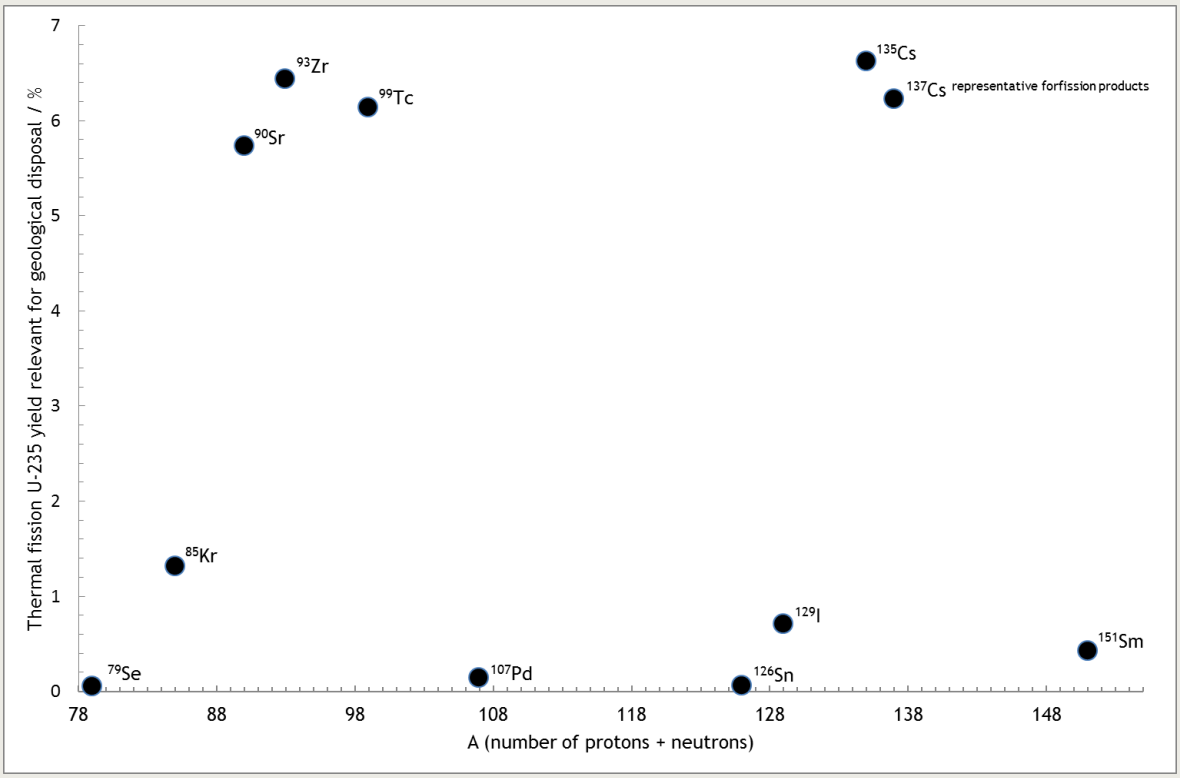

Figure 2-2. Yield from thermal fission U-235 and relevant for disposal of Dutch waste

In determining the activity of key radionuclides ^{60}Co and ^{137}Cs in the waste, the references collected as shortly as possible after the waste's generation were used. Neutron activation of fission products, e.g. ^{81}Kr, ^{145}Pm and ^{152}Eu, is not included, and therefore there are still radionuclides missing in the inventory. It is cumbersome to include neutron activation by hand, yet it can easily be incorporated by updating the nuclear library used to calculate the radionuclide inventory. Where the activity of the radionuclide was derived using the key-nuclide method, the key nuclide and the relative yield are reported in Appendix 1. The fission yields were taken from the Karlsruher Nuklidkarte [Magill, 2012]. Completion of missing radionuclides was performed for spent research reactor fuel. For processed HLW (vitrified residues and compacted metallic waste), it was used when the processing methodology of the waste gave sufficient reason to assume the presence of these radionuclides. For molybdenum waste (LILW waste arising from processing uranium targets), the reasons for their presence are also described. Chemical properties determine the separation of the radionuclides into different product and waste streams. This was used to complete the inventories, e.g. it follows that when ^{137}Cs is present in waste, ^{135}Cs must be present as well.

Information from the two IAEA documents listed below has been used to complete the inventory for radionuclides resulting from neutron activation of (additives and impurities in) materials and to obtain the necessary knowledge about how the radionuclides are expected to be present in the waste matrix:

- Application of ion exchange processes for the treatment of radioactive waste and management of spent ion exchangers [IAEA, 2002]; and
- Determination and use of scaling factors for waste characterisation in nuclear power plants [IAEA, 2009a].

2.4. Assumed decay period

To calculate the radionuclide inventory at the time of disposal, an average decay time of 130 years is assumed. This average decay time is calculated by subtracting the average production time (2000) from the start time of disposal (2130).

Generation of HLW intended for disposal in the Netherlands started in the sixties (except for spent research reactor fuel that was sent back to USA until 1988 [Dodd, 2000:p.29]). The Dodewaard nuclear power plant stopped its operations in 1997. The Borssele nuclear power plant, which started in 1973, was planned to stop operations in 2003 [EL&I, 2011:p.133]. As a result, generation of HLW was foreseen up to 2015 [EL&I, 2011:p.134]. The average production time would be in the '80s, and an average and rounded decay period for HLW of about 150 years could be assumed. The two lifetime extensions of the Borssele nuclear power plant, first up to 2014 and later 2034, as well as longer operation of the two research reactors, have shifted the average production time forward to around the turn of the century. Hence the average decay period of 130 years is assumed.

Using the average decay time, a standardised inventory of radionuclides per waste container at the time of disposal in 2130 has been derived for each waste family. The inventories are included in Appendix 1.

2.5. Assumed half-lives of selected radionuclides

There are many sources in which the half-lives of radionuclides can be found. It is beyond the scope of this report to assess which half-lives are 'correct'. Rather, the authoritative Karlsruhe Nuclide Chart has been used, which is periodically updated by Nucleonica and the JRC for the European Atomic Energy Community. In this report, half-lives and decay chains as described in the 8th edition of the Nuclide Chart [Magill, 2012] were used to derive the 2130 radionuclide inventories in Appendix 1. The half-lives were compared with the Isotope Browser from the IAEA Nuclear Data Section in November 2015. In Appendix I, the deviation between these two sources is shown for the selected radionuclides with a half-life of more than 10 years. With the exception of ^{135}Cs, ^{209}Po, ^{146}Sm, ^{121m}Sn and ^{126}Sm, the deviation in half-lives between these two sources is less than 10%.

2.6. Assumed waste matrices and packages at the time of disposal

Waste is stored at COVRA under controlled conditions, including a stable temperature, air humidity control (on average < 60% for LILW), and a dry storage environment. Heat-generating HLW is stored in an argon atmosphere and cooled by natural convection. It is therefore assumed in this report that the aging effects of conditioned waste during storage are limited. Consequently, for the waste matrix and packages, the same chemical and mechanical properties are expected after processing and at the time of disposal. This is

certainly true for depleted uranium, which is conditioned after storage prior to disposal. A possible exception is the carbonation of the first outer centimetres of concrete used for the conditioning of LILW. Carbonation is a slow process that occurs when calcium hydroxide in the cement reacts with carbon dioxide from the air and forms calcium carbonate. Considering the long storage times, the pH of the concrete pore water can fall from ca. 13 to below 10 at the time of disposal as a result of this process.

3. Heat-generating HLW

Heat-generating HLW consists of the vitrified waste from the reprocessing of spent fuel from the two nuclear power reactors in the Netherlands (Borssele and Dodewaard), the spent fuel of the two research reactors (Petten and Delft), and the spent uranium targets of molybdenum production.

3.1.Vitrified waste

3.1.1. Origin of waste and number of packages

In the Netherlands, the spent fuel from nuclear power plants (NPP) is reprocessed. Reprocessing involves extracting fissile materials (uranium and plutonium) for recycling to reduce the volume of high-level waste.

The main process for reprocessing is Plutonium Uranium Refining Extraction (PUREX), in which spent fuel is dissolved in nitric acid and tri-butyl-phosphate is used to extract the actinides [Gruppelaar, 1998]. Volatile fission products, e.g. Xe, Kr and I, can be diverted to secondary waste streams. Also, gaseous products can be formed during fuel dissolution, because of which radionuclides, e.g. ^{14}C, are not present in this vitrified waste [Davis, 1977]. The High-Level Liquid Waste (HLLW) contains fission products. Actinides and traces of plutonium and uranium are present in the liquid as well. HLLW is poured with a melted glass frit into a stainless steel container. The result is vitrified waste. Reprocessing also generates non-heat-generating HLW, the compacted hulls and ends, see section 4.1.

The choice to reprocess or not is left to the plant operator (waste generator). There are two NPPs in the Netherlands, in Dodewaard and in Borssele; both decided to reprocess their fuel. Direct disposal of spent nuclear power fuel is therefore no longer a consideration.

- Dodewaard was a Boiling Water Reactor that started operations in 1969. Dodewaard was shut down in 1997 [EA, 2014: p.29]. All spent fuel was reprocessed in the UK by BNFL (Sellafield). A total of 28 canisters of vitrified waste were returned to the Netherlands in 2010. This represented all waste from the complete operational life of Dodewaard. This number also includes the vitrified waste canisters that were exchanged for hulls and ends.

- Borssele is a pressurised Water Reactor that started its operational life in 1973 and will operate until 2034 [EA, 2014: p.65]. The spent fuel is reprocessed in France by AREVA (Cogéma [EZ, 1997:p.4]). All waste from reprocessing should be returned to the Netherlands before the end of 2052 [EA, 2014: p.15-16]. The total number of vitrified waste canisters for the complete life cycle of the plant is estimated at 450.

3.1.2. Waste container and matrix

The technical specifications of this waste are laid down in contracts with the reprocessing facility (waste producer). Examples of these specifications are a maximum in actinide mass, activity of some radionuclides, and glass composition. Waste is stored in the Netherlands that has been processed in France and in England. As for the waste matrix, guaranteed compositions of glass are presented in Table 3-1. In France, the fission products, zirconium, actinides and metal particles are included in the parameters related to the glass composition as oxides. In the UK, the guaranteed fission product and actinide oxides content are a minimum of 7.5 wt% and a maximum of 19 wt% with respect to the glass

weight. Figure 3-1 shows the schematics for this waste from France: Colis Standard de Déchets-vitrified (CSD-v).

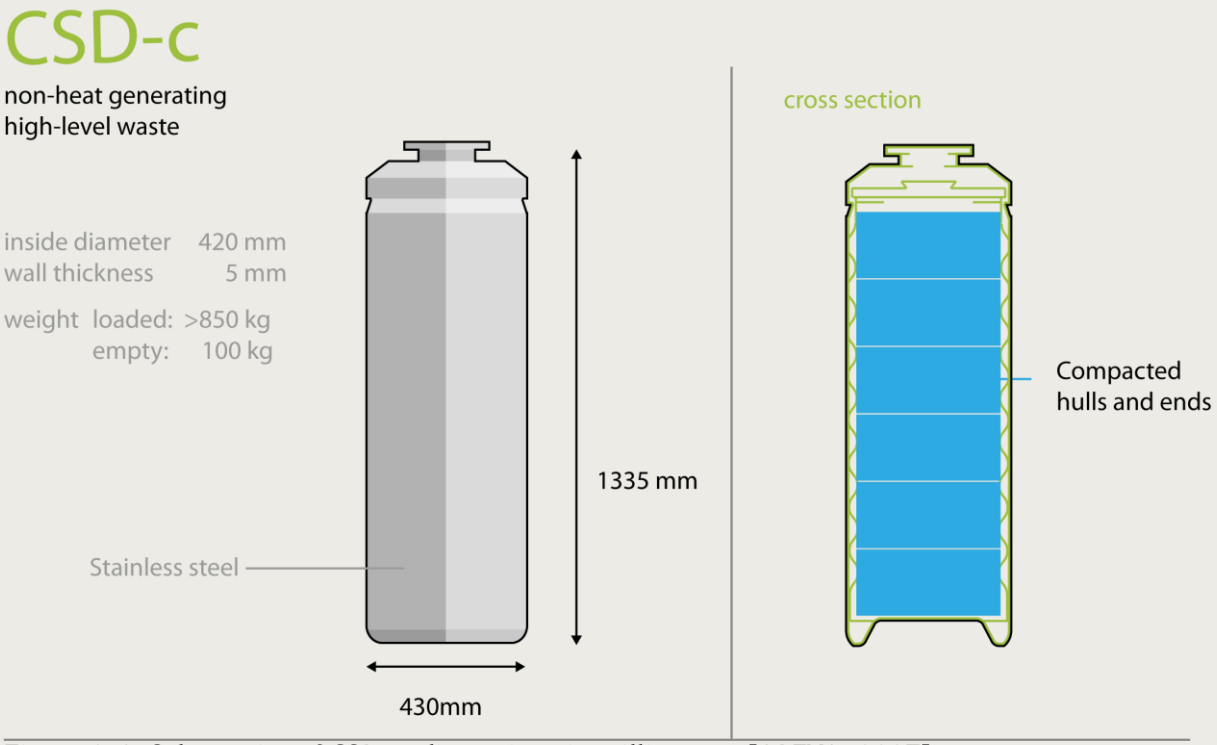

Figure 3-1. Schematics of CSD-v; dimensions in millimetres [AREVA, 2007]

Table 3-1. Guaranteed composition of the waste matrix glass stored in the Netherlands

Compound in wt%	*France*		*United Kingdom*	
	Minimum	*Maximum*	*Minimum*	*Maximum*
SiO_2	42.4	51.7	54	65
B_2O_3	12.4	16.5	21	31
Al_2O_3	3.6	6.6		
Na_2O	8.1	11	12	18
Li_2O	1.6	2.4		
Fe_2O_3	0	4.5		
NiO	0	0.5		
Cr_2O_3	0	0.6		
P_2O_5	0	1		
ZnO	2.2	2.8		
CaO	3.5	4.8		
RuO_2+Rh+Pd	0	3		
Oxides	4.2 *	18.5 *		

*(fission products + Zr + actinides) + metal particles

The radionuclides can be assumed to be uniformly distributed in this waste matrix. A matrix with a weight of about 400 kg is contained in a waste container of about 100 kg. A matrix that includes radionuclides is cooled for 24 hours before the top of the container (the dome including the mushroom) is welded.

There are no guaranteed parameters for the steel used for the waste containers. Stainless steel with a wall thickness of 5 mm was used in France and the UK. In the Netherlands, stainless steel type X12 CrNi 23.13 as per NF EN 10095 with additional specified values (including %C≤ 0.08) (or equivalent) is presented in the waste specification as typical.

The majority of these waste containers to be disposed of have the characteristics of waste processed in France. It is therefore assumed that each waste container contains the same radionuclide content as described in Appendix 1; the waste containers differ only in glass composition. In Table 3-1 the guaranteed composition made in France is sufficiently detailed. Further details on the glass produced by BNFL can be found in Appendix 2.

3.1.3. Radionuclides per waste container and heat output

Appendix 1 gives a detailed inventory of radionuclides per CSD-v. It is assumed that this inventory applies to both French and UK canisters. The methodology for determining the inventory is explained below.

Table 3-2 shows the guaranteed maximum values for uranium, plutonium, caesium and strontium per waste container. These maxima are obtained from the technical specifications for waste processed in France [AREVA, 2007] and in the United Kingdom [BNFL, 2001].

Table 3-2. Guaranteed maximum values per waste container stored in the Netherlands

Property	*Unit*	*France*	*United Kingdom*
Uranium	grams	4500	2000
Plutonium	grams	110	200
Cs-137	TBq	6600	8000
Sr-90	TBq	4625	5500

The average isotopic composition of uranium and plutonium for each waste container (shown in Appendix 2) and the ingrowth of daughters are used to determine the activity of these actinides. The comment for the values in Appendix 1 is *'max weight and isotopic'*.

In the technical specifications, values in activity typical for a waste container are also provided. The typical values used are given in Appendix 1, indicated as *'typical [AREVA, 2007]'*. For each waste container, the measured or calculated content of some radionuclides is provided. From a batch of 28 containers delivered to COVRA in 2012 from La Hague, the maximum radionuclide content is shown in Appendix 1, indicated as *'max batch 28 containers'*. This batch is used as a representative inventory for disposal within OPERA. These containers were produced between 1995 and 2002. Some actinides and one fission product is not in the reported inventory. A radionuclide content for the actinides ^{227}Ac, ^{242m}Am, ^{243}Cm, ^{246}Cm, ^{247}Cm, ^{248}Cm, ^{244}Pu, ^{229}Th, ^{230}Th, ^{232}U, ^{234}U and daughter of the uranium series ^{226}Ra is reported for compacted hulls and ends (CSD-c) but not for vitrified waste.

The content of plutonium isotopes in vitrified waste appears to be empirically similar, per waste container, to the activity of these isotopes in this compacted waste. Their origin is different and therefore it cannot be presumed that the amount of these actinides is similar to the content of vitrified waste. Compared to compacted hulls and ends, the activities of ^{241}Am and ^{243}Am are about 10^4 times larger for vitrified waste. For actinides for which no activity has been reported to COVRA, it is assumed that the activity in the vitrified waste is 10^4 times greater than that in the compacted hulls and ends (except californium). ^{231}Pa is not reported in reprocessed waste, but is assumed to be present in reprocessed waste because this radionuclide is reported for spent research reactor fuel.

Except for ^{151}Sm, for every non-volatile, longer-lived fission product indicated in Figure 2-2, the radionuclide content was reported; the ^{137}Cs activity in Table 3-2 was used to complete the radionuclide content for ^{151}Sm. In Appendix 1, as an example for the non-volatile fission products in Figure 2-2, the activity expected from the guaranteed maximum from

^{137}Cs activity is presented along with the maximum found in the batch. The difference between the activities is less than one order of magnitude. Neutron-activated fission products such as ^{145}Pm and ^{152}Eu are expected to be present in the inventory as well but are not reported.

At the time of disposal in 2130, the reported radionuclide content will contribute about 90% of the total activity of the waste container. The total activity is the summation of the activity of radionuclides for which an activity is determined, see Appendix 1.

The maximum heat production is guaranteed to be less than 2,000 W upon arrival on COVRA's premises. The heat production upon disposal — after a storage period of at least 100 years — is expected to be maximally 200 W for each container.

3.2. Spent research reactor fuel

3.2.1. Origin of waste and number of packages

The spent fuel from research reactors is not reprocessed. Spent research reactor fuel was previously sent back to the USA [Tozser, 2014] as part of the larger US nuclear non-proliferation policy. This ended in 1996 when a fee was required for accepting foreign spent fuel from so-called 'high-economy countries', including the Netherlands (Foreign Research Reactor Spent Nuclear Fuel Acceptance Programme) [Messick, 2006]. Storing spent research reactor fuel in the Netherlands became the preferred option at that time.

There are three research reactors that produce or have produced spent fuel for storage at COVRA [EA, 2014:p.65]:

- The High-Flux Reactor (HFR) 45 MW_{th} in Petten. The HFR is a tank-in-pool type research reactor. The core is composed of 33 fuel assemblies and 6 control assemblies. The HFR of the Institute for Energy (IE) of the Joint Research Centre (JRC) of the European Commission (EC) has been in operation since 1961. The Nuclear Research and Consultancy Group (NRG) operate the reactor.
- The Low-Flux Reactor (LFR) 30kW_{th} in Petten. The LFR started operations in 1960 and was shut down in 2010. The LFR is property of the Nuclear Research and Consultancy Group (NRG). The LFR was used mainly for the production of neutrons for biological and physical research.
- The Hoger Onderwijs Reactor (HOR) 2 MW_{th} in Delft. The HOR is an open-pool type research reactor, using MTR-fuel assemblies and low-enriched Uranium-235 (< 20%) as fuel. The core is composed of 20 fuel assemblies and 4 control assemblies. The HOR reactor of the Interfaculty Reactor Institute (IRI) of Delft University of Technology has been in operation since 1963.

Following international efforts to minimise and eventually eliminate the use of Highly Enriched Uranium (HEU), the conversion from HEU to low-enriched uranium (LEU) fuel was completed in 2005 for HOR and in 2006 for the HFR; the LFR was closed. The conversion of radioisotope production targets to low-enriched uranium is still ongoing. In the disposal concept, 150 ECN waste containers are intended to be disposed of [Verhoef, 2014a]. It is presumed that they all contain spent research reactor fuel. The number of waste containers containing HEU spent research reactor fuel is considered to be 30, and 120 waste containers are assumed to have LEU spent research reactor fuel.

3.2.2. Waste container and matrix

For the inventory it is assumed that all spent research reactor fuel has the same characteristics as the fuel used for the HFR (a single family). The spent fuel from the other

two research reactors in the Netherlands — LFR in Petten and HOR in Delft — compare closely to the fuel from the HFR. Differences include the ^{235}U-mass per fuel and control element, and the number of plates per fuel assembly. These differences are not considered to have a large impact on the source term for spent fuel. A distinction, however, is made between HEU and LEU fuel.

Table 3-3 shows the main fuel characteristics of fuel used at the High Flux Reactor in Petten, and Figure 3-2 shows the schematics of two different spent fuel elements (HEU and LEU) and the ECN container.

Table 3-3. Main fuel characteristics of the High Flux Reactor in Petten

Characteristic	*HEU*	*LEU*	*Source*
Fuel type	UAl_x	U_3Si_2	Ahlf, 1993 & Thijssen, 2006
Density [g cm^{-3}]	1	4.8	Thijssen, 2006
Fuel thickness [mm]	0.51	0.76	Ahlf, 1993 & Dodd, 2000 & NRG, 2012
Fuel length [mm]	600	600	Ahlf, 1993 & Dodd, 2000 & NRG, 2012
Enrichment BOI [%]	93	19.75	Ahlf, 1993 & Thijssen, 2006
^{235}U mass in fuel / control element [g] BOI	450 /310	550 / 440	Ahlf, 1993 & Thijssen, 2006
^{235}U mass in fuel / control element [g] EOI	177 /119	200 / 225	Dodd, 2000 & NRG, 2012
Max burn-up [%]	55%	70% / 60%	Dodd, 2000 & NRG, 2012
Number of plates in fuel / control element	23 / 19	20 / 17	Ahlf, 1993 & Thijssen, 2006
Framing materials	Al	AG3NE or equivalent	Dodd, 2000 & NRG, 2012
Waste matrix	Al	Al	Ahlf, 1993 & Dodd, 2000 & Thijssen, 2006, NRG, 2012
thickness of Al cladding on both sides for inner /outer fuel plate	0.38 / 0.57	0.38 / 0.57	Ahlf, 1993 & Dodd, 2000 & NRG, 2012
Burnable poison	1000 mg ^{10}B in Al side plates	40 Cd wires (diameter 0.5 mm)	Ahlf, 1993 Thijssen, 2006

BOI : Begin Of Irradiation, EOI: End of Irradiation

Figure 3-1 shows that the radionuclides are homogeneously distributed, as is the case for vitrified waste. In the spent research reactor fuel the radionuclides are present in the fuel meat of the fuel plate in the fuel assembly.

The Highly Enriched Uranium (HEU) fuel assemblies (horizontal cross section 81 mm × 77 mm, height 924 mm) contains 23 vertically arranged, parallel, curved fuel plates with a height of 625 mm. Each HEU plate consists of a layer of aluminium-uranium-alloy meat with a thickness of 0.51 mm, aluminium cladding with a thickness of 0.38 mm for the inner plates, and 0.57 mm for the outer plates. The fuel meat is thicker in LEU elements, viz. 76 mm. The length of both the HEU and LEU fuel inside the plates is 600 mm [Ahlf, 1993:p.9].

The thicknesses of the outer and inner fuel plates are respectively 1.65 and 1.27 mm for HEU and 1.90 and 1.52 mm for LEU. Each fuel assembly contains 2 outer fuel plates; all other fuel plates are inner plates. The larger thickness of fuel plates for LEU has resulted in fewer fuel plates per fuel element. Figure 3-2 shows cross sections of the assembled HEU and LEU fuel element. The neutron absorbers ^{10}B in an aluminium matrix for HEU and cadmium wires for LEU are clearly visible. For each HEU fuel assembly, two flat side plates, together containing 1,000 mg ^{10}B, were used [Ahlf, 1993: p.9].

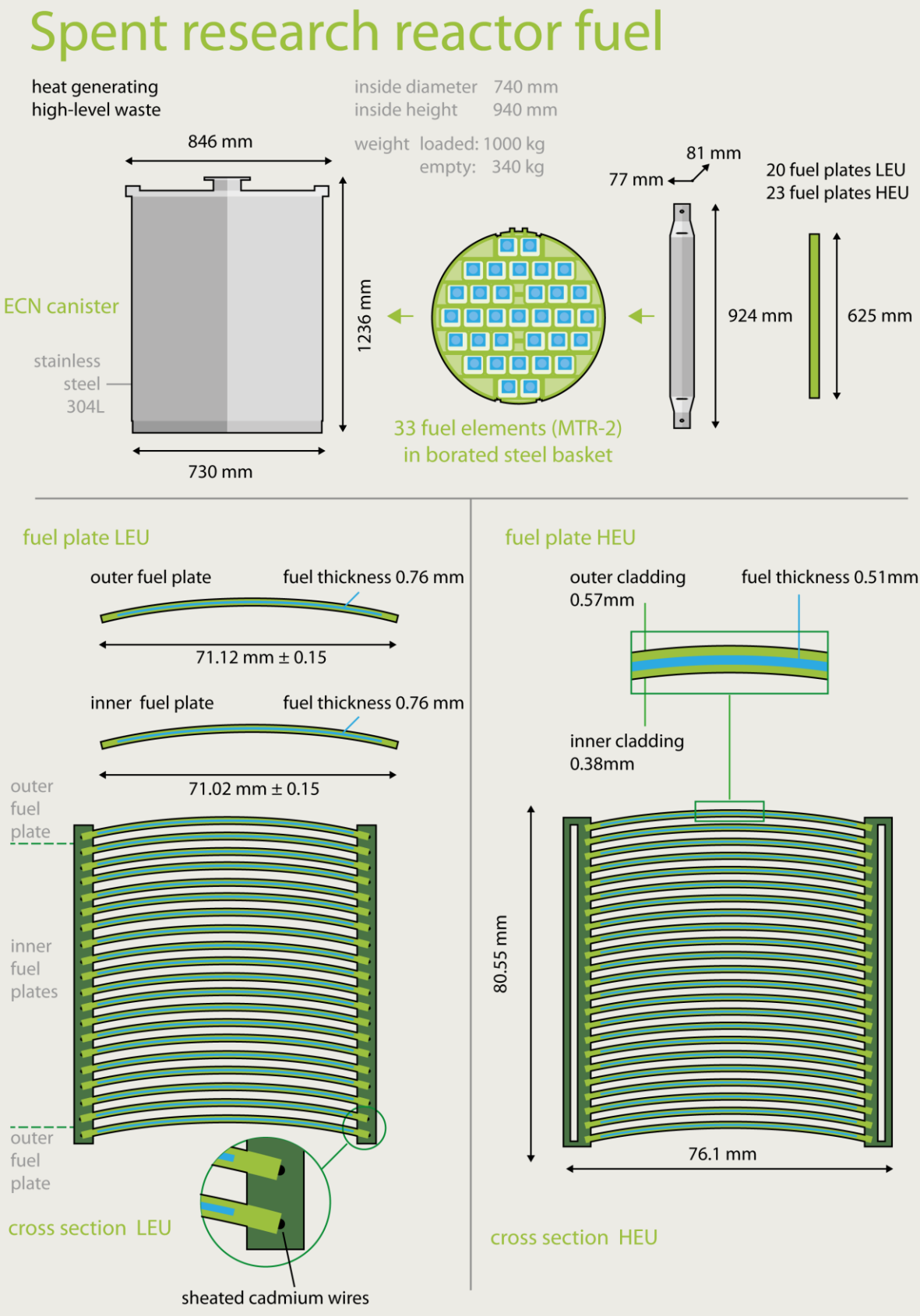

Figure 3-2. Schematics of spent fuel and an ECN container; dimensions in millimetres [Kaa,1996 & NRG, 2012]

The typical weight of a loaded canister is 1,000 kg. An empty waste container weighs 340 kg. The wall thickness is 5 mm. The lid on the container is welded, and the canister is filled with helium to check the weld and to verify the canister's integrity during storage. An additional ring is put on the weld in order to accommodate the tension forces during lifting the container at the mushroom.

The OPERA supercontainer is adopted from the ONDRAF/NIRAS supercontainer concept developed in the Belgian research programme on geological disposal of radioactive waste. In OPERA, the possibility of disposing of two canisters in a supercontainer is being investigated, see Figure 3-3 above [Verhoef, 2014a]. Because spent fuel from research reactors is stored without reprocessing, it contains fissile material. An important design consideration for disposal packages with fissile material is the critical mass. Critical mass is the smallest amount of fissile material needed for a sustained nuclear chain reaction (criticality). The critical mass of a fissile material depends, among others, upon its shape, its enrichment, its purity, and its surroundings (i.e., whether a moderator is present). The critical mass for non-irradiated HEU fuel is 16.7 kg, and more than 200 kg for non-irradiated LEU fuel [Kaa, 1997]. The conservative estimates for irradiated fuel in the previous research programme were 14.4 kg for HEU fuel and 43 kg for LEU fuel [Dodd, 2000:p.21]. At first glance these values appear to conflict in the literature, because critical mass usually increases at larger burn-up for spent fuel.

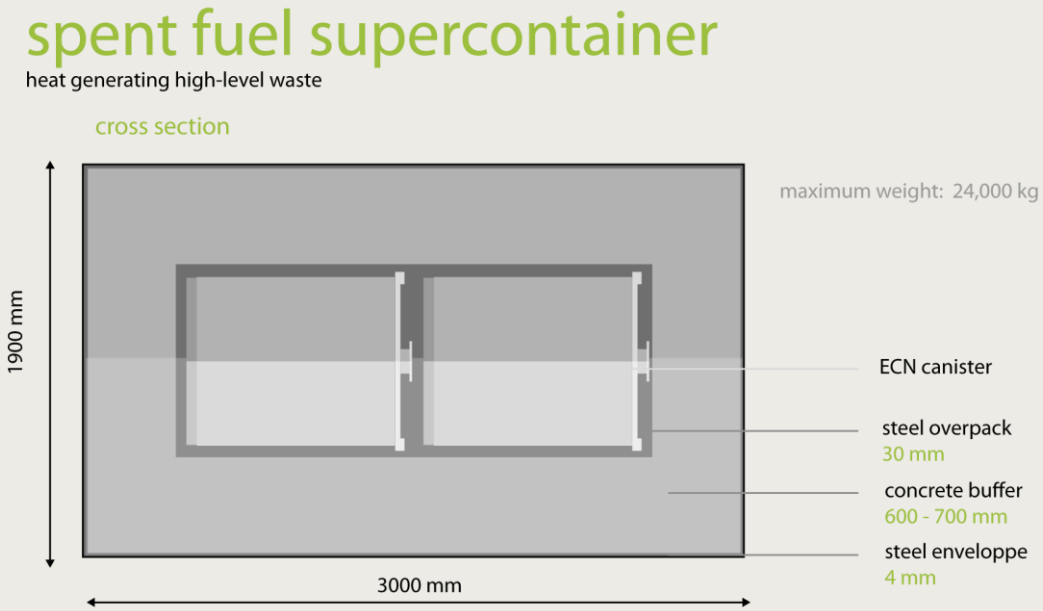

Figure 3-3. Schematics of a disposal container for spent fuel ('super container'); dimensions in millimetres.

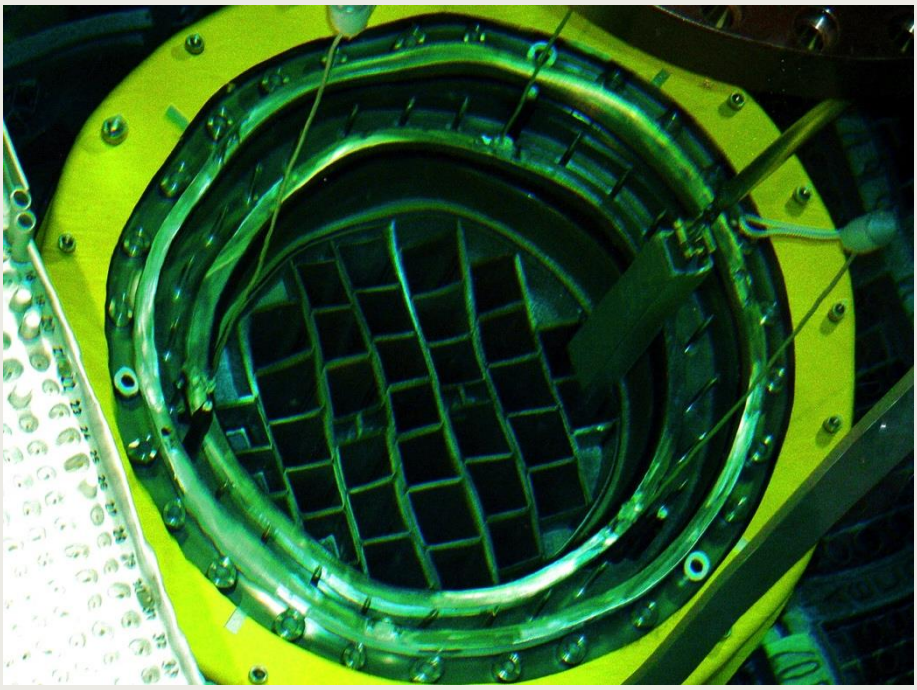

Figure 3-4. A borated steel basket is visible during the loading of the fuel and control elements in the CASTOR MTR2 transport cask.

To ensure that criticality cannot occur, the amount of fissile material should be below the critical mass. An ECN canister loaded with 33 spent fuel elements contains 16 kg of fissile material in the case of HEU fuel, or 97 kg in the case of LEU fuel, above the reported critical masses of 14.4 kg for HEU and 43 kg for LEU [Dodd, 2000:p.21]. In the ECN canisters the fuel elements are loaded in baskets of borated steel, ensuring that criticality cannot occur. Nevertheless, it would be worthwhile to investigate the disposal of a single ECN canister in a supercontainer.

3.2.3. Radionuclides per waste container and heat output

Appendix 1 gives a detailed inventory of radionuclides per ECN canister, for both HEU and LEU fuels. Further information on the actinides in the fuel and control elements can be found in Appendix 3. Below, the methodology for determining the inventory is explained.

In the previous research programme CORA, a nuclide inventory for irradiated HEU fuel and control elements was established [Dodd, 2000:p.35-37]. In Appendix 1, the activity of radionuclides per waste container can be found. These radionuclides are determined by multiplying the activity of each radionuclide for an irradiated HEU fuel element by 33 after 130 years [Dodd, 2000: p.36], as described in the previous research programme, because each waste container can contain a maximum of 33 fuel and control elements. The control elements contain less uranium than the fuel elements, and as a result also fewer actinides and fission products. A safety assessment of the disposal of spent research reactor fuel was not performed in the previous two Dutch research programmes, and to simplify the input for the research tasks in OPERA it is therefore assumed that the waste container holds 33 fuel elements. The activity at the time of disposal of a waste container with 33 LEU fuel elements is about two times larger than that of a container with 33 HEU elements, mainly because the 6 times larger initial total uranium content results in a twenty times larger activity in plutonium and a higher amount of fissioned atoms per fuel element.

Except for ^{85}Kr, for every longer-lived fission product indicated in Figure 2-2, a radionuclide content was reported; the ^{137}Cs activity calculated after one month was used to complete the radionuclide content for ^{85}Kr for irradiated HEU fuel. For irradiated LEU fuel, more of these long-lived fission products were not reported: namely ^{135}Cs, ^{129}I, ^{107}Pd, ^{79}Se and ^{126}Sn. To begin with, for missing neutron-activated fission products and radionuclides that can result from fission, as well as for neutron activation of radionuclides in the inventory for irradiated LEU fuel, it is assumed that the activity is similar to that of irradiated HEU fuel.

An activity for every radionuclide expected from the actinide series was not available. The activity of some actinides is determined from the weight described in Appendix 3. These values are indicated with *'EOI [NRG, 2012]'*. The inventory by Dodd et al. was used to complete the missing actinides, in which the differences regarding the uranium isotopic content and the amount between HEU and LEU were roughly taken into account. For actinides the activity is presumed to be 10 times larger for LEU than for HEU; for fission products a similar content is assumed for HEU and LEU. For other missing actinides, actinides in vitrified waste were considered. For uranium and plutonium, a higher content in HEU is expected than in vitrified waste because these actinides are extracted from the waste. The opposite is expected for actinides such as curium and americium that are not extracted when producing the vitrified waste.

In addition to the fission products and actinides, radionuclides resulting from neutron activation also need to be taken into account. For an irradiated HEU element, radionuclides resulting from neutron activation of impurities are reported by Dodd et al. as being ^{14}C and ^{36}Cl. For irradiated LEU fuel, these radionuclides are not reported. To begin with, the same activity is presumed for irradiated LEU. For an HEU element, neutron activation of burnable poison (^{10}B) and matrix (Al) is not expected to result in radionuclides that are relevant for geological disposal. This is not the case for LEU fuel elements. ^{32}Si can be expected to be present in spent LEU due to activation of the fuel U_3Si_2, and ^{113m}Cd is expected to be present by the natural isotopic composition of cadmium. The activity has not been calculated for either radionuclide.

At the time of disposal in 2130, the reported radionuclide content will contribute about 99.7% to the total activity of a waste container filled with HEU fuel elements and 92% of one filled with LEU fuel elements. This total activity is the summation of the activities of the radionuclides for which an activity is determined in Appendix 1.

After a storage period of at least 100 years, the heat production for each fuel element is less than 1 W [NRG, 2012], by which the maximum heat production for each waste container can be assumed to be 33W.

3.3.Uranium collection filters

3.3.1. Origin of waste and number of packages

The medical isotope ^{99m}Tc is a widely used for medical diagnostics. With a half-life of only 6 hours, ^{99m}Tc must be produced near to where it is used. In hospitals, the technetium is produced by the decay of ^{99}Mo. Mallinckrodt in Petten is the second largest producer of molybdenum for medical uses in the world.

The first step in the production of medical molybdenum is the irradiation of an HEU target. The UAl_x-dispersion targets used in the Netherlands are produced by CERCA [IAEA, 2013:p.6]. A sodium hydroxide solution is used to dissolve the target, including the aluminium cladding and the uranium-aluminium-alloy meat. Dissolution produces an alkali

solution containing sodium molybdenum and a solid residue. The solid residue contains uranium and most of the fission products except the alkali metals, iodine, fission gases, alkaline earths, and the elements that can act as either an acid or base such as molybdenum and aluminium. The solid residue is filtered from the solution. The filtrate is collected in a precipitation vessel in which uranium precipitates in the pores of a stainless steel filter (sinter metal) as $Na_2U_2O_7 \bullet 6H_2O$. The residue from a number of batches is collected in a collection filter (UCW filter). These filters are dried and subsequently packaged for storage at COVRA.

Degradation and corrosion processes of the waste matrix during geological disposal are not investigated in OPERA. Considering this and the relatively small number of packages expected, uranium collection filters are not considered in the inventory for the calculation of the source term within OPERA. It is recommended that future research programmes investigate an end-point management of this waste. This management can include investigating the degradation and corrosion processes for geological disposal, but also a processing methodology for the waste in which a product is obtained with a homogeneous distribution of radionuclides in a matrix with properties familiar for geological disposal, e.g. one similar to vitrified waste (CSD-v).

3.3.2. Waste matrix and container

Figure 3-5 shows the collection filter, the aluminium drum and the ECN canister.

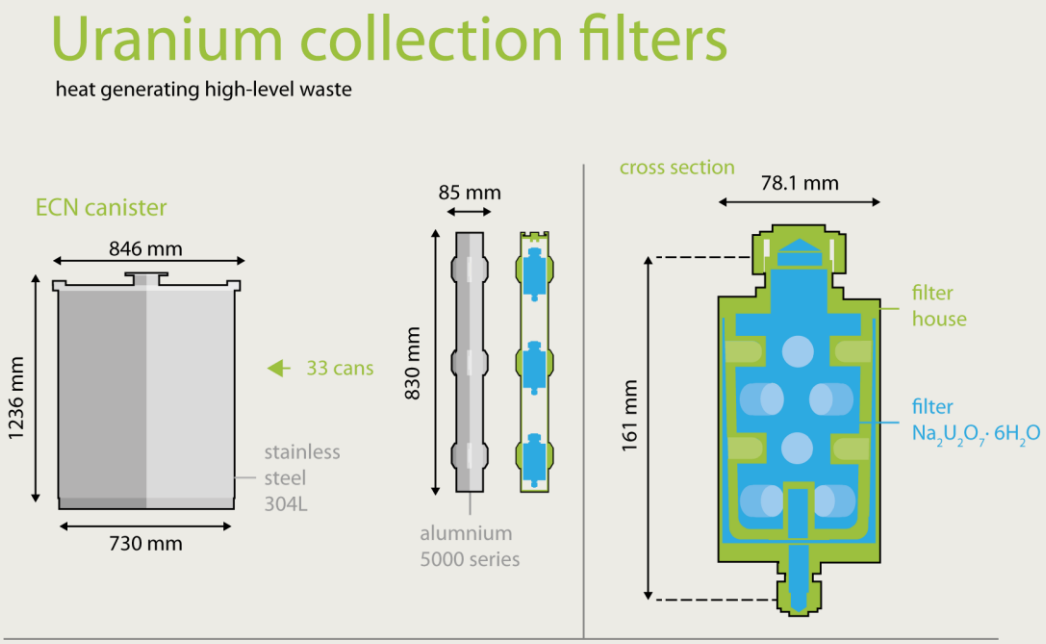

Figure 3-5. Schematics of Uranium collection filter and ECN container; dimensions in millimetres [NRG, 2009].

Three of these filters are collected in an aluminium drum. These drums are filled with helium (700-1,000 mbar at room temperature) and welded to be gastight [NRG, 2009]. Corrosion-resistant aluminium from the 5,000 series is used as material for the drums. The maximum number of drums per waste container is 33, by which the maximum number of uranium collection filters that can be contained in a waste container is 99. The radionuclide inventory per waste container is shown in Appendix 1. Each waste container with uranium collection filters contains 18 kilograms of uranium. This uranium content is greater than that of a waste container with spent HEU fuel but less than that of a waste

container with spent LEU fuel. This waste contains fissile material for which the critical mass has not yet been defined.

3.3.3. Radionuclides per waste container and heat output

As a first approximation with which to complete the missing radionuclides, radionuclides from the actinide series for which the activity is not calculated are assumed to be equal to that of HEU. The inventory is described in Appendix 1 because these details are important for the LILW characterisation of processed molybdenum waste.

4. Non-heat-generating HLW

All radioactive waste that is stored in COVRA's HABOG facility (treatment and storage building for high level waste) is categorised in the Netherlands as high-level waste. Heat-generating HLW and non-heat-generating HLW are stored in separate compartments of the HABOG. Non-heat-generating HLW is generated by the reprocessing of waste other than the vitrified residues. It also includes waste from research on reactor fuel and some decommissioning, legacy and other waste.

At present there is no internationally endorsed classification of this waste. Non-heat-generating HLW falls into the IAEA category of intermediate-level waste [IAEA, 2009b]. It is similar to types of waste considered by the Swiss waste management organisation as Intermediate Level Waste [NAGRA,2002; p.99] and by the Belgian waste management organisation as category B waste: low-level and medium-level conditioned waste containing long-lived radionuclides [NIROND, 2013; p.350].

4.1. Compacted hulls and ends

4.1.1. Origin of waste and number of packages

The two main[c] types of waste resulting from reprocessing of nuclear power spent fuel are heat-generating vitrified waste, described in section 3.1, and the non-heat-generating compacted hulls and ends, described here. Only compacted hulls and ends from the PWR in Borssele are stored by COVRA; the compacted hulls and ends from Dodewaard have been exchanged with the waste producer for a small amount of additional vitrified waste [Kers, 2004].

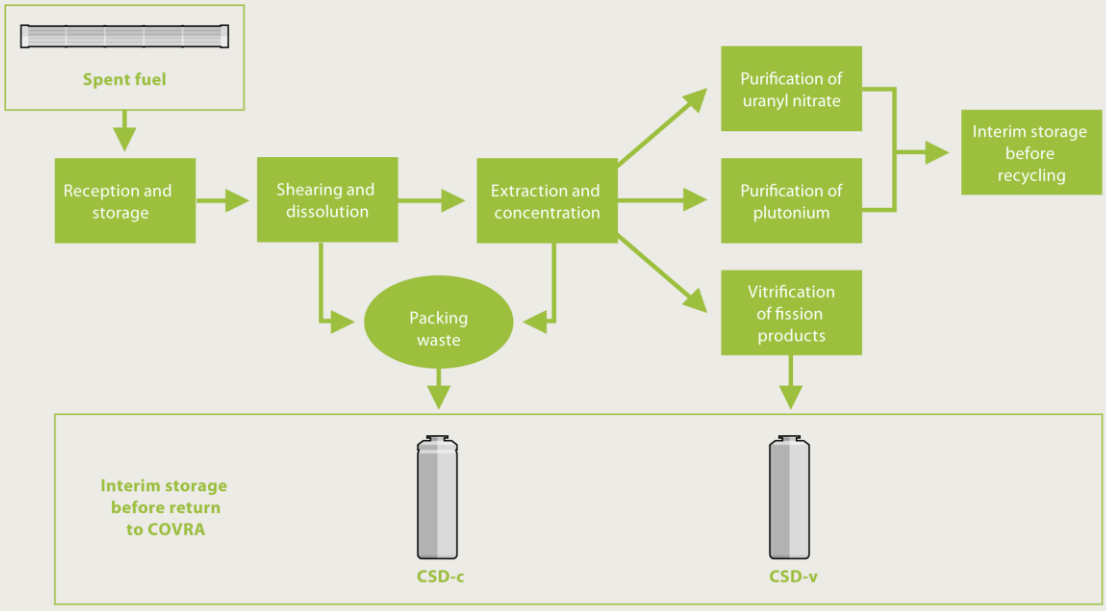

Figure 4-1. Schematics of AREVA PUREX process for reprocessing of spent fuel from the Netherlands [AREVA,2006].

[c] In addition, a small amount of non-heat-generating HLW has been produced, resulting from the vitrification of rinsing fluids (CSD-b). After installation of the new Cold Crucible Melter in 2010, allowing for higher processing temperatures, rinsing liquids are processed along with the other vitrified waste (CSD-v). Considering the small number of canisters, namely 2 CSD-b canisters for the Netherlands, these are not considered in OPERA.

In the reprocessing process, metal parts of the spent fuel assemblies from nuclear power plants are cut, rinsed and dried. A can of about 90 litres is filled with either hulls or end pieces. The hulls are made of zircaloy, and other pieces of the metal parts are usually made of Inconel. End pieces are solid stainless steel sections. Drums with waste arising from reprocessing fuels, such as pumps, stirrers and filters, are primarily made of stainless steel. All drums are compacted with a pressure of 25,000 kN. The resulting pucks are loaded into waste containers with similar outer dimensions as those used for loading vitrified waste. The containers Colis Standard Déchets-compacted (CSD-c) are welded closed and checked, one by one, for contamination. The residual (void) space around the compacted waste is about 20% of the apparent waste volume.

4.1.2. Waste matrix and container

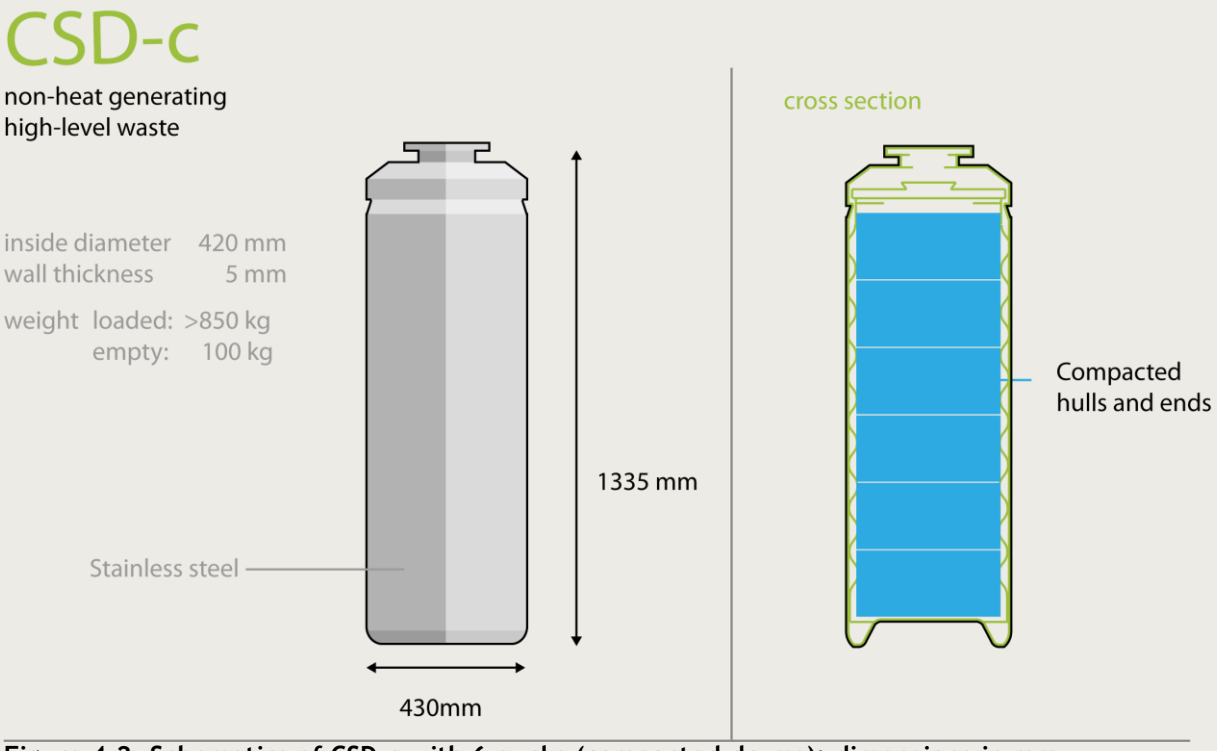

Figure 4-2. Schematics of CSD-c with 6 pucks (compacted drums); dimensions in mm.

The number of pucks varies from canister to canister. On average, it is assumed that there are 7 to 8 pucks in a CSD-c canister. This average comes from a study of a batch of 24 containers that were produced in 2008 and delivered to COVRA in 2011. The canisters in the batch each contained seven or eight compacted pucks.

The guaranteed maximum in height and diameter are 1.345 and 440 mm, respectively. There are no guaranteed amounts of metal parts of spent nuclear fuel and waste arising from reprocessing, but usually a waste container holds 90 wt% metal parts of spent fuel and 10 wt% waste arising from fuel reprocessing. In Appendix 4, a description in weights typical for reprocessing is presented. The guaranteed maximum regarding the weight of the waste container including waste is 850 kg.

The waste matrix contains nuclides of two different sources: contamination from the fuel and activation products. These are distributed differently in the matrix. The radionuclides

resulting from contamination can be assumed to be present on metallic surfaces except caesium and iodine isotopes, as they can diffuse into the cladding [IAEA, 1985:p.5 & Denryoky,1981].

Only activation products can be assumed to be homogeneously distributed in the waste matrices zircaloy and inconel. These radionuclides resulting from neutron capture are indicated in Appendix 1 as 'neutron cap'. The compacted hulls investigated in OPERA are only from a pressurised water reactor. Zircaloy can therefore be further specified as zircaloy-4 [Davis, 1977:p.10]. In Zircaloy-4, nickel has been reduced from 0.03 to 0.08 wt% until impurity levels to reduce hydrogen uptake during operational nuclear reactor conditions [Gras, 2014:p.3].

4.1.3. Radionuclides per waste container and heat output

Appendix 1 gives a detailed inventory of radionuclides per CSD-c canister. The methodology for determining the inventory is explained below.

Guaranteed parameters are only available for fission products and actinides. These radionuclides and series are shown in Table 4-1.

Table 4-1. Maximum guaranteed activity per waste container upon arrival in the Netherlands

ß	Activity [TBq]	α	Activity [TBq]
Pu-241	75	Plutonium	3.3
Cs-137	65	Cu-244	2
Sr-90 + Y-90	115	α-emitters with a half-life of more than 50 years	4.2

Other plutonium isotopes are determined from the typical isotopic composition provided by the waste processing facility. This isotopic composition is presented in Appendix 4.

There is no reason to presume a different distribution of the fission products than the one described for the waste families vitrified waste and spent fuel, except for volatile fission products. In addition, the super-compacted drums may contain waste from further extraction and concentration activities, and volatile fission products, e.g. iodine waste, are captured in the filters. The content of ^{129}I in compacted waste is about 5 times larger than expected from thermal fission of ^{235}U using the maximum ^{137}Cs content in Table 4-1. ^{135}Cs, ^{151}Sm, ^{90}Sr and ^{99}Tc were not reported. These radionuclides are assumed to be present in a fractional abundance similar to ^{137}Cs as a result of thermal fission of ^{235}U.

The guaranteed maximum regarding the heat production after producing these waste containers is 90 W [AREVA, 2001]. Eighty percent of this heat emission is caused by the decay of ^{60}Co. After a storage period of at least 100 years, the heat contribution is assumed to be negligible. A maximum heat production of 18 W can assumed to be characteristic for all waste containers intended for disposal in 2130.

4.2.Other high-level waste

4.2.1. Origin of waste and number of packages

Originally, the national radioactive waste storage facility was located at the Petten site. There still is an amount of legacy high-level waste present in the Waste Storage Facility building. This waste, resulting from four decades of nuclear research, exists of fuel material residues (spent uranium targets and irradiated fuel) and fission and activation products. Legacy and other waste may also be present in other facilities and may be

generated during the dismantling and decommissioning of the different nuclear facilities in the Netherlands. The maximum amount of legacy, decommissioning and other waste is estimated to be 200 packages.

The composition of legacy, decommissioning and other waste is not yet known to COVRA. In Petten, there is a process underway to recover, separate and characterise the legacy waste prior to packaging and shipment to a conditioning facility. It is possible to make a radiological characterisation of the waste generated in the period from 1961 through 1988 that is temporarily stored in Petten. For this study, the composition of the waste is taken to be representative for all legacy, decommissioning and other waste.

4.2.2. Waste matrix and container

For the legacy waste from Petten we consider two types of materials [NRG, 2011:p.17]:

- Neutron-activated ferro and non-ferro metals from dismantled experiments and from claddings and other (non-fissile) parts of irradiated fuel elements; and
- Organic material contaminated with activated metal and volatile fission products such as caesium. This organic material consists of plastic foils, tissues (paper), cloths used to clean hot-cell filters and so-called 'table-cloths' previously made from PVC, to collect the rubble and debris during cutting, sawing and other mechanical operations in the hot-cell.

Apart from the weight content of fissile material, a quantitative inventory of irradiated metals and organic material was not made. It is therefore assumed that the content of irradiated metal can be derived from the fraction of the ^{94}Nb activity of the compacted waste, which then becomes 20.9 kg (see section 4.2.2). The amount of organic material per waste container is assumed to be 78.6 kg.

Figure 4-3. Schematics of ECN canister with (super)compacted pucks; dimensions in mm.

A better estimate of the waste matrices requires information from the irradiation experiments. Technical details, e.g. size and length, can be found in the 'blue book' published in 1993 [Ahlf, 1993] and in the latest public accessible version, from 2005 [McGarry, 2005].

At the moment, apart from CSD-c, only for this waste have arrangements been made for storage as HLW on COVRA's premises. The 200 ECN containers with other non-heat generating HLW are therefore presumed to be of this type. The waste will be processed by Belgoprocess. The cementitious composition used for the conditioning of the waste is not expected to deviate significantly from the composition made by COVRA, as shown in Table 5-4.

4.2.3. Radionuclides per waste container

Appendix 1 gives a detailed inventory of radionuclides per ECN canister. Below, the methodology for determining the inventory is explained.

The maximum content of fissile material for unprocessed waste is characterised as 0.5 wt%. About 100 kg of waste is expected to be in each waste container, and therefore 0.5 kg of (irradiated) fissile material is assumed. Additionally, the waste contains several encapsulated sources with ^{60}Co, ^{137}Cs and ^{192}Ir. An inventory of radionuclides in this HLW was made for the transport to COVRA. The ratio between the ^{94}Nb content and other neutron-activated radionuclides as found for compacted waste has been used to complete the missing radionuclides with the maximum ^{94}Nb activity as reported to COVRA. These values are indicated as 'Nb-94 compacted waste'. Fissile material is taken into account by presuming that this material has HEU characteristics because of the generation period in which HEU fuel elements were used. A weight of 0.5 kg is equal to one irradiated HEU element, and as a first approximation the inventory of one element is used to complete the missing radionuclides. Under this assumption, the radionuclide ^{90}Sr has the largest contribution to the activity of the waste container at the time of disposal.

At the time of disposal in 2130, the reported radionuclide content will contribute about 30% of the total activity of the waste container. This total activity is the summation of the activity of radionuclides for which an activity is determined in Appendix 1.

5. LILW

5.1.(TE)NORM: depleted uranium

5.1.1. Origin of waste and number of packages

The largest waste family by volume is depleted uranium. Depleted uranium is generated by URENCO during uranium enrichment activities. URENCO in Almelo has been enriching uranium since 1973, using the ultracentrifuge technique. The tails that remain after the enrichment process are not considered as waste as long as they are available for re-enrichment. When re-enrichment is considered not economically feasible, the tails are converted to solid uranium oxide (U_3O_8) in France and stored at COVRA.

Depleted uranium is also generated as a result of reprocessing. As the uranium recovered from reprocessing is mostly ^{238}U with about 1% ^{235}U, it needs to be converted and re-enriched for reuse in new fuel. The uranium also contains small amounts of ^{232}U and ^{236}U, which are formed during or following neutron capture in the reactor, and increase with higher burn-up levels. ^{232}U, a decay product of ^{236}Pu, increases with storage time in used fuel, reaching a maximum at about ten years. Being lighter than ^{238}U, ^{232}U and ^{236}U tend to concentrate in the enriched rather than depleted uranium. Because small amounts end up in the depleted uranium, the radiation dose rate[d] is higher than that of U_3O_8 from the enrichment of fresh uranium.

The uranium oxide is stored in standardised 3.5 m^3 containers (DV-70). After the period of interim storage, depleted uranium is conditioned with concrete for disposal. A total of 9,060 containers are to be disposed of in 2130. COVRA has not received any depleted uranium from reprocessing.

5.1.2. Waste matrix and container

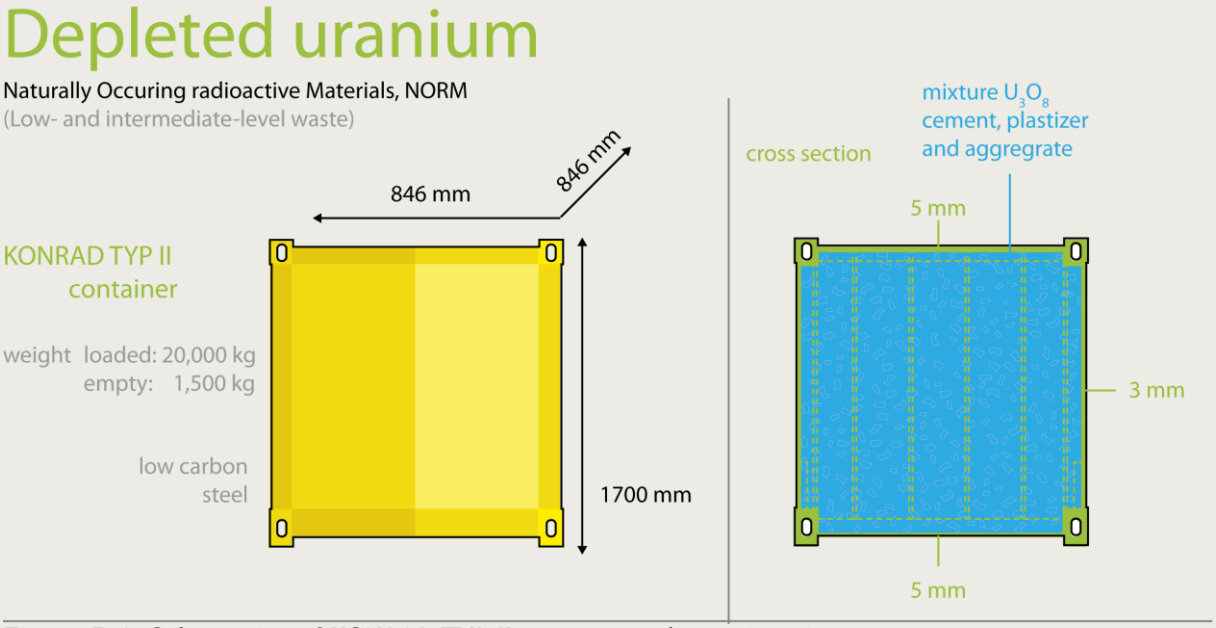

Figure 5-1. Schematics of KONRAD TYP II container; dimensions in mm.

[d] The isotopes ^{232}U and ^{236}U have shorter half-lives than ^{235}U and ^{238}U. One of the daughter products of ^{232}U - ^{208}Tl emits strong gamma radiation.

The proposed conditioning method for the depleted uranium is described in the cementitious materials report [Verhoef, 2014b]. The table below shows the recipe for the conditioning of U_3O_8 with Portland cement.

Table 5-1. Concrete composition for the disposal of depleted uranium

Component	Type		
Cement	CEM I/42.5 N SR O LA (LH)	365	kg m^{-3}
Water	-	175	kg m^{-3}
Plasticiser	TM OFT-II B84/39 CON. 35% (BT-SPL)	3.3	kg m^{-3}
Fine aggregate	U_3O_8: 0-4 mm	2664	kg m^{-3}
Coarse aggregate	Limestone: 2-8 mm	911	kg m^{-3}

N = Usual initial strength, SR = Sulphate resistance, LA = Low Alkali content, (LH = Low Hydration heat)

The uranium oxide - Portland cement mixture is to be contained in Konrad type II containers. The radionuclide inventory per waste container after 130 years is listed in Appendix 1. It is assumed that the Konrad container is made of (low) carbon steel with the specification DIN EN 10025-2 S355J2+N.

5.1.3. Radionuclides per waste container

The average specific activity per kilogram of uranium is shown in Table 5-2 and is based on the depleted uranium in storage at COVRA. No depleted uranium from reprocessing is considered in this report.

Table 5-2. Activity of depleted U_3O_8

Nuclide	Specific activity [MBq/kg U]
U-232	0.15
U-234	15
U-235	0.3
U-236	3.55
U-238	13

5.2. Compacted waste

5.2.1. Origin of waste and number of packages

Annually in the Netherlands, some two hundred organisations produce LILW, varying from nuclear power plants and research establishments to all sorts of industries and hospitals. Most of them generate only small volumes of low- and intermediate-level waste. These small volumes, however, cover a wide range of waste forms: solids, liquids of all natures, slurries, animal carcasses, machines, equipment, sealed sources, etc. LILW also arises from the dismantling of nuclear and other installations. This waste consists mostly of concrete and metals.

Most of the volume of LILW collected by COVRA is solid, compactable waste. Its volume is reduced by compacting the waste-containing drums with a 1,500-tonne super compactor. The compacted drums (pucks) are then transferred to drums with a larger diameter and conditioned with concrete. The super compactor has been in operation since 1992. In this report, it is assumed that the waste conditioned before 1992 has characteristics comparable to the waste conditioned from 1992 onwards.

In this report it is assumed that all LILW collected belongs to the compacted waste family, with the exception of depleted uranium (section 5.1) and processed liquid waste either from molybdenum production (section 5.3) or with spent ion exchangers (section 5.4). A total amount of 140,000 drums are being considered for disposal.

5.2.2. Waste matrix and container

Solid waste is collected by COVRA in drums with a volume of about 90 litres. The thickness of the respective wall, floor and lid is 1 mm. The maximum weight of these drums is 75 kg [COVRA, 2013:p.4]. The technical specifications of COVRA require that the waste collected as solid waste for compaction have a maximum moisture content of 1 vol%. Opening the drums is not part of the routine waste acceptance procedure, but is done occasionally. Based on these inspections no relation could be identified between the reported radionuclides and matrices. The solid waste to be compacted by COVRA contains among others contaminated materials. Conservatively, all radionuclides are assumed to be present on the surfaces. The waste is mainly comprised of cellulosics, plastic and metal. Other types of solid waste are generated by COVRA when processing liquid waste:

- Radionuclides are precipitated from inorganic liquid waste, and the resulting sludge is put in 90-litre drums, dried and perforated before compression.
- The organics in organic liquid waste are thermally decomposed at 900°C. The resulting smoke is sprayed with water. The spraying results in a reduction in temperature and two inorganic liquid wastes: one dense form, a sludge; and a more fluid form similar to inorganic liquid waste. Most radionuclides are present in the sludges. A minor part is collected in filters. Full filters are put in 90-litre drums and compressed. These filters are made of cloth (e.g. cotton, wool), glass fibre, steel and aluminium.

Based on expert judgement, a tentative average composition quantification of compacted waste is listed in Table 5-3.

Table 5-3. A tentative, average composition of the compacted waste family

Matrix	Specification	Fraction of matrix	Volume [%]	Margin [%]
Organic materials	Cellulosics (cloth, paper, tissue) organic sludge etcetera	1	35	20
Metals	Steel	0.75	30	20
	Aluminium	0.25		
Plastics	Halogenated	0.10	25	20
	Non-halogenated	0.90		
Others	Glass, rubber, concrete, inorganic adsorption materials, salts et cetera	1	10	10

The perforated drums are compacted with a pressure of 15,000 kN. The resulting pucks are loaded in 200-litre waste drums. On average between 4-7 pucks are loaded per 200-litre waste drum; about 250 kilograms of waste. The residual space around the compact waste is filled with concrete; a maximum in size of coarse aggregate of 8 mm is used to fill small voids. About 250 kilograms of concrete is casted in these waste drums. The maximum total weight of 200-litre drums is 750 kilograms. A layer of 4-5 cm concrete surrounds the compacted waste. At the bottom and the top of the container are massive concrete discs of at least 10 cm. The concrete contains the radionuclides, provides shielding and gives the necessary mechanical strength to the waste package. Figure 5-2 shows the schematics of a compacted-waste drum.

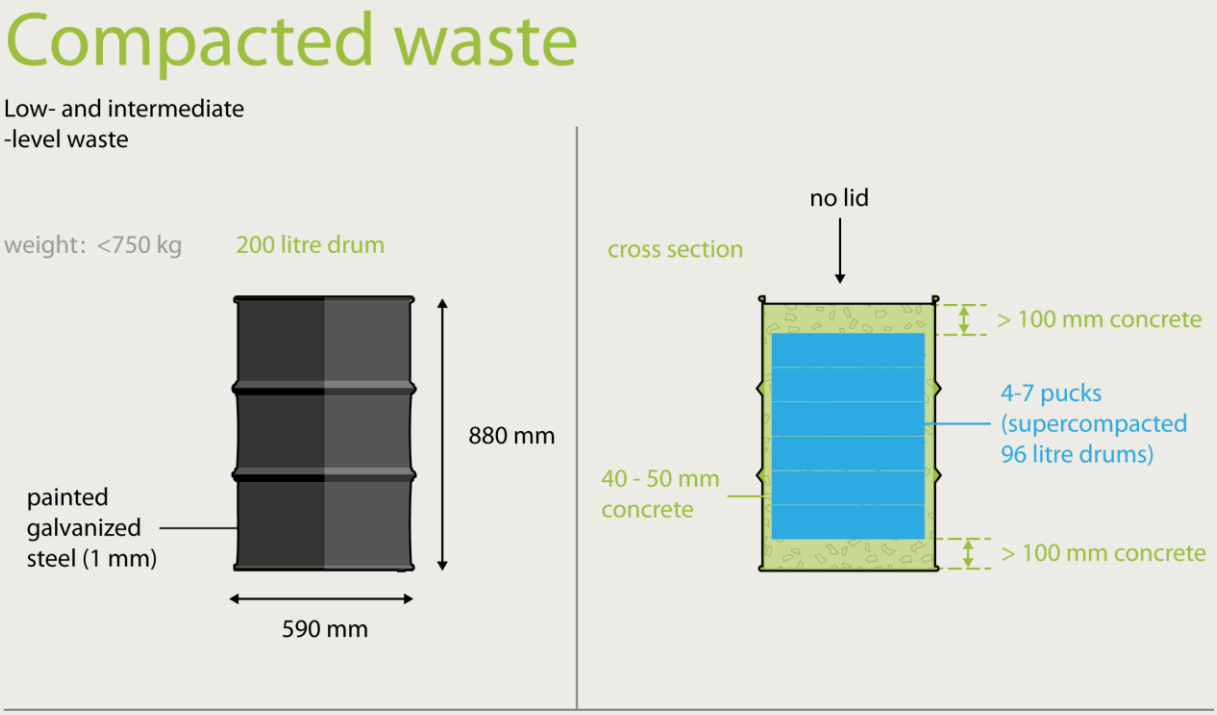

Figure 5-2. Schematics of LILW waste drum with super-compacted pucks; dimensions in mm

The concrete used for conditioning is made of blast furnace slag cement, water, aggregates and plasticiser. The composition of this concrete is published [Verhoef, 2014b] and is shown here for the completion of this waste family.

Table 5-4. Concrete composition for the disposal of compacted waste in 200-litre drums

Component	Type		
Cement	CEM III/B 42.5 N LH SR 0	407-430	kg m^{-3}
Water	-	175-185	kg m^{-3}
Plasticiser	TM OFT-II B84/39 CON. 35% (BT-SPL)	3-5	kg m^{-3}
Fine aggregate	Quartz sand : 0-4 mm	819-972	kg m^{-3}
Coarse aggregate	Quartz gravel : 2-8 mm	891-763	kg m^{-3}

LH = Low Hydration heat; SR = Sulphate resistance

Compared to the moisture content of the concrete, the moisture content in waste pucks can be considered negligible. It is assumed that these 200-litre drums will all be processed by COVRA according to the process described above.

The wall thickness of the 200-litre drums is 1 mm including the bottom. The drum has no lid, to allow inspection of the concrete during storage as shown in Figure 5-2. The drum bottom is made of steel with the technical specifications DX51D – EN10142; the drum walls are made of sheet steel with the specification DC01 – EN10130. The drums walls are galvanised with a 40 micrometres thickness of zinc. The containers are varnished with vinyl copolymer resin.

5.2.3. Radionuclides per waste container

Appendix 1 gives a detailed inventory of radionuclides per drum. Below, the methodology for determining the inventory is explained.

A nuclide vector is derived from the quarterly reporting made by COVRA. The vector is the average of the reported vectors of over 10,000 drums with conditioned waste (also including 400-, 600-, 1,000- and 1,500-litre drums).

As a first approximation, the activity of each nuclide is this summation divided by the number of 200-litre drums, and this average activity is therefore larger than the average activity of these drums from the reported activity by waste generators. Many radionuclides present in the previously described waste families are not reported to be present in this waste. These radionuclides have the comment 'not reported' in Appendix 1. It cannot be assumed that this waste does not contain these radionuclides, but a sound methodology to complete the activity of these radionuclides has not been made. A radionuclide inventory for the 400-, 600- and 1500-litre drums will not be made in OPERA.

5.3.Processed liquid molybdenum waste

5.3.1. Origin of waste and number of packages

Technetium-99m is used in approximately 85% of nuclear medicine diagnostic imaging procedures worldwide. Almost all the ^{99m}Tc for this purpose is obtained from the radioactive decay of ^{99}Mo produced by processing irradiated uranium targets.

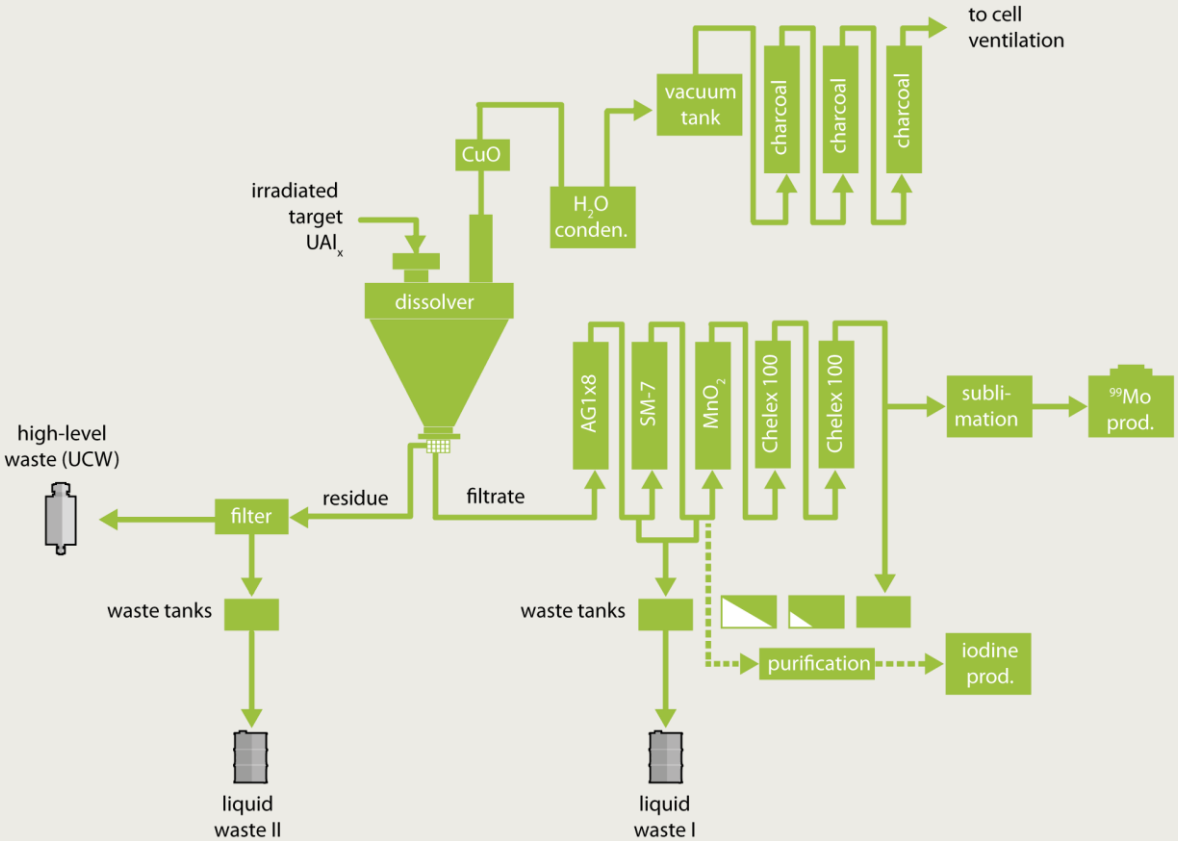

Figure 5-3. Flow diagram of ^{99}Mo production facility in the Netherlands [IAEA, 1998a:p.12][e]

[e] Target supplier has been replaced by HLW in order to avoid confusion with the present status in which material from the spent target is not sent back to the target supplier. Liquid wastes I and II are added for clarity.

The separation process results in several waste streams [Kleef, 2001] and is based on the technology developed by Sameh [IAEA, 2013:p.6]. The spent fuel from the reactor operation and uranium collection filters is discussed in sections 3.2 and 3.3. This section describes the waste family resulting from the conditioning of the two (highly) alkaline waste streams. Note that only waste streams from the molybdenum production route based on Highly Enriched Uranium (93%) irradiated targets is discussed; a molybdenum production route from Low Enriched Uranium targets is currently under development.

In the OPERA disposal concept 12,000 1,000-litre containers are intended for disposal. These concrete containers contain processed molybdenum waste, processed spent ion exchange waste and radioactive sources. For OPERA, processed molybdenum and ion exchange wastes are taken into account because the radiation sources have made up less than 1.1 vol% of this processed waste in the past 5 years.

From the quarterly reporting and application forms for the collection of waste processed at the nuclear power plant in the past 5 years, it can be deduced that the amount of processed molybdenum waste is about $^{2}/_{3}$ of the total number of 1,000-litre containers. For the OPERA disposal concept, it is therefore assumed there are 8,000 1,000-litre containers with processed liquid molybdenum waste; whereas 6,000 containers utilise magnetite as a coarse aggregate and 2,000 containers utilise quartz as aggregate. The remaining 4,000 containers thus contain liquid waste with spent ion exchangers.

5.3.2. Waste matrix and container

The two (highly) alkaline waste streams, liquids I and II, are processed with a mobile cementation plant. A mobile cementation plant is used to mix the waste with mortar in 200-litre drums. Each drum with processed waste contains a sacrificial stirrer. This stirrer is made of steel. This stirrer is shown in Figure 5-4. The radionuclides can be assumed to be uniformly distributed in this waste matrix of cementitious mortar due to the mixing. The content of liquid waste mixed with mortar in a 200-litre drum including the cementitious composition is shown in Table 5-5.

Table 5-5. 180 litres of processed molybdenum waste

Component	Type	Weight [kg]
Cement	CEMIII/B 42.5 LH HS	155
Fine aggregate in suspension	EMSAC 500 S 50 wt% water 50% silica	74
Retarder	CUGLA MMV con.25% BT	1.6
Admixture	Zeolite / Klino type 1 $(Na_2K)_6(Al_6Si_{30}O_{72})_{20}\bullet H_2O$	11
Waste stream	See I and II in Table 5-7	53 for I
		92 for II
Water	Only for waste stream I	39

Zeolite is added to the cementitious composition in order to limit the leaching of caesium. The conditioned waste should provide leach resistance in the hypothetical case that an above-ground storage unit were flooded with sea water.

One 200-litre drum with processed waste is placed in one 1,000-litre concrete container. Figure 5-4 shows the top of the 200-litre drum in this 1,000-litre container before concrete — with the composition same as compacted waste in Table 5-4 — is poured on top of the 200-litre drum.

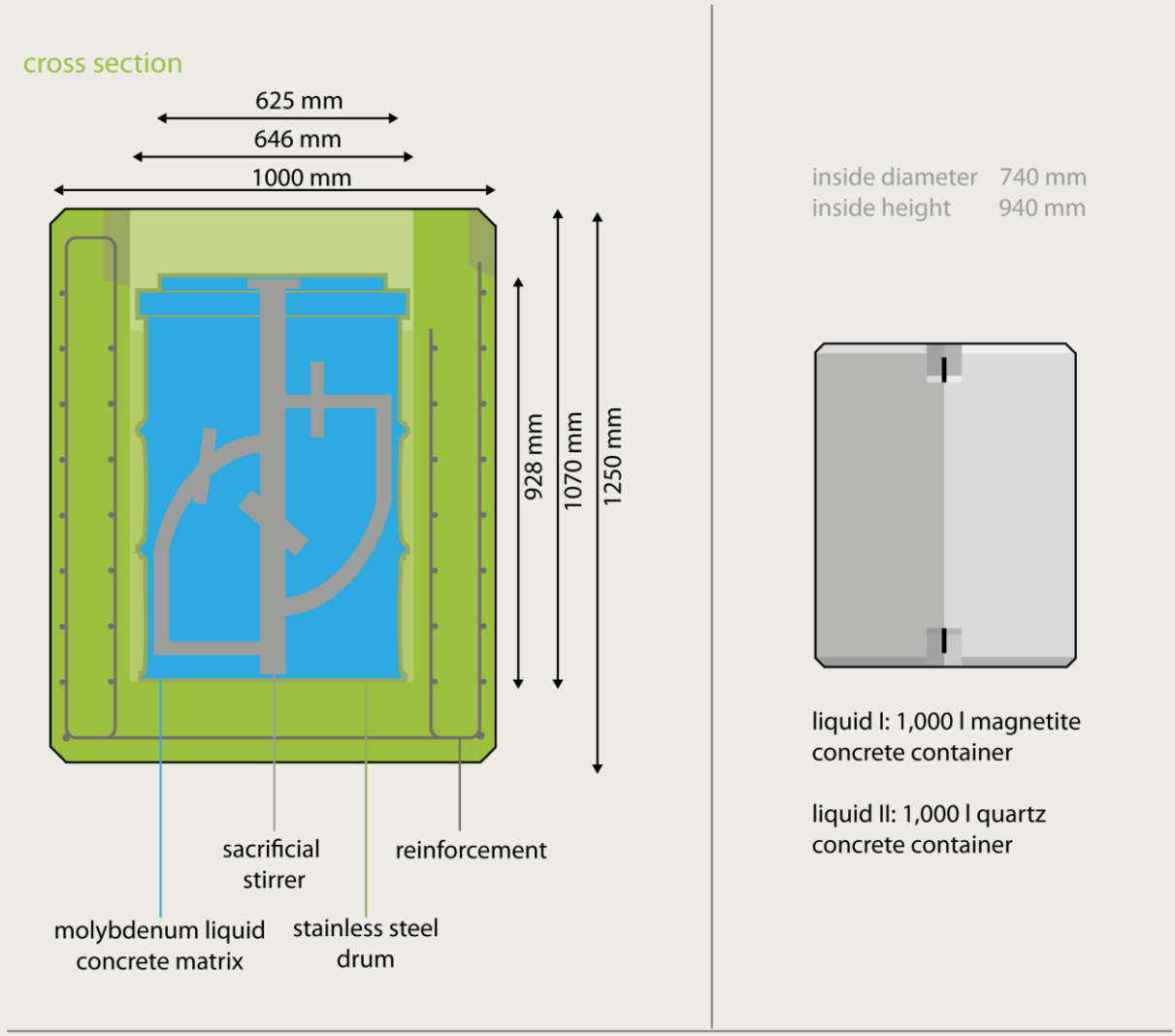

Figure 5-4. 200 litre drum with processed waste in a concrete container; dimensions in mm

The aggregates for the (reinforced) concrete waste container in which the 200-litre drum is placed are based on the required shielding at storage. The aggregates in this container are: magnetite (Fe_3O_4) for processed waste stream I, and quartz for processed waste stream II. A reinforcement lid is put on the 200-litre drums; the 1,000-litre container is filled up with concrete to remove empty volume. The reinforcement grid in the concrete containers is shown in Figure 5-4. The reinforcement steel used is FeB 500 HKN. The (reinforced) concrete wall thickness is 18 centimetres [EPZ, 1985]. The grid at the outside of the container is covered with 30 mm of concrete. Table 5-6 shows the composition of the concrete containers used. The containers are made by Romein Beton in Dodewaard.

Table 5-6. Composition of the reinforced concrete containers per m^3 without waste

Component	Type	1000-litre container for waste stream II [kg/m^3]	1000-litre container for waste stream I [kg/m^3]
Cement	CEM III/A 52.5 N	360	350
	CEM I/52.5 R	-	40
Fine aggregate	Quartz sand : 0-4 mm	820	760
Coarse aggregate	Quartz gravel: 4-16 mm	871	1775
	Magnetite gravel: 2-16 mm		
Superplasticiser	Glenium 51 35%	2.95	1.7
Filler	Foamcarb	200	0
Water		162	154

The strength class of the magnetite container is C35/45; that of the quartz container is B55. With the data in Table 5-7 and Table 5-5, it can be deduced that about $^3/_4$ of the 1,000-litre containers contains magnetite as aggregates and $^1/_4$ contain quartz.

5.3.3. Radionuclides per waste container

Appendix 1 gives a detailed inventory of radionuclides per 1,000-litre container. Below, the methodology for determining the inventory is explained.

The inventory of radionuclides in the waste streams can be more easily understood from Figure 5-3.

Aluminium and fission products soluble in caustic soda, such as alkaline and alkaline earth cations, as well as antimony, iodine, tellurium, tin and molybdenum, are dissolved in the dissolver tank. Uranium mainly remains solid ('residue' in Figure 5-3) and is collected in filters. The solution containing aluminium and fission products passes through a column filled with a strong-base ion exchanger (AG1×8). Molybdenum is quantitatively adsorbed on this ion exchanger, together with a significant proportion of other fission products. Cationic impurities such as caesium, strontium and barium pass through the column, together with hydroxyaluminate, telluric and iodine anions [IAEA, 1998:p.12] and become the first liquid alkaline waste stream ('Liquid waste I' in Figure 5-3). Only strong gamma emitters are reported to COVRA, and therefore only the activity of ^{137}Cs relevant for geological disposal is registered in COVRA's database. The other radionuclides of which the activity is registered are ^{95}Nb, ^{95}Zr, ^{125}Sb, ^{103}Ru, ^{106}Ru and ^{134}Cs; these radionuclides, including their daughters, have a half-life of less than 10 years and are therefore not included in Appendix 1. ^{90}Sr is also contained in this waste [IAEA, 1998:p.12] and is included in Appendix 1, but this radionuclide is not reported to COVRA upon collection of this waste because it is not detected with gamma spectrometry. It is therefore assumed that the radionuclide content of these radionuclides can be obtained from the chain yield in the thermal fission of ^{235}U, as shown in Figure 2-2, and for elements that are soluble in alkaline media.

After several purification processes, the filters with uranium are collected. Uranium precipitates as $Na_2U_2O_7 \bullet 6H_2O$ in the collection filter and becomes high-level waste (see section 3.3). The filters also contain the insoluble fission products ruthenium, zirconium, niobium, and the lanthanides, but also Am and Pu [IAEA, 1998:p.13]. In the precipitation process, HNO_3 and H_2O_2 are added to the precipitation vessel ('waste tanks' in Figure 5-3), and the liquid passing through the collection filter becomes the second alkaline waste stream ('Liquid waste stream II' in Figure 5-3). The activity of ^{103}Ru and ^{106}Ru reported to COVRA is several hundred times greater than that of ^{137}Cs in this waste stream.

The main characteristics of these two waste streams are shown in Table 5-7. The other liquid waste streams are treated by COVRA together with inorganic liquid waste from hospitals and laboratories (see section 5.2).

Table 5-7. Main characteristics of the alkaline waste streams

Component	Unit	Waste stream I	Reference	Waste stream II	Reference
Liquid produced after production 111 TBq ^{99}Mo	litre	13.3	[IAEA,1998]	8.7	[IAEA,1998]
NaOH	gram / litre	244	[IAEA,1998]	102	[IAEA,1998]
Aluminium	gram / litre	Max 38	[COVRA,2010]	-	
$NaNO_3$	gram / litre	-	[IAEA,1998]	29.6	[IAEA,1998]
Uranium	gram / litre	0.050	[IAEA,1998]	4 × waste stream I	Common practice
^{137}Cs	GBq per 44 litres	Max 600	[COVRA,2010] A_2 value	1 / 500 of waste stream I	Common practice
^{106}Ru	GBq per 44 litres			Max 200	[COVRA,2010] A_2 value
Main type of radionuclides		Fission products soluble in alkaline media		Fission products insoluble in alkaline media, lanthanides, uranium, americium and plutonium	

A_2 values are radionuclide-specific maximum activities that a vessel of radioactive waste may contain upon collection by COVRA. These values are published on the website of COVRA in COVRA's technical specifications.

For each waste container with waste stream I, it is assumed that the maximum ^{137}Cs activity upon processing the waste is 723 GBq. This activity is presumed to represent 6.221% fissioned ^{235}U atoms. The other fission products soluble in alkaline media are derived from this activity. For the actinides, lanthanides and fission products insoluble in alkaline media it is assumed that they have the same nuclide vector as determined for the uranium collection filters. Each 200-litre drum contains 2.7 grams of uranium regarding the data in Table 5-7 and Table 5-5. Each waste container with uranium collection filters contains 18,253 grams of uranium (18.253 kilograms). The activity is presumed to be the activity of the radionuclides in the waste container of the uranium collection filters times the ratio in weight of uranium thus activity$_{\text{radionuclide uranium collection filter}}$ × 2.7/18,253.

For waste stream II, it is assumed that the liquid waste contains a fraction of insoluble fission products present in the waste container of the uranium collection filter. Based on the data in Table 5-7 and Table 5-5, each 200-litre drum contains 18.4 grams of uranium. The activity is presumed to be the activity of the radionuclides in the waste container of the uranium collection filter times the ratio in weight of uranium.

5.4.Processed liquid waste with spent ion exchangers

5.4.1. Origin of waste and number of packages

Ion exchangers are very effective at transferring the radioactive content of a large volume of liquid into a small volume of solid [IAEA, 2002]. All nuclear reactors in the Netherlands and nuclear facilities for molybdenum production have these waste streams, but only spent

ion exchangers from Borssele nuclear power plants are conditioned with cement. Smaller quantities of resin arise from the two research reactors. Most of this is in storage awaiting processing. Therefore, only ion exchangers from nuclear power plants are discussed in this section.

From the quarterly reporting and application forms for the collection of waste processed at the nuclear power plant over the past 5 years, it can be deduced that the number of containers with sludge and spent ion exchange resin is one-third of the total number of 1,000-litre containers. Thus 4,000 of the 1,000-litre concrete waste containers are presumed to contain sludge and ion exchange resin conditioned at the nuclear power plant.

5.4.2. Waste matrix and container

In the Borssele nuclear power plant, corrosion products and fission products enter the water used to cool the reactor from the fuel cladding. Ion exchangers are used to clean the water. The Borrssele nuclear power plant is a pressurised water reactor (PWR) where the reactivity is controlled by controlling the concentration of boron in the coolant [IAEA, 1985:p.3]. Ion exchange resins in bead and powdered forms are used in the running PWR. Both types of spent resins are conditioned together with sludge in a cementitious matrix with the same type of mobile cementation unit as is used for processing the molybdenum liquids, by which each waste container with processed waste contains a sacrificial stirrer [EPZ,1985] as shown in Figure 5-4 in this report. The main characteristics of these waste streams are shown in Table 5-8.

Table 5-8. Main characteristics of processed waste streams

Component	Waste stream I	Waste stream II
Ion exchange resin	Powder max 4.3 kg (wet 8.6 kg) per waste container	Powder max 2.7 kg (wet 5.4 kg) per 200-litre drum
		Beads max 16.8 kg (wet 40 kg) per 200-litre drum
Max activity per waste container	^{60}Co : 10 GBq	A_2 values [COVRA,2013] ^{60}Co: max 400 GBq ^{137}Cs: max 600 GBq

Examples of the chemical components of spent powder resin with sludge are shown in Appendix 6. Further details of the ion exchange resins are unknown, and it is therefore presumed that the most common form of synthetic ion exchange resin used is namely polystyrene divinylbenzene in powdered form with diameters of 5-150 micrometres or in beads with diameters of 0.5-2 millimetres [IAEA,2002:p.19-20]. Table 5-9 shows the cementitious composition of processed liquid waste similar to that of molybdenum liquid waste poured into a 200-litre drum.

Table 5-9. Processed waste per 200-litre drum

Component	Type	Weight for waste stream with powder and sludge [kg]	Weight for waste stream with beads, powder and sludge [kg]
Spent ion exchanger	Resin beads	-	16.8 dry resin
			23.2 absorbed water
	Resin powder mixed with sludge	94	45
Cement	CEMIII/B 42.5 LH HS	242	181
Plasticiser	Sikament	-	3.6
Retarder	Sika retarder	2.4	1.8
Admixture	Zeolite / Klino type 1 $(Na_2K)_6(Al_6Si_{30}O_{72})_{20}•H_2O$	12.1	9.1
Admixture	$Ca(OH)_2$	15	12
Water	-	97	63.5

Considering the high activity of the resins, it is assumed that 1,000-litre concrete containers are used with magnetite as aggregate for shielding. These containers from Romein beton were described in the previous section.

5.4.3. Radionuclides per waste container

Appendix 1 gives a detailed inventory of a radionuclides container of processed liquid waste with spent ion exchangers. Below, the methodology for determining the inventory is explained.

Carbon-14 is fixed in the ion exchange resins, in addition to the fission products (e.g. by diffusion of caesium and iodine through cladding), activation products and boron. Similarly to the previously described processed molybdenum waste, the activity of only a few radionuclides are reported to COVRA, namely: ^{137}Cs, ^{134}Cs and ^{60}Co. Scaling factors of ion exchange resins for PWRs from ANDRA in 1999 [IAEA, 2009a:p.72] were used to complete the missing radionuclides. The used scaling factors are shown in Appendix 5.

In the past 5 years only waste containers containing processed waste with an activity higher than 10 GBq have been delivered to COVRA. It is therefore assumed in OPERA that 4,000 waste containers contain bead-based spent ion exchangers. As a first approximation, it is assumed that the waste containers have the maximum ^{60}Co and ^{137}Cs activity upon collection by COVRA. With the radiological characterisation that has now been , only the technical specifications made in agreements made between COVRA and the waste generator will contribute to the waste's assumed activity at the time of disposal in 2130. The total activity is the summation of the activity of radionuclides for which an activity is determined in Appendix 1. The major contribution after 130 years — with the methodology using scaling factors — will be from ^{63}Ni. This radionuclide is not reported and is therefore not included in COVRA's database for all collected LILW.

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Appendix 1: Radionuclides

Table A - 1 Safety relevant radionuclides and their half-lives [Magill,2012]

Radionuclide	half-life [years]	Daugther radionuclide	Half-life different from IAEA Nuclear data section [%]
Ac-226	0	Ra-226	
Ac-227	21.773		0
Ag-108 m	437.7		0
Am-241	432.2		0
Am-242m	141		0
Am-243	7370		0
Ba-133	10.551		0
Be-10	1387000		9
Bi-207	31.55		0
Bi-214	0	Pb-210	
C-14	5730		1
Ca-41	103000		1
Cd-113 m	14.6		3
Cf-249	351		0
Cf-251	898		0
Cf-252	2.6	Cm-248	
Cl-36	301000		0
Cm-241	0	Am-241	
Cm-243	29.1		0
Cm-244	18.1		0
Cm-245	8500		1
Cm-246	4760		0
Cm-247	15600000		0
Cm-248	340000		2
Co-60 #	5.2711		0
Cs-135	2000000		15
Cs-137	30.08		0
Eu-152	13.33		1
Eu-152 m	0		3
H-3	12.312		0
Ho-166m	1200		0
I-129	16100000		2
K-40	1.25E+09		0
Kr-81	230000		0
Kr-85	10.76		0
Mo-93	4000		0
Mo-99	0	Tc-99	
Nb-93 m	16.12		0
Nb-94	20000		1
Ni-59	75000		1
Ni-63	100		1
Np-237	2144000		0
Pa-231	32760		0
Pa-233	0	U-233	
Pa-234	0	U-234	
Pb-202	52500		0

Pb-210	22.3		0
Pb-214	0	Pb-210	
Pd-107	6500000		0
Pm-145	17.7		0
Po-209	102		23
Pu-238	87.74		0
Pu-239	24110		0
Pu-240	6563		0
Pu-241	14.35		0
Pu-242	375000		0
Pu-244	80000000		0
Ra-226	1600		0
Re-186m	200000		0
Se-79	327000		10
Si-32	153		0
Sm-146	68000000		51
Sm-151	96.6		7
Sn-121m	50		12
Sn-126	198000		16
Sr-90	28.9		0
Tc-99	210000		1
Tc-99 m	0	Tc-99	
Th-229	7932		1
Th-230	75400		0
Th-231	0	Pa-231	
Th-234	4.24E-05	U-233	
Ti-44	58.9		0
U-232	68.9		0
U-233	159200		0
U-234	245500		0
U-235	7.04E+08		0
U-236	23420000		0
U-238	4.47E+09		0
U-239	0	Pu-239	
Zr-93	1640000		2

[#] Representative activation products

Table A - 2 Activity per COGEMA waste container after 130 years decay from production

Radionuclide	CSD-V		CSD-C		
	Activity [Bq]	comments/source value	Activity [Bq]	source value	comments
Ac-226					
Ac-227	6.07E+02	10,000 × compacted waste	< 1 Bq	max batch 24 containers	surface
Ag-108 m			1.63E+03	max batch 24 containers	neutron cap
Am-241	1.06E+14	typical [AREVA, 2007]	5.33E+10	max batch 24 containers	surface
Am-242m	1.56E+12	10,000 × compacted waste	1.56E+08	max batch 24 containers	surface
Am-243	2.57E+12	typical [AREVA, 2007]	5.93E+08	max batch 24 containers	surface
Ba-133					
Be-10					
Bi-207					
Bi-214					
C-14		secondary waste stream	1.38E+10	typical [AREVA, 2001]	neutron cap
Ca-41			2.95E+06	max batch 24 containers	neutron cap
Cd-113 m					
Cf-249			3.27E+03	max batch 24 containers	surface
Cf-251			< 1 Bq	max batch 24 containers	surface
Cf-252					
Cl-36		secondary waste stream	6.31E-04	max batch 24 containers	neutron cap
Cm-241					
Cm-243	1.27E+11	10,000 × compacted waste	1.27E+07	max batch 24 containers	surface
Cm-244	2.21E+12	typical [AREVA, 2007]	1.38E+10	max guaranteed [AREVA, 2001]	surface
Cm-245	2.90E+09	max batch 28 containers	1.09E+07	max batch 24 containers	surface
Cm-246	4.77E+10	10,000 × compacted waste	4.77E+06	max batch 24 containers	surface
Cm-247	2.63E+05	10,000 × compacted waste	2.63E+01	max batch 24 containers	surface
Cm-248	1.62E+06	10,000 × compacted waste	1.62E+02	max batch 24 containers	surface
Co-60					
Cs-135	3.01E+10	max batch 28 containers	1.04E+09	max Cs-137, 6.62%	surface
Cs-137	3.30E+14	max guaranteed [AREVA, 2007]	3.25E+12	max guaranteed [AREVA, 2001]	surface
Eu-152		not reported	3.87E+06	max batch 24 containers	neutron cap
Eu-152 m					
H-3		secondary waste stream	9.96E+09	max batch 24 containers	surface
Ho-166m					
I-129		secondary waste stream	5.30E+07	max batch 24 containers	surface
K-40					
Kr-81		secondary waste stream			
Kr-85		secondary waste stream	< 1 Bq	max batch 24 containers	surface
Mo-93			5.79E+09	max batch 24 containers	neutron cap
Mo-99					
Nb-93 m					
Nb-94			5.55E+10	max batch 24 containers	neutron cap

Ni-59			3.59E+11	max batch 24 containers	neutron cap
Ni-63			1.76E+13	max batch 24 containers	neutron cap
Np-237	4.80E+10	typical [AREVA, 2007]	7.80E+06	max batch 24 containers	surface
Pa-231		not reported			
Pa-233					
Pa-234					
Pb-202					
Pb-210					
Pb-214					
Pd-107	6.78E+09	max batch 28 containers	6.74E+06	max batch 24 containers	surface
Pm-145		not reported		not reported	
Po-209					
Pu-238	4.78E+11	max weight and isotopic	1.20E+12	max act Pu-241 and isotopic	surface
Pu-239	1.44E+11	max weight and isotopic	2.14E+11	max act Pu-241 and isotopic	surface
Pu-240	2.31E+11	max weight and isotopic	3.68E+11	max act Pu-241 and isotopic	surface
Pu-241	7.35E+10	max weight and isotopic	1.41E+11	max guaranteed [AREVA,2001]	surface
Pu-242	1.01E+09	max weight and isotopic	2.09E+09	max act Pu-241 and isotopic	surface
Pu-244	4.91E+05	10,000 × compacted waste	4.91E+01	max batch 24 containers	surface
Ra-226	1.03E+02	10,000 × compacted waste	1.03E-02	max batch 24 containers	surface
Re-186m					
Se-79	2.01E+10	max batch 28 containers	5.50E+07	typical [AREVA,2001]	surface
Si-32					
Sm-146					
Sm-151	5.47E+13	max Cs-137, 0.4204%	5.38E+11	max Cs-137, 0.4204%	surface
Sn-121m					
Sn-126	3.80E+10	max batch 28 containers	8.83E+07	max batch 24 containers	surface
Sr-90	2.05E+14	max guaranteed	2.76E+12	max Cs-137,5.73%	surface
Tc-99	1.25E+12	max batch 28 containers	9.17E+09	max Cs-137,6.132%	surface
Tc-99 m					
Th-229	1.17E+04	10,000 × compacted waste	1.17E+00	max batch 24 containers	surface
Th-230	1.68E+05	10,000 × compacted waste	1.68E+01	max batch 24 containers	surface
Th-231					
Th-234					
Ti-44					
U-232	2.95E+09	10,000 × compacted waste	2.95E+05	max batch 24 containers	surface
U-233	3.20E+06	10,000 × compacted waste	3.20E+02	max batch 24 containers	surface
U-234	4.77E+08	max weight and isotopic	3.06E+06	max batch 24 containers	surface
U-235	2.88E+06	max weight and isotopic	1.25E+06	max batch 24 containers	surface
U-236	4.21E+07	max weight and isotopic	1.21E+07	max batch 24 containers	surface
U-238	5.53E+07	max weight and isotopic	1.88E+07	max batch 24 containers	surface
U-239					
Zr-93	1.05E+11	max batch 28 containers	8.91E+09	max batch 24 containers	mainly neutron cap

Table A - 3 Radionuclide content from batch and derived from ^{137}Cs presuming no neutron activation of fission products

	^{99}Tc	^{135}Cs	^{79}Se	^{93}Zr	^{107}Pd	^{126}Sn
Max batch 28 containers	1.25E+12	3.01E+10	2.01E+10	1.05E+11	6.78E+09	3.80E+10
From ^{137}Cs activity	9.32E+11	1.06E+11	4.75E+09	1.25E+11	6.84E+08	9.57E+09

Table A - 4 Activity per ECN container - filled with 33 spent research reactor fuel elements - after 130 years decay from neutron irradiation

Radionuclide	Spent High Enriched Uranium Fuel		Spent Low Enriched Uranium Fuel	
	Activity [Bq]	comments/source value	Activity [Bq]	comments/source value
Ac-226				
Ac-227				
Ag-108 m				
Am-241	1.03E+12	[Dodd,2000]	1.38E+13	[NRG,2012]
Am-242m				
Am-243	2.18E+09	[Dodd,2000]	3.54E+10	EOI [NRG,2012]
Ba-133				
Be-10				
Bi-207				
Bi-214				
C-14	6.57E+07	[Dodd,2000]	6.57E+07	same as HEU
Ca-41				
Cd-113 m				not calculated
Cf-249				
Cf-251				
Cf-252				
Cl-36	0.00E+00	[Dodd,2000]		same as HEU
Cm-241				
Cm-243	1.27E+09	100 × compacted waste	1.27E+10	1,000 × compacted waste
Cm-244	1.94E+09	one -tenth LEU	1.94E+10	[NRG,2012]
Cm-245	1.48E+07	[Dodd,2000]	1.48E+08	ten times HEU
Cm-246	1.85E+06	[Dodd,2000]	1.85E+07	ten times HEU
Cm-247	2.63E+03	100 × compacted waste	2.63E+04	1,000 × compacted waste
Cm-248	1.62E+04	100 × compacted waste	1.62E+05	1,000 compacted waste
Co-60				
Cs-135	1.35E+09	[Dodd,2000]	2.10E+10	Cs-137 (one month), 6.62%
Cs-137	4.39E+13	[Dodd,2000]	6.53E+13	[NRG,2012]
Eu-152				
Eu-152 m				
H-3				
Ho-166m				
I-129	2.04E+08	[Dodd,2000]	2.78E+08	Cs-137 (one month), 0.706%
K-40				
Kr-81				
Kr-85	1.19E+11	Cs-137 (one month), 1.310%	3.63E+10	[NRG,2012]
Mo-93				
Mo-99				
Nb-93 m				
Nb-94	1.34E+06	[Dodd,2000]	1.34E+06	same as HEU
Ni-59				
Ni-63	7.06E+03	[Dodd,2000]	7.06E+03	same as HEU
Np-237	2.76E+09	[Dodd,2000]	4.24E+09	EOI [NRG,2012]
Pa-231	1.41E+06	[Dodd,2000]	1.41E+06	same as HEU
Pa-233				
Pa-234				
Pb-202				
Pb-210				
Pb-214				
Pd-107	1.04E+08	[Dodd,2000]	1.36E+08	Cs-137 (one month), 0.1393%
Pm-145		not calculated		not calculated

Po-209				
Pu-238	4.95E+12	[Dodd,2000]	8.25E+12	[NRG,2012]
Pu-239	9.11E+10	[Dodd,2000]	1.56E+12	[NRG,2012]
Pu-240	6.83E+10	[Dodd,2000]	1.47E+12	[NRG,2012]
Pu-241	7.00E+10	[Dodd,2000]	9.44E+11	[NRG,2012]
Pu-242	3.37E+08	[Dodd,2000]	5.29E+09	EOI [NRG,2012]
Pu-244	4.91E+06	100,000 × compacted waste	4.91E+07	1,000,000 × compacted waste
Ra-226	8.78E+05	[Dodd,2000]	8.78E+05	same as HEU
Re-186m				
Se-79	2.97E+09	[Dodd,2000]	9.46E+08	Cs-137 (one month), 0.0487%
Si-32				not calculated
Sm-146				
Sm-151	6.73E+11	[Dodd,2000]	1.09E+13	Cs-137 (one month), 0.4204%
Sn-121m				
Sn-126	2.57E+09	[Dodd,2000]	1.90E+11	Cs-137 (one month), 0.0594%
Sr-90	3.80E+13	[Dodd,2000]	5.31E+13	[NRG,2012]
Tc-99	1.11E+11	[Dodd,2000]	1.11E+11	same as HEU
Tc-99 m				
Th-229	1.31E+04	[Dodd,2000]	1.31E+04	same as HEU
Th-230	3.22E+07	[Dodd,2000]	3.22E+07	same as HEU
Th-231				
Th-234				
Ti-44				
U-232	2.95E+10	100,000 × compacted waste	2.95E+11	1,000,000 × compacted waste
U-233	1.83E+06	[Dodd,2000]	1.83E+06	same as HEU
U-234	2.81E+10	[Dodd,2000]	4.29E+10	[NRG,2012]
U-235	4.72E+08	[Dodd,2000]	5.32E+08	EOI [NRG,2012]
U-236	3.80E+09	[Dodd,2000]	6.79E+09	EOI [NRG,2012]
U-238	1.01E+07	[Dodd,2000]	9.39E+08	EOI [NRG,2012]
U-239				
Zr-93	1.67E+10	[Dodd,2000]	1.67E+10	same as HEU

Table A - 5 Activity per ECN container with uranium collection filter and legacy waste after 130 years decay from cooling

Radionuclide	Uranium collection filters		Legacy waste	
	Activity [Bq]	comments / source value	Activity [Bq]	comments / source value
Ac-226				
Ac-227				
Ag-108 m			6.44E+01	Nb-94 compacted waste
Am-241	8.52E+08	[NRG,2009]+Pu-241	2.44E+09	[NRG,2011]
Am-242m				
Am-243	2.18E+09	same as HEU	6.61E+07	1 HEU element
Ba-133				
Be-10				
Bi-207				
Bi-214				
C-14		secondary waste stream	5.44E+08	Nb-94 compacted waste
Ca-41			1.17E+05	Nb-94 compacted waste
Cd-113 m				
Cf-249				
Cf-251				
Cf-252				
Cl-36		secondary waste stream	< 1 Bq	Nb-94 compacted waste
Cm-241				
Cm-243	1.27E+09	same as HEU	3.84E+09	1 HEU element
Cm-244	1.94E+09	same as HEU	5.88E+07	1 HEU element
Cm-245	1.48E+07	same as HEU	4.47E+05	1 HEU element
Cm-246	1.85E+06	same as HEU	5.61E+04	1 HEU element
Cm-247	2.63E+03	same as HEU	7.97E+03	1 HEU element
Cm-248	1.62E+04	same as HEU	4.92E+04	1 HEU element
Co-60				
Cs-135		LILW: molybdenum waste I	4.09E+07	1 HEU element
Cs-137		LILW: molybdenum waste I	5.80E+11	[NRG,2011]
Eu-152	8.72E+04	[NRG,2009]	1.53E+05	Nb-94 compacted waste
Eu-152 m				
H-3		secondary waste stream	4.52E+10	[NRG,2011]
Ho-166m	3.56E+04	[NRG,2009]		
I-129	2.06E+00	from Te-129m filter and LILW: molybdenum waste I	6.18E+06	1 HEU element
K-40				
Kr-81		secondary waste stream		
Kr-85		secondary waste stream	3.62E+09	1 HEU element
Mo-93			2.28E+08	Nb-94 compacted waste
Mo-99				
Nb-93 m				
Nb-94		not calculated	2.19E+09	[NRG,2011]
Ni-59			1.42E+10	Nb-94 compacted waste
Ni-63	1.05E+06	[NRG,2009]	1.10E+11	[NRG,2011]
Np-237	1.47E+07	[NRG,2009]	8.37E+07	1 HEU element
Pa-231	5.54E+04	[NRG,2009]		
Pa-233				
Pa-234				
Pb-202				
Pb-210				
Pb-214				
Pd-107	3.10E+07	[NRG,2009]	3.16E+06	1 HEU element
Pm-145	4.59E+01	[NRG,2009]		

Po-209				
Pu-238	2.43E+08	[NRG,2009]	1.50E+11	1 HEU element
Pu-239	2.08E+10	[NRG,2009]	2.76E+09	1 HEU element
Pu-240	1.12E+09	[NRG,2009]	2.07E+09	1 HEU element
Pu-241	4.90E+07	[NRG,2009]	2.12E+09	1 HEU element
Pu-242	1.03E+04	[NRG,2009]	1.02E+07	1 HEU element
Pu-244			1.49E+04	1 HEU element
Ra-226			2.66E+04	1 HEU element
Re-186m				
Se-79	4.65E+07	[NRG,2009]	9.00E+07	1 HEU element
Si-32				
Sm-146	1.11E+00	[NRG,2009] from Pm-146		
Sm-151	7.49E+11	[NRG,2009]	2.04E+10	1 HEU element
Sn-121m				
Sn-126	3.03E+08	[NRG,2009]	7.79E+07	1 HEU element
Sr-90	4.27E+12	[NRG,2009]	1.15E+12	1 HEU element
Tc-99	1.48E+10	[NRG,2009]	8.40E+10	[NRG,2011]
Tc-99 m				
Th-229	1.31E+04	same as HEU	3.96E+02	1 HEU element
Th-230	3.22E+07	same as HEU	9.77E+05	1 HEU element
Th-231				
Th-234				
Ti-44				
U-232	2.84E+03	[NRG,2009] + Pu-236	8.93E+07	1 HEU element
U-233	1.87E+04	[NRG,2009]	5.54E+04	1 HEU element
U-234	9.62E+06	[NRG,2009]	8.53E+08	1 HEU element
U-235	1.27E+09	[NRG,2009]	1.43E+07	1 HEU element
U-236	4.59E+08	[NRG,2009]	1.15E+08	1 HEU element
U-238	2.65E+07	[NRG,2009]	3.06E+05	1 HEU element
U-239				
Zr-93	1.95E+09	[NRG,2009]	3.52E+08	Nb-94 compacted waste

Table A - 6 Activity per 200 litre drum processed molybdenum waste (200 litre drums contained in 1000 litre concrete containers) 130 years after collecting the waste

Radionuclide	Molybdenum waste stream I		Molybdenum waste stream II	
	Activity [Bq]	comments value	Activity [Bq]	comments value
Ac-226				
Ac-227				
Ag-108 m				
Am-241	1.24E+05	fraction uranium collection filters	8.59E+05	fraction uranium collection filters
Am-242m				
Am-243	3.17E+05	fraction uranium collection filters	2.20E+06	fraction uranium collection filters
Ba-133				
Be-10				
Bi-207				
Bi-214				
C-14				
Ca-41				
Cd-113 m				
Cf-249				
Cf-251				
Cf-252				
Cl-36				
Cm-241				
Cm-243	1.84E+05	fraction uranium collection filters	1.28E+06	fraction uranium collection filters
Cm-244	2.82E+05	fraction uranium collection filters	1.96E+06	fraction uranium collection filters
Cm-245	2.14E+03	fraction uranium collection filters	1.49E+04	fraction uranium collection filters
Cm-246	2.69E+02	fraction uranium collection filters	1.87E+03	fraction uranium collection filters
Cm-247	3.82E-01	fraction uranium collection filters	2.65E+00	fraction uranium collection filters
Cm-248	2.36E+00	fraction uranium collection filters	1.64E+01	fraction uranium collection filters
Co-60				
Cs-135	1.16E+07	Cs-137 (upon collection), 6.62%		
Cs-137	3.62E+10	upon collection max A2		
Eu-152	1.27E+01	fraction uranium collection filters	8.79E+01	fraction uranium collection filters
Eu-152 m				
H-3				
Ho-166m				
I-129	1.53E+05	Cs-137 (upon collection), 0.706%		
K-40				
Kr-81				
Kr-85				
Mo-93				
Mo-99				
Nb-93 m				
Nb-94		not calculated		not calculated
Ni-59				
Ni-63			1.06E+03	fraction uranium collection filters
Np-237	2.13E+03	fraction uranium collection filters	1.48E+04	fraction uranium collection filters
Pa-231	8.04E+00	fraction uranium collection filters	5.58E+01	fraction uranium collection filters
Pa-233				
Pa-234				

Pb-202				
Pb-210				
Pb-214				
Pd-107	4.51E+03	fraction uranium collection filters	3.13E+04	fraction uranium collection filters
Pm-145			4.63E-02	fraction uranium collection filters
Po-209				
Pu-238	3.53E+04	fraction uranium collection filters	2.45E+05	fraction uranium collection filters
Pu-239	3.01E+06	fraction uranium collection filters	2.09E+07	fraction uranium collection filters
Pu-240	1.62E+05	fraction uranium collection filters	1.13E+06	fraction uranium collection filters
Pu-241	7.11E+03	fraction uranium collection filters	4.94E+04	fraction uranium collection filters
Pu-242	1.49E+00	fraction uranium collection filters	1.04E+01	fraction uranium collection filters
Pu-244				
Ra-226				
Re-186m				
Se-79	5.20E+05	Cs-137 (upon collection), 0.0487%	4.69E+04	fraction uranium collection filters
Si-32				
Sm-146				
Sm-151	1.09E+08	fraction uranium collection filters	7.55E+08	fraction uranium collection filters
Sn-121m				
Sn-126	1.05E+06	Cs-137 (upon collection), 0.0594	3.05E+05	fraction uranium collection filters
Sr-90	3.07E+10	Cs-137 (upon delivery), 5.73%	4.30E+09	fraction uranium collection filters
Tc-99		filtered		
Tc-99 m				
Th-229				
Th-230				
Th-231				
Th-234				
Ti-44				
U-232				
U-233	2.71E+00	fraction uranium collection filters	1.88E+01	fraction uranium collection filters
U-234	1.40E+03	fraction uranium collection filters	9.70E+03	fraction uranium collection filters
U-235	1.84E+05	fraction uranium collection filters	1.28E+06	fraction uranium collection filters
U-236	6.67E+04	fraction uranium collection filters	4.63E+05	fraction uranium collection filters
U-238	3.84E+03	fraction uranium collection filters	2.67E+04	fraction uranium collection filters
U-239				
Zr-93	2.83E+05	fraction uranium collection filters	1.97E+06	fraction uranium collection filters

Table A - 7 Activity per waste container for (processed) depleted uranium, spent ion exchanger and compacted waste 130 years after collecting the waste.

Radionuclide	Depleted uranium	Spent ion exchanger		Compacted 90 litre drums with waste	
	Activity [Bq]	Activity [Bq]	comments value	Activity [Bq]	comments / source value
Ac-226					
Ac-227				1.31E+02	[COVRA,2012]
Ag-108 m		3.26E+08	[IAEA,2002]	1.50E+03	[COVRA,2012]
Am-241				1.50E+08	[COVRA,2012]
Am-242m					
Am-243				3.65E+02	[COVRA,2012]
Ba-133				1.82E+01	[COVRA,2012]
Be-10		8.00E+04	[IAEA,2002]	1.79E+04	[COVRA,2012]
Bi-207				3.41E+02	[COVRA,2012]
Bi-214					
C-14		7.09E+09	[IAEA,2002]	1.98E+07	[COVRA,2012]
Ca-41		2.00E+06	[IAEA,2002]		not reported
Cd-113 m					reported activity upon collection after 30 years < 1 MBq
Cf-249				1.32E+02	[COVRA,2012]
Cf-251					
Cf-252					
Cl-36		4.00E+06	[IAEA,2002]	4.53E+04	[COVRA,2012]
Cm-241					not reported
Cm-243					not reported
Cm-244				1.41E+04	[COVRA,2012]
Cm-245					not reported
Cm-246					not reported
Cm-247					not reported
Cm-248				1.83E+00	not reported, deduced from Cf-252
Co-60		1.51E+04	A2 value upon collection		
Cs-135		3.00E+06	[IAEA,2002]		not reported
Cs-137		3.00E+10	A2 value upon collection	9.16E+08	[COVRA,2012]
Eu-152				2.48E+02	[COVRA,2012]
Eu-152 m					
H-3		2.04E+04	from conditioning and waste characteristics	1.08E+07	[COVRA,2012]
Ho-166m					not reported
I-129		6.00E+05	[IAEA,2002]	2.22E+03	[COVRA,2012]
K-40				4.22E+04	[COVRA,2012]
Kr-81				2.84E+01	[COVRA,2012]
Kr-85				1.36E+04	[COVRA,2012]
Mo-93		3.91E+05	[IAEA,2002]		reported activity upon collection after 30 years < 1 MBq
Mo-99					
Nb-93 m				1.06E-01	[COVRA,2012]
Nb-94		4.78E+07	[IAEA,2002]	1.53E+03	[COVRA,2012]
Ni-59		4.39E+08	[IAEA,2002]	1.18E+04	[COVRA,2012]
Ni-63		2.27E+11	[IAEA,2002]	3.70E+08	[COVRA,2012]
Np-237				2.84E+02	[COVRA,2012]
Pa-231				3.97E+02	[COVRA,2012]
Pa-233					
Pa-234					
Pb-202					reported activity upon collection after 30

					years < 1 MBq
Pb-210				7.71E+03	[COVRA,2012]
Pb-214					
Pd-107		6.00E+04	[IAEA,2002]		not reported
Pm-145					not reported
Po-209					reported activity upon collection after 30 years < 1 MBq
Pu-238				6.91E+07	[COVRA,2012]
Pu-239				1.26E+07	[COVRA,2012]
Pu-240				8.06E+05	[COVRA,2012]
Pu-241				3.68E+03	[COVRA,2012]
Pu-242				1.99E+06	[COVRA,2012]
Pu-244					reported activity upon collection after 30 years < 1 MBq
Ra-226				6.69E+06	[COVRA,2012]
Re-186m				2.37E+04	[COVRA,2012]
Se-79		2.40E+06	[IAEA,2002]		not reported
Si-32					reported activity upon collection after 30 years < 1 MBq
Sm-146					reported activity upon collection after 30 years < 1 MBq
Sm-151				1.65E+04	[COVRA,2012]
Sn-121m		1.98E+06	[IAEA,2002]	5.30E+02	[COVRA,2012]
Sn-126		5.40E+06	[IAEA,2002]		not reported
Sr-90		6.11E+07	[IAEA,2002]	1.94E+07	[COVRA,2012]
Tc-99		6.00E+06	[IAEA,2002]	8.88E+05	[COVRA,2012]
Tc-99 m					
Th-229				2.25E+02	[COVRA,2012]
Th-230				2.84E+01	[COVRA,2012]
Th-231					
Th-234					
Ti-44					reported activity upon collection after 30 years < 1 MBq
U-232	4.68E+08			1.69E+02	[COVRA,2012]
U-233	0.00E+00			2.04E+03	[COVRA,2012]
U-234	1.73E+11			5.15E+05	[COVRA,2012]
U-235	3.46E+09			1.35E+06	[COVRA,2012]
U-236	4.10E+10			8.52E+01	[COVRA,2012]
U-238	1.50E+11			3.83E+07	[COVRA,2012]
U-239					reported activity upon collection after 30 years < 1 MBq
Zr-93		2.00E+05	[IAEA,2002]		not reported

Appendix 2: Further details vitrified waste

Table A - 8 Average isotopic composition from a batch of 28 waste containers stored since 2012

Uranium		Plutonium	
A	% U	A	% Pu
234	0.017 ± 0.001	238	1.913 ± 0.378
235	0.801 ± 0.070	239	57.151 ± 2.273
236	0.391 ± 0.048	240	25.350 ± 0.617
238	98.792 ± 0.069	241	9.307 ± 0.502
		242	6.278 ± 0.908

Table A - 9 Composition of glass from a batch of 28 canisters produced in 2004 from BNFL

Compound	Min [wt%]	Max [wt%]	Assumption inactive oxide composition[#]	
			Min [wt%]	Max [wt%]
SiO_2	44.76	46.22		
B_2O_3	16.33	16.93		
Al_2O_3	1.54	2.11	1.50	2.40
Na_2O	7.99	8.25		
Fe_2O_3			1.44	2.05
NiO			1.38	1.67
Cr_2O_3			1.38	1.68
P_2O_5				
Li_2O	3.87	4.13		
MgO	1.43	1.57	1.42	1.91
Gd_20_5			1.53	2.56
RuO_2+Rh+Pd				
oxides(fission products)	12.50	13.65		
oxides (actinides)	0.42	0.68		

[#]The content of SiO_2, B_2O_3, Al_2O_3, Na_2O, Li_2O, MgO, inactive oxides and oxides with fission products and actinides is provided for each batch. The fission products and actinides have been generated before 2004 e.g. in case of GKN between 1969 and 1997. The incorporation rate of waste oxides and inactive oxides are determined. An example of the elemental composition is provided in the manual [BNFL, 2001]. From this example, assumptions have been made for the inactive oxide composition.

Appendix 3: Further details spent research reactor fuel

Table A - 10 Selection of actinides for research reactor fuel and control elements [Dodd, 2000 & NRG, 2012]

Nuclide [gram]	*Fuel element*				*Control Element*			
	HEU		*LEU*		*HEU*		*LEU*	
	BOI	*EOI*	*BOI*	*EOI*	*BOI*	*EOI*	*BOI*	*EOI*
U-234	5.1	3.3	7.3	5.0	2.9	2.0	5.9	4.6
U-235	450	177	561.0	201.6	310	119	452.3	223.8
U-236	0.8	50	29.1	85.9	0.4	33	23.5	59.4
U-238	27.9	24.4	2353.2	2287.4	20.1	18	1897.3	1862.5
Np-237		3.3		4.93	2			2.60
Pu-238		0.70		1.09	0.33			0.44
Pu-239		1.2		20.71	0.74			13.79
Pu-240		0.25		5.35	0.16			2.87
Pu-241		0.28		3.81	0.15			1.87
Pu-242		0.071		1.1	0.038			0.39
Am-241		0.01		0.133	0.0045			0.058
Am-243		0.0091		0.145	0.0039			0.040

Appendix 4: Further details compacted waste

Table A - 11 Typical composition in weight waste matrix

Metal part	Metal	BOI per metric ton uranium [kg]	EOI per container [kg]
Hulls	Zircaloy 0.18-0.24 wt% Fe < 20 ppm Co	261	393
Other fuel metallic parts	Inconel 12-14 wt% Fe <1200 ppm Co	13	19
Waste arising from reprocessing			116

Table A - 12 Typical isotopic composition of plutonium [AREVA, 2001]

A	% Pu
238	3
239	53
240	25
241	11
242	8

Appendix 5: Further details spent ion exchange resins

Table A - 13 Validated scaling factors for PWRs from ANDRA [IAEA, 2009a:p.72]

Radionuclide	Key nuclide	Scaling Factor 1999
Be-10	Co-60	2×10^{-7}
C-14	Co-60	1.8×10^{-2}
Cl-36	Co-60	1×10^{-5}
Ca-41	Co-60	5×10^{-6}
Ni-59	Co-60	1.1×10^{-3}
Ni-63	Co-60	1.4
Se-79	Cs-137	4×10^{-6}
Sr-90	Cs-137	2.3×10^{-3}
Mo-93	Co-60	1×10^{-6}
Zr-93	Co-60	5×10^{-7}
Nb-94	Co-60	1.2×10^{-4}
Tc-99	Cs-137	1×10^{-5}
Pd-107	Cs-137	1×10^{-7}
Ag-108m	Co-60	1×10^{-3}
Sn-121m	Cs-137	2×10^{-5}
Sn-126	Cs-137	9×10^{-6}
I-129	Cs-137	1×10^{-6}
Cs-135	Cs-137	5×10^{-6}
Sm-151	Cs-137	7×10^{-4}

Table A - 14 Examples of spent powder ion exchange resin with sludge

Parameter	Unit	Example 1	Example 2
Specific density	kg m^{-3}	1.23	1.18
Undissolved solid matter: spent powder ion exchange resin	g l^{-1}	77.6	32
Dissolved solid matter: sludge	g l^{-1}	374.4	237
pH	g l^{-1}	11.4	11.2
Fluorine	g l^{-1}	0.244	0.126
Chlorine	g l^{-1}	3.92	1.505
Sulphate	g l^{-1}	29.1	21.1
Sodium	g l^{-1}	90	42.3
Lithium	g l^{-1}	0.10	0.078
Potassium	g l^{-1}	4.6	2.4
Boron	g l^{-1}	21.5	18.4
^{60}Co activity	kBq m^{-3}	6.34×10^{7}	8.26×10^{7}
^{3}H activity	kBq m^{-3}	8.41×10^{5}	8.13×10^{5}

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