nagra National Cooperative for the Disposal of Radioactive Waste

TECHNICAL REPORT 01-01

Model Radioactive Waste Inventory for Reprocessing Waste and Spent Fuel

December 2002

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Summary

This report describes a model inventory concerning spent fuel (SF), high level vitrified ("glass") waste from reprocessing (HLW) and long lived intermediate (ILW) wastes. The inventory describes the conditioned and packaged SF, HLW and ILW that are expected to be produced by the 5 operational NPPs (3.2 GW(e) over 60 years (192 GWa(e)), along with a more general 300 GWa(e) scenario) arising from both reprocessed and unreprocessed spent fuel, including mixed oxide (MOX), obtained by the recycling of Uranium and Plutonium obtained from reprocessing. In addition, ILW from other sources are considered to be included in the overall uncertainties and not modelled in any detail.

The waste arisings are described by 17 waste sorts, each defined using an average waste package description. Radionuclide activities, other radiological characteristics, material content, specific properties and predicted maximum values of the various attributes of the packages are provided in a data base of the waste sorts. This data is presented and discussed, along with a short description of the origin of the raw waste.

The summation of the waste volumes, activities and materials gives an overview of the repository waste contents. Uncertainties associated with the repository inventory have been analysed. The results provide an indication of the possible additional waste volume arisings to be considered for repository planning.

Zusammenfassung

Der vorliegende Bericht beschreibt ein modellhaftes Inventar, das für abgebrannten Brennstoff ("SF/spent fuel"), hochaktive Abfälle aus der Wiederaufarbeitung (HAA) sowie langlebige mittelaktive Abfälle (LMA) entwickelt wurde. Das Inventar umfasst die konditionierten und verpackten Abfälle, die aufgrund einer 60-jährigen Betriebszeit der 5 schweizerischen Kernkraftwerke (192 GWa(e) bei 3.2 GW(e)) und im Rahmen eines alternativen Szenarios von 300 GWa(e) zu erwarten sind. Bei den Abfällen handelt es sich um nicht wiederaufgearbeiteten Brennstoff (inkl. MOX-Brennstoff) und die Rückstände aus der Wiederaufarbeitung. Langlebige mittelaktive Abfälle aus anderen Quellen werden im Rahmen von konservativen Betrachtungen des Volumens berücksichtigt.

Die Abfälle werden in 17 Abfallsorten unterteilt, die jeweils durch ein mittleres Abfallgebinde repräsentiert werden mit Angaben zu Aktivitäten der Radionuklide und weiteren radiologischen Eigenschaften, Bestandteilen und sonstigen Eigenschaften. Alle Daten sind in einer Datenbank gespeichert und werden hier – zusammen mit einer Beschreibung der Herkunft der Rohabfälle – präsentiert.

Den Abschluss bildet die Zusammenfassung des gesamten Abfallvolumens und des gesamten Aktivitäts- und Material-Inventars. Zusammen mit einer konservativen Betrachtung bilden diese Daten die Planungsgrundlage für die in einem geologischen Tiefenlager zu lagernden Abfälle.

Résumé

Ce rapport présente un inventaire-type des éléments combustibles usés (SF), déchets de haute activité vitrifiés (HLW) et déchets de moyenne activité à vie longue (ILW). L'inventaire décrit les SF, HLW et ILW, conditionnés et mis en conteneurs, que l'on estime devoir être produits par les 5 réacteurs nucléaires des centrales en exploitation, selon deux scénarios: l'un basé sur une production de 192 GWa(e) (3.2 GW(e) sur une période de 60 ans) et un autre, plus général, utilisant le chiffre de 300 GWa(e). Les déchets proviennent des éléments combustibles retraités et non retraités, ainsi que du combustible MOX (Métal OXyde), obtenu par le recyclage de l'uranium et du plutonium issus du retraitement. Par ailleurs, les déchets moyennement radio-actifs émanant d'autres sources sont pris en compte dans les marges générales d'incertitude, mais ne sont pas décrits en détail dans l'inventaire.

Les déchets sont divisés en 17 catégories. Chaque catégorie est définie par les caractéristiques moyennes des colis qu'elle comprend. Une base de données rassemble l'information concernant l'activité des radionucléides, les autres caractéristiques radiologiques, les matériaux présents, les propriétés spécifiques et les valeurs maximales prévues pour les différentes caractéristiques des colis. Ces données font l'objet d'une présentation et d'explications, et sont accompagnées d'un bref descriptif indiquant l'origine des déchets.

Considérés dans leur ensemble, les volumes de déchets, les niveaux d'activité et les matériaux permettent de se représenter le contenu du dépôt. Les incertitudes concernant l'inventaire du dépôt font l'objet d'une analyse séparée. Les résultats de cette analyse permettent d'estimer les volumes de déchets supplémentaires que le dépôt devrait éventuellement accueillir.

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1 Introduction

1.1 Objective

For the disposal of the Swiss radioactive waste, two repositories are planned. One repository is called the L/ILW repository, and will be for low and intermediate level wastes that do not contain significant amounts of long-lived radionuclides ("SMA" waste: Schwach- und Mittel-aktive Abfälle). The other repository will be for intermediate level waste ILW that contain significant amounts of long-lived radionuclides ("LMA" waste: Langlebige Mittelaktive Ab-fälle), high level waste HLW ("HAA" waste: Hochaktive Abfälle) and spent fuel SF ("BE": Abgebrannte Brennelemente). For this report the English notation for radioactive wastes will be adopted, and hence this repository is designated as the SF/HLW/ILW repository.

This report presents a model inventory for the SF/HLW/ILW repository and is based on all information available, up to and including December 2001.

The objective is to quantify and characterise all SF/HLW/ILW Swiss waste packages that have been produced or are expected to be produced in the future. These data are used for the design and safety assessment of the SF/HLW/ILW repository.

1.2 Basic features of the inventory

Due to only a part of the expected SF arising being contracted for reprocessing at BNFL (British Nuclear Fuels Limited) and COGEMA (Compagnie Générale des Matières Nucléaires) and because for the model inventory it is assumed that no further reprocessing takes place, the option of direct disposal of SF has to be considered.

Within the model inventory the radioactive wastes are represented as waste sorts, defined by a typical (representative) waste package combined with an estimation of the number of these waste packages. The ILW inventory is composed of 8 waste sorts arising from 2 production scenarios and the HLW inventory of 2 (1 BNFL and 1 COGEMA) glass waste sorts. The SF inventory describes 7 waste sorts, consisting of UO_2 and UO_2 + MOX spent fuel.

For additional ILW, not arising from reprocessing operations, only an estimate of the total volume has to be considered as the inventory is expected to be negligible with respect to that arising from the reprocessing waste.

The choice of 17 waste sorts is a result of a compromise between distinguishing the wastes in relation to their significant properties for disposal purposes (such as activities, materials and containers) and not considering, for the purposes of the current stage of the disposal planning, the non-significant variations occurring in real conditioned waste (such as variations in the raw waste, e.g. fuel element designs, etc.).

1.3 Structure of the report

Chapter 2 presents the assumptions made for the model inventory, i.e. for the waste production scenarios and waste descriptions.

Chapter 3 describes, for reprocessing wastes, the raw wastes and the waste sorts and examines the methods used for the determination of the data.

Chapter 4 describes the background of the characterisation of spent fuel and presents the resulting waste sorts.

Chapter 5 presents the overall SF, HLW and ILW inventories.

2 Assumptions made for the Model Inventory

2.1 Waste production scenario

The primary waste production scenario for the model inventory only considers the installed 3.2 GW(e) of electricity generated by the 5 operational nuclear power plants (Tab. 2.1).

Tab. 2.1: Commercial nuclear reactors currently operating in Switzerland (SVA 2002)

Name	Reactor type	Output	Operating since
KKB-I: Beznau I	PWR (Westinghouse)	365 MW(e)	1969
KKB-II: Beznau II	PWR (Westinghouse)	365 MW(e)	1971
KKM: Mühleberg	BWR (General Electric)	355 MW(e)	1971
KKG: Gösgen	PWR (Kraftwerk Union)	970 MW(e)	1979
KKL: Leibstadt	BWR (General Electric)	1'145 MW(e)	1984

For the purposes of this study two scenarios are considered. In the "Reference Case" the five reactors in operation are each assumed to operate for 60 years (equivalent to 192 GWa(e)). For a second case (limited) data for a 300 GWa(e) power system are provided to study the impacts of larger volumes and inventories for the safety assessment and design of the SF/HLW/ILW repository (NAGRA 2002a, NAGRA 2002b).

From the total mass of spent UO_2 fuel arising from the 192 GWa(e) or 300 GWa(e) scenario only 1'195 t_{IHM}[1] is contracted for reprocessing. Assuming that reprocessing wastes will only arise from the existing contracts and no further reprocessing will take place, the remaining UO_2 fuel is assumed to undergo direct disposal.

Under this assumption the reprocessing of 1'195 t_{IHM} of SF results in 145 t_{IHM} of Mixed Oxide (MOX) fuel, which after irradiation is assumed to be disposed of directly with the remaining UO_2 spent fuel.

For the derivation of the source term for a safety analysis (NAGRA 2002a), an average burnup of 48 GWd/t_{IHM} has been assumed for both scenarios. In addition to this, for a sensitivity analysis, spent fuel with of 55, 65 and 75 GWd/t_{IHM} is considered.

In the case of reprocessing wastes, two scenarios were chosen to model the expected arisings of these wastes. The first scenario, described as the "Cemented waste option", assumes full return of only formally specified wastes from BNFL and COGEMA. The second scenario, described as the "High force compaction option", assumes that (1) for COGEMA full compaction of hulls & ends and technological wastes takes place and (2) BNFL will substitute all LLW wastes for a small amount of additional HLW glass.

[1] The unit t_{IHM} refers to tons of initial heavy metal (mass before irradiation).

2.2 Waste sort description

As already mentioned (Chapter 1.2) in the model inventory the radioactive wastes are categorised into individual waste sorts. A waste sort is described by a representative waste package consisting of average values and an estimation of the number of these waste packages that could be reasonably expected to arise. A waste sort is defined by similar:

- type of raw waste (e.g. sludges, hulls and ends, nuclide inventory, etc.)
- type of conditioning material (cement, bitumen, glass, etc.)
- type of canister or container (180 l stainless steel flask, SF canister, overpack, etc.)

Maximum values are also produced to cover the (conservative) distribution of individual package values. Further, the waste sorts correspond as far as possible to expected arisings of real conditioned waste. The system used for designating the waste sorts is as follows:

BE ("Abgebrannte Brennelemente") for SF that has not been reprocessed followed by a number, to specify the fuel type and burnup.

WA ("Wiederaufarbeitungsabfälle") for HLW and ILW arising from the spent fuel reprocessing, followed by an abbreviation of the reprocessor, "COG" for COGEMA, "BNF" for BNFL, a number defining the type of raw waste and in some instances a letter (A) to indicate compaction: for instance WA-COG-4 (cemented hulls and ends) and WA-COG-4A (compacted hulls and ends).

3 Reprocessing Waste

3.1 Waste description

As mentioned in Chapter 2, the amount of spent fuel assumed to be reprocessed is 1'195 t_{IHM}. The spent fuel is reprocessed to recover the uranium and plutonium for recycling in a reactor as mixed oxide (MOX) fuel assemblies.

The fuel assemblies are chopped giving rise to sections of Zircaloy fuel rods and metallic end caps and the spent fuel is dissolved in an acid solution. The remaining Zircaloy hulls are either conditioned in cement together with the ends (waste sort WA-4) or compacted (waste sort WA-4A). Uranium and plutonium are extracted from the solution leaving a highly active "liquor", which contains the fission products and other actinides formed in the spent fuel, as well as some residual uranium and plutonium. The "liquor" is calcined and conditioned with a glass frit and additives giving rise to the high level vitrified residue (waste sort WA-1). Low level technological wastes from the operation (LLW and hence not characterised in this report for SF/HLW/ILW) of the plant are packaged in relatively thick walled containers, or metallic containers with an additional internal cement wall.

The other waste streams differ between COGEMA and BNFL due to the use of different reprocessing operations. For COGEMA the residues within the dissolver filters (metallic fines) and from the centrifugation of the fuel solution, before separation of uranium and plutonium, are routed to the vitrification process. In the case of BNFL, these residues form the centrifuge cake slurry which is then conditioned in cement (waste sort WA-BNF-7). Multiple element bottles (MEBs) are used for the transport to, and the storage of, fuel assemblies in the ponds. The crud arising from the MEBs, as well as the pond filter crud, is mixed by BNFL with the barium carbonate waste. The barium carbonate contains mainly radioactive carbon and iodine, arising from the scrubbing of the vitrification process off-gas. The cruds and barium carbonate are conditioned together in cement (waste sort WA-BNF-2).

Liquid wastes arising at COGEMA were initially treated for water purification and the resulting precipitates and slurries conditioned in bitumen (waste sort WA-COG-2). However, in recent years the production of bitumen waste has essentially ceased with the liquid wastes now being routed to the high level vitrification process. The COGEMA technological waste, which cannot be declared as LLW due to its high α-emitter content, is packaged in appropriate containers (waste sort WA-COG-6). However, in the future, high α-technological waste and hulls and ends will be compacted separately and the pucks placed into flasks of the same dimensions as those used for the vitrified residues. These two new waste sorts (WA-COG-6A and WA-COG-4A, respectively) have not yet been formally approved for return to Switzerland. Therefore, for this study two scenarios (see Chapter 2.1) are presented, one "Cemented waste option" describing the approved cemented waste sorts WA-COG-4 and WA-COG-6 and the "High force compaction option" describing the new compacted waste sorts WA-COG-4A and WA-COG-6A.

The resulting waste sorts that are defined by the COGEMA and BNFL reprocessing waste specifications and included in this inventory are:

- WA-COG-1: HLW vitrified residues
- WA-COG-2: Precipitates and sludges in bitumen
- WA-COG-4: Hulls and ends in cement (Cemented waste option scenario only)
- WA-COG-6: α-emitting technological waste (Cemented waste option scenario only)
- WA-COG-4A: Compacted hulls and ends (High force compaction option scenario only)
- WA-COG-6A: Compacted α-emitting technological waste (High force compaction option scenario only)

- WA-BNF-1: HLW vitrified residues
- WA-BNF-2: MEB (Multiple Element Bottle) crud and barium carbonate ($BaCO_3$) in cement
- WA-BNF-4: Hulls and ends in cement
- WA-BNF-7: Centrifuge cake slurry in cement

3.2 Waste characterisation

For the BNFL and COGEMA ILW inventory the waste sorts have been based on specifications and additional information supplied by BNFL and COGEMA. For the new COGEMA compacted ILW (WA-COG-4A and WA-COG-6A) the total radionuclide and raw waste material inventories are modelled to be the same as the corresponding total inventories obtained for the equivalent cemented waste sorts (Appendix A).

The vitrified residues (WA-BNF-1 and WA-COG-1) were characterised from first principles using ORIGEN2 (CROFF 1983) to calculate the nuclide and stable element data. These calculations were performed, based on specifications and additional information supplied by BNFL and COGEMA. The individual container inventories, both material and radionuclide content and the heat output and other radiological data, for the BNFL and COGEMA waste sorts are given in Appendix A of this report. Further, this appendix also provides the long term decay of the average HLW glass flask heat output, from the time of production.

3.3 Waste container numbers and volumes

For BNFL and COGEMA the rounded numbers of containers and their associated conditioned (packaged) volumes are given in Tab 3.1. The conditioned volumes can be calculated from "conditioned volumes/container" data which are based on the HLW/ILW container concept as defined in (NAGRA 2002b). The ILW are grouped into ILW-1 and ILW-2 (WA-COG-2) for reasons discussed in (NAGRA 2002a and b). With a loading of appr. 400 kg glass/flask (with 730 flasks, see Tab. 3.1) the total mass of HLW is expected to be 292 tons.

Tab. 3.1: Rounded number of containers and conditioned volumes[1] for reprocessing wastes

Repository Type	Waste Sort	Cemented waste option		High force compaction option	
		No. Cont.	Volume[2] [m^3]	No. Cont.	Volume[2] [m^3]
HLW	WA-BNF-1	270	375	270	375
	WA-COG-1	460	640	460	640
ILW	*ILW-1:*				
	WA-BNF-2	30	98	30	98
	WA-BNF-4	270	954	270	954
	WA-BNF-7	170	498	170	498
	WA-COG-4	320	964	-	-
	WA-COG-6	510	1'534	-	-
	WA-COG-4A	-	-	450	724
	WA-COG-6A	-	-	580	917
	ILW-2:				
	WA-COG-2	380	304	380	304
HLW	**Total**	**730**	**1'015**	**730**	**1'015**
ILW	**Total**	**1'680**	**4'360**	**1'880**	**3'490**

[1] Waste drums containing ILW packaged in disposal containers, stainless steel flasks with HLW glass packaged in steel HLW canisters (NAGRA 2002a).

[2] Volumes have been calculated with exact number of containers and finally rounded.

3.4 Uncertainties

3.4.1 ILW

When modelling inventories and especially for those wastes that may arise in the future, assumptions must be made concerning waste arising scenarios. Furthermore, as no L/ILW repository is, as yet, in operation it is difficult to judge those wastes that are currently considered as L/ILW but could be excluded on the basis of waste acceptance criteria, e.g. limits on alpha specific activity, etc..

It is intended in this section, to highlight some of the uncertainties that should be considered at this time. These are in no particular order, as follows:

- Extension of reprocessing contracts which would lead to more ILW and HLW and less SF waste: as the disposal of spent fuel is considered to be more conservative than the equivalent amount of reprocessing wastes, it is not considered necessary to examine this within this project.
- A reserve ILW volume, for each scenario, has been assumed for any additional wastes excluded from an L/ILW repository due to potential (conservative) future acceptance criteria. These volumes could result from e.g. waste from medicine, industry and research (MIR) as described in (NAGRA 2001) or future spent fuel conditioning facilities, etc. The assumed reserve volumes associated with each scenario are 2'900 m^3 for the 60 years and 3'800 m^3 for the 300 GWa(e) scenario.

- Substitution of BNFL/ILW residues for additional HLW glass: Depending on British government policy it may eventually become possible to substitute all BNFL ILW residues for an additional, radiologically equivalent, amount of HLW glass.
- Numbers of flasks of COGEMA compacted wastes (WA-COG-4A and WA-COG-6A) are considered, due to the recent start-up of the process, to be uncertain. However, to reduce the effect of this uncertainty, the total radionuclide and material inventories have been calculated so as to be the same as the equivalent COGEMA cemented wastes.

3.4.2 HLW

The uncertainties for HLW are as follows:

- Substitution of BNFL/ILW residues for additional HLW glass: Depending on British government policy it may eventually become possible to substitute all BNFL ILW residues for an equivalent of HLW glass. It is currently estimated that this could lead to around 15 % of additional flasks with HLW glass. However, as the principle of substitution is the return of a radiologically equivalent amount of waste, this estimate could vary depending on the actual HLW glass produced by BNFL.
- As in the case of ILW the extension of reprocessing contracts would lead to more ILW and HLW and less SF waste: again the disposal of spent fuel is considered to be more conservative than the equivalent amount of reprocessing wastes, and it is not considered necessary to examine this explicitly within this project.

3.5 Total volumes of HLW and ILW

The total volumes of HLW and ILW (as discussed in Tab. 3.1 and in Chapter 3.4) are summarized in Tab. 3.2.

Tab. 3.2: Total conditioned volumes for HLW and ILW

Repository Type	Cemented waste option		High force compaction option	
	60 years [m^3]	300 GWa(e) [m^3]	60 years [m^3]	300 GWa(e) [m^3]
HLW				
Total	**1'015**	**1'015**	**1'015**	**1'015**
ILW				
Reprocessing	4'360	4'360	3'490	3'490
Reserve	2'900	3'800	2'900	3'800
Total[1]	**7'300**	**8'200**	**6'400**	**7'300**

[1] The total volumes for ILW are rounded.

4 Spent Fuel (SF)

4.1 Mass of UO_2 and MOX spent fuel

The assumed masses of spent fuel for 40 year operation of the Swiss reactors, are given in Tab. 4.1. The data are based on a burnup of 48 GWd/t_{IHM} and form the basis for the calculations for the 60 year and 300 GWa(e) scenarios.

Tab. 4.1: Estimated masses of spent fuel arising from 40 years of reactor operations (excluding spent fuel sent for reprocessing)

	UO_2 Fuel Weight [t_{IHM}]	**MOX Fuel** Weight [t_{IHM}]
KKB	270	68
KKG	420	60
KKL	829	17
KKM	168	- [1]
PWR	690	128
BWR	997	17
Total	**1'687**	**145**

[1] It should be noted that KKM will not irradiate any MOX fuel in its reactor.

For the 60 year (192 GWa(e)) and 300 GWa(e) scenarios it is necessary to estimate the amount of spent fuel that will arise in the time period after 40 years of reactor operations. Therefore, to estimate the mass of UO_2 spent fuel, in terms of t_{IHM}, produced by each reactor (see Tab. 2.1), the following formula can be used:

$$t_{IHM}\,(a) = \frac{THL \times a \times 365.25 \times LF}{BU} \qquad \text{(Equation 1)}$$

a: Number of years
THL: Thermal power [GW] of the reactor
LF: Full power days / 365.25 of the reactor
BU: Spent fuel burnup [GWd/t_{IHM}]

As for the fuel arising for direct disposal over 40 years reactor operations, an average burnup of 48 GWd/t_{IHM} has been assumed for both the 60 year (192 GWa(e)) scenario (additional 20 years operation time) and for the additional 108 GWa(e) under the 300 GWa(e) scenario. The reason for not using higher burnup values is that they would lead to lower (less conservative) volumes of SF requiring disposal. However, for sensitivity studies within the safety analysis (NAGRA 2002a) burnups of 55, 65 up to 75 GWd/t_{IHM} are considered in this report.

For the 20 years reactor life time extension scenario, the data for the thermal power and full power days for each reactor (HSK 2001, BFE 2001) are given in Tab. 4.2. The full power days data were calculated as the average values from the last 10 years ("LF 10 years") of each reactors operation. To take account of any increase in reactor performance in future reactor operations and not to underestimate the amount of spent fuel, it was decided to increase the full power day values by approx. 5 %. The data of Tab. 4.2 ("THL and LF (+ 5 %)") are used in Equation 1 to calculate the t_{IHM} for the 60 years scenario.

Tab. 4.2: Performance data for the Swiss nuclear power plants (HSK 2001, BFE 2001)

Power plant	Thermal Power THL [GW]	Net electric power [GW]	LF 10 years	LF (+ 5 %)
KKB	2.260	0.73	0.86	0.90
KKG	3.002	0.97	0.92	0.97
KKL	3.515	1.145	0.86	0.90
KKM	1.097	0.355	0.86	0.90
Total	**9.874**	**3.200**	**-**	**-**

For the additional 108 GWa(e) or an equivalent of 33.75 additional years operation (108 GWa(e) / 3.2 GW(e)), the same thermal power key (thermal power "THL" and burnup "BU") is used in Equation 1 but as the performance under this scenario is difficult to estimate, a general value of LF = 0.93 was assumed.

The results of these calculations are given in Tab. 4.3.

Tab. 4.3: Calculated masses of UO_2 spent fuel from an additional 20 years and 108 GWa(e) (relative to Tab. 4.1) and total masses for 60 years and 300 GWa(e) of reactor operations (excluding spent fuel sent for reprocessing)

Reactor	20 years [t_{IHM}]	108 GWa(e) [t_{IHM}]	60 years [t_{IHM}]	300 GWa(e) [t_{IHM}]
KKB	310	540	580	1'120
KKG	443	717	863	1'580
KKL	482	840	1'311	2'151
KKM	150	262	318	580
PWR	753	1'257	1'443	2'700
BWR	632	1'102	1'629	2'731
Total	**1'385**	**2'359**	**3'072**	**5'431**

4.2 Representative fuel assembly data

For each of the Swiss reactors, fuel assemblies of slightly different designs have been used. Therefore, for the purposes of spent fuel disposal studies, average values have been estimated and are presented in Tab. 4.4 and 4.5 below.

Tab. 4.4: Assumed dimensions and masses of average UO_2 spent fuel assemblies

UO_2	**KKL (BWR)**	**KKM (BWR)**	**KKB (PWR)**	**KKG (PWR)**
U [t_{IHM}]	0.178	0.173	0.325	0.433
Length [m]	4.481	4.474	3.518	4.293
Width [m]	0.139 × 0.139	0.139 × 0.139	0.198 × 0.198	0.215 × 0.215
Weight [t]	0.294 [2]	0.27 [1]	0.480	0.666

[1] No fuel channel is included as these are currently disposed of as L/ILW.

[2] SVEA96 with an integral fuel channel.

Tab. 4.5: Assumed dimensions and masses of average MOX spent fuel assemblies

MOX	**KKL (BWR) [1]**	**KKB (PWR)**	**KKG (PWR)**
MOX [t_{IHM}]	0.175	0.32	0.427
Length [m]	4.481	3.518	4.293
Width [m]	0.139 × 0.139	0.198 × 0.198	0.215 × 0.215
Weight [t]	0.289 [2]	0.473	0.655

[1] It should be noted that KKM will not utilise MOX fuel.

[2] SVEA96 with an integral fuel channel.

4.3 Definition of representative disposal canisters

Based on Tab. 4.1, 4.3, 4.4 and 4.5, the number of fuel assemblies arising under the 60 years reactor operations and 300 GWa(e) scenario and the resulting number of disposal canisters can be derived. The conceptual canister designs with 4 PWR or 9 BWR fuel assemblies can be found in (JOHNSON & KING 2003). The length of the disposal canisters (which are different due to the different length of the fuel assemblies (Tab. 4.3, 4.4)) are given in Tab. 4.6 together with the number of disposal canisters.

For the optimum distribution of heat output, only 1 MOX spent fuel assembly will be placed in any given disposal canister (JOHNSON et al. 2002) together with 3 (PWR) resp. 8 (BWR) UO_2 fuel assemblies (with masses defined in Tab. 4.4 and 4.5).

Tab. 4.6: Number and length of disposal canisters

Reactor	UO_2		UO_2 + MOX	Length of Canister [m]
	No. Can. 60 years	No. Can. 300 GWa(e)	No. Can.	
KKB	286	702	213	3.95
KKG	393	806	141	4.73
KKL	732	1'257	97	4.92
KKM	204	372	-	4.92

To reduce the number of disposal canisters that require modelling within the safety analysis (NAGRA 2002a), it was decided to model a reduced number of canisters whose overall characteristics sufficiently represent the masses of spent fuel expected to be disposed of under the 60 years reactor operations and the 300 GWa(e) scenario. As mentioned previously, any uncertainties associated with the spent fuel burnup will be analysed via a sensitivity study.

Based on analysis of the fuel assembly data, see Tab. 4.4 and 4.5, conceptual BWR and PWR canisters were designed (JOHNSON & KING 2003). To sufficiently represent the characteristics of this spent fuel, 3 basic canister types with 48 GWd/t_{IHM} SF are modelled:

- BE-1 - BWR UO_2, containing 9 spent fuel assemblies
- BE-2 - PWR UO_2 + PWR MOX, containing 3 UO_2 and 1 MOX spent fuel assemblies
- BE-3 - PWR UO_2, containing 4 UO_2 spent fuel assemblies

As can be seen in Tab. 4.1 and Tab. 4.6, the majority of MOX spent fuel is PWR, therefore for the purposes of this project, the small amount of BWR MOX is modelled as PWR spent fuel and included in the BE-2 canister types.

For the determination of the numbers of canisters required to dispose of the spent fuel that has not been contracted for reprocessing, it is assumed that all canisters can be completely filled. Finally the numbers of basic canister types were derived (first filling the UO_2/MOX canisters) for each reactor and then rounded numbers of average canisters determined, as presented in Tab. 4.7.

Tab. 4.7: Proposed characteristics for the reference fuels and reference canisters

Parameter	**BE-1 Model BWR UO_2 fuel canister**	**BE-2 Model UO_2/MOX fuel canister**		**BE-3 Model PWR UO_2 fuel canister**
No. Can. 60 years [1]	935	450		680
No. Can. 300 GWa(e) [1]	1'630	450		1'500
Canister length [m] [2] 60 years 300 GWa(e)	 4.92 4.92	 4.40 4.40		 4.40 4.58
Canister diameter [m]	1.05	1.05		1.05
Spent fuel data	**UO_2**	**UO_2**	**MOX**	**UO_2**
Number of FA[3] /Can.	9	3	1	4
Total fuel for 60 years [t_{IHM}]	1'491	529	145	1'052
Total fuel for 300 GWa(e) [t_{IHM}]	2'593	529	145	2'309
SF/Can. [t_{IHM}] [4]	1.593	1.173	0.322	1.556
SM/Can. [tons] [5]	0.796	0.564		0.589
Mass of SF/FA [t_{IHM}]	0.177	0.391	0.322	0.389
Average burnup [GWd/t_{IHM}]	48	48	48	48

1 Rounded number of canisters.

2 The canister length is an "equivalent canister length" as a weighted average of the real length with the number of canisters as weighting factors (see Tab. 4.6). The BE-3 canister length for the 300 GWa(e) scenario takes account of the assumption that all PWR fuel assemblies, arising after 60 years of reactor operations, will be a longer PWR type.

3 FA = Fuel assembly.

4 For calculating the masses of fuel/canister the exact (not rounded) number of canisters for the 60 years scenario is used.

5 Mass of "Structural Materials/Canister".

4.4 Summary of spent fuel waste sorts

In addition to the "reference" waste sorts (Chapter 4.3):

- BE-1 9 BWR UO_2 fuel assemblies (all 48 GWd/t_{IHM})
- BE-2 3 PWR UO_2 + 1 PWR MOX fuel assemblies (all 48 GWd/t_{IHM})
- BE-3 4 PWR UO_2 fuel assemblies (all 48 GWd/t_{IHM})

for the burnup sensitivity studies (NAGRA 2002a) the following waste sorts have been defined:

- BE-4 4 PWR UO_2 fuel assemblies (55 GWd/t_{IHM})
- BE-5 3 PWR UO_2 (55 GWd/t_{IHM}) + 1 PWR UO_2 (65 GWd/t_{IHM}) fuel assemblies
- BE-6 3 PWR UO_2 (55 GWd/t_{IHM}) + 1 PWR UO_2 (75 GWd/t_{IHM}) fuel assemblies
- BE-7 3 PWR UO_2 (48 GWd/t_{IHM}) + 1 PWR MOX (65 GWd/t_{IHM}) fuel assemblies

The detailed material, radionuclide and radiological inventory data for each waste sort 40 years after the spent fuel is discharged from the reactor, are given in Appendix C. It should be noted that for each of the waste sorts only a single total radionuclide inventory is given. The reason for this is that for canisters BE-1 to BE-3 these inventories represent the average inventory for the 60 year scenario source term, but for canisters BE-4 to BE-7 these represent inventories where the burnup has been increased to permit sensitivity studies. Therefore, no maximum inventory is provided for each spent fuel canister.

4.5 Fuel depletion calculations

4.5.1 Definition of burnup

When discussing spent fuel it is important to define what is meant by burnup as this can be used in relation to a fuel pellet (and parts thereof), rod or assembly and a reactor core. Further, the burnup varies along a fuel assembly, with the highest burnup being in the centre of the assembly. When discussing inventories in this report, these are for the average burnup of a fuel assembly which takes account of the variation in burnup, along the assembly. For certain cases, i.e. shielding and criticality, it is necessary to model the variation of burnup along a fuel assembly, in these cases a representative BWR or PWR distribution is used to adjust the fuel assembly average inventory for these calculations.

4.5.2 Selection of code

Historically, the ORIGEN2 code (CROFF 1980 and 1983), has been used within the waste management community for spent fuel calculations. However, the 1 group cross-section libraries supplied with the code are only applicable for UO_2 fuels[2] with burnups up to 50 GWd/t_{IHM} (BWR to 40 GWd/t_{IHM}) and MOX fuels[2] to 33 GWd/$t_{IHM.}$ While data for higher burnups can be generated, the level of uncertainty will increase the further the target burnup deviates from these values. As it was necessary to obtain reliable spent fuel inventories, especially in the case of MOX fuels, for much higher burnups (up to 65 GWd/t_{IHM}) it was decided to develop capabilities using the PSI fuel assembly code BOXER (PARATTE et al. 1996). Since the original BOXER code was developed to calculate only a limited number of fission products and actinide nuclide masses in spent fuel, it was necessary to first extend the capabilities of the BOXER code to obtain the masses of all disposal-relevant nuclides.

[2] It should be noted that 1 group cross-section libraries for spent fuel with higher burnups have recently (SUYAMA et al. 2001) become available and are currently under evaluation.

4.5.3 General calculation methodology

For each of the calculations the fuel and Zircaloy cladding inventories (including impurities) were calculated, using the BOXER code. The remaining structural materials (end pieces, springs, spacers etc.) were calculated using the ORIGEN2 code (CROFF 1983). The use of ORIGEN2 for the characterisation of the remaining structural materials is considered to be justified as only activation product nuclides are of any relevance in these materials and ORIGEN2 can be considered, even at higher burnups, to produce acceptable results for these nuclides.

To obtain representative average spent fuel inventories (48 GWd/t_{IHM}), especially with regard to structural materials, detailed calculations were first performed for each reactor fuel and then these were combined (weighted average) to produce a representative inventory for each reactor type, i.e. PWR and BWR. For this report only the results of the weighted average spent fuel calculations are presented.

For the fuel depletion calculations performed, to provide inventories, for the higher burnup sensitivity analysis (NAGRA 2002a), i.e. 55, 65 and 75 GWd/t_{IHM} UO_2 and 65 GWd/t_{IHM} MOX, these were based, due to ease of calculation, on a PWR fuel assembly.

4.5.4 UO_2 radionuclide and stable element inventory calculations

The results of the UO_2 weighted average inventory calculations for 48 GWd/t_{IHM} PWR and BWR fuel assemblies, at time of reactor discharge and after 40 years decay, are given in Appendix B.

For the burnup sensitivity analysis, nominal PWR spent fuel inventory calculations were performed for 55, 65 and 75 GWd/t_{IHM}, and the results at time of reactor discharge and after 40 years decay, are also given in Appendix B.

4.5.5 MOX radionuclide and stable element inventory calculations

As PWR MOX spent fuel arisings dominate, only the inventories for this fuel type have been calculated and included in the model inventory.

To obtain a representative average spent fuel inventory (48 GWd/t_{IHM}), especially with regard to structural materials and initial fuel composition, detailed calculations were first performed for each PWR reactor and MOX fuel type, and then a weighted average spent fuel derived. For this report only the results of the weighted average spent fuel calculation is presented.

The results of the 48 GWd/t_{IHM} MOX weighted average inventory calculations, at time of reactor discharge and after 40 years decay, are given in Appendix B.

For the burnup sensitivity analysis, a nominal 65 GWd/t_{IHM} MOX inventory calculation was performed and the results at time of reactor discharge and after 40 years decay are given in Appendix B.

4.5.6 Heat output

The decay of canister heat output is of specific interest for repository heat transfer calculations. Therefore, the heat output data per canister 40 years after the spent fuel is discharged from the reactor, are given for each canister type (BE-1 to BE-7) in Appendix C. It should be noted that for BE-2, data from 55 years is also given. Repository temperature evolution calculations are reported in (JOHNSON et al. 2002).

4.5.7 Criticality

Calculations for MOX and UO_2 fuel of average burnup (KÜHL et al. 2003) indicate that the reactivity, i.e. the number of excess neutrons produced by fission, decreases for the first few hundred years after discharge from the reactor due to ingrowth of Am-241, a strong neutron absorber. It gradually increases back to its initial value over hundreds of thousands of years as U-235 grows back in. An analysis performed by Nagra (KÜHL et al. 2003) shows that, when loaded with spent fuel, canisters would be sub-critical, both when intact, and after failure when the void space in the canister would eventually become filled with water, provided a burnup of at least 15 GWd/t_{IHM} is reached (22 GWd/t_{IHM} for canisters with both MOX and UO_2 fuel). In the rare cases in which the minimum burnup is not attained by a fuel assembly, the co-placement of high burnup assemblies in the canister or the use of inert filler (e.g. sand) in the void space in the canister would reduce reactivity to a sub-critical level.

4.6 Uncertainties associated with fuel inventories

4.6.1 Uncertainties associated with fuel depletion calculations

The uncertainties concerning the overall results obtained from fuel depletion code BOXER have been examined based on comparisons made with other similar codes, i.e. FISPIN (UKAEA 1995) and ORIGEN-S (HERMANN & WESTFALL 1989). It should be noted that the ORIGEN-S calculations were performed within the ECN OCTOPUS system (KLOOSTER-MAN et al. 1996). For safety relevant nuclides, it is found that the results of UO_2 and MOX calculations compare, within ~ 10 % for fission products and ~ 20 % for the major U and Pu isotopes and for Am-241 and Cm-244, which are the most important minor actinides.

Further, preliminary but soon to be extended calculated (BOXER) to measured values for MOX and UO_2 fuels have been reviewed and lead to further confidence in the results.

Concerning the ORIGEN2 code, extensive world wide experience has been gained in the use of this type of code for the prediction of UO_2 spent fuel inventories. This has been supported by comparisons of predictions with measured values such as the American ATM series (see GUENTHER et al. 1988a: 43 GWd/t_{IHM} PWR, GUENTHER et al. 1988b: 30 GWd/t_{IHM} PWR, GUENTHER et al. 1991a: 43 GWd/t_{IHM} PWR and GUENTHER et al. 1991b: 28 GWd/t_{IHM} BWR) for ORIGEN2.

Concerning the half-life and branching ratio values for the radionuclides modelled (half-lives greater than 60 days + all daughters) within these codes, the uncertainties have been quantified (McGINNES 2000) and have been found to have an negligible impact on decayed inventories for nuclides of relevance for safety assessment calculations.

With regard to the uncertainty associated with cross-section data this has to a great extent been circumvented by looking directly at calculated to measured values and to a lesser extent by comparing codes. However, a study by SKB (HÅKANSSON 2000 in Swedish) summarised in (ANDERSSON 1999) found – from the comparison of results from the SCALE and CASMO codes – total uncertainties of 20 % and 12 % for actinides and fission products, respectively.

4.6.2 General uncertainties

It is intended in this section, to highlight some of the uncertainties that should be considered at this time. These are, in no particular order, as follows:

- Burnup: The current trend is for increasing burnup. While this reduces the overall amount of fuel requiring disposal, it does result in fuels with higher levels of residual activity. Therefore, the overall impact with regard to total inventories is considered to be broadly neutral.
- In addition to commercial spent fuel, it should be noted that some MTR (Material test reactor) spent fuel from DIORIT or SAPHIR reactors at PSI could require direct disposal. Due to the low amount and high level of uncertainty regarding final dispositions in Switzerland, these fuels have not been modelled in the current study. It can be considered that any arisings of these fuels can be considered as being included in the general uncertainties.
- The current disposal scenario assumes that only the spent fuel assemblies will be inserted into the disposal canisters. However, it is likely that, in the case of BWR fuel the fuel channel may also be included and for PWR the control rod cluster assembly. The effect of these components on the spent fuel inventories is considered to be negligible, due to the domination of the inventories by the fuel and its cladding.

5 Total Waste Inventories for the SF/HLW/ILW Repositories

5.1 Total activity and material inventories

5.1.1 Inventories of ILW at time of production

The total inventory for ILW, arising under the 60 years of reactor operations scenario, is given in Tab. 5.1. It should be noted that the radionuclide inventories of both the cemented and high force compaction scenarios are assumed to be identical.

Tab. 5.1: Total ILW inventory at time of production (Cut-off 1 GBq)

Nuclide	[Bq]
H-3	2.1E+16
C-14	1.8E+13
Cl-36	6.3E+10
Ca-45	1.0E+10
Mn-54	2.3E+15
Fe-55	8.2E+16
Co-57	5.1E+14
Co-58	2.4E+12
Co-60	8.0E+16
Ni-59	1.2E+14
Ni-63	1.9E+16
Zn-65	6.6E+11
Se-79	1.3E+09
Kr-85	6.2E+14
Sr-90	5.5E+15
Y-90	5.5E+15
Zr-93	6.0E+12
Nb-93M	1.6E+12
Nb-94	3.2E+13
Nb-95	1.1E+13
Mo-93	1.4E+12
Tc-99	1.6E+12
Ru-106	1.3E+16
Rh-106	1.3E+16
Pd-107	2.3E+10

Nuclide	[Bq]
Ag-110M	9.8E+12
Sn-113	3.2E+13
Sn-119M	1.3E+16
Sn-121M	4.3E+14
Sn-123	9.7E+12
Sn-126	1.9E+10
Sb-125	2.1E+16
Te-125M	2.9E+15
I-129	4.7E+10
Cs-134	4.4E+15
Cs-135	1.2E+11
Cs-137	1.1E+16
Ba-137M	1.0E+16
Ce-144	4.2E+15
Pr-144	4.2E+15
Pm-147	2.7E+15
Sm-151	7.1E+13
Eu-152	2.0E+11
Eu-154	7.4E+14
Eu-155	1.0E+14
Ta-182	4.8E+14
U-234	5.1E+10
U-235	1.1E+09
U-236	8.8E+09
U-238	1.2E+10

Nuclide	[Bq]
Np-237	6.2E+09
Pu-238	1.1E+14
Pu-239	1.8E+13
Pu-240	2.4E+13
Pu-241	4.3E+15
Pu-242	8.0E+10
Am-241	5.7E+13
Am-242M	1.4E+11
Am-243	4.0E+11
Cm-242	1.9E+12
Cm-243	2.4E+11
Cm-244	2.5E+13
Cm-245	1.6E+09
Total α	**2.4E+14**
Total βγ	**3.2E+17**
Total Activity	**3.2E+17**

The total material inventory for ILW, arising under the 60 years of reactor operations scenario, is given in Tab. 5.2.

Tab. 5.2: Total material inventory for ILW

Waste type	**ILW**	**ILW**
Scenario	Cemented waste option [kg]	High force compaction option [kg]
Metals	**8.9E+05**	**8.2E+05**
Steels	5.6E+05	4.9E+05
Al / Zn	3.4E+03	3.4E+03
Inconel	1.6E+04	1.6E+04
Zircaloy	3.0E+05	3.0E+05
Other	3.2E+03	3.2E+03
Inorganics	**2.2E+06**	**4.9E+05**
Salts	3.0E+04	3.0E+04
Ceramics	-	-
Ashes	-	-
Glass	3.1E+02	3.1E+02
Concrete / Cement	2.1E+06	4.5E+05
Other	2.3E+03	2.3E+03
Organics	**7.4E+04**	**6.3E+04**
High Mwt Subtotal	**7.4E+04**	**6.3E+04**
Bitumen	4.7E+04	4.7E+04
IX Resins	-	-
Cellulose	5.1E+02	5.1E+02
Other	2.7E+04	2.7E+04
Low Mwt Subtotal	**8.4E+02**	**8.4E+02**
Detergents	7.5E+02	7.5E+02
Flocculants	6.5E+00	6.5E+00
Complexing/agents	9.1E+01	9.1E+01
Other	-	-
Not Specified	-	-
Total	**3.1E+06**	**1.4E+06**

It should be noted that the differences in material composition, between the cemented and high force compaction scenarios, is due to the package materials and not the raw waste which is identical in both cases.

5.1.2 Inventories of HLW at time of production

The total inventory for HLW, arising under the 60 years of reactor operations scenario, is given in Tab. 5.3.

Tab. 5.3: Total HLW inventory at time of production (Cut-off 1 GBq)

Waste sort Nuclide	**WA-COG-1 [Bq]**	**WA-BNF-1 [Bq]**	**Total [Bq]**
C-14	8.8E+10	1.9E+10	1.1E+11
Ca-41	3.2E+09	1.8E+10	2.1E+10
Ca-45	2.1E+10	2.9E+11	3.1E+11
Mn-54	2.9E+12	3.2E+13	3.5E+13
Fe-55	2.4E+14	9.0E+14	1.1E+15
Co-58	1.2E+09	9.0E+07	1.3E+09
Co-60	2.6E+15	4.0E+15	6.5E+15
Ni-59	1.1E+12	1.8E+11	1.3E+12
Ni-63	1.6E+14	2.5E+13	1.8E+14
Zn-65	4.6E+13	2.5E+12	4.8E+13
Se-79	5.6E+11	4.2E+11	9.8E+11
Sr-90	1.6E+18	1.1E+18	2.7E+18
Y-90	1.6E+18	1.1E+18	2.7E+18
Mo-93	2.4E+10	1.3E+10	3.7E+10
Zr-93	4.4E+13	3.4E+13	7.8E+13
Zr-95	5.1E+12	4.8E+16	4.8E+16
Nb-93M	9.3E+12	1.3E+13	2.3E+13
Nb-94	3.2E+11	2.6E+09	3.2E+11
Tc-99	3.9E+14	2.5E+14	6.4E+14
Rh-102	9.3E+12	2.7E+11	9.6E+12
Ru-106	9.8E+17	6.4E+17	1.6E+18
Rh-106	9.8E+17	6.4E+17	1.6E+18
Pd-107	3.0E+12	2.0E+12	5.0E+12
Ag-108M	2.7E+11	1.3E+11	4.0E+11
Ag-110M	1.4E+15	1.7E+15	3.1E+15
Cd-109	3.3E+12	5.3E+09	3.3E+12
Cd-113M	1.0E+15	5.3E+14	1.6E+15
Sn-119M	1.8E+14	1.2E+14	3.0E+14
Sn-121M	4.1E+12	2.9E+12	7.0E+12
Sn-123	3.2E+13	7.4E+14	7.7E+14
Sn-126	1.7E+13	1.4E+13	3.1E+13
Sb-124	1.4E+09	8.2E+12	8.2E+12
Sb-125	1.2E+17	4.8E+16	1.6E+17
Te-123M	6.0E+10	2.1E+11	2.7E+11

Tab. 5.3: (Cont.)

Waste sort Nuclide	**WA-COG-1 [Bq]**	**WA-BNF-1 [Bq]**	**Total [Bq]**
Te-125M	2.7E+16	1.1E+16	3.9E+16
Te-127M	3.9E+13	2.3E+15	2.3E+15
I-129	7.4E+08	5.6E+08	1.3E+09
Cs-134	9.3E+17	2.4E+17	1.2E+18
Cs-135	9.3E+12	1.0E+13	2.0E+13
Cs-137	2.2E+18	1.5E+18	3.7E+18
Ba-137M	2.1E+18	1.4E+18	3.5E+18
Ce-144	7.9E+17	1.0E+18	1.8E+18
Pr-144	7.9E+17	1.0E+18	1.8E+18
Pm-146	3.1E+13	3.7E+12	3.5E+13
Pm-147	1.1E+18	6.6E+17	1.8E+18
Sm-151	8.8E+15	6.1E+15	1.5E+16
Eu-152	1.3E+14	8.7E+13	2.2E+14
Eu-154	1.5E+17	6.1E+16	2.1E+17
Eu-155	7.4E+16	3.4E+16	1.1E+17
Gd-153	5.6E+14	1.7E+14	7.2E+14
Tb-160	2.3E+11	1.1E+13	1.1E+13
Ho-166M	8.4E+10	4.0E+10	1.2E+11
Th-228	1.6E+11	1.8E+11	3.4E+11
Th-230	1.4E+09	2.9E+09	4.4E+09
U-234	3.3E+10	5.0E+09	3.9E+10
U-235	5.6E+08	8.0E+07	6.4E+08
U-236	7.9E+09	9.3E+08	8.8E+09
U-238	8.8E+09	1.6E+09	1.0E+10
Np-235	8.8E+09	7.4E+08	9.6E+09
Np-237	7.9E+12	5.0E+12	1.3E+13
Pu-236	4.7E+09	4.2E+08	5.1E+09
Pu-238	9.8E+13	9.3E+13	1.9E+14
Pu-239	9.3E+12	1.7E+13	2.7E+13
Pu-240	1.7E+13	2.9E+13	4.6E+13
Pu-241	2.8E+15	4.0E+15	6.8E+15
Pu-242	4.6E+10	8.7E+10	1.3E+11
Am-241	1.5E+16	2.9E+16	4.4E+16
Am-242M	3.4E+14	4.0E+14	7.4E+14

Tab. 5.3: (Cont.)

Waste sort Nuclide	**WA-COG-1 [Bq]**	**WA-BNF-1 [Bq]**	**Total [Bq]**
Am-243	3.5E+14	2.3E+14	5.9E+14
Cm-242	1.9E+15	1.5E+16	1.7E+16
Cm-243	3.2E+14	2.0E+14	5.2E+14
Cm-244	3.3E+16	1.4E+16	4.7E+16
Cm-245	2.7E+12	1.7E+12	4.5E+12
Cm-246	5.1E+11	3.2E+11	8.3E+11
Total α Total βγ Total Activity	**5.1E+16 1.3E+19 1.3E+19**	**5.8E+16 9.5E+18 9.5E+18**	**1.1E+17 2.3E+19 2.3E+19**

The total material inventory for HLW, arising under the 60 years of reactor operations scenario, is given in the Tab. 5.4.

Tab. 5.4: Total material inventory for HLW

Scenario	**High force compaction [kg]**
Metals	**5.8E+04**
Steels	5.8E+04
Al / Zn	-
Inconel	-
Zircaloy	-
Other	-
Inorganics	**2.9E+05**
Salts	-
Ceramics	-
Ashes	-
Glass	2.9E+05
Concrete / Cement	-
Other	-
Organics	-
High Mwt subtotal	-
Bitumen	-
IX Resins	-
Cellulose	-
Other	-
Low Mwt subtotal	-
Detergents	-
Flocculants	-
Complexing agents	-
Other	-
Not Specified	-
Total	**3.2E+05**

To the level of accuracy shown in this table, the cemented and high force compaction scenarios give identical total material inventories.

5.1.3 Inventory of SF wastes after 40 years decay

The total inventory for SF wastes, arising over 60 years of reactor operations, is given in Tab. 5.5.

Tab. 5.5: Total SF inventory (radionuclide and stable elements) at emplacement into the repository (40 years after the spent fuel was discharged from the reactor – Cut-off 1 GBq)

Waste sort **Nuclide**	**BE-1** **[Bq]**	**BE-2** **[Bq]**	**BE-3** **[Bq]**	**Total** **[Bq]**
H-3	6.4E+15	2.9E+15	4.5E+15	1.4E+16
Be-10	1.5E+10	6.8E+09	1.0E+10	3.2E+10
C-14	9.9E+13	3.4E+13	5.7E+13	1.9E+14
Cl-36	2.0E+12	6.0E+11	1.1E+12	3.7E+12
Ar-39	1.1E+13	5.1E+12	7.4E+12	2.3E+13
Ca-41	2.7E+11	9.8E+10	1.7E+11	5.5E+11
Fe-55	1.0E+13	4.1E+12	6.6E+12	2.1E+13
Co-60	7.8E+14	3.0E+14	4.9E+14	1.6E+15
Ni-59	1.2E+14	5.3E+13	8.2E+13	2.5E+14
Ni-63	1.3E+16	5.6E+15	8.9E+15	2.7E+16
Se-79	1.4E+12	6.1E+11	1.0E+12	3.0E+12
Kr-81	2.5E+09	9.7E+08	1.6E+09	5.1E+09
Kr-85	2.5E+17	1.0E+17	1.8E+17	5.3E+17
Sr-90	2.1E+18	8.4E+17	1.5E+18	4.4E+18
Y-90	2.1E+18	8.4E+17	1.5E+18	4.4E+18
Mo-93	3.9E+11	3.5E+11	5.5E+11	1.3E+12
Zr-93	1.6E+14	6.4E+13	1.1E+14	3.3E+14
Nb-93M	1.2E+14	5.1E+13	8.5E+13	2.6E+14
Nb-94	1.2E+13	1.6E+13	2.5E+13	5.4E+13
Tc-99	1.0E+15	4.6E+14	7.2E+14	2.2E+15
Pd-107	9.3E+12	5.1E+12	6.4E+12	2.1E+13
Ag-108	1.2E+11	4.9E+10	7.9E+10	2.5E+11
Ag-108M	1.4E+12	5.5E+11	9.1E+11	2.8E+12
Cd-113M	9.5E+14	5.1E+14	6.6E+14	2.1E+15
Sn-121	1.0E+14	5.4E+13	7.0E+13	2.3E+14
Sn-121M	2.2E+14	9.1E+13	1.2E+14	4.4E+14
Sn-126	2.6E+13	1.4E+13	1.9E+13	5.9E+13
Sb-125	5.7E+13	2.7E+13	3.6E+13	1.2E+14
Sb-126	4.0E+12	2.1E+12	2.8E+12	9.0E+12
Sb-126M	2.8E+13	1.5E+13	2.0E+13	6.3E+13
Te-125M	1.4E+13	6.6E+12	8.9E+12	2.9E+13
I-129	2.3E+12	1.1E+12	1.7E+12	5.1E+12
Ba-133	1.3E+11	4.6E+10	8.2E+10	2.6E+11
Cs-134	2.0E+13	8.7E+12	1.4E+13	4.3E+13

Tab. 5.5: (Cont.)

Waste sort **Nuclide**	**BE-1** **[Bq]**	**BE-2** **[Bq]**	**BE-3** **[Bq]**	**Total** **[Bq]**
Cs-135	3.3E+13	1.8E+13	2.5E+13	7.6E+13
Cs-137	3.1E+18	1.4E+18	2.2E+18	6.7E+18
Ba-137M	3.1E+18	1.4E+18	2.2E+18	6.8E+18
Pm-145	2.4E+11	1.1E+11	1.7E+11	5.2E+11
Pm-146	1.9E+11	1.1E+11	1.4E+11	4.4E+11
Pm-147	2.8E+14	1.3E+14	2.0E+14	6.1E+14
Sm-151	1.5E+16	8.5E+15	1.1E+16	3.4E+16
Eu-150	1.3E+09	1.4E+09	1.0E+09	3.7E+09
Eu-152	4.0E+13	3.5E+13	3.6E+13	1.1E+14
Eu-154	4.0E+16	1.9E+16	2.7E+16	8.7E+16
Eu-155	1.1E+15	5.8E+14	7.4E+14	2.4E+15
Tb-157	1.6E+10	6.0E+09	1.0E+10	3.3E+10
Tb-158	1.0E+11	6.8E+10	7.0E+10	2.4E+11
Ho-166M	1.9E+12	9.1E+11	1.3E+12	4.1E+12
Hf-178M	3.9E+13	1.2E+13	1.9E+13	6.9E+13
Pt-193	4.2E+11	1.5E+11	2.8E+11	8.6E+11
Tl-204	1.2E+10	4.3E+09	7.6E+09	2.3E+10
Tl-208	6.3E+10	2.7E+10	4.9E+10	1.4E+11
Pb-212	1.8E+11	7.5E+10	1.4E+11	3.9E+11
Bi-212	1.8E+11	7.5E+10	1.4E+11	3.9E+11
Po-212	1.1E+11	4.8E+10	8.9E+10	2.5E+11
Po-216	1.8E+11	7.5E+10	1.4E+11	3.9E+11
Rn-220	1.8E+11	7.5E+10	1.4E+11	3.9E+11
Ra-224	1.8E+11	7.5E+10	1.4E+11	3.9E+11
Th-228	1.8E+11	7.5E+10	1.4E+11	3.9E+11
Th-230	3.0E+10	1.4E+10	2.3E+10	6.6E+10
Th-231	8.5E+11	3.8E+11	7.2E+11	2.0E+12
Th-234	1.8E+13	7.4E+12	1.2E+13	3.7E+13
Pa-233	3.0E+13	1.2E+13	2.1E+13	6.3E+13
Pa-234	6.0E+10	2.4E+10	3.8E+10	1.2E+11
Pa-234M	1.8E+13	7.4E+12	1.2E+13	3.7E+13
U-232	1.6E+11	7.5E+10	1.4E+11	3.8E+11
U-233	6.3E+09	2.4E+09	4.4E+09	1.3E+10
U-234	9.1E+13	4.3E+13	6.7E+13	2.0E+14
U-235	8.5E+11	3.8E+11	7.2E+11	2.0E+12
U-236	1.9E+13	7.0E+12	1.4E+13	4.0E+13
U-238	1.8E+13	7.4E+12	1.2E+13	3.7E+13
Np-237	3.0E+13	1.2E+13	2.1E+13	6.3E+13
Np-238	2.5E+12	5.1E+12	2.0E+12	9.6E+12

Tab. 5.5: (Cont.)

Waste sort Nuclide	BE-1 [Bq]	BE-2 [Bq]	BE-3 [Bq]	Total [Bq]
Np-239	2.1E+15	1.8E+15	1.4E+15	5.2E+15
Pu-238	2.4E+17	1.4E+17	1.6E+17	5.4E+17
Pu-239	2.1E+16	1.3E+16	1.5E+16	4.9E+16
Pu-240	3.7E+16	3.0E+16	2.4E+16	9.1E+16
Pu-241	1.4E+18	1.1E+18	9.9E+17	3.5E+18
Pu-242	1.8E+14	1.4E+14	1.2E+14	4.3E+14
Am-241	2.8E+17	2.2E+17	1.9E+17	6.9E+17
Am-242M	5.4E+14	1.1E+15	4.2E+14	2.0E+15
Am-243	2.1E+15	1.8E+15	1.4E+15	5.2E+15
Cm-242	4.5E+14	8.6E+14	3.5E+14	1.7E+15
Cm-243	6.3E+14	8.9E+14	4.4E+14	2.0E+15
Cm-244	7.5E+16	8.2E+16	5.1E+16	2.1E+17
Cm-245	5.2E+13	7.9E+13	3.4E+13	1.7E+14
Cm-246	1.1E+13	1.5E+13	7.8E+12	3.4E+13
Cf-250	2.1E+09	4.5E+09	1.7E+09	8.3E+09
Total α **Total βγ** **Total Activity**	**6.6E+17** **1.2E+19** **1.3E+19**	**5.0E+17** **5.8E+18** **6.3E+18**	**4.4E+17** **8.6E+18** **9.1E+18**	**1.6E+18** **2.7E+19** **2.8E+19**

Waste sort Stable Element	BE-1 [Mol]	BE-2 [Mol]	BE-3 [Mol]	Total [Mol]
Ca	2.1E+03	8.8E+02	1.4E+03	4.4E+03
Fe	1.1E+06	5.2E+05	8.1E+05	2.4E+06
Co	9.2E+02	4.8E+02	7.5E+02	2.1E+03
Ni	2.5E+05	1.8E+05	2.8E+05	7.0E+05
Se	1.2E+03	5.3E+02	8.9E+02	2.6E+03
Sr	7.7E+03	3.2E+03	5.5E+03	1.6E+04
Zr	6.8E+06	2.0E+06	3.2E+06	1.2E+07
Nb	1.3E+03	2.8E+03	4.3E+03	8.4E+03
Pd	2.2E+04	1.2E+04	1.6E+04	4.9E+04
Ag	1.5E+03	8.6E+02	1.0E+03	3.4E+03
Cd	1.9E+03	1.1E+03	1.4E+03	4.4E+03
Sn	8.0E+04	2.4E+04	3.8E+04	1.4E+05
Sb	2.8E+02	1.3E+02	1.8E+02	6.0E+02
Sm	6.4E+03	2.9E+03	4.4E+03	1.4E+04
Eu	1.6E+03	8.2E+02	1.2E+03	3.6E+03
Ho	5.5E+00	3.1E+00	3.9E+00	1.2E+01
Hf	3.5E+02	1.1E+02	1.7E+02	6.2E+02
Pb	4.1E+02	1.2E+02	1.9E+02	7.2E+02

The total material inventory for SF wastes, arising under the 60 years of reactor operations scenario, is given in Tab. 5.6.

Tab. 5.6: SF Material inventory

Waste sort **Material**	**BE-1 + BE-2 + BE-3** **[kg]**
Spent Fuel	
UO_2	3.5E+06
MOX	1.7E+05
Fuel assembly structural materials	
Nickel alloys	3.3E+04
Steel	1.9E+05
Zircaloy	1.2E+06
Al_2O_3	1.5E+03
Total	**5.1E+06**

5.2 Overall summary data

5.2.1 Masses

A summary of the material composition of the wastes (excluding any overpacks for ILW, HLW and SF) is given in Tab. 5.7.

Tab. 5.7: Comparison of the material composition of all wastes

Waste type	**ILW**	**ILW**	**HLW**	**SF**
Scenario	Cemented waste option [kg]	High force compaction option [kg]	High force compaction option [kg]	All scenarios [1] [kg]
Metals	**8.9E+05**	**8.2E+05**	**5.8E+04**	**1.4E+06**
Steels	5.6E+05	4.9E+05	5.8E+04	1.9E+05
Al / Zn	3.4E+03	3.4E+03	-	-
Inconel	1.6E+04	1.6E+04	-	3.3E+04
Zircaloy	3.0E+05	3.0E+05	-	1.2E+06
Other	3.2E+03	3.2E+03	-	-
Inorganics	**2.2E+06**	**4.9E+05**	**2.9E+05**	**3.7E+06**
Salts	3.0E+04	3.0E+04	-	1.5E+03
Ceramics	-	-	-	3.7E+06
Ashes	-	-	-	-
Glass	3.1E+02	3.1E+02	2.9E+05	-
Concrete / Cement	2.1E+06	4.5E+05	-	-
Other	2.3E+03	2.3E+03	-	-
Organics	**7.4E+04**	**6.3E+04**	-	-
High Mwt subtotal	**7.4E+04**	**6.3E+04**	-	-
Bitumen	4.7E+04	4.7E+04	-	-
IX Resins	-	-	-	-
Cellulose	5.1E+02	5.1E+02	-	-
Other	2.7E+04	2.7E+04	-	-
Low Mwt subtotal	**8.4E+02**	**8.4E+02**	-	-
Detergents	7.5E+02	7.5E+02	-	-
Flocculants	6.5E+00	6.5E+00	-	-
Complexing agents	9.1E+01	9.1E+01	-	-
Other	-	-	-	-
Not Specified	-	-	-	-
Total	**3.1E+06**	**1.4E+06**	**3.2E+05**	**5.1E+06**

[1] It should be noted that the SF masses do not include the disposal canister materials.

5.2.2 Decayed inventories

To permit a comparison between the total radionuclide inventories, the individual waste sorts (except for SF which is already decayed to 40 years after discharge), were decayed to 40 years and then summed. Tab 5.8 compares the decayed inventories for ILW-1 and ILW-2 (see Chapter 3.3), Tab. 5.9 compares the decayed total inventories of ILW, HLW and SF.

Tab. 5.8: Total ILW-1 and ILW-2 inventory 40 years from production

Waste type **Nuclide**	**ILW-1** **[Bq]**	**ILW-2** **[Bq]**	**Total** **[Bq]**
H-3	2.2E+15	4.2E+10	2.2E+15
Be-10	5.2E+06	-	5.2E+06
C-14	1.8E+13	8.0E+08	1.8E+13
Cl-36	6.3E+10	1.6E+07	6.3E+10
Fe-55	3.2E+12	9.6E+06	3.2E+12
Co-60	4.0E+14	2.0E+10	4.0E+14
Ni-59	1.2E+14	6.9E+08	1.2E+14
Ni-63	1.4E+16	6.9E+10	1.4E+16
Se-79	1.3E+09	9.2E+06	1.3E+09
Kr-85	4.8E+13	-	4.8E+13
Sr-90	2.1E+15	7.7E+12	2.1E+15
Y-90	2.1E+15	7.7E+12	2.1E+15
Mo-93	1.4E+12	2.7E+06	1.4E+12
Zr-93	5.9E+12	1.0E+11	6.0E+12
Nb-93M	6.0E+12	8.4E+10	6.1E+12
Nb-94	3.2E+13	6.5E+06	3.2E+13
Tc-99	1.6E+12	3.8E+09	1.6E+12
Ru-106	2.2E+04	-	2.2E+04
Rh-106	2.2E+04	-	2.2E+04
Pd-107	2.2E+10	4.6E+08	2.3E+10
Ag-108	5.2E+07	4.2E+06	5.6E+07
Ag-108M	6.1E+08	4.6E+07	6.6E+08
Sn-121	2.0E+14	3.4E+08	2.0E+14
Sn-121M	2.6E+14	4.2E+08	2.6E+14
Sn-126	1.6E+10	3.0E+09	1.9E+10
Sb-126M	1.6E+10	2.9E+09	1.9E+10
Sb-125	9.2E+11	3.8E+08	9.2E+11
Te-125M	2.3E+11	9.6E+07	2.3E+11
Sb-126	2.2E+09	4.2E+08	2.6E+09
I-129	4.3E+10	3.8E+09	4.7E+10

Tab. 5.8: (Cont.)

Waste type Nuclide	**ILW-1 [Bq]**	**ILW-2 [Bq]**	**Total [Bq]**
Cs-134	6.2E+09	2.5E+07	6.3E+09
Cs-135	1.2E+11	3.0E+08	1.2E+11
Cs-137	4.3E+15	3.2E+13	4.3E+15
Ba-137M	4.0E+15	3.0E+13	4.0E+15
Pm-147	7.0E+10	1.3E+08	7.0E+10
Sm-147	6.6E+04	-	6.6E+04
Sm-151	5.3E+13	2.6E+11	5.3E+13
Eu-152	2.4E+10	7.3E+08	2.5E+10
Eu-154	3.1E+13	2.4E+11	3.2E+13
Eu-155	2.9E+11	9.2E+09	3.0E+11
Ho-166M	8.1E+07	-	8.1E+07
Tl-207	3.6E+05	4.2E+04	4.1E+05
Tl-208	5.7E+07	-	5.7E+07
Pb-209	3.5E+03	-	3.5E+03
Pb-210	4.7E+04	6.1E+03	5.3E+04
Bi-210	4.7E+04	6.1E+03	5.3E+04
Pb-211	3.6E+05	4.2E+04	4.1E+05
Bi-211	3.6E+05	4.2E+04	4.1E+05
Pb-212	1.6E+08	-	1.6E+08
Bi-212	1.6E+08	-	1.6E+08
Pb-214	1.5E+05	2.0E+04	1.7E+05
Bi-214	1.5E+05	2.0E+04	1.7E+05
Po-210	4.5E+04	6.1E+03	5.1E+04
Po-211	6.4E+02	-	6.4E+02
Po-212	1.0E+08	-	1.0E+08
Bi-213	3.5E+03	-	3.5E+03
Po-213	3.4E+03	-	3.4E+03
Po-214	1.5E+05	2.0E+04	1.7E+05
Po-215	3.6E+05	4.2E+04	4.1E+05
Po-216	1.6E+08	-	1.6E+08
Po-218	1.5E+05	2.0E+04	1.7E+05
At-217	3.5E+03	-	3.5E+03
Rn-219	3.6E+05	4.2E+04	4.1E+05
Rn-220	1.6E+08	-	1.6E+08
Rn-222	1.5E+05	2.0E+04	1.7E+05

Tab. 5.8: (Cont.)

Waste type Nuclide	**ILW-1 [Bq]**	**ILW-2 [Bq]**	**Total [Bq]**
Fr-221	3.5E+03	-	3.5E+03
Fr-223	5.1E+03	5.7E+02	5.7E+03
Ra-223	3.6E+05	4.2E+04	4.1E+05
Ra-224	1.6E+08	-	1.6E+08
Ra-225	3.1E+03	-	3.1E+03
Ac-225	3.5E+03	-	3.5E+03
Ra-226	1.5E+05	2.0E+04	1.7E+05
Ac-227	3.7E+05	4.2E+04	4.1E+05
Th-227	3.6E+05	3.8E+04	4.0E+05
Th-228	1.6E+08	-	1.6E+08
Th-229	3.5E+03	-	3.5E+03
Th-230	1.8E+07	2.3E+06	2.1E+07
Th-231	1.0E+09	1.1E+08	1.1E+09
Pa-231	8.3E+05	9.6E+04	9.3E+05
Pa-233	7.4E+09	5.0E+08	7.9E+09
Th-234	1.1E+10	1.5E+09	1.2E+10
Pa-234	3.6E+07	5.0E+06	4.1E+07
Pa-234M	1.1E+10	1.5E+09	1.2E+10
U-232	1.6E+08	-	1.6E+08
U-233	1.7E+06	7.7E+04	1.8E+06
U-234	5.5E+10	6.5E+09	6.2E+10
U-235	1.0E+09	1.1E+08	1.1E+09
U-235M	1.7E+13	6.5E+11	1.8E+13
U-236	7.6E+09	1.2E+09	8.8E+09
U-237	1.4E+10	7.7E+08	1.5E+10
Np-237	7.4E+09	5.0E+08	7.9E+09
U-238	1.1E+10	1.5E+09	1.2E+10
Np-238	5.0E+08	7.3E+07	5.7E+08
Pu-238	7.6E+13	3.8E+12	8.0E+13
Np-239	3.6E+11	3.7E+10	4.0E+11
Pu-239	1.7E+13	6.5E+11	1.8E+13
Pu-240	2.4E+13	7.7E+11	2.4E+13
Pu-241	5.8E+14	3.1E+13	6.1E+14
Am-241	1.6E+14	6.9E+12	1.7E+14
Pu-242	7.6E+10	4.2E+09	8.0E+10

Tab. 5.8: (Cont.)

Waste type **Nuclide**	**ILW-1** **[Bq]**	**ILW-2** **[Bq]**	**Total** **[Bq]**
Am-242	1.0E+11	1.5E+10	1.2E+11
Am-243	3.6E+11	3.7E+10	4.0E+11
Cm-242	8.5E+10	1.3E+10	9.7E+10
Am-242M	1.0E+11	1.5E+10	1.2E+11
Cm-243	9.2E+10	1.6E+09	9.4E+10
Cm-244	5.4E+12	1.0E+11	5.5E+12
Cm-245	1.5E+09	3.7E+07	1.6E+09
Cm-246	3.1E+08	6.5E+06	3.1E+08
Total α **Total βγ** **Total Activity**	**2.9E+14** **3.1E+16** **3.1E+16**	**1.2E+13** **1.1E+14** **1.2E+14**	**3.0E+14** **3.1E+16** **3.1E+16**

Tab. 5.9: Decayed radionuclide inventories for all wastes 40 years from production

Waste type Nuclide	ILW [Bq]	HLW [Bq]	SF [Bq]	Total [Bq]
H-3	2.2E+15	-	1.4E+16	1.6E+16
Be-10	5.2E+06	1.8E+08	3.2E+10	3.2E+10
C-14	1.8E+13	1.1E+11	1.9E+14	2.1E+14
Cl-36	6.3E+10	-	3.7E+12	3.7E+12
Ar-39	-	-	2.3E+13	2.3E+13
Ca-41	-	2.1E+10	5.5E+11	5.7E+11
Fe-55	3.2E+12	4.4E+10	2.1E+13	2.4E+13
Co-60	4.0E+14	3.4E+13	1.6E+15	2.0E+15
Ni-59	1.2E+14	1.3E+12	2.5E+14	3.8E+14
Ni-63	1.4E+16	1.4E+14	2.7E+16	4.2E+16
Se-79	1.3E+09	9.8E+11	3.0E+12	4.0E+12
Kr-81	4.8E+13	-	5.1E+09	4.8E+13
Kr-85	-	-	5.3E+17	5.3E+17
Sr-90	2.1E+15	1.0E+18	4.4E+18	5.4E+18
Y-90	2.1E+15	1.0E+18	4.4E+18	5.4E+18
Mo-93	1.4E+12	3.7E+10	1.3E+12	2.7E+12
Zr-93	5.9E+12	7.8E+13	3.3E+14	4.1E+14
Nb-93M	6.0E+12	6.5E+13	2.6E+14	3.3E+14
Nb-94	3.2E+13	3.2E+11	5.4E+13	8.6E+13
Tc-99	1.6E+12	6.4E+14	2.2E+15	2.8E+15
Pd-107	2.3E+10	5.0E+12	2.1E+13	2.6E+13
Ag-108	5.6E+07	3.3E+10	2.5E+11	2.8E+11
Ag-108M	6.6E+08	3.8E+11	2.8E+12	3.2E+12
Cd-113M	-	2.3E+14	2.1E+15	2.4E+15
Sn-121	2.0E+14	3.3E+12	2.3E+14	4.3E+14
Sn-121M	2.6E+14	4.2E+12	4.4E+14	7.0E+14
Sn-126	1.9E+10	3.1E+13	5.9E+13	9.0E+13
Sb-125	9.2E+11	7.3E+12	1.2E+14	1.3E+14
Sb-126	2.6E+09	4.3E+12	9.0E+12	1.3E+13
Sb-126M	1.9E+10	3.1E+13	6.3E+13	9.4E+13
Te-125M	2.3E+11	1.8E+12	2.9E+13	3.1E+13
I-129	4.7E+10	1.3E+09	5.1E+12	5.2E+12
Ba-133	-	-	2.6E+11	2.6E+11
Cs-134	6.3E+09	1.7E+12	4.3E+13	4.4E+13

Tab. 5.9: (Cont.)

Waste type Nuclide	**ILW [Bq]**	**HLW [Bq]**	**SF [Bq]**	**Total [Bq]**
Cs-135	1.2E+11	2.0E+13	7.6E+13	9.6E+13
Cs-137	4.3E+15	1.5E+18	6.7E+18	8.2E+18
Ba-137M	4.0E+15	1.4E+18	6.8E+18	8.2E+18
Pm-145	-	-	5.2E+11	5.2E+11
Pm-146	-	2.3E+11	4.4E+11	6.7E+11
Pm-147	7.0E+10	4.5E+13	6.1E+14	6.6E+14
Sm-151	5.3E+13	1.1E+16	3.4E+16	4.5E+16
Eu-152	2.5E+10	2.7E+13	1.1E+14	1.4E+14
Eu-154	3.1E+13	9.1E+15	8.7E+16	9.6E+16
Eu-155	3.0E+11	3.2E+14	2.4E+15	2.7E+15
Tb-157	-	-	3.3E+10	3.3E+10
Tb-158	-	-	2.4E+11	2.4E+11
Ho-166M	8.1E+07	1.2E+11	4.1E+12	4.2E+12
Hf-178M	-	-	6.9E+13	6.9E+13
Pt-193	-	-	8.6E+11	8.6E+11
Tl-204	-	-	2.3E+10	2.3E+10
Tl-208	5.7E+07	2.0E+08	1.4E+11	1.4E+11
Pb-212	1.6E+08	5.7E+08	3.9E+11	3.9E+11
Bi-212	1.6E+08	5.7E+08	3.9E+11	3.9E+11
Po-212	1.0E+08	3.6E+08	2.5E+11	2.5E+11
Po-216	1.6E+08	5.7E+08	3.9E+11	3.9E+11
Rn-220	1.6E+08	5.7E+08	3.9E+11	3.9E+11
Ra-224	1.6E+08	5.7E+08	3.9E+11	3.9E+11
Th-228	1.6E+08	5.7E+08	3.9E+11	3.9E+11
Th-230	2.0E+07	4.4E+09	6.6E+10	7.1E+10
Th-231	1.1E+09	6.4E+08	2.0E+12	2.0E+12
Th-234	1.2E+10	1.0E+10	3.7E+13	3.7E+13
Pa-233	7.9E+09	1.3E+13	6.3E+13	7.6E+13
Pa-234	4.1E+07	3.5E+07	1.2E+11	1.2E+11
Pa-234M	1.2E+10	1.0E+10	3.7E+13	3.7E+13
U-232	1.6E+08	5.3E+08	3.8E+11	3.8E+11
U-233	1.8E+06	2.4E+09	1.3E+10	1.5E+10
U-234	6.2E+10	7.3E+10	2.0E+14	2.0E+14
U-235	1.1E+09	6.4E+08	2.0E+12	2.0E+12

Tab. 5.9: (Cont.)

Waste type Nuclide	**ILW [Bq]**	**HLW [Bq]**	**SF [Bq]**	**Total [Bq]**
U-236	8.8E+09	8.9E+09	4.0E+13	4.0E+13
U-238	1.2E+10	1.0E+10	3.7E+13	3.7E+13
Np-237	7.9E+09	1.3E+13	6.3E+13	7.6E+13
Np-238	5.7E+08	2.9E+12	9.6E+12	1.2E+13
Np-239	4.0E+11	5.9E+14	5.2E+15	5.8E+15
Pu-238	8.0E+13	3.5E+14	5.4E+17	5.4E+17
Pu-239	1.8E+13	2.8E+13	4.9E+16	4.9E+16
Pu-240	2.4E+13	1.5E+14	9.1E+16	9.1E+16
Pu-241	6.1E+14	1.0E+15	3.5E+18	3.5E+18
Pu-242	8.0E+10	1.4E+11	4.3E+14	4.3E+14
Am-241	1.7E+14	4.1E+16	6.9E+17	7.3E+17
Am-242M	1.2E+11	6.0E+14	2.0E+15	2.6E+15
Am-243	4.0E+11	5.9E+14	5.2E+15	5.8E+15
Cm-242	9.7E+10	5.0E+14	1.7E+15	2.1E+15
Cm-243	9.4E+10	2.0E+14	2.0E+15	2.2E+15
Cm-244	5.5E+12	1.0E+16	2.1E+17	2.2E+17
Cm-245	1.6E+09	4.5E+12	1.7E+14	1.7E+14
Cm-246	3.1E+08	8.3E+11	3.4E+13	3.5E+13
Cf-250	-	-	8.3E+09	8.3E+09
Total α **Total βγ** **Total Activity**	**3.0E+14** **3.1E+16** **3.1E+16**	**5.3E+16** **5.0E+18** **5.0E+18**	**1.6E+18** **2.7E+19** **2.8E+19**	**1.7E+18** **3.2E+19** **3.3E+19**

5.2.3 Properties of wastes

A summary of the main properties of the waste sorts is given in Tab. 5.10. It should be noted that for reprocessing wastes the maximum values are given in brackets. For spent fuel the variation in radionuclide inventory is examined via a sensitivity study using canisters BE-4, BE-5, BE-6 and BE-7.

Tab. 5.10: List of waste sorts with main properties

Waste sort	Raw waste	Material of conditioning	α activity [Bq/Cont.]	βγ activity [Bq/Cont.]	Dose Rate at 0m [Sv/h]	Heat Output [W]
WA (Reprocessing Waste): Time of production						
WA-COG-1	Vitrified residues (HLW)	Glass	1.1E+14 (4.3E+14)	2.9E+16 (4.0E+16)	2.5E+03 (3.5E+03)	2.9E+03 (4.2E+03)
WA-COG-2	Sludges/ concentrates	Bitumen	2.2E+10 (5.0E+10)	1.8E+12 (5.3E+12)	4.4E-01 (2.0E+00)	1.6E-01 (4.4E-01)
WA-COG-4	Hulls and ends	Cement	2.1E+11 (2.1E+12)	8.2E+14 (3.7E+15)	3.9E+01 (7.5E+01)	1.2E+02 (5.5E+02)
WA-COG-4A	Compacted hulls and ends	-	1.5E+11 (6.4E+12)	5.8E+14 (3.5E+15)	7.4E+01 (2.8E+02)	8.6E+01 (9.0E+01)
WA-COG-6	α tech. waste	-	4.5E+10 (6.3E+11)	2.3E+12 (3.8E+12)	3.1E-02 (4.0E-02)	2.1E-01 (7.9E-01)
WA-COG-6A	Compacted tech. waste	-	4.0E+10 (5.8E+11)	2.1E+12 (9.0E+13)	1.2E-01 (1.5E+02)	1.8E-01 (8.2E+00)
WA-BNF-1	Vitrified residues (HLW)	Glass	2.2E+14 (5.2E+14)	3.6E+16 (9.1E+16)	2.1E+03 (4.3E+03)	3.5E+03 (9.6E+03)
WA-BNF-2	MEB Crud & $BaCO_3$ slurry	Cement	3.2E+8 (1.4E+10)	5.1E+11 (1.9E+13)	9.4E-02 (4.0E+00)	7.5E-02 (2.9E+00)
WA-BNF-4	Hulls and ends	Cement	1.4E+11 (1.4E+12)	1.7E+14 (6.2E+14)	1.4E+01 (5.0E+01)	1.9E+01 (8.0E+01)
WA-BNF-7	Centrifuge cake slurry	Cement	6.0E+11 (7.4E+11)	5.4E+13 (4.5E+14)	2.1E+00 (1.0E+01)	4.8E+00 (5.5E+01)
SF (Spent Fuel): After 40 years decay						
BE-1	UO_2 spent fuel (48 GWd/t_{IHM})	-	6.9E+14	1.3E+16	9.0E-03	1.5E+03
BE-2	UO_2/MOX spent fuel (48 GWd/t_{IHM})	-	1.1E+15	1.3E+16	3.3E-02	1.8E+03
BE-3	UO_2 spent fuel (48 GWd/t_{IHM})	-	6.5E+14	1.3E+16	7.0E-03	1.5E+03
BE-4	UO_2 spent fuel (55 GWd/t_{IHM})	-	8.4E+14	1.4E+16	8.0E-03	1.8E+03
BE-5	UO_2 spent fuel (3 × 55 + 1 × 65 GWd/t_{IHM})	-	8.8E+14	1.5E+16	1.5E-02	1.8E+03
BE-6	UO_2 spent fuel (3 × 55 + 1 × 75 GWd/t_{IHM})	-	9.8E+14	1.5E+16	2.1E-02	2.0E+03
BE-7	UO_2/MOX spent fuel (3 × 48 + 1 × 65 GWd/t_{IHM})	-	1.4E+15	1.3E+16	4.4E-02	2.1E+03

5.2.4 Overall volumes

According to the current model inventory and the waste allocation assumed, the rounded ILW, HLW and SF waste sort volumes are summarised in Tab. 5.11, for the two scenarios (Cemented waste option and High force compaction option, see Chapter 2.1). The waste volume given is an estimation of the expected disposal volume, i.e. waste drums containing ILW packaged in disposal containers, stainless steel flasks with HLW glass packaged in steel HLW canisters and SF packaged in SF canisters (NAGRA 2002a).

Tab. 5.11: Total (rounded) conditioned volumes for HLW and ILW (from Tab. 3.2)

Repository Type	**Cemented Waste Option**		**High Force Compaction Option**	
	60 years [m^3]	300 GWa(e) [m^3]	60 years [m^3]	300 GWa(e) [m^3]
HLW [1]	1'015	1'015	1'015	1'015
ILW [1]	7'300	8'200	6'400	7'300
SF [2]	8'300	14'600	8'300	14'600
Total	**16'600**	**23'800**	**15'700**	**22'900**

1 See Tab. 3.2.

2 Derived from Tab. 4.7: BE-1, BE-2, BE-3 with number of canisters, canister diameter and canister length.

6 Conclusions

A model radioactive waste inventory has been developed for Swiss waste disposal projects (repository planning, safety assessment). This inventory characterises all ILW and HLW expected to arise from reprocessing of Swiss spent fuel and the spent fuel not currently contracted for reprocessing (direct disposal) over a 60 year reactor operation scenario. To permit sensitivity studies to be performed, waste volumes and radionuclide inventories for a 300 GWa(e) production scenario, and for the variation in spent fuel burnup 4 additional spent fuel waste sorts (BE-4, BE-5, BE-6 and BE-7), are provided.

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Appendix A

Inventory data for reprocessing waste sorts

List of Tables

A.1 Inventory data for WA-COG-1

Tab. A.1-1: Material inventory of WA-COG-1

Oxide	WA-COG-1 [kg/flask]	Oxide	WA-COG-1 [kg/flask]	Oxide	WA-COG-1 [kg/flask]
Li_2O	8.085	SrO	1.402	CeO_2	4.044
B_2O_3	56.576	Y_2O_3	0.822	Pr_6O_{11}	1.862
Na_2O	36.917	ZrO_2	12.529	Nd_2O_3	6.467
MgO	0.005	MoO_3	8.618	Sm_2O_3	1.304
Al_2O_3	18.116	Tc_2O_7	2.097	Eu_2O_3	0.237
SiO_2	182.809	RuO_2	4.835	Gd_2O_3	4.018
P_2O_5	1.200	Rh	0.769	U_3O_8	1.853
SO_3	0.0001	Pd	2.169	NpO_2	0.739
CaO	16.167	Ag	0.095	PuO_2	0.018
Cr_2O_3	0.4	CdO	0.19	Am_2O_3	0.404
Fe_2O_3	4.400	SnO	0.186	CmO_2	0.029
NiO	0.8	TeO_2	0.801	Misc.	0.246
ZnO	10.015	Cs_2O	3.761	**Total**	**400.0**
SeO_2	0.11	BaO	2.401		
Rb_2O	0.536	La_2O_3	1.968		

Occurrence	Material	Mass [kg/flask]
Raw waste	Actinide oxides	3.04E+00
Raw waste	Fines	4.01E+00
Raw waste	Process additives	2.61E+01
Raw waste	Fission products	5.60E+01
Total Raw waste		**8.92E+01**
Matrix material	Al_2O_3	1.34E+01
Matrix material	B_2O_3	5.66E+01
Matrix material	CaO	1.62E+01
Matrix material	Li_2O	8.08E+00
Matrix material	Na_2O	2.18E+01
Matrix material	SiO_2	1.83E+02
Matrix material	ZnO	9.95E+00
Matrix material	ZrO_2	2.17E+00
Total Matrix material		**3.11E+02**
Flask material	Steel Z15 CN 24.13	8.00E+01
Total Flask material		**8.00E+01**
Total Flask with waste	**WA-COG-1**	**4.80E+02**

Tab. A.1-2: Physical Characteristics of WA-COG-1

Parameter	**Average**	**Maximum**
Total Mass [kg]	480	550
Volume [m^3]	0.18	-
Surface to mass ratio of metallic materials	**[m^2/kg]**	**[kg]**
Steel Z15 CN 24.13	6.1E-02	8.0E+01

Tab. A.1-3: Radiological characteristics of WA-COG-1 at time of production

Parameter	**Average**	**Maximum**
0 m γ dose rate [Sv/h]	2'520	3'530
1 m γ dose rate [Sv/h]	170	230
2 m γ dose rate [Sv/h]	50	80
0 m *n* dose rate [Sv/h]	9.1E-02	3.4E-01
1 m *n* dose rate [Sv/h]	1.2E-02	4.4E-02
2 m *n* dose rate [Sv/h]	6.4E-03	2.4E-02
Heat output [W]	2'950	4'230

Tab. A.1-4: Average and maximum radionuclide inventory of WA-COG-1 at time of production

Nuclide	Average [Bq/flask]	Maximum [Bq/flask]	Nuclide	Average [Bq/flask]	Maximum [Bq/flask]
Be-10	2.0E+05	2.8E+05	Ba-137M	4.5E+15	6.4E+15
C-14	1.9E+08	2.6E+08	Ce-144	1.7E+15	2.0E+15
Ca-41	6.8E+06	9.4E+06	Pr-144	1.7E+15	2.0E+15
Ca-45	4.6E+07	6.4E+07	Pm-146	6.7E+10	9.3E+10
Mn-54	6.3E+09	8.7E+09	Pm-147	2.4E+15	3.3E+15
Fe-55	5.1E+11	7.1E+11	Sm-151	1.9E+13	3.0E+13
Co-58	2.6E+06	3.6E+06	Eu-152	2.8E+11	3.9E+11
Co-60	5.5E+12	7.6E+12	Eu-154	3.2E+14	5.2E+14
Ni-59	2.4E+09	3.3E+09	Eu-155	1.6E+14	5.3E+14
Ni-63	3.4E+11	4.7E+11	Gd-153	1.2E+12	1.7E+12
Zn-65	9.8E+10	1.4E+11	Tb-160	4.9E+08	6.8E+08
Se-79	1.2E+09	1.6E+09	Ho-166M	1.8E+08	2.5E+08
Sr-90	3.5E+15	4.6E+15	Ac-227	1.1E+05	3.1E+06
Y-90	3.5E+15	4.6E+15	Th-228	3.4E+08	5.9E+09
Zr-93	9.4E+10	1.5E+11	Th-230	3.1E+06	8.4E+06
Zr-95	1.1E+10	1.3E+10	Pa-231	8.3E+05	3.3E+06
Nb-93M	2.0E+10	4.0E+10	U-232	1.1E+06	4.4E+06
Nb-94	6.8E+08	9.4E+08	U-233	8.3E+04	3.3E+05
Mo-93	5.1E+07	7.1E+07	U-234	7.2E+07	2.7E+08
Tc-99	8.4E+11	1.2E+12	U-235	1.2E+06	3.9E+06
Ru-106	2.1E+15	2.9E+15	U-236	1.7E+07	7.1E+07
Rh-102	2.0E+10	2.8E+10	U-238	1.9E+07	5.5E+07
Rh-106	2.1E+15	2.9E+15	Np-235	1.9E+07	7.6E+07
Pd-107	6.5E+09	1.1E+10	Np-237	1.7E+10	6.8E+10
Ag-108M	5.8E+08	8.0E+08	Pu-236	1.0E+07	2.3E+08
Ag-110M	3.0E+12	9.9E+13	Pu-238	2.1E+11	3.1E+12
Cd-109	7.0E+09	9.7E+09	Pu-239	2.0E+10	1.5E+11
Cd-113M	2.2E+12	3.1E+12	Pu-240	3.6E+10	2.4E+11
Sn-119M	3.9E+11	5.4E+11	Pu-241	6.1E+12	5.2E+13
Sn-121M	8.8E+09	1.7E+12	Pu-242	9.8E+07	1.5E+09
Sn-123	6.9E+10	9.6E+10	Am-241	3.3E+13	1.4E+14
Sn-126	3.7E+10	5.6E+10	Am-242M	7.3E+11	2.9E+12
Sb-124	3.1E+06	4.3E+06	Am-243	7.6E+11	4.1E+12
Sb-125	2.5E+14	3.5E+14	Cm-242	4.1E+12	6.1E+12
Te-123M	1.3E+08	1.8E+08	Cm-243	6.9E+11	2.9E+12
Te-125M	5.9E+13	8.2E+13	Cm-244	7.1E+13	2.7E+14
Te-127M	8.4E+10	1.2E+11	Cm-245	5.9E+09	1.2E+11
I-129	1.6E+06	2.6E+08	Cm-246	1.1E+09	6.4E+10
Cs-134	2.0E+15	2.5E+15	**Total α**	**1.1E+14**	**4.3E+14**
Cs-135	2.0E+10	1.4E+11	**Total βγ**	**2.9E+16**	**4.0E+16**
Cs-137	4.8E+15	6.7E+15	**Total Activity**	**2.9E+16**	**4.0E+16**

Tab. A.1-5: Decay of average flask heat output from time of production for WA-COG-1

Years from time of production	**0**	**3**	**10**	**30**	**40**	**100**	**300**	**1'000**
Inventory Activity	**Average [Bq/flask]**	**Average [Bq/flask]**	**Average [Bq/flask]**	**Average [Bq/flask]**	**Average [Bq/flask]**	**Average [Bq/flask]**	**Average [Bq/flask]**	**Average [Bq/flask]**
Total α	1.1E+14	9.9E+13	8.3E+13	5.6E+13	4.9E+13	3.2E+13	2.2E+13	7.7E+12
Total βγ	2.9E+16	1.8E+16	1.3E+16	8.1E+15	6.4E+15	1.6E+15	1.9E+13	1.9E+12
Total Activity	2.9E+16	1.9E+16	1.4E+16	8.2E+15	6.5E+15	1.6E+15	4.1E+13	9.6E+12
Heat output	**[W/flask]**	**[W/flask]**	**[W/flask]**	**[W/flask]**	**[W/flask]**	**[W/flask]**	**[W/flask]**	**[W/flask]**
Total Heat	3.0E+03	1.7E+03	1.1E+03	6.8E+02	5.4E+02	1.5E+02	2.1E+01	7.0E+00
Total α	1.0E+02	9.2E+01	7.7E+01	5.2E+01	4.5E+01	2.9E+01	2.0E+01	6.9E+00
Total β	2.4E+03	1.1E+03	6.7E+02	3.9E+02	3.0E+02	7.1E+01	6.8E-01	7.5E-02
Total γ	4.8E+02	4.5E+02	3.8E+02	2.4E+02	1.9E+02	4.8E+01	4.9E-01	4.8E-04

Years from time of production	**3'000**	**1.0E+04**	**3.0E+04**	**1.0E+05**	**3.0E+05**	**1.0E+06**	**3.0E+06**	**1.0E+07**
Inventory Activity	**Average [Bq/flask]**	**Average [Bq/flask]**	**Average [Bq/flask]**	**Average [Bq/flask]**	**Average [Bq/flask]**	**Average [Bq/flask]**	**Average [Bq/flask]**	**Average [Bq/flask]**
Total α	1.1E+12	5.5E+11	2.2E+11	9.1E+10	1.2E+11	1.3E+11	6.8E+10	7.3E+09
Total βγ	1.8E+12	1.6E+12	1.3E+12	9.3E+11	6.0E+11	2.4E+11	9.7E+10	9.2E+09
Total Activity	2.9E+12	2.1E+12	1.5E+12	1.0E+12	7.2E+11	3.7E+11	1.7E+11	1.7E+10
Heat output	**[W/flask]**	**[W/flask]**	**[W/flask]**	**[W/flask]**	**[W/flask]**	**[W/flask]**	**[W/flask]**	**[W/flask]**
Total Heat	1.0E+00	5.2E-01	2.2E-01	1.1E-01	1.3E-01	1.3E-01	7.0E-02	7.5E-03
Total α	9.8E-01	4.7E-01	1.9E-01	8.4E-02	1.2E-01	1.3E-01	6.7E-02	7.2E-03
Total β	6.7E-02	4.9E-02	3.1E-02	2.4E-02	1.6E-02	5.9E-03	2.4E-03	2.6E-04
Total γ	4.3E-04	4.3E-04	4.2E-04	4.1E-04	3.7E-04	2.7E-04	1.1E-04	4.2E-06

A.2 Inventory data for WA-COG-2

Tab. A.2-1: Material inventory of WA-COG-2

Occurrence	**Material**	**Mass [kg/Container]**
Raw waste	Barium Sulphate ($BaSO_4$)	2.54E+01
Raw waste	Detergent	1.95E+00
Raw waste	Diatomaceous Earth	5.40E+00
Raw waste	Cobalt Sulphide (CoS)	5.40E+00
Raw waste	$Na_2SO_4 + K_2SO_4$	3.65E+00
Raw waste	Sodium Nitrate ($NaNO_3$)	2.51E+01
Raw waste	Sodium Nitrite ($NaNO_2$)	2.20E+00
Raw waste	Nickel Ferrocyanide	5.40E+00
Raw waste	Tecprofloc 83/A22	1.70E-02
Raw waste	Titanium Sulphate	3.22E+00
Raw waste	Tributylphosphate (TBP)	1.70E-01
Raw waste	Water	3.90E+00
Total Raw waste		**8.20E+01**
Matrix material	Bitumen 80/100	1.22E+02
Total Matrix material		**1.22E+02**
Void	Air	6.30E-02
Container material	Steel Z2 CND 17.12	2.00E+01
Total Container material		**2.00E+01**
Total Waste Container	**WA-COG-2**	**2.24E+02**

Tab. A.2-2: Physical Characteristics of WA-COG-2

Parameter	Average	Maximum
Total Mass [kg]	224	250
Volume [m^3]	0.22	-

Surface to mass ratio of metallic materials	[m^2/kg]	[kg]
Steel Z2 CND 17.12	2.00E-01	2.00E+01

Tab. A.2-3: Radiological characteristics of WA-COG-2 at time of production

Parameter	Average	Maximum
0 m γ dose rate [Sv/h]	4.4E-01	1.5E+00 [1]
1 m γ dose rate [Sv/h]	1.8E-02	6.4E-02
2 m γ dose rate [Sv/h]	5.6E-03	2.0E-02
0 m *n* dose rate [Sv/h]	3.4E-07	1.0E-06
1 m *n* dose rate [Sv/h]	-	-
2 m *n* dose rate [Sv/h]	7.6E-09	-
Heat output [W]	1.6E-01	4.4E-01

Radiological gas production from time of production		
Time [years]	**Production [l/y]**	**Cumulative Production [m^3]**
0	2.8E+00	0.0E+00
5	1.3E+00	8.5E-03
10	1.3E+00	1.5E-02
20	1.3E+00	2.8E-02
40	1.2E+00	5.2E-02
80	1.0E+00	9.6E-02
100	9.4E-01	1.2E-01

[1] Estimated maximum – COGEMA guaranteed value is 2.0E+00 Sv/h.

Tab. A.2-4: Average and maximum radionuclide inventory of WA-COG-2 at time of production

Nuclide	Average [Bq/Cont.]	Maximum [Bq/Cont.]	Nuclide	Average [Bq/Cont.]	Maximum [Bq/Cont.]
H-3	1.0E+09	2.9E+09	Cs-135	7.7E+05	3.7E+06
C-14	2.1E+06	5.9E+06	Cs-137	2.1E+11	9.9E+11
Cl-36	4.1E+04	1.2E+05	Ba-137M	2.0E+11	9.4E+11
Mn-54	1.8E+07	5.3E+07	Ce-144	8.7E+10	1.8E+11
Fe-55	6.4E+08	1.8E+09	Pr-144	8.7E+10	1.8E+11
Co-57	1.8E+08	5.3E+08	Pm-147	1.3E+10	3.8E+10
Co-60	1.0E+10	2.9E+10	Sm-151	9.0E+08	4.3E+09
Ni-59	1.8E+06	5.3E+06	Eu-152	1.5E+07	4.4E+07
Ni-63	2.4E+08	6.8E+08	Eu-154	1.5E+10	4.4E+10
Zn-65	8.7E+08	2.5E+09	Eu-155	8.2E+09	2.4E+10
Se-79	2.4E+04	7.1E+04	U-234	1.6E+07	1.1E+08
Sr-90	5.3E+10	2.1E+11	U-235	2.9E+05	2.2E+06
Y-90	5.3E+10	2.1E+11	U-236	3.2E+06	2.5E+07
Zr-93	2.7E+08	7.6E+08	U-238	3.8E+06	2.9E+07
Mo-93	7.0E+03	3.3E+04	Np-237	1.1E+06	2.5E+06
Nb-93M	4.9E+07	1.4E+08	Np-239	9.6E+07	2.1E+08
Nb-94	1.7E+04	8.0E+04	Pu-238	1.4E+10	3.2E+10
Nb-95	3.1E+09	8.8E+09	Pu-239	1.7E+09	3.8E+09
Tc-99	1.0E+07	2.9E+07	Pu-240	2.0E+09	4.5E+09
Ru-106	2.2E+11	4.6E+11	Pu-241	5.6E+11	1.3E+12
Rh-106	2.2E+11	4.6E+11	Pu-242	1.1E+07	2.5E+07
Pd-107	1.2E+06	3.5E+06	Am-241	2.9E+09	6.5E+09
Ag-108M	1.3E+05	6.2E+05	Am-242M	4.9E+07	1.1E+08
Ag-110M	1.9E+09	8.9E+09	Am-243	9.6E+07	2.1E+08
Sn-119M	1.1E+08	5.2E+08	Cm-242	1.5E+08	3.5E+08
Sn-121M	1.9E+06	9.2E+06	Cm-243	1.1E+07	2.4E+07
Sn-123	1.1E+08	5.2E+08	Cm-244	1.2E+09	2.7E+09
Sn-126	7.7E+06	3.7E+07	Cm-245	9.6E+04	2.2E+05
Sb-125	2.3E+10	6.6E+10	Cm-246	1.7E+04	3.8E+04
Te-125M	2.8E+09	7.9E+09	**Total α**	**2.2E+10**	**5.0E+10**
I-129	1.0E+07	2.9E+07	**Total βγ**	**1.8E+12**	**5.3E+12**
Cs-134	4.6E+10	1.3E+11	**Total Activity**	**1.8E+12**	**5.3E+12**

A.3 Inventory data for WA-COG-4

Tab. A.3-1: Material inventory of WA-COG-4

Occurrence	Material	Mass [kg/Container]
Raw waste	Inconel (X-750)	3.18E+01
Raw waste	Steel	1.03E+02
Raw waste	Zircaloy	6.38E+02
Total Raw waste		**7.73E+02**
Matrix material	Fontainebleau Sand 0.3	3.53E+02
Matrix material	Fluidifier LL923	9.00E+00
Matrix material	Water	4.94E+02
Matrix material	Cement CPA 55 HTS	1.39E+03
Total Matrix material		**2.25E+03**
Void	Air	2.15E-01
Container material	Steel Z2 CN 18.10	3.48E+02
Container material	Steel Basket E24	1.22E+02
Total Container material		**4.70E+02**
Total Waste Container	**WA-COG-4**	**3.49E+03**

Tab. A.3-2: Physical Characteristics of WA-COG-4

Parameter	Average	Maximum
Total Mass [kg]	3'490	4'400
Volume [m^3]	1.50	-
Surface to mass ratio of metallic materials	**[m^2/kg]**	**[kg]**
Inconel (X-750)	1.40E-01	3.18E+01
Steel	4.30E-03	1.03E+02
Steel Z2 CN 18.10	4.00E-02	3.48E+02
Steel Basket E24	4.50E-02	1.22E+02
Zircaloy	4.40E-01	6.38E+02

Tab. A.3-3: Radiological characteristics of WA-COG-4 at time of production

Parameter	Average	Maximum
0 m γ dose rate [Sv/h]	3.9E+01	1.9E+02 [1]
1 m γ dose rate [Sv/h]	6.5E+00	3.2E+01
2 m γ dose rate [Sv/h]	2.2E+00	1.1E+01
0 m *n* dose rate [Sv/h]	3.7E-06	1.0E-04
1 m *n* dose rate [Sv/h]	-	1.2E-05
2 m *n* dose rate [Sv/h]	-	-
Heat output [W]	1.2E+02	5.5E+02

Radiological gas production from time of production		
Time [years]	**Production [l/y]**	**Cumulative Production [m^3]**
0	1.0E+03	0.0E+00
5	5.0E+02	3.6E+00
10	2.6E+02	5.5E+00
20	8.2E+01	7.0E+00
40	1.8E+01	7.8E+00
80	8.9E+00	8.2E+00
100	7.7E+00	8.4E+00

[1] Estimated maximum – COGEMA guaranteed value is 7.5E+01 Sv/h at time of transport.

Tab. A.3-4: Average and maximum radionuclide inventory of WA-COG-4 at time of production

Nuclide	Average [Bq/Cont.]	Maximum [Bq/Cont.]	Nuclide	Average [Bq/Cont.]	Maximum [Bq/Cont.]
H-3	2.7E+13	8.1E+13	Cs-134	1.2E+13	3.5E+13
Be-10	1.6E+04	7.8E+04	Cs-135	3.0E+08	9.0E+08
C-14	1.0E+10	5.0E+10	Cs-137	1.5E+13	3.9E+13
Cl-36	3.0E+07	9.0E+07	Ba-137M	1.4E+13	3.7E+13
Ca-45	3.2E+07	1.6E+08	Ce-144	1.3E+13	3.8E+13
Mn-54	7.0E+12	3.5E+13	Pr-144	1.3E+13	3.8E+13
Fe-55	2.4E+14	1.2E+15	Pm-147	7.8E+12	2.3E+13
Fe-59	1.7E+06	8.5E+06	Sm-151	2.0E+11	6.0E+11
Co-57	1.4E+12	7.0E+12	Eu-152	3.5E+08	1.1E+09
Co-58	7.5E+09	3.7E+10	Eu-154	5.0E+11	1.5E+12
Co-60	2.2E+14	1.1E+15	Eu-155	2.5E+11	7.5E+11
Ni-59	2.4E+11	1.2E+12	Ho-166M	2.6E+05	7.8E+05
Ni-63	4.1E+13	2.1E+14	Ta-182	1.5E+12	4.5E+12
Zn-65	1.0E+09	5.2E+09	U-232	2.8E+05	2.8E+06
Se-79	3.0E+06	8.9E+06	U-233	7.7E+02	7.5E+03
Kr-85	1.8E+12	5.4E+12	U-234	5.2E+07	5.0E+08
Sr-90	1.4E+13	3.1E+13	U-235	9.3E+05	8.8E+06
Y-90	1.4E+13	3.1E+13	U-236	1.2E+07	1.1E+08
Zr-93	1.0E+10	3.1E+10	U-238	1.3E+07	1.3E+08
Nb-93M	9.3E+08	4.6E+09	Np-237	4.9E+06	6.0E+07
Nb-94	4.1E+10	2.1E+11	Np-239	3.6E+08	4.4E+09
Nb-95	2.9E+10	1.5E+11	Pu-238	1.1E+11	1.0E+12
Mo-93	9.5E+08	4.7E+09	Pu-239	1.3E+10	1.2E+11
Tc-99	4.1E+09	1.2E+10	Pu-240	1.8E+10	1.7E+11
Ru-106	3.5E+13	1.1E+14	Pu-241	4.3E+12	5.2E+13
Rh-106	3.5E+13	1.1E+14	Pu-242	6.9E+07	6.6E+08
Pd-107	6.3E+07	1.9E+08	Am-241	4.1E+10	5.0E+11
Ag-108M	2.0E+06	6.0E+06	Am-242M	1.8E+08	2.2E+09
Ag-110M	2.8E+10	8.4E+10	Am-243	3.6E+08	4.4E+09
Sn-113	1.0E+11	5.0E+11	Cm-242	5.3E+09	6.3E+10
Sn-119M	3.9E+13	1.9E+14	Cm-243	1.9E+08	2.3E+09
Sn-121M	1.9E+10	9.7E+10	Cm-244	2.0E+10	2.4E+11
Sn-123	3.0E+10	1.5E+11	Cm-245	1.2E+06	1.5E+07
Sn-126	2.1E+07	1.1E+08	Cm-246	2.3E+05	2.8E+06
Sb-125	5.2E+13	2.6E+14	**Total α**	**2.1E+11**	**2.1E+12**
Te-125M	6.3E+12	3.2E+13	**Total βγ**	**8.2E+14**	**3.7E+15**
I-129	1.0E+08	3.0E+08	**Total Activity**	**8.2E+14**	**3.7E+15**

A.4 Inventory data for WA-COG-6

Tab. A.4-1: Material inventory of WA-COG-6

Occurrence	Material	Mass [kg/Container]
Raw waste	Aluminium	6.40E+00
Raw waste	Lead	2.00E-01
Raw waste	Chelating Agents	5.00E-02
Raw waste	Glass	6.00E-01
Raw waste	Rubber	6.00E+00
Raw waste	Copper	6.00E+00
Raw waste	Air	2.48E-01
Raw waste	Polyethylene	1.20E+01
Raw waste	PVC	1.20E+01
Raw waste	Stainless Steel	2.97E+02
Raw waste	Steel	8.50E+01
Raw waste	Rubble	4.00E-01
Raw waste	Cellulose	1.00E+00
Raw waste	Zinc	2.00E-01
Total Raw waste		**4.27E+02**
Raw waste containers	Polyethylene	1.06E+01
Raw waste containers	Steel	6.53E+01
Total Raw waste containers		**7.60E+01**
Matrix material	Iron Fibres	1.03E+01
Matrix material	Filling Material SC2000	2.06E+01
Matrix material	Gravel Sand	3.15E+02
Matrix material	Microsilica	6.85E+00
Matrix material	Rehobuild 2000	2.40E+00
Matrix material	Sand	2.37E+02
Matrix material	Water	5.83E+01
Matrix material	Cement CLC 45	1.48E+02
Total Matrix material		**7.97E+02**
Container material	Iron Fibres	1.42E+01
Container material	Filling Material SC2000	2.84E+01
Container material	Gravel Sand	4.35E+02
Container material	Microsilica	9.45E+00
Container material	Rehobuild 2000	3.31E+00
Container material	Sand	3.27E+02
Container material	Water	8.04E+01
Container material	Cement CLC 45	2.04E+02
Container material	Steel Basket	1.00E+02
Total Container material		**1.20E+03**
Total Waste Container	**WA-COG-6**	**2.50E+03**

Tab. A.4-2: Physical Characteristics of WA-COG-6

Parameter	Average	Maximum
Total Mass [kg]	2'500	4'000
Volume [m^3]	1.20	-

Surface to mass ratio of metallic materials	[m^2/kg]	[kg]
Aluminium	5.00E-02	6.41E+00
Iron fibres	1.00E+00	2.45E+01
Copper	1.50E-01	5.98E+00
Stainless Steel	1.20E-02	2.97E+02
Steel	1.50E-01	2.50E+02
Zinc	1.00E-01	2.00E-01

Tab. A.4-3: Radiological characteristics of WA-COG-6 at time of production

Parameter	Average	Maximum
0 m γ dose rate [Sv/h]	3.1E-02	4.0E-02
1 m γ dose rate [Sv/h]	2.3E-03	1.0E-02
2 m γ dose rate [Sv/h]	8.2E-04	1.1E-03
0 m *n* dose rate [Sv/h]	2.9E-06	5.0E-05
1 m *n* dose rate [Sv/h]	-	5.0E-06
2 m *n* dose rate [Sv/h]	-	-
Heat output [W]	2.1E-01	7.9E-01

Radiological gas production from time of production		
Time [years]	**Production [l/y]**	**Cumulative Production [m^3]**
0	3.0E+01	0.0E+00
5	2.8E+01	1.5E-01
10	2.7E+01	2.8E-01
20	2.5E+01	5.4E-01
40	2.1E+01	1.0E+00
80	1.8E+01	1.8E+00
100	1.7E+01	2.1E+00

Tab. A.4-4: Average and maximum radionuclide inventory of WA-COG-6 at time of production

Nuclide	Average [Bq/Cont.]	Maximum [Bq/Cont.]
C-14	1.2E+07	1.6E+07
Mn-54	2.1E+07	2.7E+07
Co-60	3.3E+09	4.3E+09
Ni-59	3.5E+06	4.6E+06
Ni-63	2.0E+09	2.6E+09
Se-79	3.1E+05	4.0E+05
Sr-90	3.8E+11	4.9E+11
Y-90	3.8E+11	4.9E+11
Zr-93	2.8E+07	3.6E+07
Nb-94	1.8E+06	2.3E+06
Tc-99	2.0E+08	2.6E+08
Ru-106	2.1E+10	2.8E+10
Rh-106	2.1E+10	2.8E+10
Pd-107	1.8E+06	2.3E+06
Sn-126	1.2E+07	1.6E+07
Sb-125	5.3E+09	6.8E+09
Te-125M	1.3E+09	1.7E+09
I-129	4.5E+03	5.8E+03
Cs-134	3.6E+10	4.6E+10
Cs-135	7.1E+06	9.1E+06
Cs-137	5.7E+11	7.3E+11
Ba-137M	5.4E+11	7.0E+11
Ce-144	1.1E+09	1.4E+09
Pr-144	1.1E+09	1.4E+09

Nuclide	Average [Bq/Cont.]	Maximum [Bq/Cont.]
Pm-147	1.3E+11	1.7E+11
Sm-151	7.7E+09	9.9E+09
Eu-154	7.6E+09	9.8E+09
U-232	2.1E+04	1.2E+05
U-233	1.6E+01	9.6E+01
U-234	2.3E+06	1.3E+07
U-235	1.8E+04	1.1E+05
U-236	2.3E+05	1.3E+06
U-238	3.7E+05	2.2E+06
Np-237	2.7E+06	1.6E+07
Pu-238	7.8E+09	4.6E+10
Pu-239	1.2E+09	7.3E+09
Pu-240	1.7E+09	1.0E+10
Pu-241	1.9E+11	1.1E+12
Pu-242	6.2E+06	3.6E+07
Am-241	1.7E+10	2.8E+11
Am-243	2.2E+08	3.5E+09
Cm-242	5.3E+07	8.6E+08
Cm-243	1.3E+08	2.2E+09
Cm-244	1.7E+10	2.8E+11
Total α	**4.5E+10**	**6.3E+11**
Total βγ	**2.3E+12**	**3.8E+12**
Total Activity	**2.3E+12**	**4.4E+12**

A.5 Inventory data for WA-COG-4A

Tab. A.5-1: Material inventory of WA-COG-4A

Occurrence	Material	Mass [kg/Container]
Raw waste	Inconel (X-750)	2.26E+01
Raw waste	Air	5.27E-02
Raw waste	Steel	7.31E+01
Raw waste	Zircaloy	4.54E+02
Total Raw waste		**5.50E+02**
Matrix material	None	-
Total Matrix material		-
Void	Air	2.42E-02
Container material	Steel Z15 CN 24.13	8.00E+01
Container material	Stainless Steel lid & springs	9.20E+01
Total Container material		**1.72E+02**
Total Waste Container	**WA-COG-4A**	**7.22E+02**

Tab. A.5-2: Physical Characteristics of WA-COG-4A

Parameter	Average	Maximum
Total Mass [kg]	722	850
Volume [m^3]	0.18	-

Surface to mass ratio of metallic materials	[m^2/kg]	[kg]
Inconel (X-750)	1.40E-01	2.26E+01
Stainless Steel lid & springs	5.80E-02	9.20E+01
Steel	4.30E-03	7.34E+01
Steel Z15 CN 24.13	6.10E-02	8.00E+01
Zircaloy	4.40E-01	4.54E+02

Tab. A.5-3: Radiological characteristics of WA-COG-4A at time of production

Parameter	Average	Maximum
0 m γ dose rate [Sv/h]	7.4E+01	2.8E+02 [1]
1 m γ dose rate [Sv/h]	5.3E+00	2.0E+01
2 m γ dose rate [Sv/h]	1.5E+00	6.0E+00
0 m *n* dose rate [Sv/h]	6.5E-05	1.0E-03
1 m *n* dose rate [Sv/h]	4.6E-06	7.8E-05
2 m *n* dose rate [Sv/h]	1.2E-06	2.0E-05
Heat output [W]	8.6E+01	9.0E+01 [2]

Radiological gas production from time of production		
Time [years]	**Production [3] [l/y]**	**Cumulative Production [3] [m^3]**
0	9.4E+01	0.0E+00
5	8.0E+01	4.3E-01
10	7.4E+01	8.1E-01
20	6.6E+01	1.5E+00
40	5.7E+01	2.7E+00
80	4.4E+01	4.7E+00
100	3.9E+01	5.5E+00

[1] Estimated maximum – COGEMA guaranteed value is 1.5E+02 Sv/h at time of transport.

[2] COGEMA guaranteed value at time of transport.

[3] Assumes the presence of 35 g of residual water.

Tab. A.5-4: Average and maximum radionuclide inventory of WA-COG-4A at time of production

Nuclide	Average [Bq/Cont.]	Maximum [Bq/Cont.]
H-3	1.9E+13	1.2E+14
Be-10	1.1E+04	6.9E+04
C-14	7.1E+09	4.3E+10
Cl-36	2.1E+07	1.3E+08
Ca-45	2.3E+07	1.4E+08
Mn-54	5.0E+12	3.0E+13
Fe-55	1.7E+14	1.0E+15
Fe-59	1.2E+06	7.4E+06
Co-57	1.0E+12	6.1E+12
Co-58	5.3E+09	3.3E+10
Co-60	1.6E+14	9.5E+14
Ni-59	1.7E+11	1.0E+12
Ni-63	2.9E+13	1.8E+14
Zn-65	7.1E+08	4.3E+09
Se-79	2.1E+06	1.3E+07
Kr-85	1.3E+12	7.8E+12
Sr-90	1.0E+13	5.8E+13
Y-90	1.0E+13	5.8E+13
Zr-93	7.1E+09	4.3E+10
Nb-93M	6.6E+08	4.0E+09
Nb-94	2.9E+10	1.8E+11
Nb-95	2.1E+10	1.3E+11
Mo-93	6.8E+08	4.1E+09
Tc-99	2.9E+09	1.8E+10
Ru-106	2.5E+13	1.5E+14
Rh-106	2.5E+13	1.5E+14
Pd-107	4.5E+07	2.7E+08
Ag-108M	1.4E+06	8.7E+06
Ag-110M	2.0E+10	1.2E+11
Sn-113	7.1E+10	4.3E+11
Sn-119M	2.8E+13	1.7E+14
Sn-121M	1.4E+10	8.2E+10
Sn-123	2.1E+10	1.3E+11
Sn-126	1.5E+07	9.1E+07
Sb-125	3.7E+13	2.3E+14
Te-125M	4.5E+12	2.7E+13
I-129	7.1E+07	4.3E+08
Cs-134	8.5E+12	5.2E+13
Cs-135	2.1E+08	1.3E+09
Cs-137	1.1E+13	6.5E+13
Ba-137M	1.0E+13	6.1E+13
Ce-144	9.2E+12	5.6E+13
Pr-144	9.2E+12	5.6E+13
Pm-147	5.5E+12	3.4E+13
Sm-151	1.4E+11	8.7E+11
Eu-152	2.5E+08	1.5E+09
Eu-154	3.6E+11	2.2E+12
Eu-155	1.8E+11	1.1E+12
Ho-166M	1.8E+05	1.1E+06
Ta-182	1.1E+12	6.5E+12
U-232	2.0E+05	6.6E+06
U-233	5.5E+02	1.8E+04
U-234	3.7E+07	1.2E+09
U-235	6.6E+05	2.2E+07
U-236	8.5E+06	2.8E+08
U-238	9.2E+06	3.1E+08
Np-237	3.5E+06	1.2E+08
Pu-238	7.8E+10	2.6E+12
Pu-239	9.2E+09	3.1E+11
Pu-240	1.3E+10	4.2E+11
Pu-241	3.1E+12	7.5E+13
Pu-242	4.9E+07	1.6E+09
Am-241	2.9E+10	9.6E+11
Am-242M	1.3E+08	4.2E+09
Am-243	2.6E+08	8.5E+09
Cm-242	3.8E+09	1.2E+11
Cm-243	1.4E+08	4.5E+09
Cm-244	1.4E+10	2.0E+12
Cm-245	8.5E+05	2.8E+07
Cm-246	1.6E+05	5.4E+06
Total α	**1.5E+11**	**6.4E+12**
Total βγ	**5.8E+14**	**3.5E+15**
Total Activity	**5.8E+14**	**3.5E+15**

A.6 Inventory data for WA-COG-6A

Tab. A.6-1: Material inventory of WA-COG-6A

Occurrence	**Material**	**Mass [kg/Container]**
Raw waste	Aluminium	5.72E+00
Raw waste	Lead	1.79E-01
Raw waste	Chelating Agents	4.46E-02
Raw waste	Glass	5.33E-01
Raw waste	Rubber	5.33E+00
Raw waste	Copper	5.33E+00
Raw waste	Air	5.37E-02
Raw waste	Polyethylene	1.07E+01
Raw waste	PVC	1.07E+01
Raw waste	Stainless Steel	2.65E+02
Raw waste	Steel	7.58E+01
Raw waste	Rubble	3.58E-01
Raw waste	Cellulose	8.92E-01
Raw waste	Zinc	1.79E-01
Total Raw waste		**3.81E+02**
Matrix material	None	-
Total Matrix material		-
Void	Air	2.42E-02
Container material	Steel Z15 CN 24.13	8.00E+01
Container material	Stainless Steel lid & springs	9.20E+01
Total Container material		**1.72E+02**
Total Waste Container	**WA-COG-6A**	**5.53E+02**

Tab. A.6-2: Physical Characteristics of WA-COG-6A

Parameter	Average	Maximum
Total Mass [kg]	553	850
Volume [m^3]	0.18	-

Surface to mass ratio of metallic materials	[m^2/kg]	[kg]
Aluminium	5.0E-02	5.72E+00
Copper	1.5E-01	5.33E+00
Stainless Steel	2.3E-02	3.57E+02
Steel	1.5E-02	7.58E+01
Steel Z15 CN 24.13	6.1E-02	8.00E+01
Zinc	1.0E-01	1.79E-01

Tab. A.6-3: Radiological characteristics of WA-COG-6A at time of production

Parameter	Average	Maximum
0 m γ dose rate [Sv/h]	1.2E-01	5.4E+00 [1]
1 m γ dose rate [Sv/h]	8.4E-03	3.8E-01
2 m γ dose rate [Sv/h]	2.6E-03	1.2E-01
0 m *n* dose rate [Sv/h]	1.0E-05	1.4E-04
1 m *n* dose rate [Sv/h]	7.0E-07	1.1E-05
2 m *n* dose rate [Sv/h]	2.0E-07	2.8E-06
Heat output [W]	1.8E-01	7.5E+00

Radiological gas production from time of production		
Time [years]	**Production [2] [l/y]**	**Cumulative Production [2] [m^3]**
0	3.6E+01	0.0E+00
5	3.4E+01	1.7E-01
10	3.2E+01	3.4E-01
20	3.0E+01	6.5E-01
40	2.6E+01	1.2E+00
80	2.1E+01	2.1E+00
100	2.0E+01	2.5E+00

[1] Estimated maximum – COGEMA guaranteed value is 1.5E+02 Sv/h at time of transport.

[2] Assumes the presence of 35 g of residual water.

Tab. A.6-4: Average and maximum radionuclide inventory of WA-COG-6A at time of production

Nuclide	Average [Bq/Cont.]	Maximum [Bq/Cont.]
C-14	1.1E+07	5.0E+08
Mn-54	1.9E+07	8.8E+08
Co-60	2.9E+09	1.4E+11
Ni-59	3.1E+06	1.5E+08
Ni-63	1.8E+09	8.4E+10
Se-79	2.8E+05	1.3E+07
Sr-90	3.4E+11	1.6E+13
Y-90	3.4E+11	1.6E+13
Zr-93	2.5E+07	1.2E+09
Nb-94	1.6E+06	7.6E+07
Tc-99	1.8E+08	8.4E+09
Ru-106	1.9E+10	8.8E+11
Rh-106	1.9E+10	8.8E+11
Pd-107	1.6E+06	7.6E+07
Sn-126	1.1E+07	5.0E+08
Sb-125	4.7E+09	2.2E+11
Te-125M	1.2E+09	5.5E+10
I-129	4.0E+03	1.9E+05
Cs-134	3.2E+10	1.5E+12
Cs-135	6.3E+06	3.0E+08
Cs-137	5.1E+11	2.4E+13
Ba-137M	4.8E+11	2.3E+13
Ce-144	9.8E+08	4.6E+10
Pr-144	9.8E+08	4.6E+10

Nuclide	Average [Bq/Cont.]	Maximum [Bq/Cont.]
Pm-147	1.2E+11	5.5E+12
Sm-151	6.9E+09	3.2E+11
Eu-154	6.8E+09	3.2E+11
U-232	1.9E+04	1.1E+05
U-233	1.4E+01	8.6E+01
U-234	2.1E+06	1.2E+07
U-235	1.6E+04	9.7E+04
U-236	2.1E+05	1.2E+06
U-238	3.3E+05	2.0E+06
Np-237	2.4E+06	1.4E+07
Pu-238	7.0E+09	4.2E+10
Pu-239	1.1E+09	6.4E+09
Pu-240	1.5E+09	9.1E+09
Pu-241	1.7E+11	1.0E+12
Pu-242	5.5E+06	3.3E+07
Am-241	1.5E+10	2.6E+11
Am-243	2.0E+08	3.3E+09
Cm-242	4.7E+07	8.1E+08
Cm-243	1.2E+08	2.0E+09
Cm-244	1.5E+10	2.6E+11
Total α	**4.0E+10**	**5.8E+11**
Total βγ	**2.1E+12**	**9.0E+13**
Total Activity	**2.1E+12**	**9.1E+13**

A.7 Inventory data for WA-BNF-1

Tab. A.7-1: Material inventory of WA-BNF-1

Oxide	WA-COG-1 [kg/flask]	Oxide	WA-COG-1 [kg/flask]	Oxide	WA-COG-1 [kg/flask]
Li_2O	16.010	SrO	1.730	CeO_2	5.548
B_2O_3	67.136	Y_2O_3	1.115	Pr_6O_{11}	2.560
Na_2O	33.300	ZrO_2	9.394	Nd_2O_3	8.876
MgO	5.588	MoO_3	9.506	Sm_2O_3	1.859
Al_2O_3	5.525	Tc_2O_7	2.352	Eu_2O_3	0.277
SiO_2	183.781	RuO_2	5.075	Gd_2O_3	14.622
P_2O_5	0.606	Rh	0.873	U_3O_8	0.572
SO_3	0.089	Pd	2.430	NpO_2	0.836
CaO	0.038	Ag	0.141	PuO_2	0.057
Cr_2O_3	0.679	CdO	0.239	Am_2O_3	1.093
Fe_2O_3	3.531	SnO	0.187	CmO_2	0.022
NiO	0.427	TeO_2	1.120	Misc.	0.310
ZnO	0.067	Cs_2O	5.183	**Total**	**400.0**
SeO_2	0.151	BaO	3.643		
Rb_2O	0.739	La_2O_3	2.713		

Occurrence	Material	Mass [kg/flask]
Raw waste	Actinide oxides	2.60E+00
Raw waste	Anions	5.41E-01
Raw waste	Cladding residues	5.62E+00
Raw waste	Corrosion products	3.46E+00
Raw waste	Process additives	2.17E+01
Raw waste	Fission products	7.41E+01
Total Raw waste		**1.08E+02**
Matrix material	B_2O_3	6.72E+01
Matrix material	Li_2O	7.89E+00
Matrix material	Na_2O	3.33E+01
Matrix material	SiO_2	1.84E+02
Total Matrix material		**2.92E+02**
Flask material	Steel Z15 CN 24.13	8.00E+01
Total Flask material		**8.00E+01**
Total Flask with waste	**WA-BNF-1**	**4.80E+02**

Tab. A.7-2: Physical Characteristics of WA-BNF-1

Parameter	Average	Maximum
Total Mass [kg]	480	550
Volume [m^3]	0.18	-

Surface to mass ratio of metallic materials	[m^2/kg]	[kg]
Steel Z15 CN 24.13	6.1E-02	8.0E+01

Tab. A.7-3: Radiological characteristics of WA-BNF-1 at time of production

Parameter	Average	Maximum
0 m γ dose rate [Sv/h]	2'120	4'250 [1]
1 m γ dose rate [Sv/h]	140	280 [1]
2 m γ dose rate [Sv/h]	45	90
0 m *n* dose rate [Sv/h]	1.3E-01	3.0E-01
1 m *n* dose rate [Sv/h]	7.2E-03	2.4E-02
2 m *n* dose rate [Sv/h]	2.2E-03	7.3E-03
Heat output [W]	3'540	9'600

[1] Estimated maximum – BNFL guaranteed value is 4.5E+03 and 3.4E+02 Sv/h at 0 and 1 m, respectively.

Tab. A.7-4: Average and maximum radionuclide inventory of WA-BNF-1 at time of production

Nuclide	**Average [Bq/flask]**	**Maximum [Bq/flask]**	**Nuclide**	**Average [Bq/flask]**	**Maximum [Bq/flask]**
Be-10	3.2E+05	1.2E+06	Ba-137M	5.4E+15	7.6E+15
C-14	7.1E+07	2.7E+08	Ce-144	3.9E+15	1.5E+16
Ca-41	6.9E+07	2.6E+08	Pr-144	3.9E+15	1.5E+16
Ca-45	1.1E+09	4.2E+09	Pm-146	1.4E+10	5.3E+10
Mn-54	1.2E+11	4.6E+11	Pm-147	2.5E+15	9.5E+15
Fe-55	3.4E+12	1.3E+13	Sm-151	2.3E+13	8.7E+13
Co-58	3.4E+05	1.3E+06	Eu-152	3.3E+11	1.3E+12
Co-60	1.5E+13	5.7E+13	Eu-154	2.3E+14	8.7E+14
Ni-59	6.7E+08	2.5E+09	Eu-155	1.3E+14	4.9E+14
Ni-63	9.3E+10	3.5E+11	Gd-153	6.3E+11	2.4E+12
Zn-65	9.3E+09	3.5E+10	Tb-160	4.2E+10	1.6E+11
Se-79	1.6E+09	6.1E+09	Ho-166M	1.5E+08	5.7E+08
Sr-90	4.1E+15	5.5E+15	Ac-227	4.7E+05	1.8E+06
Y-90	4.1E+15	5.5E+15	Th-228	6.7E+08	2.7E+09
Zr-93	1.3E+11	4.9E+11	Th-230	1.1E+07	4.4E+07
Zr-95	1.8E+14	6.8E+14	Pa-231	2.0E+06	8.0E+06
Nb-93M	5.0E+10	1.9E+11	U-232	2.7E+05	1.1E+06
Nb-94	9.9E+06	3.8E+07	U-233	1.2E+05	4.8E+05
Mo-93	5.0E+07	1.9E+08	U-234	1.9E+07	7.8E+07
Tc-99	9.4E+11	3.6E+12	U-235	3.0E+05	1.2E+06
Rh-102	1.0E+09	3.8E+09	U-236	3.5E+06	1.4E+07
Rh-106	2.4E+15	9.1E+15	U-238	6.0E+06	2.5E+07
Ru-106	2.4E+15	9.1E+15	Np-235	2.8E+06	1.1E+07
Pd-107	7.5E+09	2.8E+10	Np-237	1.9E+10	7.6E+10
Ag-108M	5.0E+08	1.9E+09	Pu-236	1.6E+06	6.4E+06
Ag-110M	6.3E+12	2.4E+13	Pu-238	3.5E+11	1.4E+12
Cd-109	2.0E+07	7.6E+07	Pu-239	6.6E+10	2.7E+11
Cd-113M	2.0E+12	7.6E+12	Pu-240	1.1E+11	5.0E+11
Sn-119M	4.5E+11	1.7E+12	Pu-241	1.5E+13	5.6E+13
Sn-121M	1.1E+10	4.2E+10	Pu-242	3.3E+08	1.0E+09
Sn-123	2.8E+12	1.1E+13	Am-241	1.1E+14	1.5E+14
Sn-126	5.1E+10	1.9E+11	Am-242M	1.5E+12	6.0E+12
Sb-124	3.1E+10	1.2E+11	Am-243	8.8E+11	3.5E+12
Sb-125	1.8E+14	6.8E+14	Cm-242	5.8E+13	8.1E+13
Te-123M	8.0E+08	3.0E+09	Cm-243	7.6E+11	3.9E+12
Te-125M	4.2E+13	1.6E+14	Cm-244	5.4E+13	2.8E+14
Te-127M	8.5E+12	3.2E+13	Cm-245	6.5E+09	3.3E+10
I-129	2.1E+06	8.0E+06	Cm-246	1.2E+09	6.2E+09
Cs-134	9.1E+14	3.5E+15	**Total α**	**2.2E+14**	**5.2E+14**
Cs-135	3.9E+10	1.5E+11	**Total βγ**	**3.6E+16**	**9.1E+16**
Cs-137	5.7E+15	8.0E+15	**Total Activity**	**3.6E+16**	**9.2E+16**

Tab. A.7-5: Decay of average flask heat output from time of production for WA-BNF-1

Years from time of production	0	3	10	30	40	100	300	1'000
Inventory Activity	Average [Bq/flask]	Average [Bq/flask]	Average [Bq/flask]	Average [Bq/flask]	Average [Bq/flask]	Average [Bq/flask]	Average [Bq/flask]	Average [Bq/flask]
Total α	2.2E+14	1.6E+14	1.5E+14	1.3E+14	1.2E+14	9.8E+13	7.0E+13	2.3E+13
Total βγ	3.6E+16	2.1E+16	1.6E+16	9.6E+15	7.6E+15	1.9E+15	2.3E+13	2.3E+12
Total Activity	3.6E+16	2.1E+16	1.6E+16	9.7E+15	7.7E+15	2.0E+15	9.3E+13	2.6E+13
Heat output	[W/flask]	[W/flask]	[W/flask]	[W/flask]	[W/flask]	[W/flask]	[W/flask]	[W/flask]
Total Heat	3.5E+03	1.8E+03	1.4E+03	8.5E+02	6.9E+02	2.3E+02	6.5E+01	2.1E+01
Total α	2.1E+02	1.5E+02	1.4E+02	1.1E+02	1.1E+02	8.9E+01	6.3E+01	2.1E+01
Total β	2.8E+03	1.2E+03	7.6E+02	4.5E+02	3.5E+02	8.3E+01	8.1E-01	9.1E-02
Total γ	5.7E+02	5.3E+02	4.5E+02	2.9E+02	2.3E+02	5.7E+01	5.8E-01	7.1E-04

Years from time of production	3'000	1.0E+04	3.0E+04	1.0E+05	3.0E+05	1.0E+06	3.0E+06	1.0E+07
Inventory Activity	Average [Bq/flask]	Average [Bq/flask]	Average [Bq/flask]	Average [Bq/flask]	Average [Bq/flask]	Average [Bq/flask]	Average [Bq/flask]	Average [Bq/flask]
Total α	1.9E+12	6.8E+11	3.0E+11	1.5E+11	2.1E+11	2.2E+11	1.2E+11	1.2E+10
Total βγ	2.2E+12	1.9E+12	1.5E+12	1.2E+12	7.8E+11	3.6E+11	1.5E+11	1.4E+10
Total Activity	4.1E+12	2.6E+12	1.8E+12	1.3E+12	9.9E+11	5.8E+11	2.7E+11	2.6E+10
Heat output	[W/flask]	[W/flask]	[W/flask]	[W/flask]	[W/flask]	[W/flask]	[W/flask]	[W/flask]
Total Heat	1.8E+00	6.4E-01	3.0E-01	1.7E-01	2.3E-01	2.3E-01	1.2E-01	1.3E-02
Total α	1.7E+00	5.8E-01	2.6E-01	1.4E-01	2.1E-01	2.2E-01	1.2E-01	1.2E-02
Total β	8.2E-02	6.0E-02	4.0E-02	3.2E-02	2.2E-02	9.6E-03	4.2E-03	4.4E-04
Total γ	5.9E-04	5.9E-04	5.8E-04	5.6E-04	5.1E-04	3.7E-04	1.5E-04	5.8E-06

A.8 Inventory data for WA-BNF-2

Tab. A.8-1: Material inventory of WA-BNF-2

Occurrence	Material	Mass [kg/Container]
Raw waste	Barium Carbonate ($BaCO_3$)	4.80E+01
Raw waste	Barium Nitrate ($BaNO_3$)	6.40E-04
Raw waste	Crud	1.00E+00
Raw waste	Inactive substances	1.00E+00
Raw waste	Sodium Hydroxide $Na(OH)_2$	1.00E+00
Raw waste	Sodium Nitrate ($NaNO_3$)	1.00E+00
Raw waste	Sodium Nitrite ($NaNO_2$)	2.00E+00
Raw waste	Pond wall coating material	6.00E+00
Raw waste	Water	2.59E+02
Total Raw waste		**3.20E+02**
Matrix material	Blast Furnace Slag (BFS)	5.34E+02
Matrix material	Ord. Portland Cement (OPC)	5.70E+01
Total Matrix material		**5.91E+02**
Void	Air	1.33E-02
Capping Cement	Ord. Portland Cement (OPC)	1.20E+01
Capping Cement	Pulverised Fly Ash (PFA)	3.60E+01
Capping Cement	Water	2.10E+01
Total Capping Cement		6.90E+01
Container material	Steel mixing paddle	3.00E+01
Container material	Steel 316 L	1.05E+02
Total Container material		**1.35E+02**
Total Waste Container	**WA-BNF-2**	**1.12E+03**

Tab. A.8-2: Physical Characteristics of WA-BNF-2

Parameter	Average	Maximum
Total Mass [kg]	1'115	1'400
Volume [m^3]	0.56	-

Surface to mass ratio of metallic materials	[m^2/kg]	[kg]
Steel Mixing Paddle	1.7E-02	3.00E+01
Steel 316 L	7.9E-02	1.05E+02

Tab. A.8-3: Radiological characteristics of WA-BNF-2 at time of production

Parameter	Average	Maximum
0 m γ dose rate [Sv/h]	9.4E-02	3.6E+00[1]
1 m γ dose rate [Sv/h]	9.6E-03	3.7E-01
2 m γ dose rate [Sv/h]	3.3E-03	1.3E-01
0 m *n* dose rate [Sv/h]	1.0E-08	1.0E-03
1 m *n* dose rate [Sv/h]	-	-
2 m *n* dose rate [Sv/h]	3.9E-10	-
Heat output [W]	7.5E-02	2.9E+00

Radiological gas production from time of production		
Time [years]	**Production[2] [l/y]**	**Cumulative Production[2] [m^3]**
0	1.1E+00	0.0E+00
5	6.7E-01	4.6E-03
10	3.9E-01	7.2E-03
20	1.6E-01	9.7E-03
40	8.2E-02	1.2E-02
80	6.9E-02	1.5E-02
100	6.6E-02	1.6E-02

[1] Estimated maximum – BNFL guaranteed value is 4.0E+00 Sv/h.

[2] Assumes the presence of 35 g of residual water.

Tab. A.8-4: Average and maximum radionuclide inventory of WA-BNF-2 at time of production

Nuclide	**Average [Bq/Cont.]**	**Maximum [Bq/Cont.]**	**Nuclide**	**Average [Bq/Cont.]**	**Maximum [Bq/Cont.]**
H-3	6.0E+08	2.2E+10	Eu-154	1.8E+08	6.5E+09
C-14	2.8E+11	1.0E+13	Eu-155	9.5E+07	3.5E+09
Mn-54	2.8E+07	1.0E+09	U-234	6.2E+04	1.8E+06
Co-57	1.1E+09	3.9E+10	U-235	1.0E+03	3.0E+04
Co-60	1.7E+11	6.5E+12	U-235M	1.7E+07	4.9E+08
Ni-59	3.4E+08	1.2E+10	U-236	1.4E+04	4.0E+05
Ni-63	3.4E+10	1.2E+12	U-237	1.0E+05	3.0E+06
Se-79	7.8E+02	2.9E+04	U-238	1.6E+04	4.7E+05
Kr-85	3.7E+08	1.3E+10	Np-239	7.8E+05	3.4E+07
Sr-90	3.0E+09	1.1E+11	Pu-238	1.2E+08	5.3E+09
Y-90	3.0E+09	1.1E+11	Pu-239	1.8E+07	8.1E+08
Mo-93	3.0E+05	1.1E+07	Pu-240	2.7E+07	1.2E+09
Tc-99	8.3E+05	3.0E+07	Pu-241	4.2E+09	1.9E+11
Ru-106	2.7E+07	9.9E+08	Pu-242	7.8E+04	3.5E+06
Rh-106	2.7E+07	9.9E+08	Am-241	9.5E+07	4.2E+09
Sb-125	3.1E+07	1.1E+09	Am-242	3.5E+05	1.5E+07
Te-125M	6.9E+06	2.5E+08	Am-242M	3.5E+05	1.5E+07
I-129	3.2E+08	1.2E+10	Am-243	7.8E+05	3.4E+07
Cs-134	2.6E+08	9.3E+09	Cm-242	2.9E+05	1.3E+07
Cs-135	1.9E+04	7.0E+05	Cm-243	6.6E+05	2.9E+07
Cs-137	4.3E+09	1.7E+11	Cm-244	5.9E+07	2.6E+09
Ba-137M	4.1E+09	1.6E+11	Cm-245	5.9E+03	2.6E+05
Ce-144	6.4E+06	2.3E+08	Cm-246	1.5E+03	6.7E+04
Pr-144	6.4E+06	2.3E+08	**Total α**	**3.2E+08**	**1.4E+10**
Pm-147	5.1E+08	1.9E+10	**Total βγ**	**5.1E+11**	**1.9E+13**
Eu-152	1.8E+05	6.6E+06	**Total Activity**	**5.1E+11**	**1.9E+13**

A.9 Inventory data for WA-BNF-4

Tab. A.9-1: Material inventory of WA-BNF-4

Occurrence	**Material**	**Mass [kg/Container]**
Raw waste	Fuel residues	1.00E+00
Raw waste	Inconel (X-750)	2.10E+01
Raw waste	Steel 304	5.70E+01
Raw waste	Zircaloy	3.63E+02
Total Raw waste		**4.42E+02**
Matrix material	Blast Furnace Slag (BFS)	5.55E+02
Matrix material	Ord. Portland Cement (OPC)	8.40E+01
Matrix material	Water	2.30E+02
Total Matrix material		**8.69E+02**
Void	Air	1.39E-02
Capping Cement	Ord. Portland Cement (OPC)	1.20E+01
Capping Cement	Pulverised Fly Ash (PFA)	3.60E+01
Capping Cement	Water	2.10E+01
Total Capping Cement		6.90E+01
Container material	Steel basket 304 L	3.20E+01
Container material	Steel 316 L	9.80E+01
Total Container material		**1.30E+02**
Total Waste Container	**WA-BNF-4**	**1.51E+03**

Tab. A.9-2: Physical Characteristics of WA-BNF-4

Parameter	**Average**	**Maximum**
Total Mass [kg]	1'510	2'000
Volume [m^3]	0.56	-

Surface to mass ratio of metallic materials	**[m^2/kg]**	**[kg]**
Inconel (X-750)	1.4E-01	2.1E+01
Steel 304	4.4E-03	5.7E+01
Steel 304 L	8.2E-02	3.20E+01
Steel 316 L	7.6E-02	9.80E+01
Zircaloy	4.4E-01	3.63E+02

Tab. A.9-3: Radiological characteristics of WA-BNF-4 at time of production

Parameter	**Average**	**Maximum**
0 m γ dose rate [Sv/h]	1.4E+01	5.0E+01
1 m γ dose rate [Sv/h]	1.6E+00	5.6E+00[1]
2 m γ dose rate [Sv/h]	5.0E-01	1.8E+00
0 m *n* dose rate [Sv/h]	3.7E-06	1.1E-05[2]
1 m *n* dose rate [Sv/h]	-	9.4E-07
2 m *n* dose rate [Sv/h]	-	-
Heat output [W]	1.9E+01	8.0E+01

Radiological gas production from time of production		
Time [years]	**Production[3] [l/y]**	**Cumulative Production[3] [m^3]**
0	2.0E+02	0.0E+00
5	1.1E+02	7.4E-01
10	6.3E+01	1.2E+00
20	2.7E+01	1.6E+00
40	1.2E+01	1.9E+00
80	6.8E+00	2.3E+00
100	5.6E+00	2.4E+00

[1] Estimated maximum – BNFL guaranteed value is 4.0E+00 Sv/h.

[2] Estimated maximum – BNFL guaranteed value is 1.0E-03 Sv/h.

[3] Assumes the presence of 35 g of residual water.

Tab. A.9-4: Average and maximum radionuclide inventory of WA-BNF-4 at time of production

Nuclide	Average [Bq/Cont.]	Maximum [Bq/Cont.]	Nuclide	Average [Bq/Cont.]	Maximum [Bq/Cont.]
H-3	4.5E+13	9.9E+13	Pr-144	1.5E+10	3.3E+10
C-14	2.2E+10	7.8E+10	Pm-147	3.9E+11	8.4E+11
Cl-36	2.0E+08	6.9E+08	Sm-151	7.9E+09	1.7E+10
Mn-54	1.6E+10	5.5E+10	Eu-152	2.6E+08	5.7E+08
Fe-55	1.5E+13	5.3E+13	Eu-154	2.1E+12	4.6E+12
Co-57	2.2E+11	7.8E+11	Eu-155	4.5E+10	9.7E+10
Co-60	3.4E+13	1.2E+14	Th-228	4.0E+05	1.0E+06
Ni-59	1.7E+11	5.9E+11	U-232	5.0E+05	1.3E+06
Ni-63	2.2E+13	7.7E+13	U-233	1.2E+03	3.1E+03
Se-79	5.6E+05	1.2E+06	U-234	8.2E+07	2.1E+08
Kr-85	1.5E+11	3.4E+11	U-235	2.3E+06	5.7E+06
Sr-90	1.9E+12	4.2E+12	U-235M	1.8E+10	4.5E+10
Y-90	1.9E+12	4.2E+12	U-236	9.8E+06	2.5E+07
Zr-93	1.0E+10	2.2E+10	U-237	6.4E+07	1.6E+08
Nb-93M	4.8E+09	1.0E+10	U-238	2.0E+07	5.1E+07
Nb-94	6.9E+10	2.4E+11	Np-237	6.8E+06	4.8E+07
Mo-93	4.2E+09	1.5E+10	Np-239	2.8E+08	2.0E+09
Tc-99	4.6E+08	9.9E+08	Pu-238	4.6E+10	3.2E+11
Ru-106	1.7E+12	6.5E+13	Pu-239	1.5E+10	1.1E+11
Rh-106	1.7E+12	6.5E+13	Pu-240	1.6E+10	1.1E+11
Pd-107	1.9E+06	4.2E+06	Pu-241	2.1E+12	1.4E+13
Ag-110M	2.8E+06	6.1E+06	Pu-242	4.0E+07	2.8E+08
Sn-119M	6.7E+07	2.4E+08	Am-241	4.7E+10	3.2E+11
Sn-121M	1.6E+12	5.8E+12	Am-242M	1.6E+08	1.1E+09
Sn-126	2.4E+06	5.1E+06	Am-243	2.8E+08	2.0E+09
Sb-125	1.4E+13	2.9E+13	Cm-242	1.4E+08	3.4E+09
Te-125M	3.0E+12	6.6E+12	Cm-243	2.4E+08	5.9E+09
I-129	8.0E+05	1.7E+06	Cm-244	2.0E+10	5.0E+11
Cs-134	5.4E+11	8.5E+12	Cm-245	2.6E+06	6.4E+07
Cs-135	6.6E+07	1.4E+08	Cm-246	4.6E+05	1.2E+07
Cs-137	1.4E+13	3.0E+13	**Total α**	**1.4E+11**	**1.4E+12**
Ba-137M	1.3E+13	2.9E+13	**Total βγ**	**1.7E+14**	**6.2E+14**
Ce-144	1.5E+10	3.3E+10	**Total Activity**	**1.7E+14**	**6.2E+14**

A.10 Inventory data for WA-BNF-7

Tab. A.10-1: Material inventory of WA-BNF-7

Occurrence	Material	Mass [kg/Container]
Raw waste	Fuel Solution	4.00E-01
Raw waste	C.C.S. Liquor (Water)	2.62E+02
Raw waste	Crud	6.00E-02
Raw waste	Insoluble Fission Products	1.17E+01
Raw waste	Insoluble Plutonium	2.00E-02
Raw waste	Zircaloy fines	1.00E-03
Total Raw waste		**2.74E+02**
Matrix material	Blast Furnace Slag (BFS)	6.25E+02
Matrix material	Lime ($Ca(OH)_2$)	5.00E+00
Matrix material	Ord. Portland Cement (OPC)	6.00E+01
Total Matrix material		**6.90E+02**
Void	Air	1.33E-02
Capping Cement	Ord. Portland Cement (OPC)	1.20E+01
Capping Cement	Pulverised Fly Ash (PFA)	3.60E+01
Capping Cement	Water	2.10E+01
Total Capping Cement		6.90E+01
Container material	Steel Mixing Paddle	3.00E+01
Container material	Steel 316 L	1.05E+02
Total Container material		**1.35E+02**
Total Waste Container	**WA-BNF-7**	**1.17E+03**

Tab. A.10-2: Physical Characteristics of WA-BNF-7

Parameter	Average	Maximum
Total Mass [kg]	1'170	1'400
Volume [m^3]	0.56	-

Surface to mass ratio of metallic materials	[m^2/kg]	[kg]
Steel Mixing Paddle	1.7E-02	3.00E+01
Steel 316 L	7.9E-02	1.05E+02
Zircaloy Fines	4.00E-01	1.00E-03

Tab. A.10-3: Radiological characteristics of WA-BNF-7 at time of production

Parameter	Average	Maximum
0 m γ dose rate [Sv/h]	2.1E+00	8.3E+00 [1]
1 m γ dose rate [Sv/h]	2.4E-01	9.7E-01 [1]
2 m γ dose rate [Sv/h]	8.1E-02	3.3E-01
0 m *n* dose rate [Sv/h]	5.9E-07	7.4E-07 [2]
1 m *n* dose rate [Sv/h]	4.8E-08	6.5E-08
2 m *n* dose rate [Sv/h]	-	-
Heat output [W]	4.8E+00	5.5E+01

Radiological gas production from time of production		
Time [years]	**Production [3] [l/y]**	**Cumulative Production [3] [m^3]**
0	1.1E+02	0.0E+00
5	6.8E+01	4.0E-01
10	6.5E+01	7.3E-01
20	6.3E+01	1.4E+00
40	5.7E+01	2.6E+00
80	4.8E+01	4.7E+00
100	4.4E+01	5.6E+00

[1] Estimated maximum – BNFL guaranteed value is 1.0E+01 and 1.0E+00 Sv/h for 0 and 1 m, respectively.

[2] Estimated maximum – BNFL guaranteed value is 1.0E-03 Sv/h.

[3] Assumes the presence of 35 g of residual water.

Tab. A.10-4: Average and maximum radionuclide inventory of WA-BNF-7 at time of production

Nuclide	Average [Bq/Cont.]	Maximum [Bq/Cont.]	Nuclide	Average [Bq/Cont.]	Maximum [Bq/Cont.]
H-3	3.6E+10	4.2E+10	U-235	4.5E+05	5.4E+05
C-14	5.2E+06	6.0E+06	U-235M	4.8E+10	5.8E+10
Cl-36	5.5E+04	6.4E+04	U-236	6.0E+06	7.2E+06
Se-79	3.8E+05	4.3E+05	U-237	4.7E+07	5.6E+07
Sr-90	1.3E+12	1.5E+12	U-238	7.1E+06	8.5E+06
Y-90	1.3E+12	1.5E+12	Np-237	6.0E+06	7.2E+06
Tc-99	3.2E+08	3.7E+08	Np-239	3.4E+08	4.1E+08
Ru-106	7.2E+12	2.0E+14	Pu-238	3.2E+11	3.9E+11
Rh-106	7.2E+12	2.0E+14	Pu-239	4.9E+10	6.0E+10
Pd-107	1.6E+06	1.9E+06	Pu-240	7.6E+10	9.3E+10
Sn-126	1.3E+07	1.5E+07	Pu-241	1.2E+13	1.4E+13
Sb-125	6.2E+11	7.2E+11	Pu-242	2.4E+08	3.0E+08
Te-125M	1.4E+11	1.6E+11	Am-241	1.3E+11	1.6E+11
I-129	6.0E+05	6.9E+05	Am-242	1.5E+08	1.8E+08
Cs-134	2.2E+12	2.5E+12	Am-242M	1.5E+08	1.8E+08
Cs-135	8.2E+06	9.5E+06	Am-243	3.4E+08	4.0E+08
Cs-137	1.1E+13	1.3E+13	Cm-242	1.3E+08	1.5E+08
Ba-137M	1.1E+13	1.2E+13	Cm-243	2.8E+08	3.4E+08
Ce-144	4.4E+10	5.1E+10	Cm-244	2.6E+10	3.1E+10
Pr-144	4.4E+10	5.1E+10	Cm-245	2.6E+06	3.1E+06
Pm-147	2.2E+11	2.5E+11	Cm-246	6.6E+05	7.9E+05
Eu-152	7.7E+07	8.8E+07	**Total α**	**6.0E+11**	**7.4E+11**
Eu-154	7.7E+10	8.9E+10	**Total βγ**	**5.4E+13**	**4.5E+14**
Eu-155	4.2E+10	4.8E+10	**Total Activity**	**5.5E+13**	**4.5E+14**
U-234	3.0E+07	3.7E+07			

Appendix B

Results of fuel depletion inventory calculations for model fuel assemblies

Fuel with impurities (Fuel + imp.)

Structural Materials (Struct. Mat.)

List of Tables

Tab. B.1: Discharge radioactive (Bq/t_{IHM}) and stable (mol/t_{IHM}) inventories for PWR UO_2 model fuel assembly for an average burnup of 48 GWd/t_{IHM} (Cut-off 1 GBq)

Fuel type Burnup Inventory Mass [kg/t_{IHM}] Time [a]	**PWR UO_2** 48 GWd/t_{IHM} Fuel + imp. 1'000.0 0	**PWR UO_2** 48 GWd/t_{IHM} Struct. Mat. 378.8 0	**PWR UO_2** 48 GWd/t_{IHM} Total 1'378.8 [1] 0
Nuclide	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**
H-3	4.2E+13	1.9E+10	4.2E+13
C-14	4.0E+10	2.4E+10	6.4E+10
Cl-36	5.2E+08	5.3E+08	1.1E+09
Ar-39	7.8E+09	1.5E+06	7.8E+09
Fe-55	1.5E+12	1.6E+14	1.6E+14
Co-60	5.5E+12	8.5E+13	9.0E+13
Ni-59	5.4E+08	7.9E+10	8.0E+10
Ni-63	7.3E+10	1.1E+13	1.1E+13
Se-79	1.0E+09	3.1E+03	1.0E+09
Kr-85	2.2E+15	4.3E+09	2.2E+15
Sr-90	3.8E+15	6.3E+09	3.8E+15
Y-90	4.0E+15	4.1E+13	4.0E+15
Zr-93	9.5E+10	6.6E+09	1.0E+11
Nb-93M	7.2E+09	4.7E+08	7.6E+09
Nb-94	1.2E+08	2.4E+10	2.4E+10
Tc-99	7.0E+11	1.2E+08	7.0E+11
Pd-107	6.2E+09	4.6E+04	6.2E+09
Cd-113M	4.2E+12	5.5E+07	4.2E+12
Sn-121M	1.4E+11	5.4E+10	2.0E+11
Sn-126	1.9E+10	1.3E+05	1.9E+10
Sb-125	7.2E+14	5.3E+13	7.7E+14
Te-125M	1.5E+14	1.1E+13	1.6E+14
I-129	1.7E+09	7.5E+03	1.7E+09
Cs-134	9.6E+15	3.3E+10	9.6E+15
Cs-135	2.5E+10	7.7E+04	2.5E+10
Cs-137	5.6E+15	1.9E+10	5.6E+15
Ba-133	1.1E+09	7.0E+01	1.1E+09
Ba-137M	5.3E+15	1.8E+10	5.3E+15
Pm-146	1.9E+10	6.2E+04	1.9E+10
Pm-147	7.6E+15	2.8E+10	7.6E+15
Sm-151	1.4E+13	5.8E+07	1.4E+13
Eu-152	2.7E+11	9.5E+05	2.7E+11
Eu-154	6.1E+14	2.6E+09	6.1E+14
Eu-155	2.4E+14	1.1E+09	2.4E+14
Ho-166M	1.2E+09	4.1E+03	1.2E+09
Hf-178M	1.9E+09	4.2E+10	4.4E+10
Tl-204	1.1E+10	1.7E-16	1.1E+10
U-234	4.4E+10	1.5E+03	4.4E+10
U-235	6.9E+08	1.3E+02	6.9E+08
U-236	1.3E+10	2.5E+03	1.3E+10
U-238	1.1E+10	1.0E+04	1.1E+10
Np-237	1.8E+10	5.5E+03	1.8E+10

Tab. B.1: (Cont.)

Fuel type Burnup Inventory Mass [kg/t_{IHM}] Time [a]	**PWR UO_2** 48 GWd/t_{IHM} Fuel + imp. 1'000.0 0	**PWR UO_2** 48 GWd/t_{IHM} Struct. Mat. 378.8 0	**PWR UO_2** 48 GWd/t_{IHM} Total 1'378.8 [1] 0
Nuclide	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**
Pu-238	1.9E+14	2.2E+08	1.9E+14
Pu-239	1.4E+13	8.9E+07	1.4E+13
Pu-240	2.3E+13	1.3E+08	2.3E+13
Pu-241	6.5E+15	5.2E+10	6.5E+15
Pu-242	1.1E+11	1.1E+06	1.1E+11
Am-241	7.3E+12	5.8E+07	7.3E+12
Am-242M	4.9E+11	3.9E+06	4.9E+11
Am-243	1.3E+12	1.5E+07	1.3E+12
Cm-242	2.6E+15	2.4E+10	2.6E+15
Cm-243	1.1E+12	1.0E+07	1.1E+12
Cm-244	2.2E+14	2.6E+09	2.2E+14
Cm-245	3.2E+10	3.8E+05	3.2E+10
Cm-246	7.4E+09	9.9E+04	7.4E+09
Total α **Total βγ** **Total Activity**	**3.0E+15** **2.5E+17** **2.6E+17**	**2.7E+10** **2.7E+15** **2.7E+15**	**3.0E+15** **2.6E+17** **2.6E+17**
Stable Elements			
Element	**[mol/t_{IHM}]**	**[mol/t_{IHM}]**	**[mol/t_{IHM}]**
Ca	1.1E+00	2.1E-01	1.3E+00
Fe	1.1E+00	7.6E+02	7.7E+02
Co	1.7E-02	6.9E-01	7.0E-01
Ni	4.8E-01	2.6E+02	2.6E+02
Se	8.8E-01	2.3E-06	8.8E-01
Sr	5.3E+00	1.0E-03	5.3E+00
Zr	4.3E+01	3.0E+03	3.0E+03
Nb	1.2E-02	4.0E+00	4.1E+00
Pd	1.5E+01	9.3E-05	1.5E+01
Ag	1.0E+00	8.3E-06	1.0E+00
Cd	1.3E+00	1.3E-03	1.3E+00
Sn	5.3E-01	3.5E+01	3.6E+01
Sb	1.5E-01	2.3E-02	1.7E-01
Sm	4.2E+00	1.7E-05	4.2E+00
Eu	1.1E+00	5.2E-06	1.1E+00
Ho	3.7E-03	2.4E-08	3.7E-03
Hf	7.1E-03	1.5E-01	1.6E-01
Pb	5.4E-03	1.7E-01	1.8E-01

[1] The total mass of fuel + structural materials is 1'378.8 + 134.0 (mass of oxygen) = 1'512.8 kg.

Tab. B.2: Radioactive (Bq/t_{IHM}) and stable (mol/t_{IHM}) inventories for PWR UO_2 model fuel assembly for an average burnup of 48 GWd/t_{IHM} after 40 years decay (Cut-off 1 GBq)

Fuel type Burnup Inventory Mass [kg/t_{IHM}] Time [a]	**PWR UO_2** 48 GWd/t_{IHM} Fuel + imp. 1'000.0 40	**PWR UO_2** 48 GWd/t_{IHM} Struct. Mat. 378.8 40	**PWR UO_2** 48 GWd/t_{IHM} Total 1'378.8 [1] 40
Nuclide	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**
H-3	4.4E+12	2.0E+09	4.4E+12
C-14	4.0E+10	2.4E+10	6.4E+10
Cl-36	5.2E+08	5.3E+08	1.1E+09
Ar-39	7.0E+09	1.4E+06	7.0E+09
Fe-55	5.8E+07	6.2E+09	6.3E+09
Co-60	2.9E+10	4.4E+11	4.7E+11
Ni-59	5.4E+08	7.9E+10	8.0E+10
Ni-63	5.5E+10	8.4E+12	8.5E+12
Se-79	1.0E+09	3.1E+03	1.0E+09
Kr-85	1.7E+14	3.3E+08	1.7E+14
Sr-90	1.4E+15	2.4E+09	1.4E+15
Y-90	1.4E+15	2.4E+09	1.4E+15
Zr-93	9.5E+10	6.6E+09	1.0E+11
Nb-93M	7.5E+10	5.6E+09	8.1E+10
Nb-94	1.2E+08	2.4E+10	2.4E+10
Tc-99	7.0E+11	1.2E+08	7.0E+11
Pd-107	6.2E+09	4.6E+04	6.2E+09
Cd-113M	6.3E+11	8.2E+06	6.3E+11
Sn-121	6.6E+10	2.5E+10	9.1E+10
Sn-121M	8.5E+10	3.3E+10	1.2E+11
Sn-126	1.9E+10	1.3E+05	1.9E+10
Sb-126	2.7E+09	1.8E+04	2.7E+09
Sb-126M	1.9E+10	1.3E+05	1.9E+10
Sb-125	3.2E+10	2.4E+09	3.4E+10
Te-125M	7.9E+09	5.8E+08	8.5E+09
I-129	1.7E+09	7.5E+03	1.7E+09
Cs-134	1.4E+10	4.7E+04	1.4E+10
Cs-135	2.5E+10	7.7E+04	2.5E+10
Cs-137	2.2E+15	7.6E+09	2.2E+15
Ba-137M	2.1E+15	7.2E+09	2.1E+15
Pm-147	1.9E+11	7.1E+05	1.9E+11
Sm-151	1.0E+13	4.4E+07	1.0E+13
Eu-152	3.4E+10	1.2E+05	3.4E+10
Eu-154	2.6E+13	1.1E+08	2.6E+13
Eu-155	7.1E+11	3.3E+06	7.1E+11
Ho-166M	1.2E+09	4.0E+03	1.2E+09
Hf-178M	7.8E+08	1.7E+10	1.8E+10
Th-234	1.1E+10	1.0E+04	1.1E+10
Pa-233	2.0E+10	1.8E+04	2.0E+10
Pa-234M	1.1E+10	1.0E+04	1.1E+10
U-234	6.4E+10	3.5E+04	6.4E+10
U-235	6.9E+08	1.3E+02	6.9E+08
U-235M	1.4E+13	8.9E+07	1.4E+13

Tab. B.2: (Cont.)

Fuel type Burnup Inventory Mass [kg/t_{IHM}] Time [a]	**PWR UO_2** 48 GWd/t_{IHM} Fuel + imp. 1'000.0 40	**PWR UO_2** 48 GWd/t_{IHM} Struct. Mat. 378.8 40	**PWR UO_2** 48 GWd/t_{IHM} Total 1'378.8 [1] 40
Nuclide	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**
U-236	1.3E+10	2.5E+03	1.3E+10
U-237	2.3E+10	1.8E+05	2.3E+10
U-238	1.1E+10	1.0E+04	1.1E+10
Np-237	2.0E+10	1.8E+04	2.0E+10
Np-238	1.9E+09	1.5E+04	1.9E+09
Np-239	1.3E+12	1.5E+07	1.3E+12
Pu-238	1.5E+14	2.5E+08	1.5E+14
Pu-239	1.4E+13	8.9E+07	1.4E+13
Pu-240	2.3E+13	1.3E+08	2.3E+13
Pu-241	9.4E+14	7.5E+09	9.4E+14
Pu-242	1.1E+11	1.1E+06	1.1E+11
Am-241	1.8E+14	1.5E+09	1.8E+14
Am-242	4.0E+11	3.2E+06	4.0E+11
Am-242M	4.0E+11	3.2E+06	4.0E+11
Am-243	1.3E+12	1.5E+07	1.3E+12
Cm-242	3.3E+11	2.6E+06	3.3E+11
Cm-243	4.2E+11	3.9E+06	4.2E+11
Cm-244	4.8E+13	5.6E+08	4.8E+13
Cm-245	3.2E+10	3.8E+05	3.2E+10
Cm-246	7.4E+09	9.9E+04	7.4E+09
Total α **Total βγ** **Total Activity**	**4.2E+14** **8.4E+15** **8.8E+15**	**2.5E+09** **9.1E+12** **9.1E+12**	**4.2E+14** **8.4E+15** **8.8E+15**
Stable Elements			
Element	**[mol/t_{IHM}]**	**[mol/t_{IHM}]**	**[mol/t_{IHM}]**
Ca	1.1E+00	2.1E-01	1.3E+00
Fe	1.1E+00	7.6E+02	7.7E+02
Co	1.7E-02	6.9E-01	7.0E-01
Ni	4.8E-01	2.6E+02	2.6E+02
Se	8.8E-01	2.3E-06	8.8E-01
Sr	5.3E+00	1.0E-03	5.3E+00
Zr	4.8E+01	3.0E+03	3.0E+03
Nb	1.2E-02	4.0E+00	4.1E+00
Pd	1.5E+01	9.3E-05	1.5E+01
Ag	1.0E+00	8.3E-06	1.0E+00
Cd	1.3E+00	1.3E-03	1.3E+00
Sn	5.3E-01	3.5E+01	3.6E+01
Sb	1.5E-01	2.3E-02	1.7E-01
Sm	4.2E+00	1.7E-05	4.2E+00
Eu	1.2E+00	5.2E-06	1.2E+00
Ho	3.7E-03	2.4E-08	3.7E-03
Hf	7.1E-03	1.5E-01	1.6E-01
Pb	5.4E-03	1.7E-01	1.8E-01

[1] The total mass of fuel + structural materials is 1'378.8 + 134.0 (mass of oxygen) = 1'512.8 kg.

Tab. B.3: Discharge radioactive (Bq/t_{IHM}) and stable (mol/t_{IHM}) inventories for BWR UO_2 model fuel assembly for an average burnup of 48 GWd/t_{IHM} (Cut-off 1GBq)

Fuel type Burnup Material Mass [kg/t_{IHM}] Time [a]	**BWR UO_2** 48 GWd/t_{IHM} Fuel + imp. 1'000 0	**BWR UO_2** 48 GWd/t_{IHM} Struct. Mat. 502.1 0	**BWR UO_2** 48 GWd/t_{IHM} Total 1'502.1 [1] 0
Nuclide	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**
H-3	4.2E+13	3.2E+10	4.2E+13
C-14	4.2E+10	3.6E+10	7.8E+10
Cl-36	5.4E+08	8.9E+08	1.4E+09
Ar-39	8.0E+09	2.4E+06	8.0E+09
Fe-55	1.6E+12	1.8E+14	1.8E+14
Co-60	5.9E+12	9.5E+13	1.0E+14
Ni-59	5.6E+08	7.9E+10	8.0E+10
Ni-63	7.6E+10	1.1E+13	1.1E+13
Se-79	1.0E+09	5.0E+03	1.0E+09
Kr-85	2.2E+15	7.0E+09	2.2E+15
Sr-90	3.8E+15	1.0E+10	3.8E+15
Y-90	4.0E+15	7.4E+13	4.0E+15
Zr-93	9.4E+10	1.1E+10	1.1E+11
Nb-93M	6.4E+09	6.8E+08	7.1E+09
Nb-94	1.3E+08	8.2E+09	8.3E+09
Tc-99	7.0E+11	4.2E+07	7.0E+11
Pd-107	6.4E+09	7.4E+04	6.4E+09
Cd-113M	4.3E+12	9.2E+07	4.3E+12
Sn-121M	1.5E+11	9.4E+10	2.4E+11
Sn-126	1.9E+10	2.0E+05	1.9E+10
Sb-125	7.6E+14	9.3E+13	8.5E+14
Te-125M	1.6E+14	2.0E+13	1.8E+14
I-129	1.7E+09	1.2E+04	1.7E+09
Cs-134	1.0E+16	5.5E+10	1.0E+16
Cs-135	2.3E+10	1.1E+05	2.3E+10
Cs-137	5.6E+15	3.1E+10	5.6E+15
Ba-133	1.2E+09	1.1E+02	1.2E+09
Ba-137M	5.3E+15	2.9E+10	5.3E+15
Pm-146	1.9E+10	9.8E+04	1.9E+10
Pm-147	7.6E+15	4.6E+10	7.6E+15
Sm-151	1.4E+13	9.0E+07	1.4E+13
Eu-152	2.2E+11	1.3E+06	2.2E+11
Eu-154	6.3E+14	4.3E+09	6.3E+14
Eu-155	2.5E+14	1.9E+09	2.5E+14
Ho-166M	1.3E+09	6.6E+03	1.3E+09
Hf-178M	2.1E+09	6.2E+10	6.4E+10
Tl-204	1.2E+10	1.1E-16	1.2E+10
U-234	4.0E+10	1.2E+03	4.0E+10
U-235	5.7E+08	1.5E+02	5.7E+08

Tab. B.3: (Cont.)

Fuel type Burnup Material Mass [kg/t_{IHM}] Time [a]	**BWR UO_2** 48 GWd/t_{IHM} Fuel + imp. 1'000 0	**BWR UO_2** 48 GWd/t_{IHM} Struct. Mat. 502.1 0	**BWR UO_2** 48 GWd/t_{IHM} Total 1'502.1 [1] 0
Nuclide	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**
U-236	1.3E+10	3.8E+03	1.3E+10
U-238	1.2E+10	1.4E+04	1.2E+10
Np-237	1.8E+10	8.4E+03	1.8E+10
Pu-238	2.0E+14	3.3E+08	2.0E+14
Pu-239	1.4E+13	1.4E+08	1.4E+13
Pu-240	2.5E+13	1.9E+08	2.5E+13
Pu-241	6.6E+15	8.3E+10	6.6E+15
Pu-242	1.2E+11	1.8E+06	1.2E+11
Am-241	6.6E+12	8.2E+07	6.6E+12
Am-242M	4.4E+11	5.5E+06	4.4E+11
Am-243	1.4E+12	2.4E+07	1.4E+12
Cm-242	2.6E+15	3.8E+10	2.6E+15
Cm-243	1.1E+12	1.7E+07	1.1E+12
Cm-244	2.3E+14	4.2E+09	2.3E+14
Cm-245	3.5E+10	6.6E+05	3.5E+10
Cm-246	7.6E+09	1.6E+05	7.6E+09
Total α **Total βγ** **Total Activity**	**3.1E+15** **2.7E+17** **2.8E+17**	**4.3E+10** **4.6E+15** **4.6E+15**	**3.1E+15** **2.8E+17** **2.8E+17**
Stable Elements			
Element	**[mol/t_{IHM}]**	**[mol/t_{IHM}]**	**[mol/t_{IHM}]**
Ca	1.1E+00	3.2E-01	1.4E+00
Fe	1.1E+00	7.4E+02	7.4E+02
Co	1.6E-02	6.0E-01	6.2E-01
Ni	4.8E-01	1.6E+02	1.6E+02
Se	8.8E-01	3.7E-06	8.8E-01
Sr	5.2E+00	1.6E-03	5.2E+00
Zr	4.2E+01	4.6E+03	4.6E+03
Nb	1.2E-02	8.6E-01	8.8E-01
Pd	1.5E+01	1.5E-04	1.5E+01
Ag	1.0E+00	1.3E-05	1.0E+00
Cd	1.3E+00	2.0E-03	1.3E+00
Sn	5.4E-01	5.4E+01	5.4E+01
Sb	1.5E-01	3.9E-02	1.9E-01
Sm	4.3E+00	2.7E-05	4.3E+00
Eu	1.1E+00	8.4E-06	1.1E+00
Ho	3.7E-03	3.8E-08	3.7E-03
Hf	7.1E-03	2.3E-01	2.4E-01
Pb	5.4E-03	2.6E-01	2.7E-01

[1] The total mass of fuel + structural materials is 1'502.1 + 134.0 (mass of oxygen) = 1'636.1 kg.

Tab. B.4: Radioactive (Bq/t_{IHM}) and stable (mol/t_{IHM}) inventories for BWR UO_2 model fuel assembly for an average burnup of 48 GWd/t_{IHM} after 40 years decay (Cut-off 1GBq)

Fuel Type	**BWR UO_2**	**BWR UO_2**	**BWR UO_2**
Burnup	48 GWd/t_{IHM}	48 GWd/t_{IHM}	48 GWd/t_{IHM}
Inventory	Fuel + imp.	Struct. Mat.	Total
Mass [kg/t_{IHM}]	1'000.0	502.1	1'502.1 [1]
Time [a]	40	40	40
Nuclide	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**
H-3	4.4E+12	3.4E+09	4.4E+12
C-14	4.2E+10	3.6E+10	7.8E+10
Cl-36	5.4E+08	8.9E+08	1.4E+09
Ar-39	7.2E+09	2.2E+06	7.2E+09
Fe-55	6.2E+07	7.0E+09	7.1E+09
Co-60	3.1E+10	4.9E+11	5.2E+11
Ni-59	5.6E+08	7.9E+10	8.0E+10
Ni-63	5.8E+10	8.4E+12	8.5E+12
Se-79	1.0E+09	5.0E+03	1.0E+09
Kr-85	1.7E+14	5.3E+08	1.7E+14
Sr-90	1.4E+15	3.8E+09	1.4E+15
Y-90	1.4E+15	3.8E+09	1.4E+15
Zr-93	9.4E+10	1.1E+10	1.1E+11
Nb-93M	7.4E+10	8.9E+09	8.3E+10
Nb-94	1.3E+08	8.2E+09	8.3E+09
Tc-99	7.0E+11	4.2E+07	7.0E+11
Pd-107	6.4E+09	7.4E+04	6.4E+09
Cd-113M	6.4E+11	1.4E+07	6.4E+11
Sn-121	7.0E+10	4.4E+10	1.1E+11
Sn-121M	9.1E+10	5.7E+10	1.5E+11
Sn-126	1.9E+10	2.0E+05	1.9E+10
Sb-126M	1.9E+10	2.0E+05	1.9E+10
Sb-125	3.4E+10	4.2E+09	3.8E+10
Te-125M	8.3E+09	1.0E+09	9.3E+09
Sb-126	2.7E+09	2.8E+04	2.7E+09
I-129	1.7E+09	1.2E+04	1.7E+09
Cs-134	1.4E+10	7.9E+04	1.4E+10
Cs-135	2.3E+10	1.1E+05	2.3E+10
Cs-137	2.2E+15	1.2E+10	2.2E+15
Ba-137M	2.1E+15	1.2E+10	2.1E+15
Pm-147	1.9E+11	1.2E+06	1.9E+11
Sm-151	1.0E+13	6.7E+07	1.0E+13
Eu-152	2.7E+10	1.6E+05	2.7E+10
Eu-154	2.7E+13	1.8E+08	2.7E+13
Eu-155	7.4E+11	5.6E+06	7.4E+11
Ho-166M	1.3E+09	6.4E+03	1.3E+09
Hf-178M	8.6E+08	2.5E+10	2.6E+10
Th-234	1.2E+10	1.4E+04	1.2E+10
Pa-233	2.0E+10	2.9E+04	2.0E+10
Pa-234M	1.2E+10	1.4E+04	1.2E+10
U-234	6.1E+10	5.2E+04	6.1E+10
U-235	5.7E+08	1.5E+02	5.7E+08
U-235M	1.4E+13	1.4E+08	1.4E+13
U-236	1.3E+10	3.8E+03	1.3E+10

Tab. B.4: (Cont.)

Fuel Type Burnup Inventory Mass [kg/t_{IHM}] Time [a]	**BWR UO_2** 48 GWd/t_{IHM} Fuel + imp. 1'000.0 40	**BWR UO_2** 48 GWd/t_{IHM} Struct. Mat. 502.1 40	**BWR UO_2** 48 GWd/t_{IHM} Total 1'502.1 [1] 40
Nuclide	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**
U-237	2.3E+10	2.9E+05	2.3E+10
U-238	1.2E+10	1.4E+04	1.2E+10
Np-237	2.0E+10	2.9E+04	2.0E+10
Np-238	1.7E+09	2.2E+04	1.7E+09
Np-239	1.4E+12	2.4E+07	1.4E+12
Pu-238	1.6E+14	3.8E+08	1.6E+14
Pu-239	1.4E+13	1.4E+08	1.4E+13
Pu-240	2.5E+13	1.9E+08	2.5E+13
Pu-241	9.6E+14	1.2E+10	9.6E+14
Pu-242	1.2E+11	1.8E+06	1.2E+11
Am-241	1.9E+14	2.3E+09	1.9E+14
Am-242	3.6E+11	4.5E+06	3.6E+11
Am-242M	3.6E+11	4.5E+06	3.6E+11
Am-243	1.4E+12	2.4E+07	1.4E+12
Cm-242	3.0E+11	3.7E+06	3.0E+11
Cm-243	4.2E+11	6.6E+06	4.2E+11
Cm-244	5.0E+13	9.1E+08	5.0E+13
Cm-245	3.5E+10	6.6E+05	3.5E+10
Cm-246	7.6E+09	1.6E+05	7.6E+09
Total α **Total βγ** **Total Activity**	**4.4E+14** **8.3E+15** **8.7E+15**	**4.0E+09** **9.2E+12** **9.2E+12**	**4.4E+14** **8.3E+15** **8.7E+15**
Stable Elements			
Element	**[mol/t_{IHM}]**	**[mol/t_{IHM}]**	**[mol/t_{IHM}]**
Ca	1.1E+00	3.2E-01	1.5E+00
Fe	1.1E+00	7.5E+02	7.5E+02
Co	1.6E-02	6.0E-01	6.2E-01
Ni	4.8E-01	1.6E+02	1.7E+02
Se	8.7E-01	3.7E-06	8.7E-01
Sr	5.2E+00	1.5E-03	5.2E+00
Zr	4.8E+01	4.6E+03	4.6E+03
Nb	1.2E-02	8.6E-01	8.7E-01
Pd	1.5E+01	1.5E-04	1.5E+01
Ag	1.0E+00	1.3E-05	1.0E+00
Cd	1.3E+00	2.0E-03	1.3E+00
Sn	5.5E-01	5.4E+01	5.5E+01
Sb	1.5E-01	3.9E-02	1.9E-01
Sm	4.3E+00	2.7E-05	4.3E+00
Eu	1.1E+00	8.4E-06	1.1E+00
Ho	3.7E-03	3.8E-08	3.7E-03
Hf	7.1E-03	2.3E-01	2.4E-01
Pb	5.4E-03	2.7E-01	2.7E-01

[1] The total mass of fuel + structural materials is 1'502.1 + 134.0 (mass of oxygen) = 1'636.1 kg.

Tab. B.5: Discharge radioactive (Bq/t_{IHM}) and stable (mol/t_{IHM}) inventories for PWR UO_2 model fuel assembly for an average burnup of 55 GWd/t_{IHM} (Cut-off 1 GBq)

Fuel type Burnup Inventory Mass [kg/t_{IHM}] Time [a]	**PWR UO_2** 55 GWd/t_{IHM} Fuel + imp. 1'000.00 0	**PWR UO_2** 55 GWd/t_{IHM} Struct. Mat. 402.8 0	**PWR UO_2** 55 GWd/t_{IHM} Total 1'402.8 [1] 0
Nuclide	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**
H-3	4.8E+13	2.4E+10	4.8E+13
C-14	4.6E+10	2.9E+10	7.5E+10
Cl-36	6.1E+08	6.7E+08	1.3E+09
Ar-39	7.9E+09	2.1E+06	7.9E+09
Fe-55	1.7E+12	1.8E+14	1.8E+14
Co-60	6.3E+12	9.6E+13	1.0E+14
Ni-59	6.3E+08	8.0E+10	8.1E+10
Ni-63	8.6E+10	1.1E+13	1.1E+13
Se-79	1.2E+09	3.9E+03	1.2E+09
Kr-85	2.5E+15	5.4E+09	2.5E+15
Sr-90	4.3E+15	7.9E+09	4.3E+15
Y-90	4.5E+15	5.1E+13	4.6E+15
Zr-93	1.1E+11	8.2E+09	1.2E+11
Nb-93M	8.4E+09	5.9E+08	9.0E+09
Nb-94	1.4E+08	2.3E+10	2.3E+10
Tc-99	7.6E+11	1.2E+08	7.6E+11
Pd-107	7.4E+09	5.8E+04	7.4E+09
Ag-108M	1.0E+09	2.2E+04	1.0E+09
Cd-113M	5.0E+12	7.2E+07	5.0E+12
Sn-121M	1.7E+11	6.6E+10	2.4E+11
Sn-126	2.2E+10	1.6E+05	2.2E+10
Sb-125	8.4E+14	6.4E+13	9.0E+14
Te-125M	1.8E+14	1.4E+13	1.9E+14
I-129	1.9E+09	9.3E+03	1.9E+09
Cs-134	1.3E+16	4.4E+10	1.3E+16
Cs-135	2.6E+10	8.9E+04	2.6E+10
Cs-137	6.3E+15	2.4E+10	6.3E+15
Ba-133	1.3E+09	1.1E+02	1.3E+09
Ba-137M	6.0E+15	2.3E+10	6.0E+15
Pm-146	1.6E+10	6.1E+04	1.6E+10
Pm-147	7.0E+15	2.9E+10	7.0E+15
Sm-151	1.7E+13	7.3E+07	1.7E+13
Eu-152	2.7E+11	1.0E+06	2.7E+11
Eu-154	7.4E+14	3.4E+09	7.4E+14
Eu-155	2.8E+14	1.4E+09	2.8E+14
Ho-166M	1.5E+09	6.2E+03	1.5E+09
Hf-178M	2.1E+09	4.6E+10	4.8E+10
Tl-204	1.3E+10	4.6E-16	1.3E+10
U-234	4.2E+10	1.8E+03	4.2E+10

Tab. B.5: (Cont.)

Fuel type Burnup Inventory Mass [kg/t_{IHM}] Time [a]	**PWR UO_2** 55 GWd/t_{IHM} Fuel + imp. 1'000.00 0	**PWR UO_2** 55 GWd/t_{IHM} Struct. Mat. 402.8 0	**PWR UO_2** 55 GWd/t_{IHM} Total 1'402.8 [1] 0
Nuclide	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**
U-235	5.1E+08	1.1E+02	5.1E+08
U-236	1.4E+10	2.6E+03	1.4E+10
U-238	1.1E+10	1.1E+04	1.1E+10
Np-237	2.6E+10	7.4E+03	2.6E+10
Pu-238	3.0E+14	2.9E+08	3.0E+14
Pu-239	1.3E+13	8.7E+07	1.3E+13
Pu-240	2.6E+13	1.3E+08	2.6E+13
Pu-241	6.7E+15	5.5E+10	6.7E+15
Pu-242	1.4E+11	1.4E+06	1.4E+11
Am-241	7.0E+12	5.7E+07	7.0E+12
Am-242M	4.5E+11	3.8E+06	4.5E+11
Am-243	1.9E+12	2.1E+07	1.9E+12
Cm-242	3.1E+15	3.0E+10	3.1E+15
Cm-243	1.4E+12	1.4E+07	1.4E+12
Cm-244	3.5E+14	4.1E+09	3.5E+14
Cm-245	4.5E+10	5.4E+05	4.5E+10
Cm-246	1.1E+10	1.6E+05	1.1E+10
Total α **Total βγ** **Total Activity**	**3.8E+15** **2.8E+17** **2.9E+17**	**3.4E+10** **3.2E+15** **3.2E+15**	**3.8E+15** **2.9E+17** **2.9E+17**
Stable Elements			
Element	**[mol/t_{IHM}]**	**[mol/t_{IHM}]**	**[mol/t_{IHM}]**
Ca	1.1E+00	2.3E-01	1.4E+00
Fe	1.1E+00	7.6E+02	7.6E+02
Co	1.6E-02	6.7E-01	6.8E-01
Ni	4.8E-01	2.3E+02	2.3E+02
Se	1.0E+00	2.9E-06	1.0E+00
Sr	6.0E+00	1.2E-03	6.0E+00
Zr	4.8E+01	3.3E+03	3.3E+03
Nb	1.2E-02	3.4E+00	3.4E+00
Pd	1.8E+01	1.2E-04	1.8E+01
Ag	1.2E+00	1.0E-05	1.2E+00
Cd	1.6E+00	1.4E-03	1.6E+00
Sn	6.2E-01	3.9E+01	3.9E+01
Sb	1.7E-01	2.8E-02	2.0E-01
Sm	4.9E+00	2.1E-05	4.9E+00
Eu	1.4E+00	6.7E-06	1.4E+00
Ho	4.4E-03	3.2E-08	4.4E-03
Hf	7.2E-03	1.7E-01	1.7E-01
Pb	5.4E-03	1.9E-01	1.9E-01

[1] The total mass of fuel + structural materials is 1'402.8 + 134.0 (mass of oxygen) = 1'536.8 kg.

Tab. B.6: Radioactive (Bq/t_{IHM}) and stable (mol/t_{IHM}) inventories for PWR UO_2 model fuel assembly for an average burnup of 55 GWd/t_{IHM} after 40 years decay (Cut-off 1 GBq)

Fuel type Burnup Inventory Mass [kg/t_{IHM}] Time [a]	**PWR UO_2** 55 GWd/t_{IHM} Fuel + imp. 1'000.0 40	**PWR UO_2** 55 GWd/t_{IHM} Struct. Mat. 402.8 40	**PWR UO_2** 55 GWd/t_{IHM} Total 1'402.8 [1] 40
Nuclide	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**
H-3	5.1E+12	2.5E+09	5.1E+12
C-14	4.6E+10	2.9E+10	7.5E+10
Cl-36	6.1E+08	6.7E+08	1.3E+09
Ar-39	7.1E+09	1.9E+06	7.1E+09
Fe-55	6.6E+07	7.0E+09	7.1E+09
Co-60	3.3E+10	5.0E+11	5.3E+11
Ni-59	6.3E+08	8.0E+10	8.1E+10
Ni-63	6.5E+10	8.4E+12	8.5E+12
Se-79	1.2E+09	3.9E+03	1.2E+09
Kr-85	1.9E+14	4.1E+08	1.9E+14
Sr-90	1.6E+15	3.0E+09	1.6E+15
Y-90	1.6E+15	3.0E+09	1.6E+15
Zr-93	1.1E+11	8.2E+09	1.2E+11
Nb-93M	8.7E+10	6.9E+09	9.4E+10
Nb-94	1.4E+08	2.3E+10	2.3E+10
Tc-99	7.6E+11	1.2E+08	7.6E+11
Pd-107	7.4E+09	5.8E+04	7.4E+09
Cd-113M	7.5E+11	1.1E+07	7.5E+11
Sn-121	8.0E+10	3.1E+10	1.1E+11
Sn-121M	1.0E+11	4.0E+10	1.4E+11
Sn-126	2.2E+10	1.6E+05	2.2E+10
Sb-125	3.8E+10	2.9E+09	4.1E+10
Sb-126	3.1E+09	2.2E+04	3.1E+09
Sb-126M	2.2E+10	1.6E+05	2.2E+10
Te-125M	9.2E+09	7.0E+08	9.9E+09
I-129	1.9E+09	9.3E+03	1.9E+09
Cs-134	1.9E+10	6.3E+04	1.9E+10
Cs-135	2.6E+10	8.9E+04	2.6E+10
Cs-137	2.5E+15	9.6E+09	2.5E+15
Ba-137M	2.4E+15	9.0E+09	2.4E+15
Pm-147	1.8E+11	7.4E+05	1.8E+11
Sm-151	1.3E+13	5.4E+07	1.3E+13
Eu-152	3.4E+10	1.2E+05	3.4E+10
Eu-154	3.2E+13	1.5E+08	3.2E+13
Eu-155	8.3E+11	4.1E+06	8.3E+11
Ho-166M	1.5E+09	6.1E+03	1.5E+09
Hf-178M	8.6E+08	1.9E+10	2.0E+10
Th-234	1.1E+10	1.1E+04	1.1E+10
Pa-233	2.8E+10	2.1E+04	2.8E+10
Pa-234M	1.1E+10	1.1E+04	1.1E+10
U-234	7.3E+10	4.4E+04	7.3E+10
U-235	5.1E+08	1.1E+02	5.1E+08
U-235M	1.3E+13	8.7E+07	1.3E+13

Tab. B.6: (Cont.)

Fuel type Burnup Inventory Mass [kg/t_{IHM}] Time [a]	**PWR UO_2** 55 GWd/t_{IHM} Fuel + imp. 1'000.0 40	**PWR UO_2** 55 GWd/t_{IHM} Struct. Mat. 402.8 40	**PWR UO_2** 55 GWd/t_{IHM} Total 1'402.8 [1] 40
Nuclide	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**
U-236	1.4E+10	2.6E+03	1.4E+10
U-237	2.4E+10	2.0E+05	2.4E+10
U-238	1.1E+10	1.1E+04	1.1E+10
Np-237	2.8E+10	2.1E+04	2.8E+10
Np-238	1.8E+09	1.5E+04	1.8E+09
Np-239	1.9E+12	2.1E+07	1.9E+12
Pu-238	2.3E+14	3.2E+08	2.3E+14
Pu-239	1.3E+13	8.7E+07	1.3E+13
Pu-240	2.6E+13	1.3E+08	2.6E+13
Pu-241	9.7E+14	8.0E+09	9.7E+14
Pu-242	1.4E+11	1.4E+06	1.4E+11
Am-241	1.9E+14	1.6E+09	1.9E+14
Am-242	3.7E+11	3.1E+06	3.7E+11
Am-242M	3.7E+11	3.1E+06	3.7E+11
Am-243	1.9E+12	2.1E+07	1.9E+12
Cm-242	3.1E+11	2.6E+06	3.1E+11
Cm-243	5.4E+11	5.4E+06	5.4E+11
Cm-244	7.6E+13	8.9E+08	7.6E+13
Cm-245	4.5E+10	5.4E+05	4.5E+10
Cm-246	1.1E+10	1.6E+05	1.1E+10
Total α **Total βγ** **Total Activity**	**5.4E+14** **9.3E+15** **9.9E+15**	**3.1E+09** **9.2E+12** **9.2E+12**	**5.4E+14** **9.3E+15** **9.9E+15**
Stable Elements			
Element	**[mol/t_{IHM}]**	**[mol/t_{IHM}]**	**[mol/t_{IHM}]**
Ca	1.1E+00	2.3E-01	1.4E+00
Fe	1.1E+00	7.6E+02	7.6E+02
Co	1.6E-02	6.7E-01	6.8E-01
Ni	4.8E-01	2.3E+02	2.3E+02
Se	1.0E+00	2.9E-06	1.0E+00
Sr	6.0E+00	1.2E-03	6.0E+00
Zr	5.4E+01	3.3E+03	3.3E+03
Nb	1.2E-02	3.4E+00	3.4E+00
Pd	1.8E+01	1.2E-04	1.8E+01
Ag	1.2E+00	1.0E-05	1.2E+00
Cd	1.6E+00	1.4E-03	1.6E+00
Sn	6.2E-01	3.9E+01	3.9E+01
Sb	1.7E-01	2.8E-02	2.0E-01
Sm	4.9E+00	2.1E-05	4.9E+00
Eu	1.4E+00	6.7E-06	1.4E+00
Ho	4.4E-03	3.2E-08	4.4E-03
Hf	7.2E-03	1.7E-01	1.7E-01
Pb	5.4E-03	1.9E-01	1.9E-01

[1] The mass of fuel + structural materials is 1'402.8 + 134.0 (mass of oxygen) = 1'536.8 kg.

Tab. B.7: Discharge radioactive (Bq/t_{IHM}) inventory for a PWR UO_2 model fuel assembly for an average burnup of 65 GWd/t_{IHM} (Cut-off 1 GBq)

Fuel type Burnup Inventory Mass [kg/t_{IHM}] Time [a]	**PWR UO_2** 65 GWd/t_{IHM} Fuel + imp. 1'000.0 0	**PWR UO_2** 65 GWd/t_{IHM} Struct. Mat. 402.8 0	**PWR UO_2** 65 GWd/t_{IHM} Total 1'402.8 [1] 0
Nuclide	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**
H-3	5.6E+13	2.9E+10	5.6E+13
C-14	5.7E+10	3.5E+10	9.2E+10
Cl-36	7.5E+08	8.2E+08	1.6E+09
Ar-39	8.2E+09	2.9E+06	8.2E+09
Fe-55	2.0E+12	2.1E+14	2.1E+14
Co-60	7.4E+12	1.1E+14	1.2E+14
Ni-59	7.5E+08	9.4E+10	9.5E+10
Ni-63	1.1E+11	1.3E+13	1.3E+13
Se-79	1.4E+09	4.9E+03	1.4E+09
Kr-85	2.8E+15	6.5E+09	2.8E+15
Sr-90	4.8E+15	9.7E+09	4.8E+15
Y-90	5.0E+15	5.5E+13	5.1E+15
Zr-93	1.2E+11	1.0E+10	1.3E+11
Nb-93M	1.1E+10	8.2E+08	1.2E+10
Nb-94	1.7E+08	2.8E+10	2.8E+10
Tc-99	9.0E+11	1.5E+08	9.0E+11
Pd-107	9.6E+09	7.4E+04	9.6E+09
Ag-108M	1.3E+09	3.4E+04	1.3E+09
Cd-113M	6.4E+12	8.6E+07	6.4E+12
Sn-121M	2.2E+11	8.1E+10	3.0E+11
Sn-126	2.8E+10	2.0E+05	2.8E+10
Sb-125	1.0E+15	7.3E+13	1.1E+15
Te-125M	2.2E+14	1.6E+13	2.4E+14
I-129	2.3E+09	1.2E+04	2.3E+09
Cs-134	1.5E+16	6.2E+10	1.5E+16
Cs-135	3.1E+10	1.1E+05	3.1E+10
Cs-137	7.4E+15	3.0E+10	7.4E+15
Ba-133	1.7E+09	2.2E+02	1.7E+09
Ba-137M	7.0E+15	2.8E+10	7.0E+15
Pm-145	1.0E+09	4.6E-01	1.0E+09
Pm-146	1.8E+10	7.2E+04	1.8E+10
Pm-147	7.7E+15	3.5E+10	7.7E+15
Sm-151	1.5E+13	6.6E+07	1.5E+13
Eu-152	2.2E+11	9.2E+05	2.2E+11
Eu-154	9.2E+14	4.5E+09	9.2E+14
Eu-155	3.4E+14	1.7E+09	3.4E+14
Ho-166M	2.1E+09	1.1E+04	2.1E+09
Hf-178M	2.5E+09	5.6E+10	5.8E+10
Tl-204	1.5E+10	1.6E-14	1.5E+10
U-234	3.4E+10	2.4E+03	3.4E+10
U-235	3.1E+08	8.9E+01	3.1E+08
U-236	1.4E+10	2.9E+03	1.4E+10

Tab. B.7: (Cont.)

Fuel type Burnup Inventory Mass [kg/t_{IHM}] Time [a]	**PWR UO_2** 65 GWd/t_{IHM} Fuel + imp. 1'000.0 0	**PWR UO_2** 65 GWd/t_{IHM} Struct. Mat. 402.8 0	**PWR UO_2** 65 GWd/t_{IHM} Total 1'402.8 [1] 0
Nuclide	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**
U-238	1.1E+10	1.0E+04	1.1E+10
Np-237	2.3E+10	7.0E+03	2.3E+10
Pu-238	3.3E+14	3.9E+08	3.3E+14
Pu-239	1.3E+13	7.9E+07	1.3E+13
Pu-240	2.9E+13	1.4E+08	2.9E+13
Pu-241	7.1E+15	5.2E+10	7.1E+15
Pu-242	1.9E+11	1.8E+06	1.9E+11
Am-241	7.5E+12	5.5E+07	7.5E+12
Am-242M	4.8E+11	3.6E+06	4.8E+11
Am-243	2.8E+12	3.1E+07	2.8E+12
Cm-242	3.9E+15	3.4E+10	3.9E+15
Cm-243	2.0E+12	1.8E+07	2.0E+12
Cm-244	6.3E+14	7.5E+09	6.3E+14
Cm-245	1.0E+11	1.2E+06	1.0E+11
Cm-246	3.6E+10	4.9E+05	3.6E+10
Total α **Total βγ** **Total Activity**	**4.9E+15** **3.0E+17** **3.0E+17**	**4.2E+10** **3.5E+15** **3.5E+15**	**4.9E+15** **3.0E+17** **3.1E+17**
Stable Elements			
Element	**[mol/t_{IHM}]**	**[mol/t_{IHM}]**	**[mol/t_{IHM}]**
Ca	1.1E+00	2.3E-01	1.4E+00
Fe	1.1E+00	7.4E+02	7.4E+02
Co	1.5E-02	6.3E-01	6.4E-01
Ni	4.8E-01	2.3E+02	2.3E+02
Se	1.1E+00	3.6E-06	1.1E+00
Sr	6.7E+00	1.5E-03	6.7E+00
Zr	5.6E+01	3.3E+03	3.3E+03
Nb	1.2E-02	3.3E+00	3.3E+00
Pd	2.4E+01	1.6E-04	2.4E+01
Ag	1.5E+00	1.2E-05	1.5E+00
Cd	2.2E+00	1.4E-03	2.2E+00
Sn	7.8E-01	3.8E+01	3.9E+01
Sb	2.2E-01	3.3E-02	2.5E-01
Sm	5.7E+00	2.5E-05	5.7E+00
Eu	1.5E+00	7.8E-06	1.5E+00
Ho	5.7E-03	4.5E-08	5.7E-03
Hf	7.3E-03	1.6E-01	1.7E-01
Pb	5.4E-03	1.9E-01	1.9E-01

[1] The mass of fuel + structural materials is 1'402.8 + 134.0 (mass of oxygen) = 1'536.8 kg.

Tab. B.8: Radioactive (Bq/t_{IHM}) and stable (mol/t_{IHM}) inventories for PWR UO_2 model fuel assembly for an average burnup of 65 GWd/t_{IHM} after 40 years decay (Cut-off 1 GBq)

Fuel type Burnup Inventory Mass [kg/t_{IHM}] Time [a]	**PWR UO_2** 65 GWd/t_{IHM} Fuel + imp. 1'000.0 40	**PWR UO_2** 65 GWd/t_{IHM} Struct. Mat. 402.8 40	**PWR UO_2** 65 GWd/t_{IHM} Total 1'402.8 [1] 40
Nuclide	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**
H-3	5.9E+12	3.1E+09	5.9E+12
C-14	5.7E+10	3.5E+10	9.2E+10
Cl-36	7.5E+08	8.2E+08	1.6E+09
Ar-39	7.4E+09	2.6E+06	7.4E+09
Fe-55	7.8E+07	8.2E+09	8.3E+09
Co-60	3.8E+10	5.7E+11	6.1E+11
Ni-59	7.5E+08	9.4E+10	9.5E+10
Ni-63	8.4E+10	9.9E+12	1.0E+13
Se-79	1.4E+09	4.9E+03	1.4E+09
Kr-85	2.1E+14	4.9E+08	2.1E+14
Sr-90	1.8E+15	3.7E+09	1.8E+15
Y-90	1.8E+15	3.7E+09	1.8E+15
Zr-93	1.2E+11	1.0E+10	1.3E+11
Nb-93M	9.6E+10	8.4E+09	1.0E+11
Nb-94	1.7E+08	2.8E+10	2.8E+10
Tc-99	9.0E+11	1.5E+08	9.0E+11
Pd-107	9.6E+09	7.4E+04	9.6E+09
Ag-108M	1.2E+09	3.2E+04	1.2E+09
Cd-113M	9.6E+11	1.3E+07	9.6E+11
Sn-121	1.0E+11	3.8E+10	1.4E+11
Sn-121M	1.3E+11	4.9E+10	1.8E+11
Sn-126	2.8E+10	2.0E+05	2.8E+10
Sb-125	4.5E+10	3.3E+09	4.8E+10
Sb-126	3.9E+09	2.8E+04	3.9E+09
Sb-126M	2.8E+10	2.0E+05	2.8E+10
Te-125M	1.1E+10	8.0E+08	1.2E+10
I-129	2.3E+09	1.2E+04	2.3E+09
Cs-134	2.1E+10	8.9E+04	2.1E+10
Cs-135	3.1E+10	1.1E+05	3.1E+10
Cs-137	3.0E+15	1.2E+10	3.0E+15
Ba-137M	2.8E+15	1.1E+10	2.8E+15
Pm-147	2.0E+11	8.9E+05	2.0E+11
Sm-151	1.1E+13	4.9E+07	1.1E+13
Eu-152	2.7E+10	1.1E+05	2.7E+10
Eu-154	3.9E+13	1.9E+08	3.9E+13
Eu-155	1.0E+12	5.0E+06	1.0E+12
Ho-166M	2.1E+09	1.1E+04	2.1E+09
Hf-178M	1.0E+09	2.3E+10	2.4E+10
Th-234	1.1E+10	1.0E+04	1.1E+10
Pa-233	2.5E+10	2.0E+04	2.5E+10
Pa-234M	1.1E+10	1.0E+04	1.1E+10
U-234	6.8E+10	5.7E+04	6.8E+10
U-235	3.1E+08	8.9E+01	3.1E+08
U-235M	1.3E+13	7.9E+07	1.3E+13

Tab. B.8: (Cont.)

Fuel type Burnup Inventory Mass [kg/t_{IHM}] Time [a]	**PWR UO_2** 65 GWd/t_{IHM} Fuel + imp. 1'000.0 40	**PWR UO_2** 65 GWd/t_{IHM} Struct. Mat. 402.8 40	**PWR UO_2** 65 GWd/t_{IHM} Total 1'402.8 [1] 40
Nuclide	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**
U-236	1.4E+10	2.9E+03	1.4E+10
U-237	2.5E+10	1.8E+05	2.5E+10
U-238	1.1E+10	1.0E+04	1.1E+10
Np-237	2.5E+10	2.0E+04	2.5E+10
Np-238	1.9E+09	1.4E+04	1.9E+09
Np-239	2.8E+12	3.1E+07	2.8E+12
Pu-238	2.6E+14	4.1E+08	2.6E+14
Pu-239	1.3E+13	7.9E+07	1.3E+13
Pu-240	2.9E+13	1.4E+08	2.9E+13
Pu-241	1.0E+15	7.5E+09	1.0E+15
Pu-242	1.9E+11	1.8E+06	1.9E+11
Am-241	2.0E+14	1.5E+09	2.0E+14
Am-242	3.9E+11	2.9E+06	3.9E+11
Am-242M	3.9E+11	3.0E+06	3.9E+11
Am-243	2.8E+12	3.1E+07	2.8E+12
Cm-242	3.3E+11	2.4E+06	3.3E+11
Cm-243	7.7E+11	6.9E+06	7.7E+11
Cm-244	1.4E+14	1.6E+09	1.4E+14
Cm-245	1.0E+11	1.2E+06	1.0E+11
Cm-246	3.6E+10	4.9E+05	3.6E+10
Total α **Total βγ** **Total Activity**	**6.5E+14** **1.1E+16** **1.1E+16**	**3.8E+09** **1.1E+13** **1.1E+13**	**6.5E+14** **1.1E+16** **1.1E+16**
Stable Elements			
Element	**[mol/t_{IHM}]**	**[mol/t_{IHM}]**	**[mol/t_{IHM}]**
Ca	1.1E+00	2.3E-01	1.4E+00
Fe	1.1E+00	7.4E+02	7.4E+02
Co	1.5E-02	6.3E-01	6.4E-01
Ni	4.8E-01	2.3E+02	2.3E+02
Se	1.1E+00	3.6E-06	1.1E+00
Sr	6.7E+00	1.5E-03	6.7E+00
Zr	5.9E+01	3.3E+03	3.3E+03
Nb	1.2E-02	3.3E+00	3.3E+00
Pd	2.4E+01	1.6E-04	2.4E+01
Ag	1.5E+00	1.2E-05	1.5E+00
Cd	2.2E+00	1.4E-03	2.2E+00
Sn	7.8E-01	3.8E+01	3.9E+01
Sb	2.2E-01	3.3E-02	2.5E-01
Sm	5.7E+00	2.5E-05	5.7E+00
Eu	1.6E+00	7.8E-06	1.6E+00
Ho	5.7E-03	4.5E-08	5.7E-03
Hf	7.3E-03	1.6E-01	1.7E-01
Pb	5.4E-03	1.9E-01	1.9E-01

[1] The mass of fuel + structural materials is 1'402.8 + 134.0 (mass of oxygen) = 1'536.8 kg.

Tab. B.9: Discharge radioactive (Bq/t_{IHM}) and stable (mol/t_{IHM}) inventories for PWR UO_2 model fuel assembly for an average burnup of 75 GWd/t_{IHM} (Cut-off 1 GBq)

Fuel type Burnup Inventory Mass [kg/t_{IHM}] Time [a]	**PWR UO_2** 75 GWd/t_{IHM} Fuel + imp. 1'000.0 0	**PWR UO_2** 75 GWd/t_{IHM} Struct. Mat. 402.8 0	**PWR UO_2** 75 GWd/t_{IHM} Total 1'402.8 [1] 0
Nuclide	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**
H-3	6.3E+13	3.5E+10	6.3E+13
C-14	6.2E+10	3.9E+10	1.0E+11
Cl-36	7.9E+08	8.7E+08	1.7E+09
Ar-39	8.9E+09	3.2E+06	8.9E+09
Fe-55	2.1E+12	2.4E+14	2.4E+14
Co-60	7.7E+12	1.3E+14	1.3E+14
Ni-59	7.8E+08	1.1E+11	1.1E+11
Ni-63	1.1E+11	1.5E+13	1.5E+13
Se-79	1.6E+09	5.5E+03	1.6E+09
Kr-85	3.1E+15	7.2E+09	3.1E+15
Sr-90	5.5E+15	1.1E+10	5.5E+15
Y-90	5.8E+15	5.7E+13	5.9E+15
Zr-93	1.4E+11	1.1E+10	1.5E+11
Nb-93M	1.5E+10	1.0E+09	1.6E+10
Nb-94	1.9E+08	3.2E+10	3.2E+10
Tc-99	9.6E+11	1.7E+08	9.6E+11
Pd-107	1.1E+10	8.3E+04	1.1E+10
Ag-108M	1.4E+09	4.2E+04	1.4E+09
Cd-113M	7.2E+12	1.0E+08	7.2E+12
Sn-121M	2.4E+11	9.2E+10	3.3E+11
Sn-126	3.2E+10	2.3E+05	3.2E+10
Sb-125	1.1E+15	7.9E+13	1.2E+15
Te-125M	2.4E+14	1.8E+13	2.6E+14
I-129	2.6E+09	1.3E+04	2.6E+09
Ba-133	1.8E+09	3.0E+02	1.8E+09
Cs-134	1.9E+16	7.4E+10	1.9E+16
Cs-135	3.9E+10	1.4E+05	3.9E+10
Cs-137	8.5E+15	3.3E+10	8.5E+15
Ba-137M	8.1E+15	3.2E+10	8.1E+15
Pm-145	1.2E+09	5.9E-01	1.2E+09
Pm-146	1.7E+10	6.7E+04	1.7E+10
Pm-147	6.8E+15	3.0E+10	6.8E+15
Sm-151	2.0E+13	8.5E+07	2.0E+13
Eu-152	3.1E+11	1.2E+06	3.1E+11
Eu-154	1.1E+15	5.3E+09	1.1E+15
Eu-155	4.1E+14	2.0E+09	4.1E+14
Ho-166M	2.4E+09	1.5E+04	2.4E+09
Hf-178M	2.8E+09	6.4E+10	6.7E+10
Tl-204	1.5E+10	4.3E-14	1.5E+10
U-232	1.2E+09	1.2E+02	1.2E+09
U-234	3.7E+10	3.1E+03	3.7E+10
U-235	3.2E+08	8.9E+01	3.2E+08

Tab. B.9: (Cont.)

Fuel type Burnup Inventory Mass [kg/t_{IHM}] Time [a]	**PWR UO_2** 75 GWd/t_{IHM} Fuel + imp. 1'000.0 0	**PWR UO_2** 75 GWd/t_{IHM} Struct. Mat. 402.8 0	**PWR UO_2** 75 GWd/t_{IHM} Total 1'402.8 [1] 0
Nuclide	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**
U-236	1.6E+10	2.7E+03	1.6E+10
U-238	1.1E+10	1.0E+04	1.1E+10
Np-237	3.8E+10	8.8E+03	3.8E+10
Pu-238	5.6E+14	5.1E+08	5.6E+14
Pu-239	1.3E+13	7.8E+07	1.3E+13
Pu-240	2.8E+13	1.2E+08	2.8E+13
Pu-241	7.5E+15	5.3E+10	7.5E+15
Pu-242	2.1E+11	1.9E+06	2.1E+11
Am-241	8.3E+12	5.9E+07	8.3E+12
Am-242M	5.5E+11	3.9E+06	5.5E+11
Am-243	3.6E+12	3.7E+07	3.6E+12
Cm-242	4.4E+15	3.7E+10	4.4E+15
Cm-243	2.5E+12	2.2E+07	2.5E+12
Cm-244	9.5E+14	1.1E+10	9.5E+14
Cm-245	1.4E+11	1.7E+06	1.4E+11
Cm-246	5.5E+10	7.4E+05	5.5E+10
Total α **Total βγ** **Total Activity**	**6.0E+15** **3.1E+17** **3.1E+17**	**4.9E+10** **3.6E+15** **3.6E+15**	**6.0E+15** **3.1E+17** **3.2E+17**
Stable Elements			
Element	**[mol/t_{IHM}]**	**[mol/t_{IHM}]**	**[mol/t_{IHM}]**
Ca	1.1E+00	2.3E-01	1.4E+00
Fe	1.1E+00	7.4E+02	7.4E+02
Co	1.6E-02	6.2E-01	6.4E-01
Ni	4.8E-01	2.3E+02	2.3E+02
Se	9.8E-01	2.8E-06	9.8E-01
Sr	5.9E+00	1.2E-03	5.9E+00
Zr	4.8E+01	3.3E+03	3.3E+03
Nb	1.2E-02	3.3E+00	3.3E+00
Pd	1.8E+01	1.2E-04	1.8E+01
Ag	1.1E+00	9.9E-06	1.1E+00
Cd	1.5E+00	1.4E-03	1.5E+00
Sn	6.1E-01	3.8E+01	3.9E+01
Sb	1.7E-01	2.7E-02	2.0E-01
Sm	4.8E+00	2.0E-05	4.8E+00
Eu	1.3E+00	6.3E-06	1.3E+00
Ho	4.2E-03	3.1E-08	4.2E-03
Hf	7.2E-03	1.6E-01	1.7E-01
Pb	5.4E-03	1.9E-01	1.9E-01

[1] The mass of fuel + structural materials is 1'402.8 + 134.0 (mass of oxygen) = 1'536.8 kg.

Tab. B.10: Radioactive (Bq/t_{IHM}) and stable (mol/t_{IHM}) inventories for PWR UO_2 model fuel assembly for an average burnup of 75 GWd/t_{IHM} after 40 years decay (Cut-off 1 GBq)

Fuel type Burnup Inventory Mass [kg/t_{IHM}] Time [a]	**PWR UO_2** 75 GWd/t_{IHM} Fuel + imp. 1'000.0 40	**PWR UO_2** 75 GWd/t_{IHM} Struct. Mat. 402.8 40	**PWR UO_2** 75 GWd/t_{IHM} Total 1'402.8 [1] 40
Nuclide	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**
H-3	6.6E+12	3.7E+09	6.6E+12
C-14	6.2E+10	3.9E+10	1.0E+11
Cl-36	7.9E+08	8.7E+08	1.7E+09
Ar-39	8.0E+09	2.9E+06	8.0E+09
Fe-55	8.2E+07	9.3E+09	9.4E+09
Co-60	4.0E+10	6.8E+11	7.2E+11
Ni-59	7.8E+08	1.1E+11	1.1E+11
Ni-63	8.4E+10	1.1E+13	1.1E+13
Se-79	1.6E+09	5.5E+03	1.6E+09
Kr-85	2.4E+14	5.5E+08	2.4E+14
Sr-90	2.1E+15	4.2E+09	2.1E+15
Y-90	2.1E+15	4.2E+09	2.1E+15
Zr-93	1.4E+11	1.1E+10	1.5E+11
Nb-93M	1.1E+11	9.3E+09	1.2E+11
Nb-94	1.9E+08	3.2E+10	3.2E+10
Tc-99	9.6E+11	1.7E+08	9.6E+11
Pd-107	1.1E+10	8.3E+04	1.1E+10
Ag-108M	1.3E+09	3.9E+04	1.3E+09
Cd-113M	1.1E+12	1.5E+07	1.1E+12
Sn-121	1.1E+11	4.3E+10	1.5E+11
Sn-121M	1.4E+11	5.6E+10	2.0E+11
Sn-126	3.2E+10	2.3E+05	3.2E+10
Sb-125	4.9E+10	3.6E+09	5.3E+10
Sb-126	4.5E+09	3.2E+04	4.5E+09
Sb-126M	3.2E+10	2.3E+05	3.2E+10
Te-125M	1.2E+10	8.7E+08	1.3E+10
I-129	2.6E+09	1.3E+04	2.6E+09
Cs-134	2.7E+10	1.1E+05	2.7E+10
Cs-135	3.9E+10	1.4E+05	3.9E+10
Cs-137	3.4E+15	1.3E+10	3.4E+15
Ba-137M	3.2E+15	1.2E+10	3.2E+15
Pm-147	1.7E+11	7.6E+05	1.7E+11
Sm-151	1.5E+13	6.3E+07	1.5E+13
Eu-152	3.9E+10	1.5E+05	3.9E+10
Eu-154	4.7E+13	2.3E+08	4.7E+13
Eu-155	1.2E+12	5.9E+06	1.2E+12
Ho-166M	2.4E+09	1.5E+04	2.4E+09
Hf-178M	1.1E+09	2.6E+10	2.7E+10
Th-234	1.1E+10	1.0E+04	1.1E+10
Pa-233	4.0E+10	2.2E+04	4.0E+10
Pa-234M	1.1E+10	1.0E+04	1.1E+10
U-234	9.3E+10	7.0E+04	9.3E+10
U-235	3.2E+08	8.9E+01	3.2E+08
U-235M	1.3E+13	7.8E+07	1.3E+13

Tab. B.10: (Cont.)

Fuel type Burnup Inventory Mass [kg/t_{IHM}] Time [a]	**PWR UO_2** 75 GWd/t_{IHM} Fuel + imp. 1'000.0 40	**PWR UO_2** 75 GWd/t_{IHM} Struct. Mat. 402.8 40	**PWR UO_2** 75 GWd/t_{IHM} Total 1'402.8 [1] 40
Nuclide	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**
U-236	1.6E+10	2.7E+03	1.6E+10
U-237	2.7E+10	1.9E+05	2.7E+10
U-238	1.1E+10	1.0E+04	1.1E+10
Np-237	4.0E+10	2.2E+04	4.0E+10
Np-238	2.2E+09	1.5E+04	2.2E+09
Np-239	3.6E+12	3.7E+07	3.6E+12
Pu-238	4.2E+14	5.1E+08	4.2E+14
Pu-239	1.3E+13	7.8E+07	1.3E+13
Pu-240	3.0E+13	1.4E+08	3.0E+13
Pu-241	1.1E+15	7.7E+09	1.1E+15
Pu-242	2.1E+11	1.9E+06	2.1E+11
Am-241	2.1E+14	1.5E+09	2.1E+14
Am-242	4.5E+11	3.2E+06	4.5E+11
Am-242M	4.5E+11	3.2E+06	4.5E+11
Am-243	3.6E+12	3.7E+07	3.6E+12
Cm-242	3.7E+11	2.6E+06	3.7E+11
Cm-243	9.6E+11	8.5E+06	9.6E+11
Cm-244	2.1E+14	2.4E+09	2.1E+14
Cm-245	1.4E+11	1.7E+06	1.4E+11
Cm-246	5.5E+10	7.4E+05	5.5E+10
Total α **Total βγ** **Total Activity**	**8.9E+14** **1.2E+16** **1.3E+16**	**4.7E+09** **1.2E+13** **1.2E+13**	**8.9E+14** **1.2E+16** **1.3E+16**
Stable Elements			
Element	**[mol/t_{IHM}]**	**[mol/t_{IHM}]**	**[mol/t_{IHM}]**
Ca	1.1E+00	2.3E-01	1.4E+00
Fe	1.1E+00	7.4E+02	7.4E+02
Co	1.6E-02	6.2E-01	6.4E-01
Ni	4.8E-01	2.3E+02	2.3E+02
Se	9.8E-01	2.8E-06	9.8E-01
Sr	5.9E+00	1.2E-03	5.9E+00
Zr	5.5E+01	3.3E+03	3.3E+03
Nb	1.2E-02	3.3E+00	3.3E+00
Pd	1.8E+01	1.2E-04	1.8E+01
Ag	1.1E+00	9.9E-06	1.1E+00
Cd	1.5E+00	1.4E-03	1.5E+00
Sn	6.1E-01	3.8E+01	3.9E+01
Sb	1.7E-01	2.7E-02	2.0E-01
Sm	4.8E+00	2.0E-05	4.8E+00
Eu	1.3E+00	6.3E-06	1.3E+00
Ho	4.2E-03	3.1E-08	4.2E-03
Hf	7.2E-03	1.6E-01	1.7E-01
Pb	5.4E-03	1.9E-01	1.9E-01

[1] The mass of fuel + structural materials is 1'402.8 + 134.0 (mass of oxygen) = 1'536.8 kg.

Tab. B.11: Discharge radioactive (Bq/t_{IHM}) and stable (mol/t_{IHM}) inventories for PWR MOX model fuel assembly for an average burnup of 48 GWd/t_{IHM} (Cut-off 1 GBq)

Fuel type Burnup Inventory Mass [kg/t_{IHM}] Time [a]	**PWR MOX** 48 GWd/t_{IHM} Fuel + imp. 1'000.0 0	**PWR MOX** 48 GWd/t_{IHM} Struct. Mat. 371.2 0	**PWR MOX** 48 GWd/t_{IHM} Total 1'371.2 [1] 0
Nuclide	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**
H-3	4.1E+13	2.0E+10	4.1E+13
C-14	2.7E+10	1.5E+10	4.2E+10
Cl-36	2.4E+08	2.6E+08	5.0E+08
Ar-39	1.1E+10	6.2E+05	1.1E+10
Fe-55	8.0E+11	1.4E+14	1.4E+14
Co-60	3.8E+12	6.9E+13	7.3E+13
Ni-59	2.7E+08	7.5E+10	7.6E+10
Ni-63	3.3E+10	1.0E+13	1.0E+13
Se-79	8.7E+08	2.4E+03	8.7E+08
Kr-85	1.2E+15	3.3E+09	1.2E+15
Sr-90	1.9E+15	4.8E+09	1.9E+15
Y-90	1.9E+15	3.7E+13	1.9E+15
Zr-93	6.8E+10	5.2E+09	7.3E+10
Nb-93M	5.0E+09	3.9E+08	5.4E+09
Nb-94	1.2E+08	2.4E+10	2.4E+10
Tc-99	6.8E+11	1.2E+08	6.8E+11
Pd-107	1.4E+10	3.4E+04	1.4E+10
Cd-113M	8.2E+12	4.0E+07	8.2E+12
Sn-121M	2.8E+11	4.7E+10	3.3E+11
Sn-126	3.4E+10	9.4E+04	3.4E+10
Sb-125	1.3E+15	4.7E+13	1.3E+15
Te-125M	2.8E+14	1.0E+13	2.9E+14
I-129	2.2E+09	5.8E+03	2.2E+09
Cs-134	8.6E+15	2.2E+10	8.6E+15
Cs-135	4.5E+10	9.8E+04	4.5E+10
Cs-137	5.5E+15	1.4E+10	5.5E+15
Ba-137M	5.2E+15	1.3E+10	5.2E+15
Pm-146	4.2E+10	9.8E+04	4.2E+10
Pm-147	7.6E+15	2.3E+10	7.6E+15
Sm-151	2.9E+13	8.1E+07	2.9E+13
Eu-152	1.0E+12	1.8E+06	1.0E+12
Eu-154	8.5E+14	1.8E+09	8.5E+14
Eu-155	4.9E+14	1.1E+09	4.9E+14
Ho-166M	1.9E+09	1.5E+03	1.9E+09
Hf-178M	1.9E+09	4.0E+10	4.2E+10
Tl-204	4.9E+09	2.6E-21	4.9E+09
U-234	3.2E+09	1.5E+03	3.2E+09
U-235	8.7E+07	2.5E+02	8.7E+07
U-236	6.9E+08	2.1E+03	6.9E+08

Tab. B.11: (Cont.)

Fuel type Burnup Inventory Mass [kg/t_{IHM}] Time [a]	**PWR MOX** 48 GWd/t_{IHM} Fuel + imp. 1'000.0 0	**PWR MOX** 48 GWd/t_{IHM} Struct. Mat. 371.2 0	**PWR MOX** 48 GWd/t_{IHM} Total 1'371.2 [1] 0
Nuclide	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**
U-238	1.1E+10	1.0E+04	1.1E+10
Np-237	4.9E+09	6.3E+03	4.9E+09
Pu-238	5.3E+14	1.9E+08	5.3E+14
Pu-239	3.8E+13	1.5E+08	3.8E+13
Pu-240	1.2E+14	1.4E+08	1.2E+14
Pu-241	2.9E+16	6.8E+10	2.9E+16
Pu-242	5.5E+11	6.4E+05	5.5E+11
Am-241	8.1E+13	9.0E+07	8.1E+13
Am-242M	7.0E+12	7.0E+06	7.0E+12
Am-243	7.7E+12	6.7E+06	7.7E+12
Cm-242	1.8E+16	2.0E+10	1.8E+16
Cm-243	1.2E+13	7.4E+06	1.2E+13
Cm-244	1.8E+15	7.5E+08	1.8E+15
Cm-245	4.4E+11	1.2E+05	4.4E+11
Cm-246	7.7E+10	1.3E+04	7.7E+10
Total α **Total βγ** **Total Activity**	**2.1E+16** **2.8E+17** **3.0E+17**	**2.1E+10** **2.2E+15** **2.2E+15**	**2.1E+16** **2.8E+17** **3.0E+17**
Stable Elements			
Element	**[mol/t_{IHM}]**	**[mol/t_{IHM}]**	**[mol/t_{IHM}]**
Ca	1.1E+00	1.9E-01	1.3E+00
Fe	1.1E+00	7.7E+02	7.7E+02
Co	1.7E-02	7.0E-01	7.2E-01
Ni	4.8E-01	2.7E+02	2.7E+02
Se	6.9E-01	1.8E-06	6.9E-01
Sr	2.7E+00	1.1E-03	2.7E+00
Zr	3.0E+01	2.7E+03	2.8E+03
Nb	1.2E-02	4.3E+00	4.3E+00
Pd	2.7E+01	6.4E-05	2.7E+01
Ag	2.3E+00	6.4E-06	2.3E+00
Cd	2.9E+00	1.2E-03	2.9E+00
Sn	8.7E-01	3.2E+01	3.3E+01
Sb	2.8E-01	2.0E-02	3.0E-01
Sm	4.8E+00	1.2E-05	4.8E+00
Eu	1.5E+00	3.7E-06	1.5E+00
Ho	7.8E-03	1.1E-08	7.8E-03
Hf	7.1E-03	1.4E-01	1.5E-01
Pb	5.4E-03	1.6E-01	1.6E-01

[1] The mass of fuel + structural materials is 1'371.2 + 134.0 (mass of oxygen) = 1'505.2 kg.

Tab. B.12: Radioactive (Bq/t_{IHM}) and stable (mol/t_{IHM}) inventories for PWR MOX model fuel assembly for an average burnup of 48 GWd/t_{IHM} after 40 years decay (Cut-off 1 GBq)

Fuel type Burnup Inventory Mass [kg/t_{IHM}] Time [a]	**PWR MOX** 48 GWd/t_{IHM} Fuel + imp. 1'000.0 40	**PWR MOX** 48 GWd/t_{IHM} Struct. Mat. 371.2 40	**PWR MOX** 48 GWd/t_{IHM} Total 1'371.2 [1] 40
Nuclide	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**
H-3	4.3E+12	2.1E+09	4.3E+12
C-14	2.7E+10	1.5E+10	4.2E+10
Cl-36	2.4E+08	2.6E+08	5.0E+08
Ar-39	9.9E+09	5.6E+05	9.9E+09
Fe-55	3.1E+07	5.4E+09	5.4E+09
Co-60	2.0E+10	3.6E+11	3.8E+11
Ni-59	2.7E+08	7.6E+10	7.6E+10
Ni-63	2.5E+10	7.6E+12	7.6E+12
Se-79	8.7E+08	2.4E+03	8.7E+08
Kr-85	9.9E+13	2.5E+08	9.9E+13
Sr-90	7.2E+14	1.8E+09	7.2E+14
Y-90	7.2E+14	1.8E+09	7.2E+14
Zr-93	6.8E+10	5.2E+09	7.3E+10
Nb-93M	5.4E+10	4.5E+09	5.9E+10
Nb-94	1.2E+08	2.4E+10	2.4E+10
Tc-99	6.8E+11	1.2E+08	6.8E+11
Pd-107	1.4E+10	3.4E+04	1.4E+10
Cd-113M	1.2E+12	6.1E+06	1.2E+12
Sn-121	1.3E+11	2.2E+10	1.5E+11
Sn-121M	1.7E+11	2.9E+10	2.0E+11
Sn-126	3.4E+10	9.4E+04	3.4E+10
Sb-125	5.8E+10	2.2E+09	6.0E+10
Sb-126	4.8E+09	1.3E+04	4.8E+09
Sb-126M	3.4E+10	9.4E+04	3.4E+10
Te-125M	1.4E+10	5.3E+08	1.5E+10
I-129	2.2E+09	5.8E+03	2.2E+09
Cs-134	1.2E+10	3.1E+04	1.2E+10
Cs-135	4.5E+10	9.8E+04	4.5E+10
Cs-137	2.2E+15	5.6E+09	2.2E+15
Ba-137M	2.1E+15	5.3E+09	2.1E+15
Pm-147	1.9E+11	5.8E+05	1.9E+11
Sm-151	2.2E+13	6.0E+07	2.2E+13
Eu-152	1.2E+11	2.2E+05	1.2E+11
Eu-154	3.7E+13	7.7E+07	3.7E+13
Eu-155	1.4E+12	3.3E+06	1.4E+12
Ho-166M	1.9E+09	1.5E+03	1.9E+09
Hf-178M	7.8E+08	1.6E+10	1.7E+10
U-234	6.4E+10	3.0E+04	6.4E+10
U-235	8.7E+07	2.5E+02	8.7E+07

Tab. B.12: (Cont.)

Fuel type Burnup Inventory Mass [kg/t_{IHM}] Time [a]	**PWR MOX** 48 GWd/t_{IHM} Fuel + imp. 1'000.0 40	**PWR MOX** 48 GWd/t_{IHM} Struct. Mat. 371.2 40	**PWR MOX** 48 GWd/t_{IHM} Total 1'371.2 [1] 40
Nuclide	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**
U-236	8.3E+08	2.3E+03	8.3E+08
U-238	1.1E+10	1.0E+04	1.1E+10
Np-237	1.3E+10	2.3E+04	1.3E+10
Pu-238	4.6E+14	2.1E+08	4.6E+14
Pu-239	3.8E+13	1.5E+08	3.8E+13
Pu-240	1.2E+14	1.4E+08	1.2E+14
Pu-241	4.2E+15	9.8E+09	4.2E+15
Pu-242	5.5E+11	6.4E+05	5.5E+11
Am-241	8.7E+14	1.9E+09	8.7E+14
Am-242M	5.7E+12	5.7E+06	5.7E+12
Am-243	7.7E+12	6.7E+06	7.7E+12
Cm-242	4.7E+12	4.7E+06	4.7E+12
Cm-243	4.6E+12	2.9E+06	4.6E+12
Cm-244	3.9E+14	1.7E+08	3.9E+14
Cm-245	4.4E+11	1.2E+05	4.4E+11
Cm-246	7.7E+10	1.3E+04	7.7E+10
Total α **Total βγ** **Total Activity**	**1.9E+15** **1.0E+16** **1.2E+16**	**2.6E+09** **8.2E+12** **8.2E+12**	**1.9E+15** **1.0E+16** **1.2E+16**
Stable Elements			
Nuclide	**[mol/t_{IHM}]**	**[mol/t_{IHM}]**	**[mol/t_{IHM}]**
Ca	1.1E+00	1.9E-01	1.3E+00
Fe	1.1E+00	7.7E+02	7.7E+02
Co	1.7E-02	7.0E-01	7.2E-01
Ni	4.8E-01	2.7E+02	2.7E+02
Se	6.9E-01	1.8E-06	6.9E-01
Sr	2.7E+00	1.1E-03	2.7E+00
Zr	3.2E+01	2.7E+03	2.8E+03
Nb	1.2E-02	4.3E+00	4.3E+00
Pd	2.7E+01	6.4E-05	2.7E+01
Ag	2.3E+00	6.4E-06	2.3E+00
Cd	2.9E+00	1.2E-03	2.9E+00
Sn	8.7E-01	3.2E+01	3.3E+01
Sb	2.8E-01	2.0E-02	3.0E-01
Sm	4.8E+00	1.2E-05	4.8E+00
Eu	1.6E+00	3.7E-06	1.6E+00
Ho	7.8E-03	1.1E-08	7.8E-03
Hf	7.1E-03	1.4E-01	1.5E-01
Pb	5.4E-03	1.6E-01	1.6E-01

[1] The mass of fuel + structural materials is 1'371.2 + 134.0 (mass of oxygen) = 1'505.2 kg.

Tab. B.13: Discharge radioactive (Bq/t_{IHM}) and stable (mol/t_{IHM}) inventories for PWR MOX model fuel assembly for an average burnup of 65 GWd/t_{IHM} (Cut-off 1 GBq)

Fuel type Burnup Inventory Mass [kg/t_{IHM}] Time [a]	**PWR MOX** 65 GWd/t_{IHM} Fuel + imp. 1'000.0 0	**PWR MOX** 65 GWd/t_{IHM} Struct. Mat. 402.8 0	**PWR MOX** 65 GWd/t_{IHM} Total 1'402.8 [1] 0
Nuclide	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**
H-3	5.5E+13	3.2E+10	5.5E+13
C-14	3.6E+10	2.2E+10	5.8E+10
Cl-36	3.2E+08	4.0E+08	7.2E+08
Ar-39	1.3E+10	1.0E+06	1.3E+10
Fe-55	1.0E+12	2.1E+14	2.1E+14
Co-60	4.9E+12	1.0E+14	1.1E+14
Ni-59	3.4E+08	9.2E+10	9.2E+10
Ni-63	4.4E+10	1.3E+13	1.3E+13
Se-79	1.2E+09	3.8E+03	1.2E+09
Kr-85	1.6E+15	5.3E+09	1.6E+15
Sr-90	2.5E+15	7.7E+09	2.5E+15
Y-90	2.5E+15	5.2E+13	2.6E+15
Zr-93	9.0E+10	8.1E+09	9.8E+10
Nb-93M	7.7E+09	6.9E+08	8.4E+09
Nb-94	1.6E+08	2.8E+10	2.8E+10
Tc-99	8.9E+11	1.5E+08	8.9E+11
Pd-107	1.8E+10	5.6E+04	1.8E+10
Ag-108M	1.0E+09	2.3E+04	1.0E+09
Cd-113M	1.2E+13	6.6E+07	1.2E+13
Sn-121M	3.6E+11	7.4E+10	4.3E+11
Sn-126	4.7E+10	1.6E+05	4.7E+10
Sb-125	1.6E+15	7.0E+13	1.7E+15
Te-125M	3.7E+14	1.6E+13	3.8E+14
I-129	2.9E+09	9.4E+03	2.9E+09
Cs-134	1.4E+16	4.3E+10	1.4E+16
Cs-135	5.7E+10	1.5E+05	5.7E+10
Cs-137	7.4E+15	2.3E+10	7.4E+15
Ba-137M	7.0E+15	2.2E+10	7.0E+15
Pm-145	1.1E+09	3.6E-01	1.1E+09
Pm-146	4.8E+10	1.4E+05	4.8E+10
Pm-147	8.6E+15	3.2E+10	8.6E+15
Sm-151	3.3E+13	1.1E+08	3.3E+13
Eu-152	1.0E+12	2.3E+06	1.0E+12
Eu-154	1.3E+15	3.5E+09	1.3E+15
Eu-155	7.0E+14	1.9E+09	7.0E+14
Ho-166M	3.3E+09	3.8E+03	3.3E+09
Hf-178M	2.4E+09	5.2E+10	5.5E+10
Tl-204	6.2E+09	6.9E-20	6.2E+09
U-234	6.6E+09	2.3E+03	6.6E+09
U-235	6.6E+07	2.1E+02	6.6E+07
U-236	7.9E+08	2.6E+03	7.9E+08
U-238	1.0E+10	1.0E+04	1.0E+10

Tab. B.13: (Cont.)

Fuel type Burnup Inventory Mass [kg/t_{IHM}] Time [a]	**PWR MOX** 65 GWd/t_{IHM} Fuel + imp. 1'000.0 0	**PWR MOX** 65 GWd/t_{IHM} Struct. Mat. 402.8 0	**PWR MOX** 65 GWd/t_{IHM} Total 1'402.8 [1] 0
Nuclide	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**
Np-237	6.1E+09	8.9E+03	6.1E+09
Pu-238	9.4E+14	4.1E+08	9.4E+14
Pu-239	3.4E+13	1.6E+08	3.4E+13
Pu-240	1.3E+14	1.6E+08	1.3E+14
Pu-241	3.1E+16	9.4E+10	3.1E+16
Pu-242	8.7E+11	1.2E+06	8.7E+11
Am-241	9.0E+13	1.4E+08	9.0E+13
Am-242M	8.0E+12	1.1E+07	8.0E+12
Am-243	1.3E+13	1.6E+07	1.3E+13
Cm-242	2.6E+16	3.8E+10	2.6E+16
Cm-243	2.0E+13	1.8E+07	2.0E+13
Cm-244	3.9E+15	2.4E+09	3.9E+15
Cm-245	1.1E+12	4.0E+05	1.1E+12
Cm-246	2.5E+11	5.6E+04	2.5E+11
Total α **Total βγ** **Total Activity**	**3.1E+16** **2.8E+17** **3.7E+17**	**4.1E+10** **2.4E+15** **3.1E+15**	**3.1E+16** **2.8E+17** **3.7E+17**
Stable Elements			
Element	**[mol/t_{IHM}]**	**[mol/t_{IHM}]**	**[mol/t_{IHM}]**
Ca	1.1E+00	2.3E-01	1.4E+00
Fe	1.1E+00	7.4E+02	7.4E+02
Co	1.7E-02	6.5E-01	6.6E-01
Ni	4.8E-01	2.3E+02	2.3E+02
Se	9.0E-01	2.9E-06	9.0E-01
Sr	3.6E+00	1.7E-03	3.6E+00
Zr	4.0E+01	3.3E+03	3.3E+03
Nb	1.2E-02	3.3E+00	3.3E+00
Pd	3.8E+01	1.1E-04	3.8E+01
Ag	3.0E+00	1.0E-05	3.0E+00
Cd	4.6E+00	1.4E-03	4.6E+00
Sn	1.2E+00	3.8E+01	4.0E+01
Sb	3.8E-01	3.1E-02	4.1E-01
Sm	6.4E+00	2.0E-05	6.4E+00
Eu	2.0E+00	6.2E-06	2.0E+00
Ho	1.1E-02	2.2E-08	1.1E-02
Hf	7.2E-03	1.6E-01	1.7E-01
Pb	5.4E-03	1.9E-01	1.9E-01

[1] The mass of fuel + structural materials is 1'402.8 + 134.0 (mass of oxygen) = 1'536.8 kg.

Tab. B.14: Radioactive (Bq/t_{IHM}) and stable (mol/t_{IHM}) inventories for PWR MOX model fuel assembly for an average burnup of 65 GWd/t_{IHM} after 40 years decay (Cut-off 1 GBq)

Fuel type Burnup Inventory Mass [kg/t_{IHM}] Time [a]	**PWR MOX** 65 GWd/t_{IHM} Fuel + imp. 1'000.0 40	**PWR MOX** 65 GWd/t_{IHM} Struct. Mat. 402.8 40	**PWR MOX** 65 GWd/t_{IHM} Total 1'402.8 [1] 40
Nuclide	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**
H-3	5.8E+12	3.4E+09	5.8E+12
C-14	3.6E+10	2.2E+10	5.8E+10
Cl-36	3.2E+08	4.0E+08	7.2E+08
Ar-39	1.2E+10	9.0E+05	1.2E+10
Fe-55	3.9E+07	8.2E+09	8.2E+09
Co-60	2.5E+10	5.2E+11	5.5E+11
Ni-59	3.4E+08	9.2E+10	9.2E+10
Ni-63	3.3E+10	9.9E+12	9.9E+12
Se-79	1.2E+09	3.8E+03	1.2E+09
Kr-85	1.2E+14	4.0E+08	1.2E+14
Sr-90	9.5E+14	2.9E+09	9.5E+14
Y-90	9.5E+14	2.9E+09	9.5E+14
Zr-93	9.0E+10	8.1E+09	9.8E+10
Nb-93M	7.2E+10	6.9E+09	7.9E+10
Nb-94	1.6E+08	2.8E+10	2.8E+10
Tc-99	8.9E+11	1.5E+08	8.9E+11
Pd-107	1.8E+10	5.6E+04	1.8E+10
Cd-113M	1.8E+12	9.9E+06	1.8E+12
Sn-121	1.7E+11	3.5E+10	2.1E+11
Sn-121M	2.2E+11	4.5E+10	2.7E+11
Sn-126	4.7E+10	1.6E+05	4.7E+10
Sb-126M	4.7E+10	1.6E+05	4.7E+10
Sb-125	7.2E+10	3.1E+09	7.5E+10
Te-125M	1.8E+10	7.7E+08	1.9E+10
Sb-126	6.6E+09	2.2E+04	6.6E+09
I-129	2.9E+09	9.4E+03	2.9E+09
Cs-134	2.0E+10	6.1E+04	2.0E+10
Cs-135	5.7E+10	1.5E+05	5.7E+10
Cs-137	3.0E+15	9.2E+09	3.0E+15
Ba-137M	2.8E+15	8.7E+09	2.8E+15
Pm-147	2.2E+11	8.1E+05	2.2E+11
Sm-151	2.4E+13	8.2E+07	2.4E+13
Eu-152	1.2E+11	2.9E+05	1.2E+11
Eu-154	5.6E+13	1.5E+08	5.6E+13
Eu-155	2.1E+12	5.6E+06	2.1E+12
Ho-166M	3.3E+09	3.8E+03	3.3E+09
Hf-178M	9.8E+08	2.1E+10	2.2E+10
Th-234	1.1E+10	1.0E+04	1.1E+10
Pa-233	1.4E+10	3.3E+04	1.4E+10
Pa-234M	1.1E+10	1.0E+04	1.1E+10
U-234	1.1E+11	6.1E+04	1.1E+11
U-235	6.6E+07	2.1E+02	6.6E+07
U-235M	3.4E+13	1.6E+08	3.4E+13

Tab. B.14: (Cont.)

Fuel type Burnup Inventory Mass [kg/t_{IHM}] Time [a]	**PWR MOX** 65 GWd/t_{IHM} Fuel + imp. 1'000.0 40	**PWR MOX** 65 GWd/t_{IHM} Struct. Mat. 402.8 40	**PWR MOX** 65 GWd/t_{IHM} Total 1'402.8 [1] 40
Nuclide	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**	**[Bq/t_{IHM}]**
U-236	9.4E+08	2.8E+03	9.4E+08
U-237	1.1E+11	3.3E+05	1.1E+11
U-238	1.1E+10	1.0E+04	1.1E+10
Np-237	1.4E+10	3.3E+04	1.4E+10
Np-238	3.2E+10	4.3E+04	3.2E+10
Np-239	1.3E+13	1.6E+07	1.3E+13
Pu-238	7.8E+14	4.4E+08	7.8E+14
Pu-239	3.4E+13	1.6E+08	3.4E+13
Pu-240	1.3E+14	1.6E+08	1.3E+14
Pu-241	4.5E+15	1.4E+10	4.5E+15
Pu-242	8.7E+11	1.2E+06	8.7E+11
Am-241	9.3E+14	2.7E+09	9.3E+14
Am-242	6.5E+12	9.0E+06	6.5E+12
Am-242M	6.6E+12	9.0E+06	6.6E+12
Am-243	1.3E+13	1.6E+07	1.3E+13
Cm-242	5.4E+12	7.5E+06	5.4E+12
Cm-243	7.7E+12	6.9E+06	7.7E+12
Cm-244	8.4E+14	5.2E+08	8.4E+14
Cm-245	1.1E+12	4.0E+05	1.1E+12
Cm-246	2.5E+11	5.6E+04	2.5E+11
Total α **Total βγ** **Total Activity**	**2.8E+15** **1.2E+16** **1.5E+16**	**4.0E+09** **1.1E+13** **1.1E+13**	**2.8E+15** **1.2E+16** **1.5E+16**
Stable Elements			
Element	**[mol/t_{IHM}]**	**[mol/t_{IHM}]**	**[mol/t_{IHM}]**
Ca	1.1E+00	2.3E-01	1.4E+00
Fe	1.1E+00	7.4E+02	7.4E+02
Co	1.7E-02	6.5E-01	6.6E-01
Ni	4.8E-01	2.3E+02	2.3E+02
Se	9.0E-01	2.9E-06	9.0E-01
Sr	3.6E+00	1.7E-03	3.6E+00
Zr	4.7E+01	3.3E+03	3.3E+03
Nb	1.2E-02	3.3E+00	3.3E+00
Pd	3.8E+01	1.1E-04	3.8E+01
Ag	3.0E+00	1.0E-05	3.0E+00
Cd	4.6E+00	1.4E-03	4.6E+00
Sn	1.2E+00	3.8E+01	4.0E+01
Sb	3.8E-01	3.1E-02	4.1E-01
Sm	6.4E+00	2.0E-05	6.4E+00
Eu	2.0E+00	6.2E-06	2.0E+00
Ho	1.1E-02	2.2E-08	1.1E-02
Hf	7.2E-03	1.6E-01	1.7E-01
Pb	5.4E-03	1.9E-01	1.9E-01

[1] The mass of fuel + structural materials is 1'402.8 + 134.0 (mass of oxygen) = 1'536.8 kg.

Appendix C

Inventory data for spent fuel waste sorts

List of Tables

C.1 Inventory data for BE-1

Tab. C.1-1: Material inventory of BE-1

Occurrence	Material	Mass [kg/Canister]
Raw waste	Nickel alloys	8.47E+00
Raw waste	Steel	9.27E+01
Raw waste	Uranium Oxide	1.81E+03
Raw waste	Zircaloy	6.95E+02
Total Raw waste		**2.60E+03**

It should be noted that the total canister mass for BE-1 is currently estimated to be around 28'500 kg, for the purposes of repository handling calculations (NAGRA 2002b). The detailed conceptual design of the actual canisters, currently envisaged for spent fuel disposal studies, will be given in a later report (JOHNSON & KING 2003). In this report the assumed masses of construction materials will be given. Therefore, this table only gives the raw waste (fuel assemblies) total material content of the disposal canister.

Tab. C.1-2: Radiological characteristics of BE-1 at 40 years after spent fuel reactor discharge

Parameter	Average
0 m γ dose rate [mSv/h]	5
0 m *n* dose rate [mSv/h]	4
Heat output [W]	1.5E+03

Tab. C.1-3: Radionuclide inventory of BE-1 at 40 years after spent fuel reactor discharge

Nuclide	Fuel [Bq/Canister]	Structural materials [Bq/Canister]	Total [Bq/Canister]
H-3	7.0E+12	5.4E+09	7.0E+12
Be-10	1.8E+07	2.9E+03	1.8E+07
C-14	6.7E+10	5.7E+10	1.2E+11
Cl-36	8.6E+08	1.4E+09	2.3E+09
Ar-39	1.2E+10	3.5E+06	1.2E+10
Ca-41	2.2E+08	7.0E+07	2.9E+08
Fe-55	9.9E+07	1.1E+10	1.1E+10
Co-60	4.9E+10	7.8E+11	8.3E+11
Ni-59	8.9E+08	1.3E+11	1.3E+11

Tab. C.1-3: (Cont.)

Nuclide	**Fuel [Bq/Canister]**	**Structural materials [Bq/Canister]**	**Total [Bq/Canister]**
Ni-63	9.2E+10	1.3E+13	1.3E+13
Se-79	1.6E+09	8.0E+03	1.6E+09
Kr-81	2.7E+06	-	2.7E+06
Kr-85	2.7E+14	8.4E+08	2.7E+14
Rb-87	1.6E+06	4.8E+00	1.6E+06
Sr-90	2.2E+15	6.0E+09	2.2E+15
Y-90	2.2E+15	6.0E+09	2.2E+15
Mo-93	8.3E+07	3.3E+08	4.2E+08
Zr-93	1.5E+11	1.8E+10	1.7E+11
Nb-93M	1.2E+11	1.4E+10	1.3E+11
Nb-94	2.1E+08	1.3E+10	1.3E+10
Tc-99	1.1E+12	6.7E+07	1.1E+12
Pd-107	1.0E+10	1.2E+05	1.0E+10
Ag-108	1.3E+08	-	1.3E+08
Ag-108M	1.4E+09	4.3E+04	1.4E+09
Cd-113M	1.0E+12	2.2E+07	1.0E+12
Sn-121	1.1E+11	-	1.1E+11
Sn-121M	1.5E+11	9.1E+10	2.4E+11
Sn-126	3.0E+10	3.2E+05	3.0E+10
Sb-125	5.4E+10	6.7E+09	6.1E+10
Te-125M	1.3E+10	1.6E+09	1.5E+10
Sb-126	4.3E+09	4.5E+04	4.3E+09
Sb-126M	3.0E+10	-	3.0E+10
I-129	2.7E+09	1.9E+04	2.7E+09
Ba-133	1.4E+08	1.2E+01	1.4E+08
Cs-134	2.2E+10	1.3E+05	2.2E+10
Cs-135	3.7E+10	1.8E+05	3.7E+10
Cs-137	3.5E+15	1.9E+10	3.5E+15
Ba-137M	3.3E+15	1.9E+10	3.3E+15
Pm-145	2.6E+08	-	2.6E+08
Pm-146	2.1E+08	1.0E+03	2.1E+08
Pm-147	3.0E+11	1.9E+06	3.0E+11
Sm-146	5.4E+02	-	5.4E+02
Sm-147	3.0E+05	1.8E+00	3.0E+05
Sm-151	1.6E+13	1.1E+08	1.6E+13
Eu-150	1.4E+06	4.8E+00	1.4E+06
Eu-152	4.3E+10	2.6E+05	4.3E+10
Eu-154	4.3E+13	2.9E+08	4.3E+13
Eu-155	1.2E+12	8.9E+06	1.2E+12
Tb-157	1.8E+07	1.0E+01	1.8E+07
Tb-158	1.1E+08	1.3E+03	1.1E+08
Ho-166M	2.1E+09	1.0E+04	2.1E+09
Hf-178M	1.4E+09	4.0E+10	4.1E+10
Pt-193	4.5E+08	7.2E+02	4.5E+08
Tl-204	1.2E+07	-	1.2E+07
Tl-207	3.3E+05	-	3.3E+05
Tl-208	6.7E+07	2.4E+01	6.7E+07
Tl-209	3.0E+02	-	3.0E+02
Tl-210	6.1E+01	-	6.1E+01

Tab. C.1-3: (Cont.)

Nuclide	Fuel [Bq/Canister]	Structural materials [Bq/Canister]	Total [Bq/Canister]
Pb-209	1.4E+04	-	1.4E+04
Pb-211	3.3E+05	-	3.3E+05
Pb-212	1.9E+08	6.8E+01	1.9E+08
Pb-214	2.9E+05	-	2.9E+05
Pb-210	9.4E+04	-	9.4E+04
Bi-210	9.4E+04	-	9.4E+04
Bi-211	3.3E+05	-	3.3E+05
Bi-212	1.9E+08	6.8E+01	1.9E+08
Bi-213	1.4E+04	-	1.4E+04
Bi-214	2.9E+05	-	2.9E+05
Po-210	9.1E+04	-	9.1E+04
Po-211	9.2E+02	-	9.2E+02
Po-212	1.2E+08	4.3E+01	1.2E+08
Po-213	1.4E+04	-	1.4E+04
Po-214	2.9E+05	-	2.9E+05
Po-215	3.3E+05	-	3.3E+05
Po-216	1.9E+08	6.8E+01	1.9E+08
Po-218	2.9E+05	-	2.9E+05
At-217	1.4E+04	-	1.4E+04
At-218	5.7E+01	-	5.7E+01
Rn-219	3.3E+05	-	3.3E+05
Rn-220	1.9E+08	6.8E+01	1.9E+08
Rn-222	2.9E+05	-	2.9E+05
Fr-221	1.4E+04	-	1.4E+04
Fr-223	4.6E+03	-	4.6E+03
Ra-223	3.3E+05	-	3.3E+05
Ra-224	1.9E+08	6.8E+01	1.9E+08
Ra-225	1.4E+04	-	1.4E+04
Ra-226	2.9E+05	-	2.9E+05
Ra-228	3.2E+01	-	3.2E+01
Ac-225	1.4E+04	-	1.4E+04
Ac-227	3.3E+05	-	3.3E+05
Ac-228	3.2E+01	-	3.2E+01
Th-227	3.3E+05	-	3.3E+05
Th-228	1.9E+08	6.8E+01	1.9E+08
Th-229	1.4E+04	-	1.4E+04
Th-230	3.2E+07	1.6E+01	3.2E+07
Th-231	9.1E+08	2.4E+02	9.1E+08
Th-232	4.1E+01	-	4.1E+01
Th-234	1.9E+10	2.2E+04	1.9E+10
Pa-231	7.6E+05	-	7.6E+05
Pa-233	3.2E+10	4.6E+04	3.2E+10
Pa-234	6.4E+07	-	6.4E+07
Pa-234M	1.9E+10	2.2E+04	1.9E+10
U-232	1.8E+08	6.5E+01	1.8E+08
U-233	6.7E+06	4.6E+00	6.7E+06
U-234	9.7E+10	8.3E+04	9.7E+10
U-235	9.1E+08	-	9.1E+08
U-235M	2.2E+13	2.2E+08	2.2E+13

Tab. C.1-3: (Cont.)

Nuclide	Fuel [Bq/Canister]	Structural materials [Bq/Canister]	Total [Bq/Canister]
U-236	2.1E+10	6.1E+03	2.1E+10
U-237	3.7E+10	4.6E+05	3.7E+10
U-238	1.9E+10	2.2E+04	1.9E+10
Np-237	3.2E+10	4.6E+04	3.2E+10
Np-238	2.7E+09	3.5E+04	2.7E+09
Np-239	2.2E+12	3.8E+07	2.2E+12
Pu-238	2.6E+14	6.0E+08	2.6E+14
Pu-239	2.2E+13	2.2E+08	2.2E+13
Pu-240	4.0E+13	3.0E+08	4.0E+13
Pu-241	1.5E+15	1.9E+10	1.5E+15
Pu-242	1.9E+11	2.9E+06	1.9E+11
Am-241	3.0E+14	3.7E+09	3.0E+14
Am-242	5.7E+11	-	5.7E+11
Am-242M	5.7E+11	7.1E+06	5.7E+11
Am-243	2.2E+12	3.8E+07	2.2E+12
Cm-242	4.8E+11	5.9E+06	4.8E+11
Cm-243	6.7E+11	1.0E+07	6.7E+11
Cm-244	8.0E+13	1.4E+09	8.0E+13
Cm-245	5.6E+10	1.0E+06	5.6E+10
Cm-246	1.2E+10	2.6E+05	1.2E+10
Cf-250	2.2E+06	6.1E+01	2.2E+06
Total α	**6.9E+14**	**6.3E+09**	**6.9E+14**
Total βγ	**1.3E+16**	**1.5E+13**	**1.3E+16**
Total Activity	**1.4E+16**	**1.5E+13**	**1.4E+16**
Stable Elements			
Element	**[Mol/Canister]**	**[Mol/Canister]**	**[Mol/Canister]**
Ca	1.8E+00	5.1E-01	2.3E+00
Fe	1.8E+00	1.2E+03	1.2E+03
Co	2.6E-02	9.6E-01	9.8E-01
Ni	7.6E-01	2.6E+02	2.6E+02
Se	1.4E+00	5.9E-06	1.4E+00
Sr	8.3E+00	2.5E-03	8.3E+00
Zr	7.5E+01	7.2E+03	7.3E+03
Nb	1.9E-02	1.4E+00	1.4E+00
Pd	2.4E+01	2.4E-04	2.4E+01
Ag	1.6E+00	2.1E-05	1.6E+00
Cd	2.1E+00	3.1E-03	2.1E+00
Sn	8.6E-01	8.5E+01	8.6E+01
Sb	2.4E-01	6.3E-02	3.0E-01
Sm	6.8E+00	4.3E-05	6.8E+00
Eu	1.8E+00	1.3E-05	1.8E+00
Ho	5.9E-03	6.1E-08	5.9E-03
Hf	1.1E-02	3.6E-01	3.7E-01
Pb	8.6E-03	4.3E-01	4.3E-01

Tab. C.1-4: Decay of heat output from 40 years after spent fuel reactor discharge for BE-1 canisters

Years Inventory Activity	**0 Average [Bq/Can.]**	**3 Average [Bq/Can.]**	**10 Average [Bq/Can.]**	**30 Average [Bq/Can.]**	**100 Average [Bq/Can.]**	**300 Average [Bq/Can.]**	**1'000 Average [Bq/Can.]**	**3'000 Average [Bq/Can.]**
Total α	6.9E+14	6.8E+14	6.6E+14	6.1E+14	4.8E+14	3.0E+14	1.3E+14	5.5E+13
Total βγ	1.3E+16	1.2E+16	1.0E+16	6.1E+15	1.2E+15	3.8E+13	2.6E+13	2.4E+13
Total Activity	1.4E+16	1.3E+16	1.1E+16	6.7E+15	1.6E+15	3.4E+14	1.6E+14	7.9E+13
Heat output	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**
Total Heat	1.5E+03	1.4E+03	1.3E+03	9.8E+02	5.1E+02	2.7E+02	1.2E+02	4.7E+01
Years Inventory Activity		**1.0E+04 Average [Bq/Can.]**	**3.0E+04 Average [Bq/Can.]**	**1.0E+05 Average [Bq/Can.]**	**3.0E+05 Average [Bq/Can.]**	**1.0E+06 Average [Bq/Can.]**	**3.0E+06 Average [Bq/Can.]**	**1.0E+07 Average [Bq/Can.]**
Total α		3.3E+13	1.2E+13	2.5E+12	1.4E+12	8.6E+11	4.8E+11	2.2E+11
Total βγ		2.0E+13	1.2E+13	3.1E+12	1.5E+12	7.6E+11	3.8E+11	1.5E+11
Total Activity		5.2E+13	2.4E+13	5.7E+12	2.9E+12	1.6E+12	8.7E+11	3.7E+11
Heat output		**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**
Total Heat		2.8E+01	1.1E+01	2.3E+00	1.4E+00	8.6E-01	4.8E-01	2.1E-01

C.2 Inventory data for BE-2

Tab. C.2-1: Material inventory of BE-2

Occurrence	Material	Mass [kg/Canister]
Raw waste	Al_2O_3	1.23E+00
Raw waste	Inconel	2.15E+01
Raw waste	Mixed Oxide (MOX)	3.65E+02
Raw waste	Steel	8.90E+01
Raw waste	Uranium Oxide	1.33E+03
Raw waste	Zircaloy	4.52E+02
Total Raw waste		**2.26E+03**

It should be noted that the total canister mass for BE-2 is currently estimated to be around 24'000 kg, for the purposes of repository handling calculations (NAGRA 2002a). The detailed conceptual design of the actual canisters, currently envisaged for spent fuel disposal studies, will be given in a later report (JOHNSON & KING 2003). In this report the assumed masses of construction materials will be given. Therefore, this table only gives the raw waste (fuel assemblies) total material content of the disposal canister.

Tab. C.2-2: Radiological characteristics of BE-2 at 40 years after spent fuel reactor discharge

Parameter	Average
0 m γ dose rate [mSv/h]	3
0 m *n* dose rate [mSv/h]	30
Heat output [W]	1.8E+03

Tab. C.2-3: Radionuclide inventory of BE-2 at 40 years after spent fuel reactor discharge

Nuclide	Fuel [Bq/Canister]	Structural materials [Bq/Canister]	Total [Bq/Canister]
H-3	6.6E+12	3.0E+09	6.6E+12
Be-10	1.7E+07	4.5E+03	1.7E+07
C-14	5.6E+10	3.2E+10	8.8E+10
Cl-36	6.9E+08	7.0E+08	1.4E+09
Ar-39	1.1E+10	1.8E+06	1.1E+10
Ca-41	1.8E+08	3.4E+07	2.2E+08
Fe-55	7.8E+07	9.0E+09	9.1E+09
Co-60	4.1E+10	6.3E+11	6.7E+11
Ni-59	7.2E+08	1.2E+11	1.2E+11
Ni-63	7.3E+10	1.2E+13	1.2E+13
Se-79	1.4E+09	4.4E+03	1.4E+09
Kr-81	2.2E+06	-	2.2E+06
Kr-85	2.3E+14	4.7E+08	2.3E+14
Rb-87	1.5E+06	2.7E+00	1.5E+06
Sr-90	1.9E+15	3.4E+09	1.9E+15
Y-90	1.9E+15	3.4E+09	1.9E+15
Mo-93	7.3E+07	7.1E+08	7.8E+08
Zr-93	1.3E+11	9.4E+09	1.4E+11
Nb-93M	1.0E+11	8.0E+09	1.1E+11
Nb-94	1.8E+08	3.6E+10	3.6E+10
Tc-99	1.0E+12	1.8E+08	1.0E+12
Pd-107	1.1E+10	6.5E+04	1.1E+10
Ag-108	1.1E+08	-	1.1E+08
Ag-108M	1.2E+09	2.2E+04	1.2E+09
Cd-113M	1.1E+12	1.2E+07	1.1E+12
Sn-121	1.2E+11	-	1.2E+11
Sn-121M	1.5E+11	4.8E+10	2.0E+11
Sn-126	3.3E+10	1.8E+05	3.3E+10
Sb-125	5.6E+10	3.5E+09	6.0E+10
Sb-126	4.7E+09	2.5E+04	4.7E+09
Sb-126M	3.3E+10	-	3.3E+10
Te-125M	1.4E+10	8.5E+08	1.5E+10
I-129	2.7E+09	1.1E+04	2.7E+09
Ba-133	1.0E+08	6.6E+00	1.0E+08
Cs-134	2.0E+10	6.5E+04	2.0E+10
Cs-135	4.4E+10	1.2E+05	4.4E+10
Cs-137	3.3E+15	1.1E+10	3.3E+15
Ba-137M	3.1E+15	1.0E+10	3.1E+15
Pm-145	2.4E+08	-	2.4E+08
Pm-146	2.4E+08	6.9E+02	2.4E+08
Pm-147	2.8E+11	1.0E+06	2.8E+11
Sm-146	6.4E+02	-	6.4E+02
Sm-147	2.8E+05	-	2.8E+05
Sm-151	1.9E+13	7.0E+07	1.9E+13
Eu-150	3.0E+06	4.5E+00	3.0E+06
Eu-152	7.9E+10	2.1E+05	7.9E+10
Eu-154	4.2E+13	1.5E+08	4.2E+13
Eu-155	1.3E+12	4.9E+06	1.3E+12
Tb-157	1.3E+07	5.5E+00	1.3E+07

Tab. C.2-3: (Cont.)

Nuclide	Fuel [Bq/Canister]	Structural materials [Bq/Canister]	Total [Bq/Canister]
Tb-158	1.5E+08	7.0E+02	1.5E+08
Ho-166M	2.0E+09	5.2E+03	2.0E+09
Hf-178M	1.2E+09	2.5E+10	2.6E+10
Pt-193	3.4E+08	4.0E+02	3.4E+08
Tl-204	9.5E+06	-	9.5E+06
Tl-207	3.0E+05	-	3.0E+05
Tl-208	6.0E+07	1.8E+01	6.0E+07
Tl-209	2.5E+02	-	2.5E+02
Tl-210	5.5E+01	-	5.5E+01
Pb-209	1.1E+04	-	1.1E+04
Pb-210	8.3E+04	-	8.3E+04
Pb-211	3.0E+05	-	3.0E+05
Pb-212	1.7E+08	5.1E+01	1.7E+08
Pb-214	2.6E+05	-	2.6E+05
Bi-210	8.3E+04	-	8.3E+04
Bi-211	3.0E+05	-	3.0E+05
Bi-212	1.7E+08	5.1E+01	1.7E+08
Bi-213	1.1E+04	-	1.1E+04
Bi-214	2.6E+05	-	2.6E+05
Po-210	8.0E+04	-	8.0E+04
Po-211	8.6E+02	-	8.6E+02
Po-212	1.1E+08	3.2E+01	1.1E+08
Po-213	1.1E+04	-	1.1E+04
Po-214	2.6E+05	-	2.6E+05
Po-215	3.0E+05	-	3.0E+05
Po-216	1.7E+08	5.1E+01	1.7E+08
Po-218	2.6E+05	-	2.6E+05
At-217	1.1E+04	-	1.1E+04
At-218	5.2E+01	-	5.2E+01
Rn-219	3.0E+05	-	3.0E+05
Rn-220	1.7E+08	5.1E+01	1.7E+08
Rn-222	2.6E+05	-	2.6E+05
Fr-221	1.1E+04	-	1.1E+04
Fr-223	4.3E+03	-	4.3E+03
Ra-223	3.0E+05	-	3.0E+05
Ra-224	1.7E+08	5.1E+01	1.7E+08
Ra-225	1.1E+04	-	1.1E+04
Ra-226	2.6E+05	-	2.6E+05
Ra-228	2.4E+01	-	2.4E+01
Ac-225	1.1E+04	-	1.1E+04
Ac-227	3.0E+05	-	3.0E+05
Ac-228	2.4E+01	-	2.4E+01
Th-227	3.0E+05	-	3.0E+05
Th-228	1.7E+08	5.1E+01	1.7E+08
Th-229	1.1E+04	-	1.1E+04
Th-230	3.0E+07	1.0E+01	3.0E+07
Th-231	8.4E+08	2.3E+02	8.4E+08
Th-232	3.1E+01	-	3.1E+01
Th-234	1.6E+10	1.5E+04	1.6E+10

Tab. C.2-3: (Cont.)

Nuclide	**Fuel [Bq/Canister]**	**Structural materials [Bq/Canister]**	**Total [Bq/Canister]**
Pa-231	7.0E+05	-	7.0E+05
Pa-233	2.8E+10	2.9E+04	2.8E+10
Pa-234	5.4E+07	-	5.4E+07
Pa-234M	1.6E+10	1.5E+04	1.6E+10
U-232	1.7E+08	4.9E+01	1.7E+08
U-233	5.4E+06	3.0E+00	5.4E+06
U-234	9.5E+10	5.1E+04	9.5E+10
U-235	8.4E+08	-	8.4E+08
U-235M	2.9E+13	1.5E+08	2.9E+13
U-236	1.6E+10	3.7E+03	1.6E+10
U-237	5.9E+10	2.9E+05	5.9E+10
U-238	1.6E+10	1.5E+04	1.6E+10
Np-237	2.8E+10	2.9E+04	2.8E+10
Np-238	1.1E+10	2.7E+04	1.1E+10
Pu-238	3.2E+14	3.6E+08	3.2E+14
Np-239	4.0E+12	2.0E+07	4.0E+12
Pu-239	2.9E+13	1.5E+08	2.9E+13
Pu-240	6.6E+13	2.0E+08	6.6E+13
Pu-241	2.5E+15	1.2E+10	2.5E+15
Pu-242	3.1E+11	1.5E+06	3.1E+11
Am-241	4.9E+14	2.4E+09	4.9E+14
Am-243	4.0E+12	2.0E+07	4.0E+12
Am-242	2.3E+12	-	2.3E+12
Am-242M	2.3E+12	5.6E+06	2.3E+12
Cm-242	1.9E+12	4.6E+06	1.9E+12
Cm-243	2.0E+12	5.5E+06	2.0E+12
Cm-244	1.8E+14	7.1E+08	1.8E+14
Cm-245	1.8E+11	4.8E+05	1.8E+11
Cm-246	3.3E+10	1.2E+05	3.3E+10
Cf-250	9.9E+06	3.1E+01	9.9E+06
Total α	**1.1E+15**	**3.8E+09**	**1.1E+15**
Total βγ	**1.3E+16**	**1.3E+13**	**1.3E+16**
Total Activity	**1.4E+16**	**1.3E+13**	**1.4E+16**
Stable Elements			
Element	**[Mol/Canister]**	**[Mol/Canister]**	**[Mol/Canister]**
Ca	1.6E+00	3.1E-01	1.9E+00
Fe	1.6E+00	1.2E+03	1.2E+03
Co	2.5E-02	1.0E+00	1.1E+00
Ni	7.2E-01	4.0E+02	4.0E+02
Se	1.3E+00	3.3E-06	1.3E+00
Sr	7.1E+00	1.5E-03	7.1E+00
Zr	6.7E+01	4.4E+03	4.5E+03
Nb	1.8E-02	6.1E+00	6.2E+00
Pd	2.6E+01	1.3E-04	2.6E+01
Ag	1.9E+00	1.2E-05	1.9E+00
Cd	2.5E+00	1.8E-03	2.5E+00
Sn	9.0E-01	5.2E+01	5.3E+01
Sb	2.7E-01	3.3E-02	3.0E-01

Tab. C.2-3: (Cont.)

Stable Elements			
Element	**[Mol/Canister]**	**[Mol/Canister]**	**[Mol/Canister]**
Sm	6.5E+00	2.4E-05	6.5E+00
Eu	1.8E+00	7.3E-06	1.8E+00
Ho	6.9E-03	3.2E-08	6.9E-03
Hf	1.1E-02	2.3E-01	2.4E-01
Pb	8.1E-03	2.6E-01	2.6E-01

Tab. C.2-4.1: Decay of heat output from 40 years after spent fuel reactor discharge for BE-2 canisters

Years Inventory Activity	**0 Average [Bq/Can.]**	**3 Average [Bq/Can.]**	**10 Average [Bq/Can.]**	**30 Average [Bq/Can.]**	**100 Average [Bq/Can.]**	**300 Average [Bq/Can.]**	**1'000 Average [Bq/Can.]**	**3'000 Average [Bq/Can.]**
Total α Total βγ Total Activity	1.1E+15 1.3E+16 1.4E+16	1.1E+15 1.2E+16 1.3E+16	1.0E+15 9.8E+15 1.1E+16	9.5E+14 5.8E+15 6.8E+15	7.4E+14 1.0E+15 1.8E+15	4.9E+14 4.6E+13 5.4E+14	2.1E+14 3.4E+13 2.4E+14	8.5E+13 3.2E+13 1.2E+14
Heat output	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**
Total Heat	1.8E+03	1.7E+03	1.6E+03	1.2E+03	7.4E+02	4.3E+02	1.8E+02	7.2E+01
Years Inventory Activity		**1.0E+04 Average [Bq/Can.]**	**3.0E+04 Average [Bq/Can.]**	**1.0E+05 Average [Bq/Can.]**	**3.0E+05 Average [Bq/Can.]**	**1.0E+06 Average [Bq/Can.]**	**3.0E+06 Average [Bq/Can.]**	**1.0E+07 Average [Bq/Can.]**
Total α Total βγ Total Activity		4.8E+13 2.5E+13 7.4E+13	1.7E+13 1.5E+13 3.2E+13	3.3E+12 3.6E+12 6.8E+12	1.8E+12 1.7E+12 3.4E+12	1.1E+12 8.6E+11 2.0E+12	5.9E+11 4.4E+11 1.0E+12	2.1E+11 1.4E+11 3.6E+11
Heat output		**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**
Total Heat		4.1E+01	1.4E+01	2.9E+00	1.7E+00	1.1E+00	5.8E-01	2.0E-01

Tab. C.2-4.2: Decay of heat output from 55 years after spent fuel reactor discharge for BE-2 canisters

Years Inventory Activity	**0 Average [Bq/Can.]**	**3 Average [Bq/Can.]**	**10 Average [Bq/Can.]**	**30 Average [Bq/Can.]**	**100 Average [Bq/Can.]**	**300 Average [Bq/Can.]**	**1'000 Average [Bq/Can.]**	**3'000 Average [Bq/Can.]**
Total α	1.0E+15	1.0E+15	9.6E+14	8.9E+14	7.2E+14	4.7E+14	2.0E+14	8.5E+13
Total βγ	8.6E+15	7.9E+15	6.6E+15	3.9E+15	7.5E+14	4.4E+13	3.4E+13	3.2E+13
Total Activity	9.6E+15	9.0E+15	7.6E+15	4.8E+15	1.5E+15	5.2E+14	2.4E+14	1.2E+14
Heat output	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**
Total Heat	1.5E+03	1.4E+03	1.3E+03	1.1E+03	6.9E+02	4.2E+02	1.8E+02	7.2E+01
Years Inventory Activity		**1.0E+04 Average [Bq/Can.]**	**3.0E+04 Average [Bq/Can.]**	**1.0E+05 Average [Bq/Can.]**	**3.0E+05 Average [Bq/Can.]**	**1.0E+06 Average [Bq/Can.]**	**3.0E+06 Average [Bq/Can.]**	**1.0E+07 Average [Bq/Can.]**
Total α		4.8E+13	1.7E+13	3.3E+12	1.7E+12	1.1E+12	5.8E+11	2.1E+11
Total βγ		2.5E+13	1.5E+13	3.6E+12	1.6E+12	8.6E+11	4.4E+11	1.4E+11
Total Activity		7.4E+13	3.2E+13	6.8E+12	3.4E+12	2.0E+12	1.0E+12	3.5E+11
Heat output		**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**
Total Heat		4.1E+01	1.4E+01	2.9E+00	1.7E+00	1.1E+00	5.8E-01	2.0E-01

C.3 Inventory data for BE-3

Tab. C.3-1: Material inventory of BE-3

Occurrence	Material	Mass [kg/Canister]
Raw waste	Al_2O_3	1.34E+00
Raw waste	Inconel	2.21E+01
Raw waste	Steel	9.25E+01
Raw waste	Uranium Oxide	1.77E+03
Raw waste	Zircaloy	4.73E+02
Total Raw waste		**2.36E+03**

It should be noted that the total canister mass for BE-3 is currently estimated to be around 24'000 kg, for the purposes of repository handling calculations (NAGRA 2002b). The detailed conceptual design of the actual canisters, currently envisaged for spent fuel disposal studies, will be given in a later report (JOHNSON & KING 2003). In this report the assumed masses of construction materials will be given. Therefore, this table only gives the raw waste (fuel assemblies) total material content of the disposal canister.

Tab. C.3-2: Radiological characteristics of BE-3 at 40 years after spent fuel reactor discharge

Parameter	Average
0 m γ dose rate [mSv/h]	3
0 m *n* dose rate [mSv/h]	4
Heat output [W]	1.5E+03

Tab. C.3-3: Radionuclide inventory of BE-3 at 40 years after spent fuel reactor discharge

Nuclide	**Fuel [Bq/Canister]**	**Structural materials [Bq/Canister]**	**Total [Bq/Canister]**
H-3	6.9E+12	3.1E+09	6.9E+12
Be-10	1.7E+07	4.5E+03	1.7E+07
C-14	6.2E+10	3.6E+10	9.8E+10
Cl-36	8.1E+08	8.2E+08	1.6E+09
Ar-39	1.1E+10	2.2E+06	1.1E+10
Ca-41	2.2E+08	4.0E+07	2.6E+08
Fe-55	9.0E+07	9.6E+09	9.7E+09
Co-60	4.5E+10	6.8E+11	7.3E+11
Ni-59	8.4E+08	1.2E+11	1.2E+11
Ni-63	8.6E+10	1.3E+13	1.3E+13
Se-79	1.6E+09	4.8E+03	1.6E+09
Kr-81	2.3E+06	-	2.3E+06
Kr-85	2.6E+14	5.1E+08	2.6E+14
Rb-87	1.7E+06	3.0E+00	1.7E+06
Sr-90	2.2E+15	3.7E+09	2.2E+15
Y-90	2.2E+15	3.7E+09	2.2E+15
Mo-93	7.6E+07	7.3E+08	8.1E+08
Zr-93	1.5E+11	1.0E+10	1.6E+11
Nb-93M	1.2E+11	8.7E+09	1.3E+11
Nb-94	1.9E+08	3.7E+10	3.7E+10
Tc-99	1.1E+12	1.9E+08	1.1E+12
Pd-107	9.6E+09	7.2E+04	9.6E+09
Ag-108	1.2E+08	-	1.2E+08
Ag-108M	1.3E+09	2.5E+04	1.3E+09
Cd-113M	9.8E+11	1.3E+07	9.8E+11
Sn-121	1.0E+11	-	1.0E+11
Sn-121M	1.3E+11	5.1E+10	1.8E+11
Sn-126	3.0E+10	2.0E+05	3.0E+10
Sb-125	5.0E+10	3.7E+09	5.4E+10
Sb-126	4.2E+09	2.8E+04	4.2E+09
Sb-126M	3.0E+10	-	3.0E+10
Te-125M	1.2E+10	9.0E+08	1.3E+10
I-129	2.6E+09	1.2E+04	2.6E+09
Ba-133	1.2E+08	7.8E+00	1.2E+08
Cs-134	2.2E+10	7.3E+04	2.2E+10
Cs-135	3.9E+10	1.2E+05	3.9E+10
Cs-137	3.4E+15	1.2E+10	3.4E+15
Ba-137M	3.3E+15	1.1E+10	3.3E+15
Pm-145	2.5E+08	-	2.5E+08
Pm-146	2.0E+08	6.4E+02	2.0E+08
Pm-147	3.0E+11	1.1E+06	3.0E+11
Sm-146	5.3E+02	-	5.3E+02
Sm-147	3.0E+05	-	3.0E+05
Sm-151	1.6E+13	6.7E+07	1.6E+13
Eu-150	1.5E+06	3.4E+00	1.5E+06
Eu-152	5.3E+10	1.9E+05	5.3E+10
Eu-154	4.0E+13	1.7E+08	4.0E+13
Eu-155	1.1E+12	5.1E+06	1.1E+12
Tb-157	1.5E+07	6.1E+00	1.5E+07
Tb-158	1.0E+08	7.8E+02	1.0E+08

Tab. C.3-3: (Cont.)

Nuclide	Fuel [Bq/Canister]	Structural materials [Bq/Canister]	Total [Bq/Canister]
Ho-166M	1.9E+09	6.2E+03	1.9E+09
Hf-178M	1.2E+09	2.6E+10	2.8E+10
Pt-193	4.2E+08	5.3E+02	4.2E+08
Tl-204	1.1E+07	-	1.1E+07
Tl-207	3.9E+05	-	3.9E+05
Tl-208	7.3E+07	1.7E+01	7.3E+07
Tl-209	3.1E+02	-	3.1E+02
Tl-210	6.5E+01	-	6.5E+01
Pb-209	1.4E+04	-	1.4E+04
Pb-210	1.0E+05	-	1.0E+05
Pb-211	3.9E+05	-	3.9E+05
Pb-212	2.0E+08	4.7E+01	2.0E+08
Pb-214	3.1E+05	-	3.1E+05
Bi-210	1.0E+05	-	1.0E+05
Bi-211	3.9E+05	-	3.9E+05
Bi-212	2.0E+08	4.7E+01	2.0E+08
Bi-213	1.4E+04	-	1.4E+04
Bi-214	3.1E+05	-	3.1E+05
Po-210	9.8E+04	-	9.8E+04
Po-211	1.1E+03	-	1.1E+03
Po-212	1.3E+08	3.0E+01	1.3E+08
Po-213	1.4E+04	-	1.4E+04
Po-214	3.1E+05	-	3.1E+05
Po-215	3.9E+05	-	3.9E+05
Po-216	2.0E+08	4.7E+01	2.0E+08
Po-218	3.1E+05	-	3.1E+05
At-217	1.4E+04	-	1.4E+04
At-218	6.2E+01	-	6.2E+01
Rn-219	3.9E+05	-	3.9E+05
Rn-220	2.0E+08	4.7E+01	2.0E+08
Rn-222	3.1E+05	-	3.1E+05
Fr-221	1.4E+04	-	1.4E+04
Fr-223	5.4E+03	-	5.4E+03
Ra-223	3.9E+05	-	3.9E+05
Ra-224	2.0E+08	4.7E+01	2.0E+08
Ra-225	1.4E+04	-	1.4E+04
Ra-226	3.1E+05	-	3.1E+05
Ra-228	3.1E+01	-	3.1E+01
Ac-225	1.4E+04	-	1.4E+04
Ac-227	3.9E+05	-	3.9E+05
Ac-228	3.1E+01	-	3.1E+01
Th-227	3.9E+05	-	3.9E+05
Th-228	2.0E+08	4.7E+01	2.0E+08
Th-229	1.4E+04	-	1.4E+04
Th-230	3.4E+07	1.1E+01	3.4E+07
Th-231	1.1E+09	2.0E+02	1.1E+09
Th-232	4.0E+01	-	4.0E+01
Th-234	1.7E+10	1.6E+04	1.7E+10
Pa-231	9.0E+05	-	9.0E+05
Pa-233	3.1E+10	2.8E+04	3.1E+10

Tab. C.3-3: (Cont.)

Nuclide	**Fuel [Bq/Canister]**	**Structural materials [Bq/Canister]**	**Total [Bq/Canister]**
Pa-234	5.6E+07	-	5.6E+07
Pa-234M	1.7E+10	1.6E+04	1.7E+10
U-232	2.0E+08	4.5E+01	2.0E+08
U-233	6.5E+06	3.0E+00	6.5E+06
U-234	1.0E+11	5.4E+04	1.0E+11
U-235	1.1E+09	-	1.1E+09
U-235M	2.2E+13	1.4E+08	2.2E+13
U-236	2.0E+10	3.9E+03	2.0E+10
U-237	3.6E+10	2.8E+05	3.6E+10
U-238	1.7E+10	1.6E+04	1.7E+10
Np-237	3.1E+10	2.8E+04	3.1E+10
Np-238	3.0E+09	2.3E+04	3.0E+09
Np-239	2.0E+12	2.3E+07	2.0E+12
Pu-238	2.3E+14	3.9E+08	2.3E+14
Pu-239	2.2E+13	1.4E+08	2.2E+13
Pu-240	3.6E+13	2.0E+08	3.6E+13
Pu-241	1.5E+15	1.2E+10	1.5E+15
Pu-242	1.7E+11	1.7E+06	1.7E+11
Am-241	2.8E+14	2.3E+09	2.8E+14
Am-242	6.2E+11	-	6.2E+11
Am-242M	6.2E+11	5.0E+06	6.2E+11
Am-243	2.0E+12	2.3E+07	2.0E+12
Cm-242	5.1E+11	4.0E+06	5.1E+11
Cm-243	6.5E+11	6.1E+06	6.5E+11
Cm-244	7.5E+13	8.7E+08	7.5E+13
Cm-245	5.0E+10	5.9E+05	5.0E+10
Cm-246	1.2E+10	1.5E+05	1.2E+10
Cf-250	2.5E+06	4.0E+01	2.5E+06
Total α	**6.5E+14**	**4.0E+09**	**6.5E+14**
Total βγ	**1.3E+16**	**1.4E+13**	**1.3E+16**
Total Activity	**1.4E+16**	**1.4E+13**	**1.4E+16**
Stable Elements			
Element	**[Mol/Canister]**	**[Mol/Canister]**	**[Mol/Canister]**
Ca	1.7E+00	3.3E-01	2.0E+00
Fe	1.7E+00	1.2E+03	1.2E+03
Co	2.6E-02	1.1E+00	1.1E+00
Ni	7.5E-01	4.1E+02	4.1E+02
Se	1.4E+00	3.6E-06	1.4E+00
Sr	8.2E+00	1.5E-03	8.2E+00
Zr	7.5E+01	4.6E+03	4.7E+03
Nb	1.9E-02	6.3E+00	6.3E+00
Pd	2.3E+01	1.5E-04	2.3E+01
Ag	1.6E+00	1.3E-05	1.6E+00
Cd	2.0E+00	1.9E-03	2.0E+00
Sn	8.2E-01	5.5E+01	5.6E+01
Sb	2.3E-01	3.5E-02	2.7E-01
Sm	6.5E+00	2.7E-05	6.5E+00
Eu	1.7E+00	8.1E-06	1.7E+00
Ho	5.8E-03	3.7E-08	5.8E-03

Tab. C.3-3: (Cont.)

Stable Elements			
Element	**[Mol/Canister]**	**[Mol/Canister]**	**[Mol/Canister]**
Hf	1.1E-02	2.4E-01	2.5E-01
Pb	8.4E-03	2.7E-01	2.8E-01

Tab. C.3-4: Decay of heat output from 40 years after spent fuel reactor discharge for BE-3 canisters

Years	**0**	**3**	**10**	**30**	**100**	**300**	**1'000**	**3'000**
Inventory Activity	**Average [Bq/Can.]**	**Average [Bq/Can.]**	**Average [Bq/Can.]**	**Average [Bq/Can.]**	**Average [Bq/Can.]**	**Average [Bq/Can.]**	**Average [Bq/Can.]**	**Average [Bq/Can.]**
Total α	6.5E+14	6.4E+14	6.3E+14	5.8E+14	4.5E+14	2.9E+14	1.2E+14	5.2E+13
Total βγ	1.3E+16	1.2E+16	1.0E+16	6.0E+15	1.1E+15	3.7E+13	2.5E+13	2.3E+13
Total Activity	1.4E+16	1.3E+16	1.1E+16	6.6E+15	1.6E+15	3.3E+14	1.5E+14	7.5E+13
Heat output	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**
Total Heat	1.5E+03	1.4E+03	1.3E+03	9.4E+02	4.9E+02	2.6E+02	1.1E+02	4.4E+01

Years	**1.0E+04**	**3.0E+04**	**1.0E+05**	**3.0E+05**	**1.0E+06**	**3.0E+06**	**1.0E+07**
Inventory Activity	**Average [Bq/Can.]**	**Average [Bq/Can.]**	**Average [Bq/Can.]**	**Average [Bq/Can.]**	**Average [Bq/Can.]**	**Average [Bq/Can.]**	**Average [Bq/Can.]**
Total α	3.1E+13	1.2E+13	2.5E+12	1.4E+12	8.2E+11	4.6E+11	2.0E+11
Total βγ	1.9E+13	1.1E+13	3.0E+12	1.5E+12	7.3E+11	3.7E+11	1.4E+11
Total Activity	5.0E+13	2.3E+13	5.5E+12	2.8E+12	1.6E+12	8.2E+11	3.4E+11
Heat output	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**
Total Heat	2.6E+01	1.0E+01	2.2E+00	1.3E+00	8.2E-01	4.5E-01	1.9E-01

C.4 Inventory data for BE-4

Tab. C.4-1: Material inventory of BE-4

Occurrence	Material	Mass [kg/Canister]
Raw waste	Al_2O_3	1.34E+00
Raw waste	Inconel	2.21E+01
Raw waste	Steel	9.25E+01
Raw waste	Uranium Oxide	1.77E+03
Raw waste	Zircaloy	4.73E+02
Total Raw waste		**2.36E+03**

It should be noted that the total canister mass for BE-4 is not specified as it is not required for any repository handling studies as these canisters are only modelled for the purposes of safety analysis burnup sensitivity studies. However, these canisters are the same as design as BE-3 and if ever required, this mass can be used.

Tab. C.4-2: Radiological characteristics of BE-4 at 40 years after spent fuel reactor discharge

Parameter	Average
0 m γ dose rate [Sv/h]	3
0 m *n* dose rate [Sv/h]	5
Heat output [W]	1.8E+03

Tab. C.4-3: Radionuclide inventory of BE-4 at 40 years after spent fuel reactor discharge

Nuclide	Fuel [Bq/Canister]	Structural materials [Bq/Canister]	Total [Bq/Canister]
H-3	7.9E+12	3.9E+09	7.9E+12
Be-10	2.0E+07	4.7E+03	2.0E+07
C-14	7.2E+10	4.5E+10	1.2E+11
Cl-36	9.5E+08	1.0E+09	2.0E+09
Ar-39	1.1E+10	3.0E+06	1.1E+10
Ca-41	2.8E+08	5.8E+07	3.4E+08
Fe-55	1.0E+08	1.1E+10	1.1E+10
Co-60	5.1E+10	7.8E+11	8.3E+11
Ni-59	9.8E+08	1.2E+11	1.2E+11
Ni-63	1.0E+11	1.3E+13	1.3E+13
Se-79	1.9E+09	6.1E+03	1.9E+09
Kr-81	3.0E+06	-	3.0E+06
Kr-85	3.0E+14	6.4E+08	3.0E+14
Rb-87	1.9E+06	3.6E+00	1.9E+06
Sr-90	2.5E+15	4.7E+09	2.5E+15
Y-90	2.5E+15	4.7E+09	2.5E+15
Mo-93	8.4E+07	7.6E+08	8.5E+08
Zr-93	1.7E+11	1.3E+10	1.8E+11
Nb-93M	1.4E+11	1.1E+10	1.5E+11
Nb-94	2.2E+08	3.6E+10	3.6E+10
Tc-99	1.2E+12	1.9E+08	1.2E+12
Pd-107	1.2E+10	9.0E+04	1.2E+10
Ag-108	1.3E+08	-	1.3E+08
Ag-108M	1.5E+09	3.3E+04	1.5E+09
Cd-113M	1.2E+12	1.7E+07	1.2E+12
Sn-121	1.2E+11	-	1.2E+11
Sn-121M	1.6E+11	6.2E+10	2.2E+11
Sn-126	3.4E+10	2.5E+05	3.4E+10
Sb-125	5.9E+10	4.5E+09	6.4E+10
Sb-126	4.8E+09	3.4E+04	4.8E+09
Sb-126M	3.4E+10	-	3.4E+10
Te-125M	1.4E+10	1.1E+09	1.5E+10
I-129	3.0E+09	1.5E+04	3.0E+09
Ba-133	1.5E+08	1.2E+01	1.5E+08
Cs-134	3.0E+10	9.8E+04	3.0E+10
Cs-135	4.0E+10	1.4E+05	4.0E+10
Cs-137	3.9E+15	1.5E+10	3.9E+15
Ba-137M	3.7E+15	1.4E+10	3.7E+15
Pm-145	2.6E+08	-	2.6E+08
Pm-146	1.7E+08	6.4E+02	1.7E+08
Pm-147	2.8E+11	1.2E+06	2.8E+11
Sm-146	4.5E+02	-	4.5E+02
Sm-147	2.6E+05	-	2.6E+05
Sm-151	2.0E+13	8.4E+07	2.0E+13
Eu-150	1.7E+06	4.0E+00	1.7E+06
Eu-152	5.3E+10	1.9E+05	5.3E+10
Eu-154	5.0E+13	2.3E+08	5.0E+13
Eu-155	1.3E+12	6.4E+06	1.3E+12
Tb-157	1.7E+07	7.5E+00	1.7E+07
Tb-158	1.3E+08	1.1E+03	1.3E+08

Tab. C.4-3: (Cont.)

Nuclide	Fuel [Bq/Canister]	Structural materials [Bq/Canister]	Total [Bq/Canister]
Ho-166M	2.3E+09	9.5E+03	2.3E+09
Hf-178M	1.3E+09	3.0E+10	3.1E+10
Pt-193	5.3E+08	1.1E+03	5.3E+08
Tl-204	1.3E+07	-	1.3E+07
Tl-207	3.0E+05	-	3.0E+05
Tl-208	2.3E+08	2.3E+01	2.3E+08
Tl-209	3.9E+02	-	3.9E+02
Tl-210	6.2E+01	-	6.2E+01
Pb-209	1.7E+04	-	1.7E+04
Pb-210	9.5E+04	-	9.5E+04
Pb-211	3.0E+05	-	3.0E+05
Pb-212	6.4E+08	6.5E+01	6.4E+08
Pb-214	3.0E+05	-	3.0E+05
Bi-210	9.3E+04	-	9.3E+04
Bi-211	3.0E+05	-	3.0E+05
Bi-212	6.4E+08	6.5E+01	6.4E+08
Bi-213	1.7E+04	-	1.7E+04
Bi-214	3.0E+05	-	3.0E+05
Po-210	9.0E+04	-	9.0E+04
Po-211	8.1E+02	-	8.1E+02
Po-212	4.0E+08	4.2E+01	4.0E+08
Po-213	1.7E+04	-	1.7E+04
Po-214	3.0E+05	-	3.0E+05
Po-215	3.0E+05	-	3.0E+05
Po-216	6.4E+08	6.5E+01	6.4E+08
Po-218	3.0E+05	-	3.0E+05
At-217	1.7E+04	-	1.7E+04
At-218	5.9E+01	-	5.9E+01
Rn-219	3.0E+05	-	3.0E+05
Rn-220	6.4E+08	6.5E+01	6.4E+08
Rn-222	3.0E+05	-	3.0E+05
Fr-221	1.7E+04	-	1.7E+04
Fr-223	4.0E+03	-	4.0E+03
Ra-223	3.0E+05	-	3.0E+05
Ra-224	6.4E+08	6.5E+01	6.4E+08
Ra-225	1.9E+04	-	1.9E+04
Ra-226	3.0E+05	-	3.0E+05
Ra-228	3.4E+01	-	3.4E+01
Ac-225	1.7E+04	-	1.7E+04
Ac-227	3.0E+05	-	3.0E+05
Ac-228	3.4E+01	-	3.4E+01
Th-227	2.8E+05	-	2.8E+05
Th-228	6.4E+08	6.5E+01	6.4E+08
Th-229	1.9E+04	-	1.9E+04
Th-230	3.6E+07	1.4E+01	3.6E+07
Th-231	7.9E+08	1.7E+02	7.9E+08
Th-232	4.4E+01	-	4.4E+01
Th-234	1.7E+10	1.7E+04	1.7E+10
Pa-231	6.7E+05	-	6.7E+05
Pa-233	4.4E+10	3.3E+04	4.4E+10
Pa-234	5.6E+07	-	5.6E+07

Tab. C.4-3: (Cont.)

Nuclide	**Fuel [Bq/Canister]**	**Structural materials [Bq/Canister]**	**Total [Bq/Canister]**
Pa-234M	1.7E+10	1.7E+04	1.7E+10
U-232	6.2E+08	6.4E+01	6.2E+08
U-233	8.7E+06	3.4E+00	8.7E+06
U-234	1.1E+11	6.8E+04	1.1E+11
U-235	7.9E+08	-	7.9E+08
U-235M	2.0E+13	1.4E+08	2.0E+13
U-236	2.2E+10	4.0E+03	2.2E+10
U-237	3.7E+10	3.1E+05	3.7E+10
U-238	1.7E+10	1.7E+04	1.7E+10
Np-237	4.4E+10	3.3E+04	4.4E+10
Np-238	2.8E+09	2.3E+04	2.8E+09
Np-239	3.0E+12	3.3E+07	3.0E+12
Pu-238	3.6E+14	5.0E+08	3.6E+14
Pu-239	2.0E+13	1.4E+08	2.0E+13
Pu-240	4.0E+13	2.0E+08	4.0E+13
Pu-241	1.5E+15	1.2E+10	1.5E+15
Pu-242	2.2E+11	2.2E+06	2.2E+11
Am-241	3.0E+14	2.5E+09	3.0E+14
Am-243	3.0E+12	3.3E+07	3.0E+12
Am-242	5.8E+11	-	5.8E+11
Am-242M	5.8E+11	4.8E+06	5.8E+11
Cm-242	4.8E+11	4.0E+06	4.8E+11
Cm-243	8.4E+11	8.4E+06	8.4E+11
Cm-244	1.2E+14	1.4E+09	1.2E+14
Cm-245	7.0E+10	8.4E+05	7.0E+10
Cm-246	1.7E+10	2.5E+05	1.7E+10
Cf-250	3.7E+06	6.4E+01	3.7E+06
Total α	**8.4E+14**	**4.8E+09**	**8.4E+14**
Total βγ	**1.4E+16**	**1.4E+13**	**1.4E+16**
Total Activity	**1.5E+16**	**1.4E+13**	**1.5E+16**
Stable Elements			
Element	**[Mol/Canister]**	**[Mol/Canister]**	**[Mol/Canister]**
Ca	1.8E+00	3.5E-01	2.1E+00
Fe	1.7E+00	1.2E+03	1.2E+03
Co	2.5E-02	1.0E+00	1.1E+00
Ni	7.5E-01	3.6E+02	3.6E+02
Se	1.6E+00	4.6E-06	1.6E+00
Sr	9.3E+00	1.9E-03	9.3E+00
Zr	8.4E+01	5.1E+03	5.2E+03
Nb	1.9E-02	5.3E+00	5.3E+00
Pd	2.8E+01	1.9E-04	2.8E+01
Ag	1.9E+00	1.6E-05	1.9E+00
Cd	2.5E+00	2.2E-03	2.5E+00
Sn	9.6E-01	6.1E+01	6.2E+01
Sb	2.7E-01	4.3E-02	3.2E-01
Sm	7.7E+00	3.3E-05	7.7E+00
Eu	2.2E+00	1.0E-05	2.2E+00
Ho	6.8E-03	5.0E-08	6.8E-03
Hf	1.1E-02	2.6E-01	2.7E-01
Pb	8.4E-03	2.9E-01	3.0E-01

Tab. C.4-4: Decay of heat output from 40 years after spent fuel reactor discharge for BE-4 canisters

Years Inventory Activity	**0 Average [Bq/Can.]**	**3 Average [Bq/Can.]**	**10 Average [Bq/Can.]**	**30 Average [Bq/Can.]**	**100 Average [Bq/Can.]**	**300 Average [Bq/Can.]**	**1'000 Average [Bq/Can.]**	**3'000 Average [Bq/Can.]**
Total α	8.4E+14	8.2E+14	7.8E+14	7.0E+14	5.2E+14	3.1E+14	1.3E+14	5.4E+13
Total βγ	1.4E+16	1.3E+16	1.1E+16	6.7E+15	1.3E+15	3.8E+13	2.4E+13	2.3E+13
Total Activity	1.5E+16	1.4E+16	1.2E+16	7.4E+15	1.8E+15	3.5E+14	1.5E+14	7.7E+13
Heat output	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**
Total Heat	1.8E+03	1.7E+03	1.5E+03	1.1E+03	5.6E+02	2.8E+02	1.1E+02	4.6E+01
Years Inventory Activity		**1.0E+04 Average [Bq/Can.]**	**3.0E+04 Average [Bq/Can.]**	**1.0E+05 Average [Bq/Can.]**	**3.0E+05 Average [Bq/Can.]**	**1.0E+06 Average [Bq/Can.]**	**3.0E+06 Average [Bq/Can.]**	**1.0E+07 Average [Bq/Can.]**
Total α		3.2E+13	1.2E+13	2.7E+12	1.7E+12	9.4E+11	5.0E+11	2.1E+11
Total βγ		1.9E+13	1.1E+13	3.2E+12	1.7E+12	8.3E+11	4.0E+11	1.4E+11
Total Activity		5.0E+13	2.3E+13	6.0E+12	3.4E+12	1.8E+12	9.0E+11	3.4E+11
Heat output		**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**
Total Heat		2.7E+01	1.0E+01	2.5E+00	1.6E+00	9.4E-01	4.9E-01	1.9E-01

C.5 Inventory data for BE-5

Tab. C.5-1: Material inventory of BE-5

Occurrence	Material	Mass [kg/Canister]
Raw waste	Al_2O_3	1.34E+00
Raw waste	Inconel	2.21E+01
Raw waste	Steel	9.25E+01
Raw waste	Uranium Oxide	1.77E+03
Raw waste	Zircaloy	4.73E+02
Total Raw waste		**2.36E+03**

It should be noted that the total canister mass for BE-5 is not specified as it is not required for any repository handling studies as these canisters are only modelled for the purposes of safety analysis burnup sensitivity studies. However, these canisters are the same as design as BE-3 and if ever required, this mass can be used.

Tab. C.5-2: Radiological characteristics of BE-5 at 40 years after spent fuel reactor discharge

Parameter	Average
0 m γ dose rate [Sv/h] [1]	4
0 m *n* dose rate [Sv/h] [1]	11
Heat output [W]	1.8E+03

[1] γ- and *n*-dose rates are scaled from the 55 GWd/t_{IHM} results using the difference between Cs-137 and Cm-244 activities respectively.

Tab. C.5-3: Radionuclide inventory of BE-5 at 40 years after spent fuel reactor discharge

Nuclide	**Fuel [Bq/Canister]**	**Structural materials [Bq/Canister]**	**Total [Bq/Canister]**
H-3	8.2E+12	4.1E+09	8.2E+12
Be-10	2.1E+07	4.9E+03	2.1E+07
C-14	7.6E+10	4.7E+10	1.2E+11
Cl-36	1.0E+09	1.1E+09	2.1E+09
Ar-39	1.1E+10	3.2E+06	1.1E+10
Ca-41	2.9E+08	6.0E+07	3.5E+08
Fe-55	1.1E+08	1.1E+10	1.2E+10
Co-60	5.3E+10	8.1E+11	8.6E+11
Ni-59	1.0E+09	1.3E+11	1.3E+11
Ni-63	1.1E+11	1.4E+13	1.4E+13
Se-79	1.9E+09	6.5E+03	1.9E+09
Kr-81	3.3E+06	-	3.3E+06
Kr-85	3.0E+14	6.7E+08	3.0E+14
Rb-87	1.9E+06	3.8E+00	1.9E+06
Sr-90	2.6E+15	4.9E+09	2.6E+15
Y-90	2.6E+15	4.9E+09	2.6E+15
Mo-93	8.9E+07	8.0E+08	8.9E+08
Zr-93	1.8E+11	1.4E+10	1.9E+11
Nb-93M	1.4E+11	1.1E+10	1.5E+11
Nb-94	2.3E+08	3.8E+10	3.8E+10
Tc-99	1.2E+12	2.0E+08	1.2E+12
Pd-107	1.2E+10	9.6E+04	1.2E+10
Ag-108	1.4E+08	-	1.4E+08
Ag-108M	1.6E+09	3.7E+04	1.6E+09
Cd-113M	1.3E+12	1.8E+07	1.3E+12
Sn-121	1.3E+11	-	1.3E+11
Sn-121M	1.7E+11	6.6E+10	2.3E+11
Sn-126	3.7E+10	2.6E+05	3.7E+10
Sb-125	6.2E+10	4.7E+09	6.6E+10
Sb-126	5.1E+09	3.7E+04	5.1E+09
Sb-126M	3.7E+10	-	3.7E+10
Te-125M	1.5E+10	1.1E+09	1.6E+10
I-129	3.1E+09	1.6E+04	3.1E+09
Ba-133	1.6E+08	1.5E+01	1.6E+08
Cs-134	3.0E+10	1.1E+05	3.0E+10
Cs-135	4.2E+10	1.5E+05	4.2E+10
Cs-137	4.1E+15	1.6E+10	4.1E+15
Ba-137M	3.9E+15	1.5E+10	3.9E+15
Pm-145	2.8E+08	-	2.8E+08
Pm-146	1.8E+08	6.7E+02	1.8E+08
Pm-147	2.9E+11	1.2E+06	2.9E+11
Sm-146	4.7E+02	-	4.7E+02
Sm-147	2.7E+05	-	2.7E+05
Sm-151	1.9E+13	8.2E+07	1.9E+13
Eu-150	1.8E+06	4.2E+00	1.8E+06
Eu-152	5.0E+10	1.8E+05	5.0E+10
Eu-154	5.3E+13	2.5E+08	5.3E+13
Eu-155	1.4E+12	6.7E+06	1.4E+12
Tb-157	1.8E+07	8.0E+00	1.8E+07
Tb-158	1.6E+08	1.3E+03	1.6E+08

Tab. C.5-3: (Cont.)

Nuclide	Fuel [Bq/Canister]	Structural materials [Bq/Canister]	Total [Bq/Canister]
Ho-166M	2.6E+09	1.1E+04	2.6E+09
Hf-178M	1.4E+09	3.1E+10	3.2E+10
Pt-193	5.8E+08	1.9E+03	5.8E+08
Tl-204	1.4E+07	-	1.4E+07
Tl-207	2.6E+05	-	2.6E+05
Tl-208	2.1E+08	2.5E+01	2.1E+08
Tl-209	3.8E+02	-	3.8E+02
Tl-210	6.2E+01	-	6.2E+01
Pb-209	1.7E+04	-	1.7E+04
Pb-210	9.5E+04	-	9.5E+04
Pb-211	2.6E+05	-	2.6E+05
Pb-212	5.6E+08	7.0E+01	5.6E+08
Pb-214	3.0E+05	-	3.0E+05
Bi-210	9.4E+04	-	9.4E+04
Bi-211	2.6E+05	-	2.6E+05
Bi-212	5.6E+08	7.0E+01	5.6E+08
Bi-213	1.7E+04	-	1.7E+04
Bi-214	3.0E+05	-	3.0E+05
Po-210	9.1E+04	-	9.1E+04
Po-211	7.3E+02	-	7.3E+02
Po-212	3.6E+08	4.5E+01	3.6E+08
Po-213	1.7E+04	-	1.7E+04
Po-214	3.0E+05	-	3.0E+05
Po-215	2.6E+05	-	2.6E+05
Po-216	5.6E+08	7.0E+01	5.6E+08
Po-218	3.0E+05	-	3.0E+05
At-217	1.7E+04	-	1.7E+04
At-218	5.9E+01	-	5.9E+01
Rn-219	2.6E+05	-	2.6E+05
Rn-220	5.6E+08	7.0E+01	5.6E+08
Rn-222	3.0E+05	-	3.0E+05
Fr-221	1.7E+04	-	1.7E+04
Fr-223	3.7E+03	-	3.7E+03
Ra-223	2.6E+05	-	2.6E+05
Ra-224	5.6E+08	7.0E+01	5.6E+08
Ra-225	1.8E+04	-	1.8E+04
Ra-226	3.0E+05	-	3.0E+05
Ra-228	3.4E+01	-	3.4E+01
Ac-225	1.7E+04	-	1.7E+04
Ac-227	2.6E+05	-	2.6E+05
Ac-228	3.4E+01	-	3.4E+01
Th-227	2.5E+05	-	2.5E+05
Th-228	5.6E+08	7.0E+01	5.6E+08
Th-229	1.8E+04	-	1.8E+04
Th-230	3.5E+07	1.5E+01	3.5E+07
Th-231	7.2E+08	1.6E+02	7.2E+08
Th-232	4.4E+01	-	4.4E+01
Th-234	1.7E+10	1.7E+04	1.7E+10
Pa-231	6.0E+05	-	6.0E+05
Pa-233	4.2E+10	3.2E+04	4.2E+10
Pa-234	5.6E+07	-	5.6E+07

Tab. C.5-3: (Cont.)

Nuclide	**Fuel [Bq/Canister]**	**Structural materials [Bq/Canister]**	**Total [Bq/Canister]**
Pa-234M	1.7E+10	1.7E+04	1.7E+10
U-232	5.5E+08	6.8E+01	5.5E+08
U-233	8.5E+06	3.4E+00	8.5E+06
U-234	1.1E+11	7.4E+04	1.1E+11
U-235	7.2E+08	-	7.2E+08
U-235M	2.0E+13	1.3E+08	2.0E+13
U-236	2.2E+10	4.2E+03	2.2E+10
U-237	3.8E+10	3.0E+05	3.8E+10
U-238	1.7E+10	1.7E+04	1.7E+10
Np-237	4.2E+10	3.2E+04	4.2E+10
Np-238	2.8E+09	2.3E+04	2.8E+09
Np-239	3.3E+12	3.7E+07	3.3E+12
Pu-238	3.7E+14	5.3E+08	3.7E+14
Pu-239	2.0E+13	1.3E+08	2.0E+13
Pu-240	4.2E+13	2.1E+08	4.2E+13
Pu-241	1.5E+15	1.2E+10	1.5E+15
Pu-242	2.4E+11	2.3E+06	2.4E+11
Am-241	3.0E+14	2.5E+09	3.0E+14
Am-242	5.8E+11	-	5.8E+11
Am-242M	5.8E+11	4.8E+06	5.8E+11
Am-243	3.3E+12	3.7E+07	3.3E+12
Cm-242	4.9E+11	4.0E+06	4.9E+11
Cm-243	9.3E+11	9.0E+06	9.3E+11
Cm-244	1.4E+14	1.7E+09	1.4E+14
Cm-245	9.1E+10	1.1E+06	9.1E+10
Cm-246	2.7E+10	3.8E+05	2.7E+10
Cf-250	9.4E+06	1.6E+02	9.4E+06
Total α	**8.8E+14**	**5.1E+09**	**8.8E+14**
Total βγ	**1.5E+16**	**1.5E+13**	**1.5E+16**
Total Activity	**1.6E+16**	**1.5E+13**	**1.6E+16**
Stable Elements			
Element	**[Mol/Canister]**	**[Mol/Canister]**	**[Mol/Canister]**
Ca	1.8E+00	3.5E-01	2.1E+00
Fe	1.7E+00	1.2E+03	1.2E+03
Co	2.5E-02	1.0E+00	1.1E+00
Ni	7.5E-01	3.6E+02	3.6E+02
Se	1.6E+00	4.8E-06	1.6E+00
Sr	9.6E+00	2.0E-03	9.6E+00
Zr	8.8E+01	5.1E+03	5.2E+03
Nb	1.9E-02	5.3E+00	5.3E+00
Pd	3.1E+01	2.0E-04	3.1E+01
Ag	2.0E+00	1.6E-05	2.0E+00
Cd	2.7E+00	2.2E-03	2.7E+00
Sn	1.0E+00	6.1E+01	6.2E+01
Sb	2.9E-01	4.5E-02	3.4E-01
Sm	8.0E+00	3.5E-05	8.0E+00
Eu	2.3E+00	1.1E-05	2.3E+00
Ho	7.4E-03	5.5E-08	7.4E-03
Hf	1.1E-02	2.6E-01	2.7E-01
Pb	8.4E-03	2.9E-01	3.0E-01

Tab. C.5-4: Decay of heat output from 40 years after spent fuel reactor discharge for BE-5 canisters

Years Inventory Activity	**0 Average [Bq/Can.]**	**3 Average [Bq/Can.]**	**10 Average [Bq/Can.]**	**30 Average [Bq/Can.]**	**100 Average [Bq/Can.]**	**300 Average [Bq/Can.]**	**1'000 Average [Bq/Can.]**	**3'000 Average [Bq/Can.]**
Total α	8.8E+14	8.6E+14	8.2E+14	7.3E+14	5.3E+14	3.2E+14	1.3E+14	5.6E+13
Total βγ	1.5E+16	1.4E+16	1.1E+16	6.9E+15	1.3E+15	3.9E+13	2.5E+13	2.3E+13
Total Activity	1.6E+16	1.5E+16	1.3E+16	7.7E+15	1.9E+15	3.6E+14	1.5E+14	7.9E+13
Heat output	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**
Total Heat	1.8E+03	1.7E+03	1.5E+03	1.1E+03	5.7E+02	2.8E+02	1.2E+02	4.7E+01

Years Inventory Activity	**1.0E+04 Average [Bq/Can.]**	**3.0E+04 Average [Bq/Can.]**	**1.0E+05 Average [Bq/Can.]**	**3.0E+05 Average [Bq/Can.]**	**1.0E+06 Average [Bq/Can.]**	**3.0E+06 Average [Bq/Can.]**	**1.0E+07 Average [Bq/Can.]**
Total α	3.3E+13	1.2E+13	2.7E+12	1.7E+12	9.5E+11	5.0E+11	2.1E+11
Total βγ	1.9E+13	1.1E+13	3.3E+12	1.7E+12	8.4E+11	4.0E+11	1.4E+11
Total Activity	5.2E+13	2.3E+13	6.1E+12	3.5E+12	1.8E+12	9.1E+11	3.4E+11
Heat output	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**
Total Heat	2.7E+01	1.0E+01	2.5E+00	1.7E+00	9.4E-01	4.9E-01	1.9E-01

C.6 Inventory data for BE-6

Tab. C.6-1: Material inventory of BE-6

Occurrence	Material	Mass [kg/Canister]
Raw waste	Al_2O_3	1.34E+00
Raw waste	Inconel	2.21E+01
Raw waste	Steel	9.25E+01
Raw waste	Uranium Oxide	1.77E+03
Raw waste	Zircaloy	4.73E+02
Total Raw waste		**2.36E+03**

It should be noted that the total canister mass for BE-6 is not specified as it is not required for any repository handling studies as these canisters are only modelled for the purposes of safety analysis burnup sensitivity studies. However, these canisters are the same as design as BE-3 and if ever required, this mass can be used.

Tab. C.6-2: Radiological characteristics of BE-6 at 40 years after spent fuel reactor discharge

Parameter	Average
0 m γ dose rate [Sv/h] [1]	4
0 m *n* dose rate [Sv/h] [1]	17
Heat output [W]	2.0E+03

[1] γ- and *n*-dose rates are scaled from the 55 GWd/ t_{IHM} results using the difference between Cs-137 and Cm-244 activities respectively.

Tab. C.6-3: Radionuclide inventory of BE-6 at 40 years after spent fuel reactor discharge

Nuclide	**Fuel [Bq/Canister]**	**Structural materials [Bq/Canister]**	**Total [Bq/Canister]**
H-3	8.5E+12	4.4E+09	8.5E+12
Be-10	2.2E+07	5.1E+03	2.2E+07
C-14	7.8E+10	4.9E+10	1.3E+11
Cl-36	1.0E+09	1.1E+09	2.1E+09
Ar-39	1.1E+10	3.3E+06	1.1E+10
Ca-41	3.0E+08	6.2E+07	3.7E+08
Fe-55	1.1E+08	1.2E+10	1.2E+10
Co-60	5.4E+10	8.5E+11	9.0E+11
Ni-59	1.0E+09	1.4E+11	1.4E+11
Ni-63	1.1E+11	1.4E+13	1.4E+13
Se-79	2.0E+09	6.7E+03	2.0E+09
Kr-81	3.6E+06	-	3.6E+06
Kr-85	3.2E+14	6.9E+08	3.2E+14
Rb-87	2.0E+06	3.9E+00	2.0E+06
Sr-90	2.7E+15	5.1E+09	2.7E+15
Y-90	2.7E+15	5.1E+09	2.7E+15
Mo-93	9.2E+07	8.3E+08	9.2E+08
Zr-93	1.8E+11	1.4E+10	2.0E+11
Nb-93M	1.4E+11	1.2E+10	1.6E+11
Nb-94	2.4E+08	3.9E+10	4.0E+10
Tc-99	1.3E+12	2.1E+08	1.3E+12
Pd-107	1.3E+10	1.0E+05	1.3E+10
Ag-108	1.4E+08	-	1.4E+08
Ag-108M	1.6E+09	4.0E+04	1.6E+09
Cd-113M	1.3E+12	1.9E+07	1.3E+12
Sn-121	1.4E+11	-	1.4E+11
Sn-121M	1.7E+11	6.8E+10	2.4E+11
Sn-126	3.8E+10	2.8E+05	3.8E+10
Sb-125	6.3E+10	4.8E+09	6.8E+10
Sb-126	5.4E+09	3.8E+04	5.4E+09
Sb-126M	3.8E+10	-	3.8E+10
Te-125M	1.5E+10	1.2E+09	1.7E+10
I-129	3.2E+09	1.6E+04	3.2E+09
Ba-133	1.6E+08	1.7E+01	1.6E+08
Cs-134	3.3E+10	1.2E+05	3.3E+10
Cs-135	4.6E+10	1.6E+05	4.6E+10
Cs-137	4.2E+15	1.6E+10	4.2E+15
Ba-137M	4.0E+15	1.5E+10	4.0E+15
Pm-145	3.0E+08	-	3.0E+08
Pm-146	1.7E+08	6.5E+02	1.7E+08
Pm-147	2.8E+11	1.2E+06	2.8E+11
Sm-146	4.6E+02	-	4.6E+02
Sm-147	2.6E+05	-	2.6E+05
Sm-151	2.1E+13	8.8E+07	2.1E+13
Eu-150	2.0E+06	4.9E+00	2.0E+06
Eu-152	5.5E+10	2.0E+05	5.5E+10
Eu-154	5.6E+13	2.6E+08	5.6E+13
Eu-155	1.4E+12	7.1E+06	1.4E+12
Tb-157	1.9E+07	8.3E+00	1.9E+07

Tab. C.6-3: (Cont.)

Nuclide	**Fuel [Bq/Canister]**	**Structural materials [Bq/Canister]**	**Total [Bq/Canister]**
Tb-158	1.8E+08	1.4E+03	1.8E+08
Ho-166M	2.7E+09	1.3E+04	2.7E+09
Hf-178M	1.4E+09	3.2E+10	3.4E+10
Pt-193	6.0E+08	2.8E+03	6.0E+08
Tl-204	1.4E+07	-	1.4E+07
Tl-207	2.7E+05	-	2.7E+05
Tl-208	2.9E+08	2.9E+01	2.9E+08
Tl-209	4.2E+02	-	4.2E+02
Tl-210	6.4E+01	-	6.4E+01
Pb-209	1.9E+04	-	1.9E+04
Pb-210	9.6E+04	-	9.6E+04
Pb-211	2.7E+05	-	2.7E+05
Pb-212	8.0E+08	8.1E+01	8.0E+08
Pb-214	3.0E+05	-	3.0E+05
Bi-210	9.5E+04	-	9.5E+04
Bi-211	2.7E+05	-	2.7E+05
Bi-212	8.0E+08	8.1E+01	8.0E+08
Bi-213	1.9E+04	-	1.9E+04
Bi-214	3.0E+05	-	3.0E+05
Po-210	9.2E+04	-	9.2E+04
Po-211	7.4E+02	-	7.4E+02
Po-212	5.1E+08	5.2E+01	5.1E+08
Po-213	1.9E+04	-	1.9E+04
Po-214	3.0E+05	-	3.0E+05
Po-215	2.7E+05	-	2.7E+05
Po-216	8.0E+08	8.1E+01	8.0E+08
Po-218	3.0E+05	-	3.0E+05
At-217	1.9E+04	-	1.9E+04
At-218	6.1E+01	-	6.1E+01
Rn-219	2.7E+05	-	2.7E+05
Rn-220	8.0E+08	8.1E+01	8.0E+08
Rn-222	3.0E+05	-	3.0E+05
Fr-221	1.9E+04	-	1.9E+04
Fr-223	3.7E+03	-	3.7E+03
Ra-223	2.7E+05	-	2.7E+05
Ra-224	8.0E+08	8.1E+01	8.0E+08
Ra-225	2.0E+04	-	2.0E+04
Ra-226	3.0E+05	-	3.0E+05
Ra-228	3.5E+01	-	3.5E+01
Ac-225	1.9E+04	-	1.9E+04
Ac-227	2.7E+05	-	2.7E+05
Ac-228	3.5E+01	-	3.5E+01
Th-227	2.6E+05	-	2.6E+05
Th-228	8.0E+08	8.1E+01	8.0E+08
Th-229	2.0E+04	-	2.0E+04
Th-230	3.7E+07	1.6E+01	3.7E+07
Th-231	7.2E+08	1.6E+02	7.2E+08
Th-232	4.5E+01	-	4.5E+01
Th-234	1.7E+10	1.7E+04	1.7E+10

Tab. C.6-3: (Cont.)

Nuclide	**Fuel [Bq/Canister]**	**Structural materials [Bq/Canister]**	**Total [Bq/Canister]**
Pa-231	6.1E+05	-	6.1E+05
Pa-233	4.8E+10	3.3E+04	4.8E+10
Pa-234	5.6E+07	-	5.6E+07
Pa-234M	1.7E+10	1.7E+04	1.7E+10
U-232	7.8E+08	7.9E+01	7.8E+08
U-233	9.6E+06	3.5E+00	9.6E+06
U-234	1.2E+11	7.9E+04	1.2E+11
U-235	7.2E+08	-	7.2E+08
U-235M	2.0E+13	1.3E+08	2.0E+13
U-236	2.3E+10	4.1E+03	2.3E+10
U-237	3.9E+10	3.1E+05	3.9E+10
U-238	1.7E+10	1.7E+04	1.7E+10
Np-237	4.8E+10	3.3E+04	4.8E+10
Np-238	3.0E+09	2.3E+04	3.0E+09
Pu-238	4.3E+14	5.7E+08	4.3E+14
Np-239	3.6E+12	3.9E+07	3.6E+12
Pu-239	2.0E+13	1.3E+08	2.0E+13
Pu-240	4.2E+13	2.1E+08	4.2E+13
Pu-241	1.6E+15	1.2E+10	1.6E+15
Pu-242	2.5E+11	2.4E+06	2.5E+11
Am-241	3.0E+14	2.5E+09	3.0E+14
Am-242	6.1E+11	-	6.1E+11
Am-243	3.6E+12	3.9E+07	3.6E+12
Am-242M	6.1E+11	4.9E+06	6.1E+11
Cm-242	5.1E+11	4.0E+06	5.1E+11
Cm-243	1.0E+12	9.6E+06	1.0E+12
Cm-244	1.7E+14	2.0E+09	1.7E+14
Cm-245	1.1E+11	1.3E+06	1.1E+11
Cm-246	3.4E+10	4.7E+05	3.4E+10
Cf-250	1.5E+07	2.5E+02	1.5E+07
Total α	**9.8E+14**	**5.4E+09**	**9.8E+14**
Total βγ	**1.5E+16**	**1.5E+13**	**1.5E+16**
Total Activity	**1.6E+16**	**1.5E+13**	**1.6E+16**
Stable Elements			
Element	**[Mol/Canister]**	**[Mol/Canister]**	**[Mol/Canister]**
Ca	1.8E+00	3.5E-01	2.1E+00
Fe	1.7E+00	1.2E+03	1.2E+03
Co	2.5E-02	1.0E+00	1.0E+00
Ni	7.5E-01	3.6E+02	3.6E+02
Se	1.6E+00	4.5E-06	1.6E+00
Sr	9.3E+00	1.9E-03	9.3E+00
Zr	8.4E+01	5.1E+03	5.2E+03
Nb	1.9E-02	5.3E+00	5.3E+00
Pd	2.8E+01	1.8E-04	2.8E+01
Ag	1.8E+00	1.6E-05	1.8E+00
Cd	2.4E+00	2.2E-03	2.4E+00
Sn	9.6E-01	6.1E+01	6.2E+01
Sb	2.7E-01	4.3E-02	3.1E-01

Tab. C.6-3: (Cont.)

Stable Elements			
Element	**[Mol/Canister]**	**[Mol/Canister]**	**[Mol/Canister]**
Sm	7.6E+00	3.3E-05	7.6E+00
Eu	2.2E+00	1.0E-05	2.2E+00
Ho	6.8E-03	4.9E-08	6.8E-03
Hf	1.1E-02	2.6E-01	2.7E-01
Pb	8.4E-03	2.9E-01	3.0E-01

Tab. C.6-4: Decay of heat output from 40 years after spent fuel reactor discharge for BE-6 canisters

Years Inventory Activity	**0 Average [Bq/Can.]**	**3 Average [Bq/Can.]**	**10 Average [Bq/Can.]**	**30 Average [Bq/Can.]**	**100 Average [Bq/Can.]**	**300 Average [Bq/Can.]**	**1'000 Average [Bq/Can.]**	**3'000 Average [Bq/Can.]**
Total α Total βγ Total Activity	9.8E+14 1.5E+16 1.6E+16	9.6E+14 1.4E+16 1.5E+16	9.0E+14 1.2E+16 1.3E+16	7.9E+14 7.2E+15 8.1E+15	5.7E+14 1.4E+15 1.9E+15	3.3E+14 4.0E+13 3.7E+14	1.3E+14 2.5E+13 1.6E+14	5.7E+13 2.4E+13 8.0E+13
Heat output	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**
Total Heat	2.0E+03	1.9E+03	1.6E+03	1.2E+03	6.1E+02	2.9E+02	1.2E+02	4.7E+01
Years Inventory Activity		**1.0E+04 Average [Bq/Can.]**	**3.0E+04 Average [Bq/Can.]**	**1.0E+05 Average [Bq/Can.]**	**3.0E+05 Average [Bq/Can.]**	**1.0E+06 Average [Bq/Can.]**	**3.0E+06 Average [Bq/Can.]**	**1.0E+07 Average [Bq/Can.]**
Total α Total βγ Total Activity		3.3E+13 1.9E+13 5.2E+13	1.2E+13 1.1E+13 2.4E+13	2.9E+12 3.4E+12 6.4E+12	1.9E+12 1.9E+12 3.8E+12	1.0E+12 8.8E+11 1.9E+12	5.3E+11 4.2E+11 9.5E+11	2.1E+11 1.4E+11 3.5E+11
Heat output		**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**
Total Heat		2.8E+01	1.0E+01	2.6E+00	1.8E+00	1.0E+00	5.1E-01	2.0E-01

C.7 Inventory data for BE-7

Tab. C.7-1: Material inventory of BE-7

Occurrence	**Material**	**Mass [kg/Canister]**
Raw waste	Al_2O_3	1.23E+00
Raw waste	Inconel	2.15E+01
Raw waste	Mixed Oxide (MOX)	3.65E+02
Raw waste	Steel	8.90E+01
Raw waste	Uranium Oxide	1.33E+03
Raw waste	Zircaloy	4.52E+02
Total Raw waste		**2.26E+03**

It should be noted that the total canister mass for BE-7 is not specified as it is not required for any repository handling studies as these canisters are only modelled for the purposes of safety analysis burnup sensitivity studies. However, these canisters are the same as design as BE-3 and if ever required, this mass can be used.

Tab. C.7-2: Radiological characteristics of BE-7 at 40 years after spent fuel reactor discharge

Parameter	**Average**
0 m γ dose rate [Sv/h] [1]	4
0 m *n* dose rate [Sv/h] [1]	40
Heat output [W]	2.1E+03

[1] γ- and *n*-dose rates are scaled from the 48 GWd/t_{IHM} MOX results using the difference between Cs-137 and Cm-244 activities respectively.

Tab. C.7-3: Radionuclide inventory of BE-7 at 40 years after spent fuel reactor discharge

Nuclide	**Fuel [Bq/Canister]**	**Structural materials [Bq/Canister]**	**Total [Bq/Canister]**
H-3	7.0E+12	3.4E+09	7.0E+12
Be-10	1.8E+07	4.7E+03	1.8E+07
C-14	5.9E+10	3.4E+10	9.3E+10
Cl-36	7.1E+08	7.5E+08	1.5E+09
Ar-39	1.2E+10	1.9E+06	1.2E+10
Ca-41	1.9E+08	3.7E+07	2.3E+08
Fe-55	8.1E+07	9.9E+09	1.0E+10
Co-60	4.2E+10	6.8E+11	7.3E+11
Ni-59	7.4E+08	1.2E+11	1.2E+11
Ni-63	7.5E+10	1.3E+13	1.3E+13
Se-79	1.6E+09	4.9E+03	1.6E+09
Kr-81	2.4E+06	-	2.4E+06
Kr-85	2.4E+14	5.2E+08	2.4E+14
Rb-87	1.5E+06	3.0E+00	1.5E+06
Sr-90	1.9E+15	3.8E+09	1.9E+15
Y-90	2.0E+15	3.8E+09	2.0E+15
Mo-93	7.9E+07	7.4E+08	8.2E+08
Zr-93	1.4E+11	1.0E+10	1.5E+11
Nb-93M	1.1E+11	8.8E+09	1.2E+11
Nb-94	1.9E+08	3.7E+10	3.7E+10
Tc-99	1.1E+12	1.9E+08	1.1E+12
Pd-107	1.3E+10	7.2E+04	1.3E+10
Ag-108	1.1E+08	-	1.1E+08
Ag-108M	1.3E+09	2.6E+04	1.3E+09
Cd-113M	1.3E+12	1.3E+07	1.3E+12
Sn-121	1.3E+11	-	1.3E+11
Sn-121M	1.7E+11	5.3E+10	2.2E+11
Sn-126	3.7E+10	2.0E+05	3.7E+10
Sb-125	6.1E+10	3.8E+09	6.5E+10
Sb-126	5.3E+09	2.8E+04	5.3E+09
Sb-126M	3.7E+10	-	3.7E+10
Te-125M	1.5E+10	9.3E+08	1.6E+10
I-129	2.9E+09	1.2E+04	2.9E+09
Ba-133	1.1E+08	7.8E+00	1.1E+08
Cs-134	2.3E+10	7.5E+04	2.3E+10
Cs-135	4.8E+10	1.4E+05	4.8E+10
Cs-137	3.5E+15	1.2E+10	3.5E+15
Ba-137M	3.4E+15	1.1E+10	3.4E+15
Pm-145	2.6E+08	-	2.6E+08
Pm-146	2.6E+08	7.8E+02	2.6E+08
Pm-147	2.9E+11	1.1E+06	2.9E+11
Sm-146	6.8E+02	-	6.8E+02
Sm-147	2.9E+05	-	2.9E+05
Sm-151	2.0E+13	7.7E+07	2.0E+13
Eu-150	3.6E+06	5.8E+00	3.6E+06
Eu-152	7.9E+10	2.3E+05	7.9E+10
Eu-154	4.9E+13	1.8E+08	4.9E+13
Eu-155	1.5E+12	5.7E+06	1.5E+12
Tb-157	1.4E+07	6.2E+00	1.4E+07
Tb-158	2.2E+08	8.4E+02	2.2E+08

Tab. C.7-3: (Cont.)

Nuclide	Fuel [Bq/Canister]	Structural materials [Bq/Canister]	Total [Bq/Canister]
Ho-166M	2.5E+09	5.9E+03	2.5E+09
Hf-178M	1.2E+09	2.7E+10	2.8E+10
Pt-193	3.6E+08	4.2E+02	3.6E+08
Tl-204	9.7E+06	-	9.7E+06
Tl-207	3.0E+05	-	3.0E+05
Tl-208	6.4E+07	2.3E+01	6.4E+07
Tl-209	2.6E+02	-	2.6E+02
Tl-210	5.9E+01	-	5.9E+01
Pb-209	1.2E+04	-	1.2E+04
Pb-210	8.8E+04	-	8.8E+04
Pb-211	3.0E+05	-	3.0E+05
Pb-212	1.8E+08	6.4E+01	1.8E+08
Pb-214	2.8E+05	-	2.8E+05
Bi-210	8.8E+04	-	8.8E+04
Bi-211	3.0E+05	-	3.0E+05
Bi-212	1.8E+08	6.4E+01	1.8E+08
Bi-213	1.2E+04	-	1.2E+04
Bi-214	2.8E+05	-	2.8E+05
Po-210	8.5E+04	-	8.5E+04
Po-211	8.5E+02	-	8.5E+02
Po-212	1.1E+08	4.1E+01	1.1E+08
Po-213	1.1E+04	-	1.1E+04
Po-214	2.8E+05	-	2.8E+05
Po-215	3.0E+05	-	3.0E+05
Po-216	1.8E+08	6.4E+01	1.8E+08
Po-218	2.8E+05	-	2.8E+05
At-217	1.2E+04	-	1.2E+04
At-218	5.6E+01	-	5.6E+01
Rn-219	3.0E+05	-	3.0E+05
Rn-220	1.8E+08	6.4E+01	1.8E+08
Rn-222	2.8E+05	-	2.8E+05
Fr-221	1.2E+04	-	1.2E+04
Fr-223	4.2E+03	-	4.2E+03
Ra-223	3.0E+05	-	3.0E+05
Ra-224	1.8E+08	6.4E+01	1.8E+08
Ra-225	1.2E+04	-	1.2E+04
Ra-226	2.8E+05	-	2.8E+05
Ra-228	2.4E+01	-	2.4E+01
Ac-225	1.2E+04	-	1.2E+04
Ac-228	2.4E+01	-	2.4E+01
Ac-227	3.0E+05	-	3.0E+05
Th-227	3.0E+05	-	3.0E+05
Th-228	1.8E+08	6.4E+01	1.8E+08
Th-229	1.2E+04	-	1.2E+04
Th-230	3.3E+07	1.2E+01	3.3E+07
Th-231	8.3E+08	2.2E+02	8.3E+08
Th-232	3.1E+01	-	3.1E+01
Th-234	1.6E+10	1.5E+04	1.6E+10
Pa-231	7.0E+05	-	7.0E+05
Pa-233	2.8E+10	3.2E+04	2.8E+10

Tab. C.7-3: (Cont.)

Nuclide	Fuel [Bq/Canister]	Structural materials [Bq/Canister]	Total [Bq/Canister]
Pa-234	5.4E+07	-	5.4E+07
Pa-234M	1.6E+10	1.5E+04	1.6E+10
U-232	1.8E+08	6.2E+01	1.8E+08
U-233	5.5E+06	3.3E+00	5.5E+06
U-234	1.1E+11	6.1E+04	1.1E+11
U-235	8.3E+08	-	8.3E+08
U-235M	2.7E+13	1.6E+08	2.7E+13
U-236	1.6E+10	3.8E+03	1.6E+10
U-237	6.2E+10	3.2E+05	6.2E+10
U-238	1.6E+10	1.5E+04	1.6E+10
Np-237	2.8E+10	3.2E+04	2.8E+10
Np-238	1.3E+10	3.1E+04	1.3E+10
Np-239	5.7E+12	2.3E+07	5.7E+12
Pu-238	4.3E+14	4.4E+08	4.3E+14
Pu-239	2.7E+13	1.6E+08	2.7E+13
Pu-240	6.9E+13	2.0E+08	6.9E+13
Pu-241	2.6E+15	1.3E+10	2.6E+15
Pu-242	4.1E+11	1.7E+06	4.1E+11
Am-241	5.1E+14	2.6E+09	5.1E+14
Am-242	2.6E+12	-	2.6E+12
Am-242M	2.6E+12	6.7E+06	2.6E+12
Am-243	5.7E+12	2.3E+07	5.7E+12
Cm-242	2.1E+12	5.5E+06	2.1E+12
Cm-243	3.0E+12	6.8E+06	3.0E+12
Cm-244	3.3E+14	8.2E+08	3.3E+14
Cm-245	3.9E+11	5.8E+05	3.9E+11
Cm-246	8.9E+10	1.3E+05	8.9E+10
Cf-250	5.3E+07	3.4E+01	5.3E+07
Total α	**1.4E+15**	**4.3E+09**	**1.4E+15**
Total βγ	**1.4E+16**	**1.4E+13**	**1.4E+16**
Total Activity	**1.5E+16**	**1.4E+13**	**1.5E+16**
Stable Elements			
Element	**[Mol/Canister]**	**[Mol/Canister]**	**[Mol/Canister]**
Ca	1.7E+00	3.2E-01	2.0E+00
Fe	1.6E+00	1.1E+03	1.1E+03
Co	2.5E-02	1.0E+00	1.0E+00
Ni	7.2E-01	3.8E+02	3.8E+02
Se	1.3E+00	3.6E-06	1.3E+00
Sr	7.4E+00	1.7E-03	7.4E+00
Zr	7.0E+01	4.6E+03	4.6E+03
Nb	1.8E-02	5.8E+00	5.9E+00
Pd	3.0E+01	1.5E-04	3.0E+01
Ag	2.1E+00	1.3E-05	2.1E+00
Cd	3.0E+00	1.9E-03	3.0E+00
Sn	1.0E+00	5.4E+01	5.5E+01
Sb	3.0E-01	3.6E-02	3.3E-01
Sm	7.0E+00	2.7E-05	7.0E+00
Eu	2.0E+00	8.1E-06	2.0E+00
Ho	7.9E-03	3.5E-08	7.9E-03

Tab. C.7-3: (Cont.)

Stable Elements			
Element	**[Mol/Canister]**	**[Mol/Canister]**	**[Mol/Canister]**
Hf	1.1E-02	2.3E-01	2.4E-01
Pb	8.1E-03	2.7E-01	2.8E-01

Tab. C.7-4: Decay of heat output from 40 years after spent fuel reactor discharge for BE-7 canisters

Years Inventory Activity	**0 Average [Bq/Can.]**	**3 Average [Bq/Can.]**	**10 Average [Bq/Can.]**	**30 Average [Bq/Can.]**	**100 Average [Bq/Can.]**	**300 Average [Bq/Can.]**	**1'000 Average [Bq/Can.]**	**3'000 Average [Bq/Can.]**
Total α	1.4E+15	1.3E+15	1.3E+15	1.1E+15	8.2E+14	5.2E+14	2.1E+14	8.7E+13
Total βγ	1.4E+16	1.3E+16	1.0E+16	6.1E+15	1.1E+15	4.8E+13	3.4E+13	3.1E+13
Total Activity	1.5E+16	1.4E+16	1.2E+16	7.3E+15	2.0E+15	5.7E+14	2.5E+14	1.2E+14
Heat output	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**
Total Heat	2.1E+03	2.0E+03	1.8E+03	1.4E+03	8.1E+02	4.6E+02	1.9E+02	7.4E+01
Years Inventory Activity		**1.0E+04 Average [Bq/Can.]**	**3.0E+04 Average [Bq/Can.]**	**1.0E+05 Average [Bq/Can.]**	**3.0E+05 Average [Bq/Can.]**	**1.0E+06 Average [Bq/Can.]**	**3.0E+06 Average [Bq/Can.]**	**1.0E+07 Average [Bq/Can.]**
Total α		4.9E+13	1.7E+13	3.6E+12	1.8E+12	1.2E+12	6.1E+11	2.1E+11
Total βγ		2.5E+13	1.4E+13	3.7E+12	2.1E+12	9.2E+11	4.5E+11	1.5E+11
Total Activity		7.5E+13	3.1E+13	7.2E+12	3.8E+12	2.1E+12	1.0E+12	3.6E+11
Heat output		**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**	**[W/Can.]**
Total Heat		4.2E+01	1.4E+01	3.2E+00	2.0E+00	1.2E+00	6.0E-01	2.0E-01