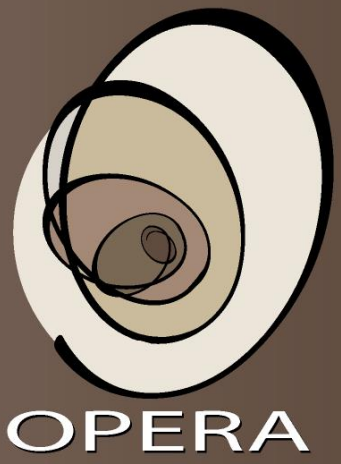

Determination of the inventory
Part A: Radionuclides

OPERA-PU-NRG1112A

Radioactive substances and ionizing radiation are used in medicine, industry, agriculture, research, education and electricity production. This generates radioactive waste. In the Netherlands, this waste is collected, treated and stored by COVRA (Centrale Organisatie Voor Radioactief Afval). After interim storage for a period of at least 100 years radioactive waste is intended for disposal. There is a world-wide scientific and technical consensus that geological disposal represents the safest long-term option for radioactive waste.

Geological disposal is emplacement of radioactive waste in deep underground formations. The goal of geological disposal is long-term isolation of radioactive waste from our living environment in order to avoid exposure of future generations to ionising radiation from the waste. OPERA (OnderzoeksProgramma Eindberging Radioactief Afval) is the Dutch research programme on geological disposal of radioactive waste.

Within OPERA, researchers of different organisations in different areas of expertise will cooperate on the initial, conditional Safety Cases for the host rocks Boom Clay and Zechstein rock salt. As the radioactive waste disposal process in the Netherlands is at an early, conceptual phase and the previous research programme has ended more than a decade ago, in OPERA a first preliminary or initial safety case will be developed to structure the research necessary for the eventual development of a repository in the Netherlands. The safety case is conditional since only the long-term safety of a generic repository will be assessed. OPERA is financed by the Dutch Ministry of Economic Affairs and the public limited liability company Electriciteits-Produktiemaatschappij Zuid-Nederland (EPZ) and coordinated by COVRA. Further details on OPERA and its outcomes can be accessed at www.covra.nl.

OPERA-PU-NRG1112A
Title: Report on the determination of the inventory
Authors: J. Hart (NRG)
Date of publication: 19 June 2015
Keywords: Radioactive Waste, Waste Characteristics

Contents

Summary

The present report provides information and the results of estimates of the radionuclide inventory of the Dutch radioactive waste that is expected to be disposed in a radioactive waste repository in the year 2130.

The data presented in this report are on the result of calculations performed by NRG. The information is based on inventory data provided by COVRA and inventory data obtained from other relevant sources. The inventory data are extrapolated to 2130 on basis of the generic OPERA disposal concept (Verhoef, 2014; Appendix). The methods used and a summary of the resulting data are given in this report, while the detailed numerical calculations have been performed in a separate Excel spread sheet.

The radionuclide inventory is composed of several parts, in analogy with the anticipated waste sections in the ultimate deep geological repository (partly named after Dutch waste classes):

- Low and intermediate level waste
- Non-heat-generating high level waste
- Spent fuel from research reactors
- Vitrified high level waste from reprocessing of spent fuel from the Dutch nuclear power plants.

Uncertainties in the respective inventories have been recognized, and where possible taken into account.

Comparison of the inventories of the different waste categories shows that the vitrified high level waste section contains the highest overall inventory, both in terms of activity and total radiotoxicity.

Samenvatting

Dit rapport bevat een beschrijving en schatting van de radionuclide inventaris van het Nederlandse radioactieve afval dat naar verwachting in het jaar 2130 in een eindberging moet worden gebracht.

De gegevens in dit rapport zijn het resultaat van berekeningen die zijn uitgevoerd door NRG. De informatie is gebaseerd op gegevens over inventarissen van COVRA en van andere relevante bronnen. De inventarisgegevens zijn geëxtrapoleerd naar het jaar 2130, rekening houdend met het generieke OPERA concept voor eindberging (Verhoef, 2014; Appendix). De toegepaste methoden en een samenvatting van de daaruit afgeleide data zijn hier beschreven; de gedetailleerde numerieke beschrijving is verricht in een afzonderlijk Excel bestand.

De radionuclide inventaris is opgebouwd uit afzonderlijke delen, in analogie met de diverse afvalsecties in de uiteindelijke eindberging (deels genoemd naar Nederlandse afval categorieën):

- Laag- en middelradioactief afval
- Niet-warmteproducerend hoogradioactief afval
- Gebruikte splijtstof uit onderzoeksreactoren
- Verglaasd hoogradioactief afval, afkomstig van de opwerking van gebruikte splijtstoffen uit de Nederlandse vermogensreactoren.

Onzekerheden in de respectievelijke inventarissen zijn benoemd, en waar mogelijk in acht genomen.

Uit de vergelijking van de inventarissen van de verschillende afvalcategorieën blijkt dat de sectie voor het verglaasde hoogradioactieve afval de grootste bijdrage heeft aan de totale radioactiviteit en de radiotoxiciteit.

1. Introduction

1.1.Background

The five-year research programme for the geological disposal of radioactive waste – OPERA - started on 7 July 2011 with an open invitation for research proposals. In these proposals, research was proposed for the tasks described in the OPERA Research Plan. The present report (*Determination of the inventory Part A: Radionuclides*, OPERA Milestone M1.1.1.2A) is one of the outcomes of the OPERA Project *OPCHAR* (OPERA Waste Characteristics), as part of Task 1.1.1, *Definition of radionuclide inventory and matrix composition*. It describes the structure and content of the *Reference list of waste inventory - Part A: Radionuclides* that has been developed within the project OPCHAR.

The present report and the accompanying *Reference list of waste inventory - Part A: Radionuclides* will serve as input for performance assessment calculations in OPERA WP7. The results presented in this report are also necessary input for OPERA Task 5.1.1 in which the future evolution of the repository and interactions with the host rock are evaluated.

1.2.Objectives

The aim of the task is to provide an estimation of the radionuclide inventory of the radioactive waste that is expected to be stored in a radioactive waste repository in the year 2130, and to provide a database containing the relevant information to be used by other OPERA participants as input for their work.

1.3.Realization

The data presented in this report are on the result of calculations performed by NRG. The information is based on inventory data provided by COVRA and inventory data obtained from other relevant sources. The inventory data are extrapolated to 2130 on basis of the generic OPERA disposal concept (Verhoef, 2014; Appendix). The methods used and a summary of the resulting data are given in this report, while the detailed numerical calculations have been performed in a separate Excel spread sheet.

1.4.Explanation contents

An overview of the anticipated waste sections of the OPERA Disposal Concept is provided in Chapter 2. The set-up of the reference database is described in Chapter 3. Chapter 4 describes the existing information for the different waste types. The extrapolated values of the inventories of the different waste sections and the considerations for these extrapolations are elucidated in Chapter 5. An evaluation of the results is provided in Chapter 6, whereas Chapter 7 contains the concluding remarks.

2. Waste categorization

The most recent waste classification scheme developed by IAEA (IAEA, 2009) covers all types of radioactive waste and provides a generic linkage with disposal options for all types of waste. That scheme is based on considerations of long-term safety, and thus, by implication, disposal of the waste. Six classes of waste have been derived and used as the basis for the classification scheme (IAEA, 2009; p. 5). The IAEA classification also covers radioactive waste having such low levels of activity concentration that it is not required to be managed or regulated as radioactive waste.

The waste classification scheme for The Netherlands is not based on a law or regulation but since 1985 it is common practice to use this classification scheme. Roughly in the Dutch waste classification scheme there are three waste categories, namely Low and Intermediate Level Waste (LILW), non-heat-generating High Level Waste (HLW non-heat-generating) and heat generating High Level Waste (HLW heat-generating). Three groups of LILW can be distinguished: LILW, LILW (NORM) and LILW (DU).

The comparison of the definitions of waste classes in both the Dutch and the IAEA's waste classification schemes led to the matrix presented in Table 2-1. The radioactive waste categories that are not required to be managed or regulated as radioactive waste have not been considered in this comparison. The table shows that in the Netherlands no distinction is made between short-lived and long-lived LILW as defined by the IAEA Safety Guide on Classification (IAEA, 2009; p. 5). The reason is that shallow land burial is not applicable for the Netherlands. All categories of waste indicated in Table 2-1, except part of the "LILW-NORM", will be disposed of in a deep geologic repository in the future. The only fraction of (TE)NORM that is foreseen for geological disposal is depleted uranium, DU (Verhoef, 2014; p.9). Due to the small amounts of radioactive waste, no separate disposal facilities for LILW and HLW are envisaged (EL&I, 2011; p.23).

Table 2-1 Comparison of the Dutch and IAEA's radioactive waste classification schemes (EL&I, 2011; p.122)

	(IAEA) Distribution %		
Waste Class Name (NL)	**LILW-SL**	**LILW-LL**	**HLW**
LILW	90	10	
LILW-NORM	100		
LILW-DU		100	
HLW, non-heat-generating		100	
HLW, heat-generating			100

In (Verhoef, 2014), an outline of the waste inventory and some of its general features are given. In the disposal layout, four waste disposal sections are distinguished (Verhoef, 2014; Fig. 5-2), partly named after the Dutch waste classes:

- low and intermediate level waste (LILW)
- non-heat producing high level waste (HLW)
- vitrified HLW
- spent fuel (SF)

The expected total number of containers of each type of waste to be disposed of in each waste disposal section is summarized in (Verhoef et al., 2011a; Appendix, Table A-1 to A-3).

Spent fuel

Spent fuel comprises spent fuel from the HOR (Hogere Onderzoeks Reactor) in Delft and the HFR (Hoge Flux Reactor) in Petten, and spent uranium targets from molybdenum production (EL&I, 2011; p.24). The very small amounts of spent fuel from the dismantled

LFR (Lage Flux Reactor) in Petten are insignificant compared to those from the HFR and HOR, as is elucidated in Section 4.3.1.

To be noted is that in the present report also spent fuel of a future replacement of the HFR has been considered. This reactor is currently indicated as PALLAS.

Vitrified HLW

Vitrified high-level waste consists of the heat-generating radioactive waste from the reprocessing of spent fuel from the two nuclear power reactors in the Netherlands (Borssele and Dodewaard).

Non-heat-generating high-level waste

Non-heat-generating HLW mainly consist of the residues from the reprocessing of spent fuel, other than the vitrified HLW. It also includes waste from research on reactor fuel, the production of isotopes, and waste resulting from the decommissioning of the nuclear power plants in the Netherlands (Verhoef, 2014; Section 4.3)

Low- and intermediate-level waste

At present, the low- and intermediate-level waste in the Netherlands is segregated in four categories A - D according to the scheme in Table 2-2 (EL&I, 2011; p.24):

Table 2-2 Low and intermediate-level waste classified by type of radioactivity

Category	*Type of radioactivity*
A	Alpha emitters
B	Beta/gamma contaminated waste from nuclear power plants
C	Beta/gamma contaminated waste from producers other than nuclear power plants with a half-life longer than 15 years
D	Beta/gamma contaminated waste from producers other than nuclear power plants with a half-life shorter than 15 years

LILW arises from activities with radioisotopes in among others industry, research institutes and hospitals. It includes lightly contaminated materials, such as tissues, plastic -, metal - or glass objects, or cloth. In addition, drums with waste conditioned in cement, originating from the nuclear power plants, contribute to the amount of LILW. The LILW is presently stored in four types of packages with volumes of 200, 600, 1000, or 1500 litres (Verhoef, 2014; Section 4.1).

(TE)NORM

The only fraction of (TE)NORM that is foreseen for geological disposal (Verhoef, 2014; Section 4.2) is depleted uranium (DU) originating from the uranium enrichment facility of URENCO (EL&I, 2011; p.22). The DU is presently stored in DV-70 containers (volume 3,5 m^3). For the purpose of OPERA it is assumed that the DU will be immobilized in concrete and finally disposed of in KONRAD type II containers (volume 4,6 m^3).

3. Set-up of the OPERA reference database

The previously indicated waste categories include a variety of radionuclides which together define a source term of the radiological safety assessment calculations in OPERA WP7 *Scenario development and performance assessment*. For the purpose of these analyses in OPERA WP7, OPCHAR distinguishes separate fractions of waste, based on the above-mentioned waste categories[1]. For each of the four waste fractions, the following parameters have been included in the OPERA reference database:

- Name of radionuclide;
- Radionuclide half-life [yr];
- Activity, expressed as [Bq];

If applicable, daughter nuclides have been taken into account. The respective sections explain in more detail if and when daughter nuclides are considered.

Only for a very limited number of radionuclides quantitative uncertainty indicators for the radioactivity could be determined (see Appendix 5 and Appendix 6). These have been taken into account when applicable. In general the uncertainty in the activities of nuclides has been accounted for by providing upper and/or maximum values for the different waste sections. As a result, the provided inventories can be regarded as an upper envelope for serving as input for the long-term safety assessment.

Selection of radionuclides considered

Although the different categories of radioactive waste contain a large variety of radionuclides, many radionuclides are not relevant for the long-term safety due to their short half-lives, and can therefore be excluded from the safety assessment to be performed in OPERA WP7 *Scenario development and performance assessment*.

In order to obtain a useful and manageable selection of radionuclides for the database to be developed in the present task, a selection criterion has been applied. The procedures and selection criteria which have been applied in Belgium, France, and Switzerland to arrive at a manageable list of radionuclides have been evaluated. Based on this information, summarized in Appendices 1 to 3, OPCHAR applied the following selection criterion:

Half-life of the radionuclides: for OPCHAR only radionuclides with half-lives longer than 10 years have been taken into account. Considering the time span of the surface storage in the Netherlands, viz. in the order of 100 years or longer, and the long travel time of radionuclides through the Boom Clay, radionuclides with short half-lives are of no relevance for the long-term safety.

Any additional selection criterion such as potential radiological impact, as applied e.g., in France (Appendix 2) and Switzerland (Appendix 3) has not been applied, as that would require either a preliminary iteration of the safety assessment, or the application of clearly defined regulations concerning the radiological impact of waste disposed in a deep geological facility. For example, the summation of activities or activity concentrations (BS, 2001; Bijlage 3) in determining the radiological impact is not applicable in this case since it is still undefined whether the summation should be performed for a single waste container, a waste section, or the total repository.

[1] Such a distinction is also to be preferred for cataloguing of the non-radiological waste matrices

Radionuclide half-life

The half-lives of the radioactive nuclides and their daughters (if applicable) taken into consideration in the waste inventory have been obtained from the OECD/NEA JEFF Database (Kellett, 2009).

Data sources for amounts and (estimated) uncertainties

The data on the amount of radionuclides are preferably based on the current inventory at COVRA, extrapolated to 2130 and the overall amount of waste defined in the OPERA reference concept (Verhoef, 2014; Appendix). Where necessary, additional data sources were used. Detailed information regarding amounts, masses and compositions of spent fuel from research reactors is considered confidential in the framework of *Artikel 10 lid 1.c* of the *Wet Openbaarheid Bestuur* (WOB, 1991). As a consequence, the information utilized as a basis for the extrapolations to 2130 for this type of radioactive waste is not provided in the present report. The sections treating the various inventories provide supplemental information about the data sources and uncertainties, if applicable.

4. Current inventory of the OPERA disposal concept

4.1. Low and Intermediate level waste section

4.1.1. Low and Intermediate level waste

The inventory of low and intermediate level waste (LILW) has been obtained from COVRA per 19 February 2013 (Huizen, 2013), further referred to as the "cutoff date", and consists of 189'584 entries in an Excel spreadsheet. The LILW inventory comprises all radionuclides stored at that time in four different container types, distinguished by volume (cf. Section 2[2]). For each container the date of emplacement has also been supplied, which is relevant for the decay and ingrowth of radionuclides, and the extrapolation to possible future inventories.

The LILW inventory contains 192 different radionuclides, many of which are not relevant to the long-term safety due to their relatively short half-lives, i.e. less than 10 years.

The following procedure has been applied to determine the radionuclides relevant for the long-term safety assessment:

- Only radionuclides with half-lives longer than 10 years have been taken into account;
- For each nuclide its decay has been calculated from the time to emplacement to the cutoff date;
- For the daughter nuclides with half-lives longer than 10 years, only nuclides have been processed further that contribute more than 1% to the already emplaced inventory of those nuclides.

This procedure results in a total of 54 radionuclides which is taken into account in the extrapolation to the year 2130. The present inventory is shown in Table 4-1.

Table 4-1 Present LILW radionuclide inventory (status February 2013), relevant for the long-term safety

Nuclide	*Half-life [yr]*	*Present Inventory [Bq]*
H-3	1.23E+01	3.03E+14
Be-10	1.60E+06	6.30E+08
C-14	5.70E+03	7.07E+11
Si-32	1.32E+02	1.83E+03
Cl-36	3.01E+05	1.59E+09
K-40	1.27E+09	1.53E+09
Ti-44	6.00E+01	2.97E+04
Ni-59	7.60E+04	4.16E+08
Ni-63	1.01E+02	3.01E+13
Kr-81	2.10E+05	9.00E+05
Kr-85	1.08E+01	1.07E+12
Sr-90	2.88E+01	1.00E+13

[2] Besides the 200, 600, 1000, and 1500-litre containers (35'431 containers in total), also two 400-litre containers are presently stored at COVRA.

Nuclide	*Half-life [yr]*	*Present Inventory [Bq]*
Nb-93m	1.63E+01	4.00E+05
Mo-93	4.00E+03	7.95E+04
Nb-94	2.00E+04	5.40E+07
Tc-99	2.14E+05	3.13E+10
Ag-108m	4.18E+02	6.37E+07
Cd-113m	1.41E+01	4.23E+03
Sn-121m	5.50E+01	1.01E+08
I-129	1.61E+07	7.82E+07
Ba-133	1.05E+01	1.81E+09
Cs-137	3.00E+01	5.61E+14
Sm-146	1.00E+08	1.00E+03
Sm-151	9.00E+01	1.35E+09
Eu-152	1.35E+01	3.83E+09
Re-187	4.35E+10	6.67E+06
Pb-202	5.20E+04	2.10E+05
Bi-207	3.18E+01	8.94E+06
Po-209	1.02E+02	2.67E+04
Pb-210	2.22E+01	1.13E+11
Ra-226	1.60E+03	2.48E+11
Ac-227	2.18E+01	2.16E+08
Th-229	7.34E+03	8.40E+06
Th-230	7.54E+04	2.22E+06
Pa-231	3.28E+04	2.36E+08
U-232	6.88E+01	2.06E+07
Th-232	1.41E+10	1.58E+11
U-233	1.59E+05	7.24E+07
U-234	2.46E+05	2.39E+10
U-235	7.04E+08	6.98E+11
U-236	2.37E+07	2.70E+06
Np-237	2.14E+06	3.91E+07
Pu-238	8.77E+01	6.50E+12
U-238	4.47E+09	1.41E+12
Pu-239	2.41E+04	4.50E+11
Pu-240	6.56E+03	2.87E+10
Pu-241	1.43E+01	3.17E+10
Am-241	4.33E+02	6.43E+12
Pu-242	3.74E+05	7.01E+10
Am-243	7.36E+03	1.26E+07
Cm-244	1.80E+01	5.30E+10
Pu-244	8.00E+07	2.00E+04
Cf-249	3.57E+02	6.03E+06

From the COVRA database it can be deduced that the activities of the respective radionuclides are far from homogeneously distributed amongst the containers. It is however beyond the scope of the present analysis to analyze this aspect in more detail.

4.1.2. TE(NORM)

Depleted uranium (DU) originating from the uranium enrichment facility of URENCO is presently stored in the form of uranium oxide (U_3O_8). DU is intended to be disposed in the

geological repository, but at present it is not yet conditioned to allow for possible future reuse of the material. For the purpose of OPERA it is assumed that the DU will be conditioned and re-packed in KONRAD type II containers for final disposal (Verhoef, 2014; Table A-2).

The radionuclide inventory as per September 2013 of the DU, stored in a total number of 2831 containers (2829 DV-70 containers, and two 200 litre containers), is listed in Table 4-2 (COVRA, 2013a; Bijlage 5c).

Table 4-2 Radionuclide inventory of the DU stored at COVRA per September 2013

Nuclide	*Half-life [yr]*	*Activity [Bq]*
U-234	2.46E+05	7.60E+13
U-235	7.04E+08	7.60E+12
U-238	4.47E+09	3.29E+14

4.2.Non-heat-generating high level waste section

It is expected that non-heat-generating HLW will include the non-heat-generating waste originated from the reprocessing of the spent fuel (such as hulls and ends from fuel assemblies) as well as the waste from dismantling and decommissioning nuclear facilities or historical wastes (Verhoef, 2014; Section 4.3). These waste types are described in the following sections.

4.2.1. Compacted hulls and ends

The compacted hulls and ends from the reprocessing process are stored in CSD-C canisters, which have the same dimension as the CSD-V canisters. The inventory of the radionuclides contained in a single CSD-C container has been obtained from COVRA (COVRA, 2013b; "CSD-C"; see also Appendix 5), and is tabulated below. Only radionuclides with half-lives longer than 10 years have been tabulated.

Table 4-3 Radionuclide inventory of a CSD-C canister at the time of production

Nuclide	*Half-life [years]*	*Inventory [Bq/canister]*
C-14	5.70E+03	1.400E+10
Ni-59	7.60E+04	3.590E+11
Ni-63	1.01E+02	2.700E+13
Se-79	3.77E+05	5.500E+07
Sr-90	2.88E+01	2.8000E+13
Zr-93	1.53E+06	3.000E+08
Mo-93	4.00E+03	5.919E+09
Nb-94	2.00E+04	5.572E+10
Tc-99	2.14E+05	2.300E+09
I-129	1.61E+07	5.300E+07
Np-237	2.14E+06	7.200E+06
Pd-107	6.50E+06	1.171E+07
Sn-126	2.30E+05	8.837E+07
Cs-135	2.30E+06	7.100E+07
Cs-137	3.00E+01	6.500E+13
Sm-151	9.00E+01	6.000E+10
Ac-227	2.18E+01	3.806E+00

Nuclide	*Half-life [years]*	*Inventory [Bq/canister]*
Pa-231	3.28E+04	2.430E+02
Th-229	7.34E+03	1.190E+00
Th-230	7.54E+04	1.681E+01
U-232	6.88E+01	1.089E+06
U-233	1.59E+05	3.203E+02
U-234	2.46E+05	3.061E+06
U-235	7.04E+08	1.247E+06
U-236	2.37E+07	1.211E+07
U-238	4.47E+09	1.883E+07
Pu-238	8.77E+01	1.647E+11
Pu-239	2.41E+04	1.564E+10
Pu-240	6.56E+03	1.963E+10
Pu-241	1.43E+01	7.500E+13
Pu-242	3.74E+05	9.853E+07
Pu-244	8.00E+07	4.909E+01
Am-241	4.33E+02	3.500E+10
Am-242m	1.41E+02	1.560E+08
Am-243	7.36E+03	3.500E+10
Cm-243	2.85E+01	2.799E+08
Cm-244	1.80E+01	9.500E+10
Cm-245	8.50E+03	1.097E+07
Cm-246	4.73E+03	4.860E+06
Cm-247	1.56E+07	2.631E+01
Cm-248	3.39E+05	1.625E+02

4.2.2. Historical wastes

This waste fraction comprises radioactive material that is presently stored at the ECN/NRG site in Petten, in the "Waste Storage Facility" (WSF). Information about the distribution of radionuclides has been reported as part of the Environmental Impact Assessment (EIS) for the *Hoog-Actief Vast Afval Verpakkings Unit* (HAVA-VU) in Petten (Timmermans, 2007). The inventories are reported in Tables 2 and 3 of that document.

Table 4-4 summarizes a conservative estimation of the inventory of non-heat-generating HLW[3] waste containing non-fissile material present in the WSF.

Table 4-4 Non-heat-generating HLW[3] waste containing non-fissile material present in the WSF on the reference date 4 March 2003

Nuclide	*Half-life [yr]*	*Present Inventory [Bq]*
H-3	1.23E+01	6.00E+13
Ni-63	1.01E+02	5.30E+12
Nb-94	2.00E+04	4.40E+10
Tc-99	2.14E+05	1.70E+12
Cs-137	3.00E+01	1.17E+14
Am-241	4.33E+02	5.00E+10

[3] Indicated in Table 2 of (Timmermans, 2003) as HAVA/MAVA, i.e. "niet-splijtstofhoudend vast radioactief afval"

In addition to non-fissile material, the WSF contains also fissile material. This fissile material will be re-packed and loaded in so-called MTR-2 containers. For the present analysis it is assumed that the column "maximum inventory spent fuel in cell" (Timmermans, 2007; Table 3) represents the maximum values of the radionuclide inventory in a single MTR-2 container. These values are presented in Table 4-5. It is expected that all radioactive containing fissile material can packed in two MTR-2 containers (Timmermans, 2007; p.62).

Table 4-5 Maximum inventory of fissionable material in an MTR-2 container

Nuclide	*Half-life [yr]*	*Present Inventory [Bq]*
Kr-85	1.08E+01	4.10E+13
Sr-90	2.88E+01	2.80E+14
Cs-137	3.00E+01	3.68E+14
U-235	7.04E+08	3.10E+08
U-238	4.47E+09	3.80E+08
Pu-238	8.77E+01	7.89E+12
Pu-239	2.41E+04	2.59E+13
Pu-240	6.56E+03	6.70E+11
Pu-241	1.43E+01	2.48E+14
Am-241	4.33E+02	6.00E+11
Am-242m	1.41E+02	8.10E+09
Am-243	7.36E+03	3.00E+10
Cm-243	2.85E+01	2.00E+10
Cm-244	1.80E+01	4.97E+12

4.2.3. Waste from dismantling and decommissioning

Decommissioning waste results from the dismantling of the two Dutch nuclear power plants (EL&I, 2011; p.62):

- The Borssele PWR (Siemens/KWU design, 515 MWe, 1366 MWth), presently operated by EPZ. The Dutch government has reached an agreement with the operator on immediate dismantling after closure, scheduled in 2033 (EL&I, 2011; p.63).
- The Dodewaard BWR (GE design, 60 MWe, 183 MWth), previously operated by GKN. The Dodewaard reactor was shut down in 1997 and is now in safe enclosure, a stage of decommissioning. All spent fuel from the Dodewaard NPP has been removed from the site. In 2003, the last batch of spent fuel from the reactor was transferred to Sellafield (UK) for reprocessing. The separated uranium and plutonium is not returned to the Netherlands. The remaining waste returned from Sellafield to the Netherlands in April 2010, for long-term storage (EL&I, 2011; p. 16). Further dismantling is foreseen to commence in 2045 (EL&I, 2011; p. 63)

At present there seems no information available concerning the radionuclide speciation and inventory of the decommissioning waste of the two Dutch NPPs. Previous studies concerning the decommissioning of the Borssele and Dodewaard NPPs usually only mention decommissioning waste, including activated materials, in terms of total weight without further speciation (e.g. Paul, 1994a, Tables related to Chapter 8; Paul, 1994b, Tables related to Chapter 8).

For the determination of the radionuclide inventory of waste from dismantling and decommissioning the relevant information of the Nagra Model Inventory has been utilized

as a basis (Alder, 1994). The reason to utilize the information in that document is that it is the only source of information providing the level of detail necessary to establish the radionuclide inventory of decommissioning waste. In addition, there is no a priori reason to assume that these data would not apply to the Dutch case. The Nagra study summarizes in detail the inventories of the different waste classes adopted in Switzerland. The NPP decommissioning waste is indicated in (Alder, 1994) as "SMA SA-NPP" waste, or "SA-KKW" waste, and consists of the following waste forms (Alder, 1994; p.30):

- core internals and reactor vessel parts,
- biological shield (concrete),
- reactor vessel parts,
- PWR primary circuits and BWR recirculation circuits,
- low contaminated components and piping,
- other primary waste (other materials) plus secondary waste (such as tools, plastic, filters, etc.),
- decontamination residues.

The total amount of SA-KKW waste has been estimated at 42'948 m^3 (Alder, 1994; p.69).

4.3. Spent fuel from research reactors

The spent fuel intended for final disposal consists of spent fuel from the three Dutch research reactors and spent uranium targets from the production of molybdenum.

The basic characteristics of the three main test and research reactors in the Netherlands, viz. the Hoge Flux Reactor (HFR); the Hoger Onderwijs Reactor (HOR) and the Lage Flux Reactor (LFR) are summarized in Table 4-6 (updated from Dodd, 2000; p.28).

Table 4-6 Test and research reactors in the Netherlands

Reactor	Type	Owner	Criticality Date	Design Power (kW)	Maximum Licensed Power (kW)
LFR*	Argonaut	NRG	28-09-60	10	30
HFR	Tank	European Commission	09-11-61	20000	45000
HOR	Pool	TUD	25-04-63	200	3000

* Reactor was shut down in 2010

The following sections provide a brief overview of each of these reactors and the utilized fuels.

4.3.1. LFR

The Lage Flux Reactor (LFR) was a small research reactor used for training and general research purposes. The LFR was owned by NRG and is located in Petten. The 30 kW reactor was not operated on a regular basis, and shut down in 2010 (EL&I, 2011; p.16).

The LFR fuel assemblies have been removed in December 2013[4]. Due to the low power and discontinuous operation of the LFR the burn-up of the fuel elements was low. The quantity and the equivalent energy production, and hence the total lifetime production of actinides and fission products of spent fuel elements from the LFR is negligible in comparison with those from the HFR, and will therefore be ignored in the present analysis.

4.3.2. HFR

The HFR is a multi-purpose research reactor used for material testing, radioisotope production and general research purposes. The HFR is owned by Institute for Energy of the Joint Research Centre of the European Commission. The reactor is located at the Petten site of the Joint Research Centre. It is operated under contract by NRG.

From the start of operation the HFR utilized highly-enriched uranium fuel assemblies, HEU, (Dodd, 2000; p. 28). Since May 2006 the HFR only uses low enriched uranium (LEU). This is in line with the worldwide move to abandon the use of high enriched uranium (HEU) for non-proliferation reasons (EL&I, 2011; p.16). After the HEU to LEU conversion the power of the HFR was uprated from 45 MWth to 50 MWth (VROM, 2010; Annex 3, Table 1).

The spent fuel elements of the research reactors are contained in a basket with circa 33 elements, both spent fuel assemblies and control assemblies (EL&I, 2011; p.23). For safeguard reasons the radionuclide inventories of HEU and LEU spent fuel assemblies are not provided here (WOB, 1991; Art.10, lid 1c). The total anticipated amount of radionuclides of all spent fuels at the time of geological disposal is provided in Section 5.3.1.

4.3.3. HOR

The Hoger Onderwijs Reactor (HOR) is a multi-purpose research reactor used for basic and applied research and education purposes. The HOR is owned by the Delft University of Technology (TUD) and operated by the Interfacultary Reactor Institute (IRI). The HOR is located in Delft. The HOR is a pool type reactor with a light water coolant and moderator. In 1966 the HOR was licensed to operate at a maximum power of 200 kW; this was increased to 500 kW in the beginning of 1967. As a result of an extensive modification and upgrading project later that year it was then possible to operate the reactor at 2 MW. Since 1969 the maximum licensed power is 3 MW (Dodd, 2000; p.30).

The reactor was originally designed for and was operated on HEU fuel. For safeguard reasons it was decided to convert the HOR to LEU fuel. This fuel type change was accomplished in 2005 (RID, 2012; p.22).

The spent fuel elements of the research reactors are contained in a basket with circa 33 elements, both spent fuel assemblies and control assemblies (EL&I, 2011; p.23). For safeguard reasons the radionuclide inventory of HEU and LEU spent fuel assemblies are not provided here (WOB, 1991; Art.10, lid 1c). The total anticipated amount of radionuclides of all spent fuels at the time of geological disposal is provided in Section 5.3.1.

[4] E.g.: http://www.world-nuclear-news.org/WR-LFR-fuel-removed-from-Petten-0612134.html; last accessed on 12 February 2014.

4.3.4. Uranium filters

Uranium filters originate from the production of molybdenum at NRG, Petten. The U-filters consist of the uranium remains of irradiated uranium targets. The filters are stored in canisters, 99 filters per canister (Verberg, 2013; p.23). Table 4-7 provides the activity of radionuclides in one canister containing HEU uranium filters with an irradiation period of 150 hours presuming a 50 MW power level of the HFR. The reported values are maximum values after a cooling period of 2 years (COVRA, 2013b; "unreprocessed", see also Appendix 4).

Table 4-7 Activity of radionuclides in HEU uranium filters in one canister

Nuclide	*Half-life [yr]*	*Activity [Bq/canister]*
Ni-63	1.01E+02	2.56E+06
Se-79	3.77E+05	4.65E+07
Sr-90	2.88E+01	9.19E+13
Zr-93	1.53E+06	1.95E+09
Tc-99	2.14E+05	1.48E+10
Np-237	2.14E+06	1.47E+07
Pd-107	6.50E+06	3.10E+07
Sn-126	2.30E+05	3.03E+08
Sm-147	1.06E+11	4.31E+03
Sm-151	9.00E+01	1.88E+12
U-232	6.88E+01	6.96E+03
U-233	1.59E+05	1.87E+04
U-234	2.46E+05	9.62E+06
U-235	7.04E+08	1.27E+09
U-236	2.37E+07	4.59E+08
U-238	4.47E+09	2.65E+07
Pu-238	8.77E+01	6.68E+08
Pu-239	2.41E+04	2.08E+10
Pu-240	6.56E+03	1.13E+09
Pu-241	1.43E+01	2.37E+10
Pu-242	3.74E+05	1.03E+04
Am-241	4.33E+02	7.99E+07
Pa-231	3.28E+04	5.54E+04

The composition of the LEU uranium targets, which will be applied in the near future for the production of molybdenum, is not yet accessible. Therefore an educated guess is performed, based on the composition of the HEU uranium targets, and taking into account that:

- the U-235 enrichment of the LEU uranium targets is about 4 times lower than that of the HEU targets;
- the amount of U-235 is equal for LEU and HEU to obtain a similar amount of produced molybdenum.

These assumption imply that:

- the amount of fission products will be equal for HEU and LEU targets
- the amount of actinides, formed from U-238, will be larger for LEU targets than for HEU targets.

4.4. Vitrified High Level Waste

According to the current contract between the operator of the Borssele NPP, EPZ, and AREVA, spent fuel from the Borssele NPP is sent to AREVA in France for reprocessing. The vitrified waste residues and the compacted hulls and ends from the reprocessing process are being returned to the Netherlands and stored at COVRA (EL&I, 2011; p.15).

Vitrified waste residues

The vitrified waste residues are stored in CSD-V canisters[5]. The inventory of the radionuclides contained in a single CSD-V container has been provided by COVRA (COVRA, 2013b; "CSD-V"; see also Appendix 6). Part of the reported values represent empirical maximum values at production (not guaranteed), and can therefore be regarded as an upper envelope.

Uncertainty estimates have not been reported for the section listing the fission products, but these arise from e.g. differences in burn-up values of the reprocessed spent fuel elements, although the associated uncertainties are considered relatively minor (e.g. McGinnes, 2002; p.17). Another source of uncertainty in the inventory lies in the formulation of reprocessing contracts, which may change over the years.

Two radionuclides in the CSD-V data provided by COVRA have been analyzed in more detail:

- Iodine-129, is was not present in the COVRA data list;
- Selenium-79, for which the listed inventory is considered too large compared to other relevant radionuclides.

Iodine-129

An important issue in the data list provided by COVRA is that the nuclide I-129, which is considered relevant for the long-term safety (e.g. ONDRAF/NIRAS, 2001; Ch11.3.8; p.43/58), is lacking. Due to its long half-life and high mobility, I-129 is always one of the largest contributors to the total dose rate in long-term safety assessments.

In order to consider the possible contribution of I-129 to the OPERA long-term safety calculations, other sources of information have been consulted for obtaining the inventory of I-129 in vitrified HLW (see also Table 4-8):

- The SAFIR-2 study, viz. the inventory of I-129 in a ZAGALC container (ONDRAF/NIRAS, 2001, Table 2-6). The SAFIR-2 report states that the level of I-129 in the very high-level vitrified waste ZAGALC (currently assumed at 1% of the initial level in the spent fuel) remains to be confirmed however. It is mentioned that the figure of 1% is unquestionably an overestimate of the true level of I-129 in the ZAGALC waste (ONDRAF/NIRAS, 2001, Table 2-6).
- The Nagra Model Inventory, more specifically the WA-COG-1 canisters (HLW vitrified residues). I-129 inventories are provided in (McGinnes, 2002): Tab.5.3 (average value) and Tab. A.1-4 (averaged and maximum value at production).

[5] CSD-V: Colis Standard de Déchets - Vitrifiés; containers as well as glass matrix are designed by the French company COGEMA (Compagnie Générale des Matières Nucléaires), presently AREVA.
CSD-C: Colis Standard de Déchets – Compactés

Table 4-8 Inventory of I-129 in a vitrified HLW canister

SAFIR-2	WA-COG-1 avg	WA-COG-1 avg2	WA-COG-1 max
[Bq/can]	[Bq/can]	[Bq/can]	[Bq/can]
1.52E+08	1.609E+06	1.600E+06	2.600E+08

On the basis of this information it has been decided to add the maximum value of the I-129 inventory in the Swiss WA-COG-1 canister, i.e. the "WA-COG-1-max" value, to the OPERA radionuclide database of vitrified HLW. This value provides the maximum inventory of I-129.

Selenium-79

Analyzing the list of radionuclides in vitrified HLW provided by COVRA it appeared that the inventory of Se-79 has been reported as a much higher value, compared to other, comparable sources of information, more specifically data provided by Nagra (McGinnes, 2002). This observation has been elucidated in Table 4-9, in which a comparison has been made of the inventories of selected fission products present in:

- PWR UO_2 Spent Fuel for an average burnup of 48 GWd/tHM (McGinnes, 2002; Tab. B.1) – "Nagra Spent Fuel";
- Vitrified HLW, i.e. the WA-COG-1 canister, for which Nagra reported data in (McGinnes, 2002; Tab. A.1-4) – "Nagra HLW";
- The CSD-V canisters containing vitrified HLW, for which the data have been provided by COVRA (COVRA, 2013b; "CSD-V") – "COVRA CSD-V"

Table 4-9 Comparison of Se-79 inventories with selected radionuclides in spent fuel and vitrified HLW

	Nagra Spent Fuel		Nagra HLW		COVRA CSD-V		
Nuclide	Inventory [Bq]	Ratio Inventory Nuclide/ Sn-126	Inventory [Bq]	Ratio Inventory Nuclide/ Sn-126	Inventory [Bq]	Ratio Inventory Nuclide/ Sn-126	**Ratio Inventory COVRA CSD-V /NAGRA HLW**
Se-79	1.00E+09	0.0526	1.20E+09	0.0324	2.01E+10	**0.5301**	**16.786**
Zr-93	9.50E+10	5.00	9.40E+10	2.54	1.05E+11	2.77	1.121
Tc-99	7.00E+11	36.8	8.40E+11	22.7	1.25E+12	33.0	1.492
Pd-107	6.20E+09	0.326	6.50E+09	0.176	6.78E+09	0.178	1.043
Sn-126	1.90E+10	1.00	3.70E+10	1.00	3.80E+10	1.00	1.027

On basis of Table 4-9 the following observations apply:

- The distribution of the respective nuclides in Nagra SF, Nagra HLW, and COVRA CSD-V are of the same order of magnitude and vary not more than about a factor of 1.5, except for the Se-79 inventory in COVRA CSD-V;
- The ratio: inventories of nuclides / inventory of Sn-126, is comparable for Nagra SF, Nagra HLW, and COVRA CSD-V, except for Se-79.
- The ratio of the Se-79 inventory in COVRA CSD-V / Nagra HLW is about 16 times larger compared to the other nuclides.

A possible explanation of this large discrepancy (Kossert, 2010) is the considerable variation of half-lives reported for Se-79 in the last decades (e.g. Kossert, 2010). Values published so far for the Se-79 half-life varied over an order of magnitude; For example, Nagra has used $1.1 \cdot 10^6$ year for the Se-79 half-life (Nagra, 2002; p.205), whereas the SAFIR-2 study applied a value of 65'000 years (ONDRAF/NIRAS, 2001; Table 2-2). The value in the JEFF database is 377'000 years (Kellett, 2009; p.71), and that in the Karlsruher Nuklidkarte is 327'000 years (Magill, 2012).

This may bring a large variation of reported inventories for Se-79, in case these inventories are obtained from mass (molar) values initially. This aspect is elucidated in Table 4-10, in which the Se-79 inventories have been estimated as molar mass per canister for several reported inventories, taking into account the Se-79 half lives in the respective studies:

- SAFIR-2, Se-79 inventory in ZAGALC vitrified HLW (ONDRAF/NIRAS, 2001; Table 2-5);
- The Se-79 inventory, average and maximum values, in vitrified HLW (WA-COG-1), reported in the NAGRA Model Inventory, (McGinnes, 2002; Tab. A.1-4);
- The values provided by COVRA (COVRA, 2013b; "CSD-V").

Taking into account the reported values of Se-79 inventories and the respective half-life values, it can be seen that the Se-79 inventory in terms of mole/canister are comparable for the NAGRA and SAFIR-2 studies. Applying a half-life value of 65000 years to the COVRA data also results in a value comparable to those studies in terms of mole/canister. Assuming a value of 377'000 years for the Se-79 half-life results however in the larger Se-79 inventory in terms of Bq/canister (Table 4-10).

Table 4-10 Comparison of Se-79 inventories in vitrified HLW

Data source	***Half-life*** [years]	***Inventory*** [Bq/can]	***Inventory*** [mole/can]
NAGRA (average)	1100000	1.20E+09	0.0998
NAGRA (maximum)	1100000	1.60E+09	**0.133**
SAFIR-2	65000	1.66E+10	0.0816
COVRA	65000[6]	2.01E+10	0.0990
COVRA	377000[7]	2.01E+10	0.574
Se-79 in vitrified HLW for OPERA (Proposal)	*377000*	*4.67E+09*	*0.133*

For the present OPERA data base it is proposed to utilize a value for the Se-79 inventory of $4.67 \cdot 10^9$ Bq, which is estimated from the maximum reported inventory (mole/can) for Se-79 in Nagra HLW (0.133 mole/canister), and Se-79 half-life of 377'000 years. The resulting inventory in terms of Bq/canister is more in line with the distribution in spent fuel, compared to the inventories of other long-lived nuclides.

Uncertainties

For several nuclides in the database provided by COVRA also the standard deviation has been reported (See also Appendix 6). For estimating the total model inventory for the year 2130, values of the standard deviation have been added to the average values. In that case, upper boundary values for these nuclides are obtained.

[6] Assumed half-life value; not reported in (COVRA, 2013b)
[7] Assumed half-life value; not reported in (COVRA, 2013b)

Resulting CSD-V inventory
Taking into account the above-mentioned considerations the resulting inventory in a CSD-V canister (COVRA, 2013b; "CSD-V") is listed in
Table 4-11.

Table 4-11 Radionuclide inventory of a CSD-V canister at the time of production

Nuclide	*Half-ife [years]*	*Inventory [Bq/canister]*
Se-79	3.77E+05	4.67E+09
Sr-90	2.88E+01	4.63E+15
Zr-93	1.53E+06	1.05E+11
Tc-99	2.14E+05	1.25E+12
Pd-107	6.50E+06	6.78E+09
Sn-126	2.30E+05	3.80E+10
I-129	1.61E+07	2.60E+08
Cs-135	2.30E+06	3.01E+10
Cs-137	3.00E+01	6.60E+15
Np-237	2.14E+06	2.37E+10
Am-241	4.33E+02	1.13E+14
Am-243	7.36E+03	1.74E+12
Cm-244	1.80E+01	1.78E+14
Cm-245	8.50E+03	2.03E+10
U-234	2.46E+05	3.74E+07
U-235	7.04E+08	6.51E+05
U-236	2.37E+07	9.69E+06
U-238	4.47E+09	1.15E+07
Pu-238	8.77E+01	2.12E+11
Pu-239	2.41E+04	2.00E+10
Pu-240	6.56E+03	3.20E+10
Pu-241	1.43E+01	5.51E+12
Pu-242	3.74E+05	1.54E+08

5. Extrapolation to future OPERA reference inventory

It is anticipated that for each category of radioactive waste uncertainties can arise in the determination of the radionuclide inventories. Three types of uncertainties can be distinguished:

1. Uncertainties in the inventory reported by the suppliers of the waste to COVRA;
2. Uncertainties by extrapolation of the inventory composition into the future;
3. Uncertainty in the spatial distribution of radionuclides in the disposal sections.

To enable a better estimation on the first mentioned potential uncertainties, OPCHAR has relied on the information available at COVRA. On the basis of the information supplied by COVRA sources of uncertainties in the radionuclide inventories have been established.

The extrapolation of the inventory composition resulting from the future deployment of additional nuclear reactors has not been addressed in Task 1.1.1, because the OPERA Safety Case is based on clear assumptions concerning the nuclear energy use in the Netherlands (Verhoef, 2011; p.3)[8]. Also any replacement of the High Flux Reactor (HFR) in Petten has not been considered in the present analysis.

The last mentioned uncertainty will be covered by reporting the inventory for each of the waste categories defined for the OPERA disposal concept. This is assumed to provide sufficient detailing with respect to the spatial distribution for the current OPERA Safety Case.

The extrapolation of the various waste categories to the year 2130 takes into account the following:

- For each considered presently stored nuclide its decay has been calculated from the specified cutoff date (in most cases 2013) to 2130.
- For daughter nuclides with half-lives longer than 10 years, only nuclides have been processed further that contribute more than 1% to the already emplaced inventory of those nuclides. For the majority of the considered radionuclides this is justified since most nuclides produce insignificant amounts of daughter activity until 2130. Only in a few cases the ingrowth of daughters may be significant, viz. for radionuclides with half-lives ranging from ten to several hundred years, and producing daughters with half-lives longer than 10 years.

Any exceptions to these general considerations are described in the appropriate sections.

5.1. Low and Intermediate level waste section

5.1.1. Low and intermediate level waste

The extrapolation of the present LILW inventory to the year 2130 has been performed on the basis of the following considerations:

- From the COVRA LILW data base, described in Section 4.1.1, it can be deduced that, during the last 30 years, the rate of emplacement of the LILW containers has been considerably constant at about 325 m^3/year.
- The rate of emplacement of the individual relevant radionuclides is not constant. However, for obtaining the OPERA radionuclide inventory in the year 2130 is has

[8] Note that Task 1.1.2 of OPCHAR is aiming at (developing tools for) analysing other future nuclear energy use scenarios

been assumed that the present average overall concentration of nuclides in the LILW will remain the same for future waste.

- At the cutoff date (19 February 2013; see Section 4.1.1) about 49% of the total LILW inventory is beta/gamma contaminated waste from nuclear power plants. The supply of this type of waste is assumed to be terminated in 2034, i.e. one year after the anticipated shut-down of the Borssele NPP, at 2033.
- The COVRA LILW data base makes no distinction between different categories of LILW. Therefore it is assumed that the individual radionuclides are uniformly distributed in the total emplaced volume at all times.

On the basis of these considerations three possible inventories have been calculated for the year 2130:

- An Upper Inventory, where it is assumed that the present rate of emplacement will continue until 2130. The anticipated termination of the emplacement of waste from nuclear power plants in 2034 is neglected, but instead is it assumed that other sources of radioactive waste will replace this waste stream. In this case the total volume of LILW in the year 2130 will be about 48'600 m^3.
- An Expected Inventory, where it is assumed that LILW emplacement will continue from present date to 2130 with a rate of 255 m^3/year, leading to a total volume of LILW by the year 2130 of about 40'500 m^3. This volume is in line with the inventory expected by COVRA (Verhoef, 2014; Table A-1).
- A Lower Inventory, where it is assumed that the emplacement of waste from nuclear power plants (comprising 49% of the total present LILW inventory) is terminated by the year 2034, i.e. one year after the shut-down of the Borssele NPP. The remaining waste categories are assumed to be continuously delivered to COVRA until 2130. This results in a total volume of LILW of about 33'200 m^3 by the year 2130.

Taking into account the selection criterion mentioned in Section 3 (half-life), and the above-mentioned considerations the radionuclides listed in Table 5-1 represent the inventory for the year 2130.

Table 5-1 Estimated LILW radionuclide inventory in the year 2130, relevant for the long-term safety

Nuclide	*Half-life [yr]*	*Upper Inventory [Bq]*	*Expected Inventory [Bq]*	*Lower Inventory [Bq]*
H-3	1.23E+01	3.00E+14	2.36E+14	1.79E+14
Be-10	1.60E+06	2.96E+09	2.46E+09	2.02E+09
C-14	5.70E+03	3.30E+12	2.75E+12	2.25E+12
Si-32	1.32E+02	3.75E+03	3.16E+03	2.64E+03
Cl-36	3.01E+05	7.49E+09	6.24E+09	5.12E+09
K-40	1.27E+09	7.20E+09	6.00E+09	4.92E+09
Ti-44	6.00E+01	8.25E+04	6.66E+04	5.24E+04
Ni-59	7.60E+04	1.95E+09	1.63E+09	1.33E+09
Ni-63	1.01E+02	9.46E+13	7.74E+13	6.19E+13
Kr-81	2.10E+05	4.23E+06	3.52E+06	2.89E+06
Kr-85	1.08E+01	9.24E+11	7.28E+11	5.52E+11
Sr-90	2.88E+01	1.94E+13	1.54E+13	1.18E+13
Mo-93	4.00E+03	3.69E+05	3.07E+05	2.52E+05

Nuclide	*Half-life [yr]*	*Upper Inventory [Bq]*	*Expected Inventory [Bq]*	*Lower Inventory [Bq]*
Nb-93m	1.63E+01	5.68E+05	4.48E+05	3.40E+05
Nb-94	2.00E+04	2.53E+08	2.11E+08	1.73E+08
Tc-99	2.14E+05	1.47E+11	1.22E+11	1.00E+11
Ag-108m	4.18E+02	2.72E+08	2.26E+08	1.84E+08
Cd-113m	1.41E+01	6.08E+03	4.79E+03	3.64E+03
Sn-121m	5.50E+01	2.41E+08	1.95E+08	1.53E+08
I-129	1.61E+07	3.67E+08	3.06E+08	2.51E+08
Ba-133	1.05E+01	1.43E+09	1.12E+09	8.53E+08
Cs-137	3.00E+01	9.01E+14	7.18E+14	5.54E+14
Sm-146	1.00E+08	4.70E+03	3.91E+03	3.21E+03
Sm-151	9.00E+01	4.15E+09	3.38E+09	2.70E+09
Eu-152	1.35E+01	4.37E+09	3.45E+09	2.62E+09
Re-187	4.35E+10	3.13E+07	2.61E+07	2.14E+07
Pb-202	5.20E+04	9.85E+05	8.21E+05	6.73E+05
Bi-207	3.18E+01	1.62E+07	1.29E+07	9.97E+06
Po-209	1.02E+02	8.87E+04	7.25E+04	5.79E+04
Pb-210	2.22E+01	1.84E+12	1.50E+12	1.19E+12
Ra-226	1.60E+03	1.14E+12	9.45E+11	7.73E+11
Ac-227	2.18E+01	3.61E+09	3.12E+09	2.69E+09
Th-229	7.34E+03	3.92E+07	3.26E+07	2.68E+07
Th-230	7.54E+04	8.37E+07	7.20E+07	6.14E+07
Th-232	1.41E+10	7.44E+11	6.20E+11	5.08E+11
Pa-231	3.28E+04	1.15E+10	9.89E+09	8.40E+09
Np-237	2.14E+06	8.84E+08	7.52E+08	6.34E+08
U-232	6.88E+01	5.39E+07	4.38E+07	3.48E+07
U-233	1.59E+05	3.40E+08	2.83E+08	2.32E+08
U-234	2.46E+05	1.14E+11	9.47E+10	7.77E+10
U-235	7.04E+08	3.28E+12	2.73E+12	2.24E+12
U-236	2.37E+07	1.27E+07	1.06E+07	8.67E+06
U-238	4.47E+09	6.63E+12	5.52E+12	4.53E+12
Pu-238	8.77E+01	1.91E+13	1.56E+13	1.25E+13
Pu-239	2.41E+04	2.11E+12	1.76E+12	1.44E+12
Pu-240	6.56E+03	1.34E+11	1.12E+11	9.15E+10
Pu-241	1.43E+01	4.26E+10	3.36E+10	2.55E+10
Pu-242	3.74E+05	3.29E+11	2.74E+11	2.25E+11
Pu-244	8.00E+07	9.39E+04	7.83E+04	6.42E+04
Am-241	4.33E+02	2.75E+13	2.28E+13	1.86E+13
Am-243	7.36E+03	5.88E+07	4.90E+07	4.01E+07
Cm-244	1.80E+01	5.81E+10	4.59E+10	3.50E+10
Cf-249	3.57E+02	2.56E+07	2.12E+07	1.73E+07

5.1.2. (TE)NORM

The extrapolation of the present DU inventory to the year 2130 has been performed on the basis of the following considerations:

- The expected volume of unconditioned depleted uranium to be finally disposed of will amount 30 000 m^3 (Verhoef, 2014; p.9).
- Decay of the uranium nuclides and ingrowth of the (relevant) daughters has been taken into account.

- It is assumed that the present distribution of uranium nuclides delivered at COVRA will not change in the forthcoming years.

Taking into account the selection criterion mentioned in Section 3 (half-life), and the above-mentioned considerations, the radionuclides inventory of DU for the year 2130 has been estimated and listed in Table 5-2.

Table 5-2 Estimated DU radionuclide inventory in the year 2130, relevant for the long-term safety

Nuclide	*Half-life [years]*	*Activity [Bq]*
Ra-226	1.60E+03	5.05E+09
Ac-227	2.18E+01	3.68E+10
Th-230	7.54E+04	2.23E+11
Pa-231	3.28E+04	5.14E+10
U-234	2.46E+05	2.30E+14
U-235	7.04E+08	2.30E+13
U-238	4.47E+09	9.98E+14

5.2.Non-heat-generating high level waste section

It is expected that non-heat-generating HLW includes compacted hulls and ends, non-heat-generating wastes from the reprocessing, waste from historical wastes, and the dismantling and decommissioning nuclear facilities. These waste types are described in the following sections.

5.2.1. Compacted hulls and ends

The extrapolation of the present inventory of non-heat-generating HLW from reprocessing such as compacted hulls and ends, to the year 2130 has been performed on the basis of the following assumptions:

- The radioactive decay of each of the nuclides mentioned in Table 4-3 has been calculated up to 2034, i.e. one year after the anticipated shut down of the Borssele NPP.
- The inventories have been estimated for a total number of 1250 CSD-C canisters (Verhoef, E., 2014; Table A-3).

Taking into account the selection criterion mentioned in Section 3 (half-life), and the above-mentioned considerations the radionuclides listed in Table 5-3 represent the total expected inventories of non-heat-generating HLW resulting from the reprocessing of spent fuels in the year 2130.

Table 5-3 Inventory of radionuclides in CSD-C canisters resulting from the reprocessing of spent fuels from NPP's

Nuclide	*Half-Life [years]*	*Inventory CSD-C [Bq]*
C-14	5.70E+03	1.72E+13
Ni-59	7.60E+04	4.48E+14
Ni-63	1.01E+02	1.50E+16
Se-79	3.77E+05	6.87E+10
Sr-90	2.88E+01	2.14E+15
Zr-93	1.53E+06	3.75E+11
Mo-93	4.00E+03	7.25E+12
Nb-94	2.00E+04	6.93E+13

Nuclide	*Half-Life [years]*	*Inventory CSD-C [Bq]*
Tc-99	2.14E+05	2.87E+12
Pd-107	6.50E+06	1.46E+10
Sn-126	2.30E+05	1.10E+11
I-129	1.61E+07	6.62E+10
Cs-135	2.30E+06	8.87E+10
Cs-137	3.00E+01	1.28E+15
Sm-151	9.00E+01	3.04E+13
Ra-226	1.60E+03	1.30E+01
Pa-231	3.28E+04	3.03E+05
Ac-227	2.18E+01	1.20E+02
Np-237	2.14E+06	9.00E+09
U-232	6.88E+01	4.18E+08
U-233	1.59E+05	4.00E+05
U-234	2.46E+05	4.82E+10
U-235	7.04E+08	1.56E+09
U-236	2.37E+07	1.51E+10
U-238	4.47E+09	2.35E+10
Pu-238	8.77E+01	8.14E+13
Pu-239	2.41E+04	1.95E+13
Pu-240	6.56E+03	2.42E+13
Pu-241	1.43E+01	4.94E+13
Pu-242	3.74E+05	1.23E+11
Pu-244	8.00E+07	6.13E+04
Th-229	7.34E+03	1.47E+03
Th-230	7.54E+04	2.10E+04
Am-241	4.33E+02	4.29E+14
Am-242m	1.41E+02	1.09E+11
Am-243	7.36E+03	4.33E+13
Cm-243	2.85E+01	2.08E+10
Cm-244	1.80E+01	1.41E+12
Cm-245	8.50E+03	1.36E+10
Cm-246	4.73E+03	5.97E+09
Cm-247	1.56E+07	3.29E+04
Cm-248	3.39E+05	2.03E+05

5.2.2. Historical wastes

Based on the inventories reported in Table 4-4 and Table 4-5 the total amount of radionuclides with half-lives longer than 10 years cumulated until 2130 has been estimated and summarized in Table 5-4.

Table 5-4 Estimated radionuclide inventory in 2130 originating from historical wastes (WSF)

Nuclide	*Half-life [years]*	*Activity [Bq]*
H-3	1.23E+01	4.76E+10
Ni-63	1.01E+02	2.21E+12
Kr-85	1.08E+01	2.28E+10
Sr-90	2.88E+01	2.63E+13
Nb-94	2.00E+04	4.38E+10
Tc-99	2.14E+05	1.70E+12
Cs-137	3.00E+01	4.55E+13

Nuclide	*Half-life [years]*	*Activity [Bq]*
Sm-147*	1.06E+11	3.17E+04
U-234	2.46E+05	3.57E+09
U-235	7.04E+08	6.26E+08
U-236	2.37E+07	4.94E+06
U-238	4.47E+09	7.60E+08
Pu-238	8.77E+01	5.83E+12
Pu-239	2.41E+04	5.15E+13
Pu-240	6.56E+03	1.35E+12
Pu-241	1.43E+01	1.07E+12
Np-237	2.14E+06	4.65E+07
Pa-231	3.28E+04	1.66E+06
Am-241	4.33E+02	1.48E+13
Am-242m	1.41E+02	8.68E+09
Am-243	7.36E+03	5.93E+10
Cm-243	2.85E+01	1.82E+09
Cm-244	1.80E+01	7.47E+10

* Ingrowth from Pm-147

5.2.3. Waste from dismantling and decommissioning

For the determination of the radionuclide inventory in waste resulting from the dismantling and decommissioning of NPPs at the time of disposal the following assumptions have been made:

- The SMA SA-NPP inventory, compiled by Nagra (Alder, 1994; Table 2) is taken as the basis for the OPERA decommissioning inventory
- The Nagra SA-NPP radionuclide inventory is rigorously scaled down by a conservative[9] factor of 600 m^3/42942 m^3 to match the anticipated amounts of waste generated by dismantling and decommissioning nuclear facilities (Verhoef, 2014; Section 4.3).
- Only radionuclides with a total inventory concentration larger than 1 Bq/g have been taken into account. This is a factor of 10 below the lowest value of exemption levels presently taken into account in the Netherlands (BS, 2001; Tabel 1). The conversion from m^3 to grams has been performed by assuming that the bulk of the decommissioning waste matrix consists of concrete having a density of 2300 kg/m^3.

The resulting radionuclide inventory of the decommissioning waste is summarized in Table 5-5.

Table 5-5 Estimated radionuclide inventory in 2130 originating from decommissioning of the two Dutch NPPs

Nuclide	*Half-life [years]*	*Activity [Bq]*
H-3	1.23E+01	1.25E+10
C-14	5.70E+03	2.07E+12
Cl-36	3.01E+05	1.19E+10
Ar-39	2.69E+02	3.15E+10
Ca-41	1.40E+05	2.51E+10

[9] The estimated volume of 600 m^3 includes waste generated by dismantling and decommissioning nuclear facilities or historical wastes not yet stored at COVRA.

Nuclide	*Half-life [years]*	*Activity [Bq]*
Ni-59	7.60E+04	1.81E+13
Ni-63	1.01E+02	1.14E+15
Sr-90	2.88E+01	3.04E+09
Nb-94	2.00E+04	7.38E+10
Mo-93	4.00E+03	1.37E+10
Tc-99	2.14E+05	3.07E+09
Ag-108m	4.18E+02	1.78E+09
Cs-137	3.00E+01	6.27E+09
Ho-166m	1.20E+03	3.30E+09
Am-241	4.33E+02	1.91E+09

The uncertainties of the inventory originating from decommissioning activities can be considerable but are difficult to establish, primarily since the inventory of Table 5-5 does not specifically relate to the Dutch situation. In addition, there are several aspects that contribute to the uncertainty, as recognized by Nagra (Alder, 1994; p.62):

a) Waste production scenario assumed
b) Development of nuclear power technology
c) Changes occurring in medicine, industry and research activities
d) Improvement of waste arising, treatment and conditioning processes
e) Waste stored and declassified as not being radioactive or recycled
f) Waste which might be exchanged with other countries
g) Wastes that are known to occur, but as yet, are not included as waste sorts
h) Changes in regulations
i) Improved characterisation of the waste

Some, if not all of these aspects may also influence the eventual radionuclide inventory in the Dutch case in 2130. Nagra mentions that a reserve of one order of magnitude is estimated for the radionuclide activities of the SMA waste (Alder, 1994; p.70). Additional reserves of 2 to 3 orders of magnitude are to be considered for a few nuclides arising in the individual waste sorts due to difficulties in the characterisation (correlation factors for BA waste and crud in a.o. SA-KKW, applicability of computer codes for unusual waste, etc.).

For the present Dutch situation, with only a single NPP left and currently no additional NPPs foreseen in the near future, the uncertainties mentioned by Nagra have not been addressed further. However, a more precise estimate is at present difficult to establish unless more detailed information about the radionuclide inventories of the decommissioning waste from the Borssele and Dodewaard NPPs would become available.

5.2.4. Summed inventory non-heat-generating high level waste

Table 5-6 sums the respective radionuclide inventories of the non-heat-generating HLW from reprocessing (Table 5-3), historical waste (Table 5-4) and that of the decommissioning waste (Table 5-5).

Table 5-6 Estimated radionuclide inventory in 2130 of non-heat-generating HLW

Nuclide	*Half-Life [years]*	*Activity*
H-3	1.23E+01	6.01E+10
C-14	5.70E+03	1.93E+13
Cl-36	3.01E+05	1.19E+10
Ar-39	2.69E+02	3.15E+10
Ca-41	1.40E+05	2.51E+10
Ni-59	7.60E+04	4.66E+14
Ni-63	1.01E+02	1.62E+16
Se-79	3.77E+05	6.87E+10
Kr-85	1.08E+01	2.28E+10
Sr-90	2.88E+01	2.16E+15
Mo-93	4.00E+03	7.26E+12
Nb-94	2.00E+04	6.95E+13
Zr-93	1.53E+06	3.75E+11
Tc-99	2.14E+05	4.58E+12
Pd-107	6.50E+06	1.46E+10
Ag-108m	4.18E+02	1.78E+09
Sn-126	2.30E+05	1.10E+11
I-129	1.61E+07	6.62E+10
Cs-135	2.30E+06	8.87E+10
Cs-137	3.00E+01	1.33E+15
Sm-151	9.00E+01	3.04E+13
Ho-166m	1.20E+03	3.30E+09
Ra-226	1.60E+03	1.30E+01
Ac-227	2.18E+01	1.20E+02
Th-229	7.34E+03	1.47E+03
Th-230	7.54E+04	2.10E+04
Pa-231	3.28E+04	1.97E+06
Np-237	2.14E+06	9.04E+09
U-232	6.88E+01	4.18E+08
U-233	1.59E+05	4.00E+05
U-234	2.46E+05	5.18E+10
U-235	7.04E+08	2.18E+09
U-236	2.37E+07	1.51E+10
U-238	4.47E+09	2.43E+10
Pu-238	8.77E+01	8.72E+13
Pu-239	2.41E+04	7.10E+13
Pu-240	6.56E+03	2.56E+13
Pu-241	1.43E+01	5.05E+13
Pu-242	3.74E+05	1.23E+11
Pu-244	8.00E+07	6.13E+04
Am-241	4.33E+02	4.44E+14
Am-242m	1.41E+02	1.18E+11
Am-243	7.36E+03	4.33E+13
Cm-243	2.85E+01	2.26E+10
Cm-244	1.80E+01	1.49E+12
Cm-245	8.50E+03	1.36E+10
Cm-246	4.73E+03	5.97E+09
Cm-247	1.56E+07	3.29E+04

Cm-248	3.39E+05	2.03E+05

5.3. Spent fuel section

5.3.1. HFR and replacement reactor

For calculating the future radionuclide inventory of the HFR and its anticipated replacement reactor, the following assumptions have been made:

- The HFR continues its present operation until 2025;
- The lifetime of the HFR replacement reactor is 50 years;
- The HFR replacement reactor utilizes the same fuel as presently the HFR;
- For the production of molybdenum, LEU uranium targets are applied;
- The spent fuel and control assemblies take 80% of the total amount of the spent fuel baskets; the remainder will hold the spent LEU uranium filters.

The inventories of the HEU and LEU spent fuel and control assemblies, and that of the HEU and LEU uranium filters at the time of disposal have been calculated from the available information for a decay time of 130 years. The results are listed in Table 5-7.

Table 5-7 Estimated radionuclide inventory originating from spent fuel and uranium filters from the HFR in the year 2130, relevant for the long-term safety

Nuclide	*Half-Life [years]*	*Activity [Bq]*		
		Spent Fuel	*U-filters*	*Sum*
C-14	5.70E+03	1.29E+13		1.29E+13
Ni-63	1.01E+02	2.24E+12	4.20E+07	2.24E+12
Se-79	3.77E+05	4.17E+11	1.45E+09	4.18E+11
Kr-85	1.08E+01	2.17E+14		2.17E+14
Sr-90	2.88E+01	1.97E+16	3.35E+14	2.01E+16
Zr-93	1.53E+06	2.31E+12	6.09E+10	2.37E+12
Nb-94	2.00E+04	2.98E+08		2.98E+08
Tc-99	2.14E+05	1.66E+13	4.61E+11	1.71E+13
Pd-107	6.50E+06	4.05E+10	9.68E+08	4.14E+10
Sn-126	2.30E+05	4.95E+11	9.45E+09	5.04E+11
I-129	1.61E+07	3.08E+10		3.08E+10
Cs-135	2.30E+06	1.87E+11		1.87E+11
Cs-137	3.00E+01	2.32E+16		2.32E+16
Sm-147	1.06E+11		1.34E+05	1.34E+05
Sm-151	9.00E+01		2.86E+13	2.86E+13
U-232	6.88E+01		1.40E+06	1.40E+06
U-233	1.59E+05	2.04E+08	5.82E+05	2.05E+08
U-234	2.46E+05	4.03E+12	3.00E+08	4.03E+12
U-235	7.04E+08	5.19E+10	3.96E+10	9.15E+10
U-236	2.37E+07	6.27E+11	1.43E+10	6.41E+11
U-238	4.47E+09	7.60E+10	2.09E+10	9.69E+10
Np-237	2.14E+06	4.08E+11	1.16E+10	4.19E+11
Pu-238	8.77E+01	9.83E+14	2.72E+11	9.83E+14
Pu-239	2.41E+04	1.25E+14	1.64E+13	1.42E+14
Pu-240	6.56E+03	1.15E+14	8.84E+11	1.16E+14
Pu-241	1.43E+01	4.17E+14	4.27E+11	4.17E+14

Nuclide	*Half-Life [years]*	*Activity [Bq]*		
		Spent Fuel	*U-filters*	*Sum*
Pu-242	3.74E+05	4.12E+11	8.10E+06	4.12E+11
Am-241	4.33E+02	1.13E+15	5.99E+11	1.13E+15
Am-243	7.36E+03	2.71E+12		2.71E+12
Cm-244	1.80E+01	5.58E+12		5.58E+12
Th-230	7.54E+04	3.64E+09		3.64E+09
Pa-231	3.28E+04	1.26E+08	1.23E+08	2.49E+08
Th-232	1.41E+10	3.48E+03		3.48E+03

A potential uncertainty of the radionuclide inventory originating from spent fuel from research reactors may arise from the anticipated replacement of the HFR. In the present analysis also spent fuel from an HFR replacement reactor has been considered. In case such a reactor will not be built, the inventory of the spent fuel section of the repository will be significantly less than indicated in Table 5-7.

5.3.2. HOR

The inventory of the HEU spent fuel and control assemblies at the time of disposal has been calculated for a decay time of 130 years. It has been assumed that the HOR will operate for additional 50 years (Kloosterman, 2013).

Table 5-8 Estimated radionuclide inventory originating from spent fuel assemblies and control elements from the HOR in the year 2130, relevant for the long-term safety

Nuclide	*Half-Life [years]*	*Activity [Bq]*
C-14	5.70E+03	3.97E+11
Ni-63	1.01E+02	6.22E+10
Se-79	3.77E+05	1.27E+10
Kr-85	1.08E+01	1.16E+11
Sr-90	2.88E+01	1.63E+14
Zr-93	1.53E+06	7.05E+10
Nb-94	2.00E+04	9.38E+06
Tc-99	2.14E+05	5.07E+11
Pd-107	6.50E+06	1.26E+09
Sn-126	2.30E+05	1.53E+10
I-129	1.61E+07	9.44E+08
Cs-135	2.30E+06	5.78E+09
Cs-137	3.00E+01	1.98E+14
Sm-151	9.00E+01	4.08E+11
U-234	2.46E+05	1.28E+11
U-235	7.04E+08	1.70E+09
U-236	2.37E+07	1.86E+10
U-238	4.47E+09	2.19E+09
Np-237	2.14E+06	1.02E+10
Pu-238	8.77E+01	1.93E+13
Pu-239	2.41E+04	3.63E+12
Pu-240	6.56E+03	3.40E+12
Pu-241	1.43E+01	2.07E+12
Pu-242	3.74E+05	1.23E+10

Nuclide	*Half-Life [years]*	*Activity [Bq]*
Am-241	4.33E+02	3.20E+13
Am-243	7.36E+03	8.09E+10
Th-230	7.54E+04	1.38E+08
Pa-231	3.28E+04	4.64E+06
Th-232	1.41E+10	1.19E+02
U-233	1.59E+05	5.74E+06

5.3.3. Total inventory of the spent fuel section

The accumulated radionuclide inventory for the year 2130, originating from the different spent fuels and uranium filters is summarized in Table 5-9. The total anticipated amount of spent fuel baskets is 150, which is in line with the inventory expected in the OPERA disposal concept (Verhoef, 2014; Table A-2).

Table 5-9 Estimated radionuclide inventory originating from spent fuel assemblies, control assemblies, and uranium filters in the year 2130, relevant for the long-term safety.

Nuclide	*Half-Life [years]*	*Activity [Bq]*		
		Spent Fuel	*U-filters*	*Sum*
C-14	5.70E+03	1.33E+13		1.33E+13
Ni-63	1.01E+02	2.30E+12	4.20E+07	2.30E+12
Se-79	3.77E+05	4.29E+11	1.45E+09	4.31E+11
Kr-85	1.08E+01	2.17E+14		2.17E+14
Sr-90	2.88E+01	1.99E+16	3.35E+14	2.02E+16
Zr-93	1.53E+06	2.38E+12	6.09E+10	2.44E+12
Nb-94	2.00E+04	3.08E+08		3.08E+08
Tc-99	2.14E+05	1.71E+13	4.61E+11	1.76E+13
Pd-107	6.50E+06	4.17E+10	9.68E+08	4.27E+10
Sn-126	2.30E+05	5.10E+11	9.45E+09	5.19E+11
I-129	1.61E+07	3.17E+10		3.17E+10
Cs-135	2.30E+06	1.93E+11		1.93E+11
Cs-137	3.00E+01	2.34E+16		2.34E+16
Sm-147	1.06E+11		1.34E+05	1.34E+05
Sm-151	9.00E+01	4.08E+11	2.86E+13	2.90E+13
U-232	6.88E+01		1.40E+06	1.40E+06
U-233	1.59E+05	2.04E+08	5.82E+05	2.05E+08
U-234	2.46E+05	4.16E+12	3.00E+08	4.16E+12
U-235	7.04E+08	5.36E+10	3.96E+10	9.32E+10
U-236	2.37E+07	6.46E+11	1.43E+10	6.60E+11
U-238	4.47E+09	7.82E+10	2.09E+10	9.91E+10
Np-237	2.14E+06	4.18E+11	1.16E+10	4.29E+11
Pu-238	8.77E+01	1.00E+15	2.72E+11	1.00E+15
Pu-239	2.41E+04	1.29E+14	1.64E+13	1.45E+14
Pu-240	6.56E+03	1.18E+14	8.84E+11	1.19E+14
Pu-241	1.43E+01	4.19E+14	4.27E+11	4.19E+14
Pu-242	3.74E+05	4.24E+11	8.10E+06	4.24E+11
Am-241	4.33E+02	1.16E+15	5.99E+11	1.16E+15
Am-243	7.36E+03	2.79E+12		2.79E+12
Cm-244	1.80E+01	5.58E+12		5.58E+12
Th-230	7.54E+04	3.78E+09		3.78E+09
Pa-231	3.28E+04	1.30E+08	1.23E+08	2.53E+08
Th-232	1.41E+10	3.60E+03		3.60E+03

5.4. Vitrified High Level Waste section

The extrapolation of the present inventory of vitrified HLW to the year 2130 has been performed on the basis of the following assumptions:
The radioactive decay of each of the nuclides mentioned in

- Table 4-11 has been calculated up to 2034, i.e. one year after the anticipated shut down of the Borssele NPP.
- The inventories have been estimated for a total number of 625 CSD-V canisters (Verhoef, E., 2014; Table A-3).

Taking into account the selection criterion mentioned in Section 3 (half-life), and the above-mentioned considerations the radionuclides listed in Table 5-10 represent the total expected inventories of vitrified waste in the year 2130.

Table 5-10 Inventory of radionuclides in CSD-V canisters in the year 2130 resulting from the reprocessing of spent fuels from NPP's

Nuclide	*Half-Life [years]*	*Inventory CSD-V [Bq]*
Se-79	3.77E+05	2.92E+12
Sr-90	2.88E+01	1.76E+17
Zr-93	1.53E+06	6.59E+13
Tc-99	2.14E+05	7.83E+14
Pd-107	6.50E+06	4.24E+12
Sn-126	2.30E+05	2.37E+13
I-129	1.61E+07	1.62E+11
Cs-135	2.30E+06	1.88E+13
Cs-137	3.00E+01	2.82E+17
Pa-231	3.28E+04	1.01E+06
Np-237	2.14E+06	1.48E+13
U-234	2.46E+05	5.20E+10
U-235	7.04E+08	4.08E+08
U-236	2.37E+07	6.55E+09
U-238	4.47E+09	7.18E+09
Pu-238	8.77E+01	5.24E+13
Pu-239	2.41E+04	1.25E+13
Pu-240	6.56E+03	3.18E+14
Pu-241	1.43E+01	1.36E+13
Pu-242	3.74E+05	9.64E+10
Th-230	7.54E+04	3.19E+07
Th-232	1.41E+10	3.52E+01
Am-241	4.33E+02	5.84E+16
Am-243	7.36E+03	1.07E+15
Cm-244	1.80E+01	1.32E+15
Cm-245	8.50E+03	1.26E+13

6. Evaluation of the results

A summarizing overview of the radionuclide inventories of the respective anticipated waste sections of the OPERA disposal concept for the year 2130 is given in Table 6-1. For the LILW waste section only the “Expected Inventory” has been provided in the table below. For the Upper and Lower inventories is referred to Table 5-1.

Table 6-1 Inventory of radionuclides of the separate waste sections for the year 2130

		Activity [Bq]						
Nuclide	***Half-life [years]***	***Spent Fuel (HFR+HOR)***	***U-filters***	***Vitrified HLW (CSD-V)***	***Compacted hulls & ends (CSD-C)***	***LILW Expected***	***Depleted Uranium***	***Other non-heat producing HLW***
H-3	1.23E+01					2.36E+14		6.01E+10
Be-10	1.60E+06					2.46E+09		
C-14	5.70E+03	1.33E+13			1.72E+13	2.75E+12		2.07E+12
Si-32	1.32E+02					3.16E+03		
Cl-36	3.01E+05					6.24E+09		1.19E+10
Ar-39	2.69E+02							3.15E+10
K-40	1.27E+09					6.00E+09		
Ca-41	1.40E+05							2.51E+10
Ti-44	6.00E+01					6.66E+04		
Ni-59	7.60E+04				4.48E+14	1.63E+09		1.81E+13
Ni-63	1.01E+02	2.30E+12	4.20E+07		1.50E+16	7.74E+13		1.14E+15
Se-79	3.77E+05	4.29E+11	1.45E+09	2.92E+12	6.87E+10			
Kr-81	2.10E+05					3.52E+06		
Kr-85	1.08E+01	2.17E+14				7.28E+11		2.28E+10
Sr-90	2.88E+01	1.99E+16	3.35E+14	1.76E+17	2.14E+15	1.54E+13		2.63E+13
Mo-93	4.00E+03				7.25E+12	3.07E+05		1.37E+10
Nb-93m	1.63E+01					4.48E+05		
Nb-94	2.00E+04	3.08E+08			6.93E+13	2.11E+08		1.18E+11
Zr-93	1.53E+06	2.38E+12	6.09E+10	6.59E+13	3.75E+11			
Tc-99	2.14E+05	1.71E+13	4.61E+11	7.83E+14	2.87E+12	1.22E+11		1.70E+12
Pd-107	6.50E+06	4.17E+10	9.68E+08	4.24E+12	1.46E+10			
Ag-108m	4.18E+02					2.26E+08		1.78E+09
Cd-113m	1.41E+01					4.79E+03		
Sn-121m	5.50E+01					1.95E+08		
Sn-126	2.30E+05	5.10E+11	9.45E+09	2.37E+13	1.10E+11			

		Activity [Bq]						
Nuclide	*Half-life [years]*	*Spent Fuel (HFR+HOR)*	*U-filters*	*Vitrified HLW (CSD-V)*	*Compacted hulls & ends (CSD-C)*	*LILW Expected*	*Depleted Uranium*	*Other non-heat producing HLW*
I-129	1.61E+07	3.17E+10		1.62E+11	6.62E+10	3.06E+08		
Ba-133	1.05E+01					1.12E+09		
Cs-135	2.30E+06	1.93E+11		1.88E+13	8.87E+10			
Cs-137	3.00E+01	2.34E+16		2.82E+17	1.28E+15	7.18E+14		4.55E+13
Sm-146	1.00E+08					3.91E+03		
Sm-147	1.06E+11		1.34E+05					3.17E+04
Sm-151	9.00E+01	4.08E+11	2.86E+13		3.04E+13	3.38E+09		
Eu-152	1.35E+01					3.45E+09		
Ho-166m	1.20E+03							3.30E+09
Re-187	4.35E+10					2.61E+07		
Pb-202	5.20E+04					8.21E+05		
Bi-207	3.18E+01					1.29E+07		
Po-209	1.02E+02					7.25E+04		
Pb-210	2.22E+01					1.50E+12		
Ra-226	1.60E+03				1.30E+01	9.45E+11	5.05E+09	
Ac-227	2.18E+01				1.20E+02	3.12E+09	3.68E+10	
Th-229	7.34E+03				1.47E+03	3.26E+07		
Th-230	7.54E+04	3.78E+09		3.19E+07	2.10E+04	7.20E +07	2.23E+11	
Th-232	1.41E+10	3.60E+03		3.52E+01		6.20E+11		
Pa-231	3.28E+04	1.30E+08	1.23E+08	1.01E+06	3.03E+05	9.89E+09	5.14E+10	1.66E+06
Np-237	2.14E+06	4.18E+11	1.16E+10	1.48E+13	9.00E+09	7.52E+08		4.65E+07
U-232	6.88E+01		1.40E+06		4.18E+08	4.38E+07		
U-233	1.59E+05	2.04E+08	5.82E+05		4.00E+05	2.83E+08		
U-234	2.46E+05	4.16E+12	3.00E+08	5.20E+10	4.82E+10	9.47E+10	2.30E+14	3.57E+09
U-235	7.04E+08	5.36E+10	3.96E+10	4.08E+08	1.56E+09	2.73E+12	2.30E+13	6.26E+08
U-236	2.37E+07	6.46E+11	1.43E+10	6.55E+09	1.51E+10	1.06E+07		4.94E+06

		Activity [Bq]						
Nuclide	*Half-life [years]*	*Spent Fuel (HFR+HOR)*	*U-filters*	*Vitrified HLW (CSD-V)*	*Compacted hulls & ends (CSD-C)*	*LILW Expected*	*Depleted Uranium*	*Other non-heat producing HLW*
U-238	4.47E+09	7.82E+10	2.09E+10	7.18E+09	2.35E+10	5.52E+12	9.98E+14	7.60E+08
Pu-238	8.77E+01	1.00E+15	2.72E+11	5.24E+13	8.14E+13	1.56E+13		5.83E+12
Pu-239	2.41E+04	1.29E+14	1.64E+13	1.25E+13	1.95E+13	1.76E+12		5.15E+13
Pu-240	6.56E+03	1.18E+14	8.84E+11	3.18E+14	2.42E+13	1.12E+11		1.35E+12
Pu-241	1.43E+01	4.19E+14	4.27E+11	1.36E+13	4.94E+13	3.36E+10		1.07E+12
Pu-242	3.74E+05	4.24E+11	8.10E+06	9.64E+10	1.23E+11	2.74E+11		
Pu-244	8.00E+07				6.13E+04	7.83E+04		
Am-241	4.33E+02	1.16E+15	5.99E+11	5.84E+16	4.29E+14	2.28E+13		1.48E+13
Am-242m	1.41E+02				1.09E+11			8.68E+09
Am-243	7.36E+03	2.79E+12		1.07E+15	4.33E+13	4.90E+07		5.93E+10
Cm-243	2.85E+01				2.08E+10			1.82E+09
Cm-244	1.80E+01	5.58E+12		1.32E+15	1.41E+12	4.59E+10		7.47E+10
Cm-245	8.50E+03			1.26E+13	1.36E+10			
Cm-246	4.73E+03				5.97E+09			
Cm-247	1.56E+07				3.29E+04			
Cm-248	3.39E+05				2.03E+05			
Cf-249	3.57E+02					2.12E+07		

The information provided in Chapter 5 has been summarized in order to provide an overview of the total inventory in terms of activity and radiotoxicity values for the anticipated waste sections of the OPERA disposal concept. The radiotoxicity equivalents for ingestion, Re_{ing}, have been obtained from the Besluit Stralingsbescherming (BS, 2001; Table 4.1). The results are summarized in Table 6-2.

Table 6-2 Overall activity and radiotoxicity of anticipated waste sections and highest contributing nuclides of the OPERA disposal concept for the year 2130

	Activity [Bq]	Most-Contributing Nuclides	Radiotoxicity [Sv]	Most-Contributing Nuclides
LILW section (upper inventory)	1.38E+15	H-3 Ni-63 Cs-137	2.51E+07	Cs-137 Pu-238 Am-241
LILW section (expected inventory)	1.10E+15	H-3 Ni-63 Cs-137	2.03E+07	Cs-137 Pu-238 Am-241
LILW Section (lower inventory)	8.52E+14	H-3 Ni-63 Cs-137	1.61E+07	Cs-137 Pu-238 Am-241
(TE)NORM	1.25E+15	U-234 U-238	5.74E+07	U-234 U-238
Non-heat-generating HLW section CSD-C	1.97E+16	Ni-63 Sr-90 Cs-137	2.03E+08	Sr-90 Am-241
Non-heat-generating HLW section Decommissioning and historical wastes	1.31E+15	Ni-63 Cs-137 Pu-239	1.91E+07	Pu-238 Pu-239 Am-241
Spent fuel and uranium filters section	4.68E+16	Sr-90 Cs-137 Pu-238 Am-241	1.40E+09	Sr-90 Cs-137 Pu-238 Am-241
Vitrified HLW section CSD-V	5.21E+17	Sr-90 Cs-137 Am-241	2.08E+10	Sr-90 Cs-137 Am-241

The results show that the vitrified waste section (CSD-V) contains the highest inventory, both in terms of activity and radiotoxicity.

7. Concluding remarks

In the layout of the OPERA disposal concept, four waste disposal sections are distinguished:

- low and intermediate level waste (LILW)
- non-heat-generating high level waste (HLW)
- spent fuel (SF)
- vitrified HLW

These waste categories include a variety of radionuclides which together define a source term for the radiological safety assessment calculations in OPERA WP7 *Scenario development and performance assessment*. For each of the four waste sections the radionuclide inventory at the time of disposal, i.e. the year 2130, has been estimated. The radionuclide inventories are provided in the form of tables including:

- Name of radionuclide;
- Radionuclide half-life [yr];
- Amount, expressed as [Bq];

If applicable, daughter nuclides have been taken into account.

For only a very limited number of radionuclides quantitative uncertainty indicators for the amount could be determined. These have been taken into account when applicable. In general the uncertainty in the amounts of nuclides has been accounted for by providing upper and/or maximum values for the different waste sections. The provided inventories can be regarded as an upper envelope and can be taken as input for the long-term safety assessment.

Comparison of the inventories of the different waste categories shows that the vitrified waste section contains the highest overall inventory, in terms of both total activity and total radiotoxicity. For the long-term safety assessment it is not obvious that the radionuclides having the largest contribution to the total inventory or overall radiotoxicity will have the highest impact on the total future dose rate in the biosphere. For the safety assessment the diffusion properties of the individual nuclides, including the retardation properties of Boom Clay, play a major role in the eventual release of nuclides into the biosphere.

The total number of spent fuel baskets, containing spent fuel elements and control assemblies from the research reactors and the uranium filters, amounts 150. This amount is in line with the OPERA disposal concept.

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Appendix 1 Belgian case

The procedure for the compilation of the radionuclide inventory and for the selection of radionuclides with a relevant radiological impact has been described in the SAFIR-2 study (ONDRAF/NIRAS, 2001).

In principle, the process of developing a waste inventory for the Belgian safety assessment has been divided into the administrative management of existing waste packages and the (prospective) management of all the streams of waste packages, present and future.

Chapter 2 of the SAFIR-2 study gives a detailed overview and estimate for the waste inventory as at the end of 1999, including the volumes of waste to be disposed of and their radionuclide content. Most safety assessments were started in early 1997 and the contemporary waste inventory available was used. Section 11.5 of the SAFIR-2 report assesses only the most active waste forms, i.e. the vitrified high-level waste (ZAGALC), the spent fuel (ZAGALS) and the hulls and end pieces (HAGALC2).

The Belgian waste inventory is reported in Section 11.3.2, "Waste and radionuclide inventory" of the SAFIR-2 study.

Section 2.1.5 "The radionuclides taken into account" of the SAFIR-2 report provides an overview of the selection procedure for the radionuclides that are considered relevant for the Belgian safety assessment.

In Belgium a distinction is made between 'groups' and 'classes' of radioactive waste. A 'group' is the most general level of classification used for conditioned wastes, and is based on the potentially suitable disposal method (ONDRAF/NIRAS, 2001; Section 2.1).

- The open group - this group refers to: 'conditioned radioactive waste with radiological characteristics, i.e. the concentration of activity of the radioelements which they contain and the half-life of those radioelements, are sufficiently low and short-lived for alternative solutions to deep disposal to be considered. Their radioactive decay must allow them to attain a radiological level that is viewed as insignificant over a period of time that is compatible with the options for monitoring that decay'. This group therefore contains waste for which the long-term strategic and technical management options are open, hence its name. The waste in this group is not considered in the Belgian long-term safety assessment.
- The geological group - the geological group refers to conditioned radioactive waste whose radiological characteristics, i.e. the concentration of activity of the radioelements which they contain and the half-life of those radioelements, are such that their permanent isolation from the biosphere is imperative, and that this therefore constitutes the only ultimate solution. This permanent isolation is currently regarded as being achievable by burial in deep and stable geological formations. All the conditioned waste discussed in the SAFIR-2 report belongs to this group.

In addition, Belgium considers a waste categorization as follows:

- Category A. This waste category belongs to the 'open' group as the concentrations of activity and the half-lives of the radionuclides present in this waste have values that are 'low'.
- Category B. This waste category encompasses low- and medium-level long-lived waste that does not meet not meet criteria that are associated with category A waste.
- Category C. These conditioned wastes contain very substantial amounts of beta and alpha emitters and generate significant amounts of heat, i.e. above 20 W/m^3 at the time of disposal. They are produced by the vitrification of the effluent arising from

reprocessing of spent fuel and other related waste, or they are spent fuel if this has not been reprocessed.
- Category R. This waste category was created to address the particular nature of the large amounts of waste contaminated by radium which make up the stock of approximately 30'000 m^3 stored on the site of UMICORE at Olen.

Additional information about the categorization is provided in (ONDRAF/NIRAS, 2001; Section 2.1).
The principles of classification of radioactive waste into different categories is important in relation to the handling, transport, and disposal procedures of the waste containers. The radiological characteristics of conditioned waste determine which group and category the waste belongs to, and mainly determine its class. As well as classification, radiological characteristics were used to evaluate the short- and long-term safety of management operations for the following aspects:
- The radiological impact of disposal due to long-lived radionuclides.
- The emission of decay energy and its consequences, i.e. thermal power, dose rate and radiolysis.
- The release of volatile radionuclides.
- The risk of criticality.

With a few exceptions (waste containing radium), all category B waste (low- and medium-level long-lived waste) and C waste (high-level waste) originates from ^{235}U or MOX fuel cycle operations, whether this cycle is open or closed. The spent fuel therefore provided the basis for selecting the radionuclides to be considered. Since the list of all radionuclides is extensive (over 150 radionuclides not counting the minor types), two selection criteria were applied to restrict the amount of radionuclides to be considered in the safety assessments:
1. A minimum half-life of the order of one year, since the management of waste currently only begins some years after the spent fuel source leaves the reactor;
2. A minimum production rate of the order of 0.001 per fission for fission products. A similar production criterion is used for activation products and actinides.

The result of the selection is shown in the table below. The 'greyed' radionuclides were considered of secondary interest. The SAFIR-2 report is not clear about the argumentation to denote radionuclides as of secondary interest. Otherwise, the origin and type of waste can often be used to disregard whole groups of radionuclides. Some examples are the following:
1. It is not necessary to consider activation products in the case of glass produced during fuel reprocessing glass from nuclear power stations since they remain virtually entirely in the structural waste.
2. Waste from the production of MOX fuel is free from radionuclides which are removed during the first extraction cycle (except for ^{241}Am which is gradually formed by radioactive decay of ^{241}Pu).
3. ^{85}Kr needs only be considered in the case of non-reprocessed fuels.

Finally, the reference t_0 used to establish the spectrum of radionuclides of a package is judged sufficiently long ahead, from the end of irradiation of the fuel till the production of the waste at reprocessing time, so that radionuclides with short half-lives (^{54}Mn, ^{144}Ce, ^{242}Cm etc.) can be regarded as being absent. Daughter products are also included if appropriate. The radionuclide inventory of the considered spent fuel types were considered after a cooling period of 50 years (ONDRAF/NIRAS, 2001; Section 11.3.2.3).

With regard to the radionuclide inventory, the SAFIR-2 study did not mention explicitly the aspect of uncertainties in that area.

Table 2-2: *Overview of some important radiological characteristics of the main radionuclides present in concerned waste.*

Nuclide	Half-life (years)	Safety*	Em.**	Nuclide	Half-life (years)	Safety*	Em.**
H-3	12.28	V	beta	Eu-155	4.96	E	beta
C-14	5730	V, L	beta	Ra-226	1600	V, L	alpha
Cl-36	3.01E+05	V, L	beta	Th-229	7340	L	alpha
Ca-41	1.03E+05	L	CE	Th-230	77000	L	alpha
Mn-54	312.7 days	E	CE	Th-232	1.41E+10	V, L	alpha
Co-60	5.271	E	beta	Pa-231	32800	L	alpha
Ni-59	75000	L	CE	U-233	1.59E+05	C, L	alpha
Ni-63	100.1	E	beta	U-234	2.45E+05	L	alpha
Se-79	65000	L	beta	U-235	7.04E+08	C, L	alpha
Kr-85	10.72	E, V	beta	U-236	2.34E+07	L	alpha
Sr-90	28.6	E	beta	U-238	4.47E+09	L	alpha
Zr-93	1.53E+06	L	beta	Np-237	2.14E+06	L	alpha
Nb-94	20300	L	beta	Pu-238	87.75	E, L	alpha
Tc-99	2.13E+05	L	beta	Pu-239	24131	C, L	alpha
Ru-106	368.2 days	E	beta	Pu-240	6569	E, L	alpha
Pd-107	6.50E+06	L	beta	Pu-241	14.4	C, L	beta
Ag-108m	127	E	CE, beta+	Pu-242	3.76E+05	L	alpha
Ag-110m	249.85 days	E	beta	Pu-244	8.26E+07	L	alpha
Sn-126	1.00E+05	L	beta	Am-241	432.2	E, L	alpha
Sb-125	2.77	E	beta	Am-242m	152	L	TI
I-129	1.57E+07	V, L	beta	Am-243	7380	L	alpha
Cs-134	2.062	E	beta	Cm-242	163.2 days	E, L	alpha
Cs-135	2.30E+06	L	beta	Cm-243	28.5	E, L	alpha
Cs-137	30.17	E	beta	Cm-244	18.11	E, L	alpha
Ce-144	284.3 days	E	beta	Cm-245	8500	L	alpha
Pm-147	2.62	E	beta	Cm-246	4750	L	alpha
Sm-151	90	E	beta	Cm-247	1.56E+07	L	alpha
Eu-154	8.8	E	beta	Cm-248	3.39E+05	L	alpha

** The 'Safety' column shows the aspect(s) affected by the radionuclide: L (long-term impact), E (energy emission), V (volatility), C (criticality).*
*** The 'Emission' column lists the dominant type of decay: emission of particles α, β, β⁻, electronic capture (EC) or isomeric transition (IT).*

References

(ONDRAF/NIRAS, 2001) ONDRAF/NIRAS, SAFIR 2 - Safety Assessment and Feasibility Interim Report 2, NIROND 2001-06 E December 2001, 1-1490.

Appendix 2 French case

The French Dossier 2005 lists information about the radiological inventory of the different waste packages as well as a procedure to select radionuclides that are relevant for the safety assessment.
In France a distinction is conventionally made between the following categories of high-level and long-lived wastes (HLLL waste), which have their own characteristics (ANDRA, 2005; Section2.1.2).

- The so-called B wastes are characterised by low or medium ß-γ activity and, as a result, by a low release of heat. They represent the largest number of packages, as well as the greatest variety of conditioning types. Their total inventory in long-lived radioisotopes, relatively lower than that of the other categories hereafter, is distributed over the large volume which they represent.
- The C wastes consist of the fission products and minor actinides separated during the recycling of the fuels. Their high ß-γ activity causes a large heat release, which decreases over time, mainly with the decrease in radioactivity of the fission products with a medium half-life (cesium 137, strontium 90). The conditioning of these wastes consists of incorporating them in a glass matrix; the confinement capability of this material is particularly high and durable under favourable physical / chemical environmental conditions.
- In addition, the spent fuels (CU waste), although they are not strictly speaking waste, but recyclable material, also have a high activity and, as a result, release a significant amount of heat. This release of heat is due to their content in fission products with a medium half-life, and plutonium and americium (mainly obtained from the disintegration of plutonium); these latter two elements lead to a slower decrease in heat over time. Other specificities are the large dimensions of fuel unloaded from the electronuclear reactors, if it were decided to dispose of them as is, as well as a larger content in fissional materials associated with the question of a criticality risk.

The radiological inventory of waste packages concerns a large list of radionuclides, including fission products (PF), activation products (PA) and actinides. Those radionuclides have been subdivided into three classes (Andra, 2005; Section 2.1.7):

- 44 short-lived radionuclides, with a half-life less than 6 years,
- 16 intermediate-lived radionuclides, with a half-life between 7 and 30 years,
- 84 long-lived radionuclides, with a half-life longer than 31 years.

Nickel 63 is a particular case due to its half-life (100 years). It is an activation product present in many packages. Its activity is especially high during the first centuries, or is even dominating in ILW/LLW B waste types.

The procedure to select relevant radionuclides for performing the French safety assessment has been elucidated as part of the Dossier 2005 (Andra, 2005; Section 5.3.1.5). This procedure is summarized in the following paragraphs.

The Dossier 2005 impact calculations were preceded by an exercise consisting in selecting 144 radionuclides with half-lives long than 6 months that are liable to be present in the waste packages, and whose initial activity is reported. From these 144 nuclides those have been selected that contribute most to the radiological impact in the normal evolution scenario (SEN; Scénario d'Évolution Normale). A preliminary calculation revealed that the high chemical retention of actinides in the Callovo-Oxfordian argillites delays the onset of the peak molar flux exiting the host formation until well after the study's million-year end-point. Over this period, the quantity of activity exiting the Callovo-Oxfordian layer is negligible, and any such activity remains almost entirely contained in the near field. Consequently, no actinides were selected for the SEN's global impact calculation.

For fission and activation products, a two-step selection process was adopted.

- The first phase involved eliminating any radionuclides with a half-life of less than 1,000 years, which all decay in the host formation. Based on the observation that the theoretical transfer times are in the region of a few hundred thousand years, or at least 100 times the half-life of such radionuclides, the radionuclides' initial activity decays by around 30 orders of magnitude at least, which would seem enough to render them negligible.
- The second step involved considering the radionuclides with half-lives longer than 1,000 years and eliminating those that could not contribute to the radiological impact. The remaining elements were organised into groups according to their class of geochemical behaviour, in order to group together the radionuclides that are expected to migrate together. Within each group, any elements with minimal activity were disregarded (for example, ^{99}Tc was selected rather than ^{98}Tc or ^{97}Tc, which are present in smaller quantities). A comprehensive calculation concerning all the radionuclides with half-lives longer than 1,000 years was performed in order to validate this approach and to confirm that it did not lead to the elimination of any important radionuclides. Many radionuclides are strongly sorbed in argillite, which further delays and reduces their contribution. This is particularly the case for ^{137}La, ^{146}Sm, ^{150}Gd, ^{154}Dy, ^{176}Lu, ^{182}Hf, ^{186m}Re, ^{208}Bi and ^{210m}Bi.

After considering all types of waste package, this approach ultimately yielded a selection of 15 radionuclides with half-lives in excess of 1,000 years for use in the SEN impact calculation, together with ^{93m}Nb (a daughter of ^{93}Zr). These nuclides are listed in the following table (copied from (Andra, 2005)).

Isotopes	From anionic/cationic form	Radioactive half-life [years]
^{129}I	anionic form	15,700,000 years
^{107}Pd	cationic form	6,500,000 years
^{135}Cs	cationic form	2,300,000 years
^{10}Be	cationic form	1,600,000 years
^{93}Zr (→ ^{93}Nb)	cationic form	1,530,000 years (TNb93m ≅ 16.4 years)
^{36}Cl	anionic form	302,000 years
^{99}Tc	cationic form	213,000 years
^{41}Ca	cationic form	103,000 years
^{126}Sn	cationic form	100,000 years
^{59}Ni	cationic form	75,000 years
^{79}Se	anionic form	65,000 years
^{94}Nb	anionic form	20,300 years
^{14}C	anionic form	5,730 years
^{93}Mo	anionic form	3,500 years
^{166}Ho	cationic form	1,200 years

Radionuclides considered in a normal evolution scenario

The aspect of uncertainties concerning waste package inventory has also been addressed in the Dossier 2005 (ANDRA, 2005; Section 6.2.4). With respect to uncertainties of the radiological inventory the following aspects apply:

- uncertainties arising strictly from the inventory, in terms of number of packages, radionuclides or toxic chemical elements, which only affect the input data for the impact calculation. They are largely covered by the bounding choices made when dimensioning the inventory model. However, it could not be demonstrated that this

model was based on the most pessimistic case. In all events, variations on any given family of packages did not seem to be sufficient to cast significant doubt over the calculations;

- uncertainties that may, through the packaging modes, have an influence on thermal aspects, package irradiation rate or radiolysis gas emission rate. These are also widely covered by the inventory model hypotheses.

The main uncertainty, in this context, relates both on the stock of older wastes which have not yet been the subject of a detailed characterisation, or whose characterisation, at the time, considered only particular radionuclides, as well on future packages whose characteristics have not been completely defined presently.

References

(ANDRA, 2005) ANDRA, Dossier 2005, Tome: Safety evaluation of a geological repository, December 2005, 1-782.

Appendix 3 Swiss case

The procedure for the selection of radionuclides which are considered relevant in the Swiss Safety Case (Opalinus clay) is described in (Nagra, 2002; Appendix 5): Selection of Safety-Relevant Radionuclides. The report describes the models, codes and data used in the l radiological safety assessment of a deep geological repository in Opalinus Clay in the potential siting area of the Zürcher Weinland in northern Switzerland and designed for the disposal of:

- spent fuel, in the form of fuel assemblies containing UO_2 or mixed-oxide (MOX) fuel,
- vitrified high-level waste (HLW) from the reprocessing of spent fuel, and
- long-lived intermediate-level waste (ILW).

In a safety assessment, in principle all radionuclides should be considered with the potential to give rise to significant radiation doses following the disposal of radioactive waste in a repository, or the parents of such radionuclides. However, the number of radionuclides present in the inventory is prohibitively large for detailed model-chain calculations to be performed for all of them. A set of radionuclides for consideration in the Swiss safety assessment has therefore been selected by means of a simple screening analysis which took account of:

- a period of complete containment of radionuclides;
- a generic duration of near field release;
- dilution in the Quaternary aquifer;
- ingestion of contaminated aquifer water (drinking water dose).

The solubility limitation in the near field, transport and sorption of radionuclides in bentonite (SF/HLW) and cementitious materials (ILW), retardation in the geological barriers, accumulation of radionuclides in the surface environment and dose pathways other than ingestion of drinking water were neglected.

The selection of safety-relevant radionuclides is an iterative process:

- In a first step, the set of safety-relevant radionuclides is selected on the basis of provisional datasets (inventory, dilution rate, etc.) and, thereafter, provisional model calculations are performed for the selected radionuclides.
- In a second step, the selection is revised and finalised, based on definitive datasets and on the results of provisional model calculations.

After the second step the definitive model calculations are performed for the final selection of safety-relevant radionuclides.

The safety-relevant radionuclides are selected according to their drinking water dose $D^{(i)}$ [mSv a^{-1}] calculated by:

$$D^{(i)} = \frac{N^{(i)} I_{wat} D_{ing}^{(i)}}{T_c Q}$$

where:

- $N^{(i)}$ [Bq] is the inventory of radionuclide *i* at failure time (i.e. end of complete containment). For SF/HLW and ILW, the failure time is set at 1'000 years and 100 years, respectively.
- T_c is a generic duration of near field release. In the case of SF/HLW, the generic release duration is taken to be 1000 years, based on extremely pessimistic assumptions regarding waste dissolution and near field transport times. In the case of ILW, a generic near field release duration of 100 years is assumed.
- Q is the water flow in Quaternary aquifer, set at 10^6 m^3 a^{-1};
- I_{wat} is the total annual drinking water intake by an individual, set at 0.73 m^3 a^{-1};

- $D_{ing}^{(i)}$ is the effective dose per unit ingestion of radionuclide *i*, [mSv Bq^{-1}].

All activation and fission products for which the dose is larger than 10^{-5} mSv a^{-1} are selected as being relevant to safety. This value is four orders of magnitude below the regulatory guideline of 0.1 mSv a^{-1}. Additionally, ^{3}H, ^{90}Sr and ^{137}Cs are included in the list of safety-relevant radionuclides. They serve as indicators for phenomena that occur at relatively early times after the end of emplacement (e.g. pinhole release for defective SF/HLW canisters). Short-lived daughter radionuclides are implicitly taken into account by adding their dose coefficients to the longer-lived parent radionuclide. This corresponds to the assumption of equilibrium between daughter and parent radionuclide. The dose contributions of some daughter radionuclides are negligible, because their dose coefficients are very small relative to the parent (e.g. ^{108}Ag, ^{137}Ba).

The actinide chains 4N, 4N+1, 4N+2 and 4N+3 are considered to be safety-relevant, irrespective of any safety criterion. Decay-chain branching (4N+2) and recombination (4N, 4N+2) are explicitly taken into account. In the Swiss selection procedure this was done using the codes STMAN, PICNIC and TAME. Short-lived daughter radionuclides are implicitly taken into account by adding their dose coefficients to the longer-lived parent radionuclides.

The basis of Nagra's database of dose coefficients is the revised Swiss "Strahlenschutzverordnung" (StSV, 1994). Data not available in (StSV, 1994) were taken from (ICRP, 1996), because (ICRP, 1996) is the original source from which the values in (StSV, 1994) were derived.

A potential source of bias arises in the treatment of ^{222}Rn in the 4N + 2 chain, which is assumed to be in radioactive equilibrium with its parent throughout the assessment model chain. ^{222}Rn is the immediate daughter of ^{226}Ra and is a noble gas with a half-life of about 4 days; it decays through a series of very short-lived radionuclides (half-lives 27 minutes or less) to ^{210}Pb (half-life 22.3 years). The assumption of radioactive equilibrium effectively ignores any transport of ^{222}Rn from its point of origin to its point of decay. This assumption is valid in the near field and geosphere, and between the majority of biosphere compartments, where transport times are considerably longer than 4 days. If ^{226}Ra is present in the soil, then a fraction of the ^{222}Rn produced could escape to the air before decaying. This could have either favourable or unfavourable consequences for safety, depending on whether the ^{222}Rn is quickly dispersed in the atmosphere, or accumulates in closed spaces. It is currently estimated that the net bias due to the assumption of ^{222}Rn equilibrium in biosphere modelling is small, but this is an issue that might deserve attention in future assessment calculations.

Nagra does not treat the aspect of uncertainties in relation to radionuclide inventories.

References

(ICRP, 1996): International Commission on Radiological Protection: Age-dependent doses to members of the public from intake of radionuclides, Part 5: Compilation of ingestion and inhalation dose coefficients. ICRP Publication 72, Annals of the ICRP 26. Pergamon Press, Oxford and New York, 1-94.

(Nagra, 2002) Nagra, Project Opalinus Clay: Models, codes and data for safety assessment. Demonstration of disposal feasibility for spent fuel, vitrified high-level waste and long-lived intermediate-level waste (Entsorgungsnachweis). Nagra Technical Report NTB 02-06. Nagra, Wettingen, Switzerland, 1-360.

(StSV, 1994): Strahlenschutzverordnung vom 22.6.1994 (Stand 4.4.2000), StSV, SR 814.501, Switzerland.

Appendix 4 Spent fuel and uranium filters

Note from the author: this Appendix copies the information from: *(COVRA, 2013b), Excel file "HLW-NRG.xlsx", Worksheet "unreprocessed", Annex to e-mail, E. Neeft, "OPERA: NRG111", 3 October 2013.*

Uraniumfilters

Year delivery to COVRA	2004	2005	2006	2007	2008	2009	2010	2011	2012		shipment	1	canister
Number of shipments	0	0	1	1	1	0	1	0	0				

presuming maximum filling rate per canister

batch of 23 canisters transported to COVRA between 2006 and 2013 and calculation

before irradiation					after irradiation		
U235	17127	gram		U235	15869.28	1799.30	gram
U(tot)	19206	gram		U(tot)	18184.07	2672.34	gram
				Pu(tot)	9.21	0	gram
Weight including actinides	max	215820	gram		max	min	

Activity radionuclides calculated by NRG with irradiation period 150 hours presuming power HFR 50 MW; maximum values after cooling period of 2 years

C-14		not indicated	
C-60		not indicated	
Ni-59		not indicated	
Ni-63	2559300	Bq/canister	
Se-79	46546000	Bq/canister	
Rb-87		not indicated	
Sr-90	9.188E+13	Bq/canister	
Zr-93	1952000000	Bq/canister	
Mo-93		not indicated	
Nb-94		not indicated	
Tc-99	14787000000	Bq/canister	
Pd-107	31032000	Bq/canister	
Sn-126	303080000	Bq/canister	
I-129		not indicated	
Te-129	248840000	Bq/canister	not requested by NRG
Te-129m	360320000	Bq/canister	not requested by NRG
Cs-135		not indicated	
Cs-137		not indicated	is contained in liquid waste; processed as LILW
Sm-147	4305.3	Bq/canister	

Sm-151	1.876E+12	Bq/canister	
Eu-154	98794000000	Bq/canister	
Cm-248		not indicated	
Pu-244		not indicated	
Cm-244		not indicated	
Pu-240	1.13E+09	Bq/canister	
U-236	4.59E+08	Bq/canister	
Th-232		not indicated	
U-232	6964.1	Bq/canister	
Cm-245		not indicated	
Pu-241	2.37E+10	Bq/canister	
Am-241	79905000	Bq/canister	
Np-237	14673000	Bq/canister	
U-233	18650	Bq/canister	
Th-229		not indicated	
Cm-246		not indicated	
Pu-242	10279	Bq/canister	
Am-242m		not indicated	
U-238	26470000	Bq/canister	
Pu-238	6.68E+08	Bq/canister	
U-234	9622800	Bq/canister	
Th-230		not indicated	
Ra-226		not indicated	
Pb-210		not indicated	
Cm-247		not indicated	
Am-243		not indicated	
Pu-239	2.08E+10	Bq/canister	
U-235	1.27E+09	Bq/canister	
U-235m	2.08E+10	Bq/canister	not requested
Pa-231	55420	Bq/canister	
Ac-227		not indicated	
Cm-242	136930	Bq/canister	not requested
Pu-236	88922	Bq/canister	not requested
U-237	581290	Bq/canister	not requested

bulk chemical composition after processing Na2U2O7*6H20 (solid)
filter stainless steel
181 Watt

Fuel elements High Flux Reactor (Petten)

Year delivery to HABOG	2004	2005	2006	2007	2008	2009	2010	2011	2012		shipment	1	canister
Number of shipments	7	1	6	4	2	2	0	1	0				

presuming maximum filling rate

per canister					per canister		
before irradiation					after irradiation		batch of 23 canisters transported to COVRA between 2003 and 2011
U235	15147	gram		U235	7799.10	4035.02	gram
U(tot)	16830	gram		U(tot)	10072.38	5293.99	gram
chemical composition UAlx/Al (solid)				Pu(tot)	58.05	18.99	gram
Weight including actinides	231000	gram			max	min	

foreseen in future LEU instead of HEU

Other radionuclides to be calculated by irradiation period and neutron flux High Flux Reactor

typical actvity 5 PBq; maximum 15PBq

Fuel elements Hoger Onderwijs Reactor (Delft)

Year delivery to HABOG	2004	2005	2006	2007	2008	2009	2010	2011	2012		shipment	1	canister
Number of shipments	2	0	0	0	0	0	0	1	0				

presuming maximum filling rate

per canister					per canister		
before irradiation					after irradation		batch of 3 canisters transported to COVRA between 2004 and 2011
U235	6270	gram		U235	2737.04	1786.85	gram
U(tot)	6748.5	gram		U(tot)	3547.47	2318.19	gram
chemical composition UAlx/Al (solid)				Pu(tot)	2.82	2.06	gram
Weight including actinides	148500	gram			max	min	

foreseen in future LEU instead of HEU

Other radionuclides to be calculated by irradiation period and neutron flux Hoger Onderwijs Reactor

typical actvity 2 PBq; maximum 5 PBq

Appendix 5 CSD-C

Notes from the author: this Appendix copies the information from: *(COVRA, 2013b), Excel file "HLW-NRG.xlsx", Worksheet "CSD-C", Annex to e-mail, E. Neeft, "OPERA: NRG111", 3 October 2013;*

Colis Standard de Dechects Vitrifies (CSD-C)

300 AQ 055

Year delivery to COVRA	2004	2005	2006	2007	2008	2009	2010	2011	2012
Number of shipments with 20 canisters	0	0	0	0	0	2	0	2	0
Number of shipments with 24 canisters	0	0	0	0	0	0	2	2	0

Compaction	25000	kN		
Composition by weight				
Zircaloy	393	kg	typical values	
Inconel	19	kg	typical values	
Stainless steel	116	kg	typical values	
Total waste	528	kg	typical values	
Total weight	700	kg	typical values	
Max	850	kg	maximum guaranteed	

For each CSD

	Act (TBq) - typical values at production							Act(TBq) - typical values at production
Cs-137	1.50E+01		6.50E+01	max			C-14	1.40E-02
Sr-90/Y-90	2.80E+01	Sr-90	1.15E+02	max	/' is or		Co-60	7.70E+01
Am-241+Am-243	3.50E-02						Ni-63	2.70E+01
Am-243	see above						Se-79	5.50E-05
Cm-244	9.50E-02						Zr-93	3.00E-04
Np-237	7.20E-06						Tc-99	2.30E-03
Pu-241	1.00E+01		7.50E+01	max			I-129	5.30E-05
							Cs-135	7.10E-05
							Sm-151	6.00E-02
							Eu-154	7.60E-01
thermal power	90	Watt	max					

Empirical maximum values at production (not guaranteed) ; 24 canisters arrival in 2011 to COVRA; canisters produced in 2008

C-14	see typical value		
Co-60	see typical value		
Ni-59	0.358998392	TBq/canis.	
Ni-63	see typical value		
Se-79	see typical value		
Rb-87	not measured		
Zr-93	see typical value		
Mo-93	0.005918734	TBq/canis.	
Nb-94	0.05571956	TBq/canis.	
Nb-95	7.86512E-37	TBq/canis	not requested by NRG
Tc-99	see typical value		
I-129	see typical value		
Np-237	see typical value		
Pd-107	1.17103E-05	TBq/canis.	
Sn-126	8.83689E-05	TBq/canis	
Sm-147	not measured		
Pm-147	0.289157905	TBq/canis	not requested by NRG
Sm-151	see typical value		
Eu-154	see typical value		
Cm-242	6.9435E-11	TBq/canis	
Cm-243	0.000279947	TBq/canis	
Cm-244	see typical value		
Cm-245	1.09657E-05	TBq/canis	
Cm-246	4.86027E-06	TBq/canis	
Cm-247	2.63141E-11	TBq/canis	
Cm-248	1.62484E-10	TBq/canis	
232-U	1.09E-06	TBq/canis	
233-U	3.2E-10	TBq/canis	
234-U	3.06E-06	TBq/canis	
235-U	1.25E-06	TBq/canis	
U-236	1.21E-05	TBq/canis	
U-238	1.88E-05	TBq/canis	
Pu-238	0.164683	TBq/canis	
Pu-239	0.015635	TBq/canis	
Pu-240	0.019632	TBq/canis	
Pu-241	typical value		
Pu-242	9.85E-05	TBq/canis	
Pu-244	4.91E-11	TBq/canis	
Th-229	1.19E-12	TBq/canis	
Th-230	1.68093E-11	TBq/canis	
Th-232	2.02E-15	TBq/canis	
Am-241	see typical value		

Am-242m	0.000155953	TBq/canis
Am-243	see typical value	
Ra-226	1.09186E-14	TBq/canis
Pb-210	not measured	
Pa-231	2.43026E-10	TBq/canis
Ac-227	3.80611E-12	TBq/canis

Appendix 6 CSD-V

Notes from the author: this Appendix copies the information from: *(COVRA, 2013b), Excel file "HLW-NRG.xlsx", Worksheet "CSD-V", Annex to e-mail, E. Neeft, "OPERA: NRG111", 3 October 2013;*

Colis Standard de Dechects Vitrifies (CSD-V)

300 AQ 016

Year delivery to COVRA	2004	2005	2006	2007	2008	2009	2010	2011	2012		shipment	28	canisters
Number of shipments	1	1	1	2	0	0	1	0	1				

Characteristics glass

Chemical composition

	Min (wt%)	Max (wt%)	average (wt%) not available			
SiO2	42.4	51.7		density	2.8	gram/cm3
B2O3	12.4	16.5				
Al2O3	3.6	6.6				
Na2O	8.1	11				
Fe2O3	0	4.5				
NiO	0	0.5				
Cr2O3	0	0.6				
P2O5	0	1				
Li2O	1.6	2.4				
ZnO	2.2	2.8				
CaO	3.5	4.8				
RuO2+Rh+Pd	0	3				
oxides(fission products+Zr+actinides)	4.2	18.5				

For each CSD

	Act (TBq) - typical values at production				
Cs-137	6200		6600	max	
Sr-90/Y-90	4100	Sr-90	4625	max	/' is or
Am-241	130				
Am-243	2.6				
Cm-244	320				
Np-237	0.048				
Pu	24				

U	0.0006				
thermal power	2000	Watt	max		

Empirical maximum values at production (not guaranteed) at production; 28 canisters arrival in 2012 to COVRA; canisters produced between 1995 and 2002

C-14	not measured		
Ce-144 +Pr-144	1580	TBq/canis.	not requested by NRG
Co-60	3.022	TBq/canis.	
Ni-59	not measured	TBq/canis.	
Ni-63	not measured		
Sb-125	89.54	TBq/canis.	not requested by NRG
Se-79	0.020143782	TBq/canis.	
Rb-87	not measured		
Ru-106 +Rh-106	1240	TBq/canis.	not requested by NRG
Sr-90	3965	TBq/canis.	
Sr-90 +Y-90	7930	TBq/canis.	not requested by NRG
Zr-93	0.10540186	TBq/canis.	
Mo-93	not measured		
Nb-94	not measured		
Tc-99	1.2529	TBq/canis.	
Pd-107	0.00678	TBq/canis.	
Sn-126	0.038	TBq/can.	
I-129	not measured		
Cs-134	1109	TBq/canis.	not requested by NRG
Cs-135	0.03008	TBq/canis.	
Cs-137	6183	TBq/canis.	
Sm-147	not measured		
Sm-151	not measured		
Eu-154	226	TBq/canis.	
Cm-248	not measured		
Pu-244	not measured		
Cm-244	reported as weight	59	gram/canister
Pu-240	see average values		
U-236	see average values		
Th-232	not measured		
U-232	not measured		
Cm-245	reported as weight	3.2	gram/canister
Pu-241	see average values		
Am-241	reported as weight	891	gram/canister
Np-237	reported as weight	910	gram/canister
U-233	not measured		
Th-229	not measured		
Cm-246	not measured		
Pu-242	see average values		

Am-242m	not measured		
U-238	see average values		
Pu-238	see average values		
U-234	see average values		
Th-230	not measured		
Ra-226	not measured		
Pb-210	not measured		
Cm-247	not measured		
Am-243	reported as weight	235	gram/canister
Pu-239	see average values		
U-235	see average values		
Pa-231	not measured		
Ac-227	not measured		

Empirical average values at production (not guaranteed); 28 canisters arrival in 2012 to COVRA; canisters produced between 1995 and 2002

average	stdev		
0.016546429	0.00088212	U-234	% ov. U
0.801046429	0.070031265	U-235	% ov. U
0.391057143	0.047705691	U-236	% ov. U
98.79201429	0.068824972	U-238	% ov. U
933.9643	404.901		gram U/canister
1.913446429	0.377809695	Pu-238	% ov. Pu
57.151375	2.27287653	Pu-239	% ov. Pu
25.34982143	0.617256178	Pu-240	% ov. Pu
9.306875	0.501872708	Pu-241	% ov. Pu
6.278375	0.908315843	Pu-242	% ov. Pu
14.67857	7.50335		gram Pu/canister
not measured		not measured	
U-232	U-233	Pu-244	

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