Technical Report

TR-10-13

Spent nuclear fuel for disposal in the KBS-3 repository

Svensk Kärnbränslehantering AB

December 2010

Svensk Kärnbränslehantering AB
Swedish Nuclear Fuel
and Waste Management Co

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

ISSN 1404-0344
SKB TR-10-13

Spent nuclear fuel for disposal in the KBS-3 repository

Svensk Kärnbränslehantering AB

December 2010

Keywords: SKBdoc id 1175231, Spent nuclear fuel, Safety report, Fuel quantities, Handling, Encapsulation, Decay power, Criticality, Initial state, Radionuclide inventory.

Preface

An important part of SKB's licence application for the construction, possession and operation of the KBS-3 repository is the safety report. The safety report addresses both safety during operation of the KBS-3 repository facility (**SR-Operation**), and the long-term safety of the KBS-3 repository (**SR-Site**).

For the construction of the KBS-3 repository SKB has defined a set of production lines:

- the spent nuclear fuel,
- the canister,
- the buffer,
- the backfill,
- the closure, and
- the underground openings.

These production lines are reported in separate *Production reports*, and in addition there is a *Repository production report presenting* the common basis for the reports.

This set of reports addresses design premises, reference design, conformity of the reference design to design premises, production and the initial state, i.e. the results of the production. Thus the reports provide input to **SR-Site** concerning the characteristics of the as built KBS-3 repository and to **SR-Operation** concerning the handling of the engineered barriers and construction of underground openings.

The preparation of the set of reports has been lead and coordinated by Lena Morén with support from Roland Johansson, Karin Pers and Marie Wiborgh.

This report has been authored by Per Grahn, Lena Morén and Marie Wiborgh.

Summary

The report is included in a set of *Production reports,* presenting how the KBS-3 repository is designed, produced and inspected. The set of reports is included in the safety report for the KBS-3 repository and repository facility. The report provides input to the assessment of the long-term safety, **SR-Site** as well as to the operational safety report, **SR-Operation**.

The report presents the spent fuel to be deposited, and the requirements on the handling and selection of fuel assemblies for encapsulation that follows from that it shall be deposited in the KBS-3 repository. An overview of the handling and a simulation of the encapsulation and the resulting canisters to be deposited are presented. Finally, the initial state of the encapsulated spent nuclear fuel is given. The initial state comprises the radionuclide inventory and other data required for the assessment of the long-term safety.

Spent nuclear fuel to be deposited in the KBS-3 repository

The major part of the spent nuclear fuel to be deposited in the KBS-3 repository consists of fuel from the operation of the twelve Swedish nuclear power plants. The presented fuel quantities are based on the spent fuel stored in the interim storage facility, Clab, and on SKB's reference scenario for the operation of the power plants. The number of BWR and PWR assemblies to be deposited, their burnup and their ages the last year of operation of the last reactor to close down in the reference scenario, are presented.

There are also minor quantities of *miscellaneous fuels,* from research and the early part of the Swedish nuclear power programme to be deposited in the KBS-3 repository. The sources and amounts of these fuels are presented.

The properties of the spent nuclear fuel impact the design of the KBS-3 repository. The design of the KBS-3 repository has in turn resulted in requirements on the handling of the spent fuel. The spent nuclear fuel dimensions, enrichment, burnup and age are properties imposing requirements on the handling.

Requirements on the handling of the spent fuel

The decay power of a fuel assembly depends on its burnup, age and the mass of uranium. The decay power will influence the temperature in the final repository. Since the temperature in the buffer need to be restricted there is a maximum allowed total decay power of the assemblies in a canister.

Criticality must always be prevented in the handling of the spent fuel. The assemblies shall be selected for encapsulation with respect to their enrichment and burnup, and the design of the canister so that criticality under no circumstances can occur in the canister.

The dimensions of the largest fuel assemblies constitute design premises for the canister. The canister in turn imposes requirements on the handling of the fuel. Small assemblies shall if necessary be provided with distance devices that prevent them from moving in the channel tubes of the insert. To avoid corrosion inside the canister insert the fuel assemblies shall be dried prior to encapsulation and the air in the insert exchanged for inert gas.

With respect to the environment in the final repository there is a maximum acceptable radiation dose rate at the canister surface. Since the radiation dose rate in similarity to the decay power depends on the burnup and age of the encapsulated assemblies the maximum allowed decay power will also restrict the radiation. Both with respect to long-term safety and the radiation protection during operation it shall be verified that the radiation dose rate at the canister surface does not exceed acceptable levels.

With respect Sweden's commitments regarding non-proliferation and the control of fissile material each sealed canister shall be marked and constitute a unit for the account of nuclear material.

The handling of the spent fuel and the canisters to be deposited

The spent nuclear fuel is accepted for transportation from the nuclear power plants and is delivered to the interim storage facility. After a period of interim storage the fuel assemblies are selected for encapsulation with respect to their burnup and age so that the decay power in the canister does not exceed the acceptable level. It is verified that criticality cannot occur in the canister for the selected assemblies and that the radiation dose rate at the canister surface does not exceed the acceptable level. The spent fuel assemblies are then transferred to the encapsulation building, dried and placed in the canister. After inspection of the assembly identities the steel lid is placed on the insert and the air inside it is changed with argon. Finally, the copper lid is placed on the canister and it is sealed.

The encapsulation of the spent fuel has been simulated based on the spent nuclear fuel to be deposited, the planned operation times of SKB's facilities and the requirements on selection of fuel assemblies for encapsulation. The simulation results in the number of canisters to be deposited and the burnup of the assemblies in the canisters.

Initial state – encapsulated spent nuclear fuel

Encapsulated spent nuclear fuel comprises the spent fuel assemblies and the gases and liquids in the cavities of the canister. The initial state refers to the properties of the encapsulated fuel when the canister is finally sealed and no more handling of individual assemblies is possible.

The radionuclide inventory at the initial state is an important input to the safety assessment. The total radionuclide inventory in the final repository ultimately depends on the total energy output from the nuclear power plants. The radionuclide inventory in individual spent fuel assemblies will mainly depend on the burnup and consequently, since the decay power also depends on the burnup, the radionuclide inventory in a canister is restricted by the maximum allowed decay power.

The radionuclide inventory in the final repository has been calculated as the sum of the calculated inventories in individual assemblies, and as the sum of the inventories inventories in a set of type canisters. The type canisters are selected based on the results from the simulation of the encapsulation to provide a representative and adequate description of the canisters' content of spent nuclear fuel. The radionuclide inventories in the type canisters are also an input to the assessment of the operational safety.

In addition to the radionuclide inventory the propensity for criticality decay power, encapsulated gases and liquids, the radiation at the canister surface and other parameters of importance for the assessment of the long-term safety are reported for the initial state.

Sammanfattning

Rapporten ingår i en grupp av *Produktionsrapporter* som redovisar hur KBS-3-förvaret är utformat, producerat och kontrollerat. Gruppen av rapporter ingår i säkerhetsredovisningen för KBS-3-förvaret och förvarsanläggningen. Rapporten levererar indata till analysen av den långsiktiga säkerheten, **SR-Site,** samt till redovisningen av driftsäkerheten, **SR-Drift**.

Rapporten redovisar det använda kärnbränsle som ska deponeras, och kraven på hantering och val av bränsleelement för inkapsling som följer av att det ska deponeras i KBS-3-förvaret. En översikt av hanteringen och en simulering av inkapslingen och det resulterande antalet kapslar som ska deponeras redovisas. Slutligen redovisas initialtillståndet för det inkapslade använda kärnbränslet. Initialtillståndet omfattar radionuklidinventariet och andra data som behövs för analysen av den långsiktiga säkerheten.

Använt kärnbränsle som ska deponeras i KBS-3-förvaret

Det använda kärnbränsle som ska deponeras i KBS-3-förvaret utgörs till största delen av bränsle från driften av de tolv svenska kärnkraftverken. De redovisade bränslemängderna är baserade på det använda kärnbränsle som lagras i mellanlagtret, Clab, och på SKB:s referensscenario för driften av kärnkraftverken. Antalet BWR och PWR-element som ska deponeras, deras utbränning och deras ålder det sista driftåret för den reaktor som stängs sist i referensscenariot redovisas.

Det finns också mindre mängder av udda bränslen, från forskning och den tidiga delen av det svenska kärnkraftsprogrammet, som ska deponeras i KBS-3-förvaret. Källor till och mängder av dessa bränslen redovisas.

Egenskaperna hos det använda kärnbränslet påverkar utformningen av KBS-3-förvaret. Utformningen av KBS-3-förvaret har i sin tur resulterat i krav på hanteringen av det använda bränslet. Det använda kärnbränslets mått, anrikning, utbränning och ålder medför krav på hanteringen.

Krav på hanteringen av det använda kärnbränslet

Resteffekten hos ett bränsleelement beror på dess utbränning, ålder och vikt uran. Resteffekten påverkar temperaturen i slutförvaret. Eftersom temperaturen i bufferten behöver begränsas finns det en maximalt tillåten total resteffekt hos elementen i en kapsel.

Kriticitet måste alltid förhindras vid hanteringen av det använda bränslet. Bränsleelementen ska väljas för inkapsling med hänsyn till sin anrikning och utbränning, och med hänsyn till kapselns utformning, så att kriticitet under inga omständigheter kan uppstå i kapseln.

Bränsleelementens mått utgör konstruktionsförutsättningar för kapseln. Kapseln ställer i sin tur krav på hanteringen av bränslet. Små element ska om nödvändigt förses med distansklossar som hindrar dem från att röra sig i insatsens bränslekanaler. För att undvika korrosion inuti kapselns insats ska bränsleelementen torkas innan de kapslas in och luften i insatsen ska bytas ut mot inert gas.

Med hänsyn till miljön i slutförvaret finns en högsta tillåten strålningsdosrat på kapselytan. Eftersom strålningsdosraten i likhet med resteffekten beror av det inkapslade elementens utbränning och ålder kommer den maximalt tillåtna resteffekten också att begränsa strålningen. Både med hänsyn till långsiktig säkerhet och strålskydd under drift ska det verifieras att strålningsdosraten på kapselytan inte överskrider tillåtna nivåer.

Med hänsyn till Sveriges åtaganden då det gäller icke-spridning och kontroll av klyvbart material ska varje försluten kapsel vara märkt och utgöra en enhet i redovisningen av kärnämne.

Hanteringen av det använda kärnbränslet och de kapslar som ska deponeras

Det använda kärnbränslet godkänns för transport från kärnkraftverken och levereras till mellanlagret. Efter en tids mellanlagring av bränsleelementen väljs de ut för inkapsling med hänsyn till sin utbränning och ålder så att resteffekten i kapseln inte överskrider den tillåtna nivån. Det verifieras att kriticitet inte kan uppstå i kapseln för de valda elementen och att strålningsdosraten på kapselytan inte överskrider den tillåtna nivån. De använda kärnbränsleelementen förs sedan över till inkapslingsbyggnaden, torkas och placeras i kapseln. Efter kontroll av elementens identitet placeras stållocket på insatsen och luften i instasen byts mot argon. Slutligen placeras kopparlocket på kapseln och den försluts.

Inkapslingen av det använda kärnbränslet har simulerats utifrån det använda kärnbränsle som ska deponeras, den planerade driften av SKB:s anläggningar och kraven på val av bränsleelement för inkapsling. Simuleringen resulterar i det antal kapslar som ska deponeras och utbränningen hos elementen i kapslarna.

Initialtillstånd – inkapslat använt kärnbränsle

Inkapslat använt kärnbränsle omfattar de använda bränsleelementen och gaserna och vätskorna i kapselns hålrum. Initialtillståndet avser det inkapslade bränslets egenskaper då kapseln slutligen försluts och ingen mer hantering av enskilda element är möjlig.

Radionuklidinventariet vid initialtillståndet utgör viktig indata till säkerhetsanalysen. Det totala radionuklidinventariet i slutförvaret beror ytterst av den totala energiproduktionen i kärnkraftverken. Radionuklidinventariet i enskilda element beror huvudsakligen på utbränningen och följaktligen, eftersom resteffekten också beror av utbränningen, begränsas radionuklidinnehållet i en kapsel av den maximalt tillåtna resteffekten.

Radionuklidinventariet i slutförvaret har beräknats som summan av de beräknade inventarierna i enskilda element, och som summan av inventarierna i en uppsättning typkapslar. Typkapslarna har valts baserat på resultaten från simuleringen av inkapslingen, för att ge en representativ och adekvat beskrivning av kapslarnas innehåll av använt kärnbränsle. Radionuklidinventariet i typkapslarna utgör också indata till analysen av driftsäkerheten.

Utöver radionuklidinventariet redovisas benägenheten för kriticitet, resteffekt, inkapslade gaser och vätskor, strålningen på kapselns yta och andra parametrar med betydelse för analysen av den långsiktiga säkerheten för initialtillståndet.

Contents

1 Introduction

1.1 General basis

1.1.1 This report

This report describes the spent nuclear fuel to be deposited in the KBS-3 repository and the handling of the spent fuel within the KBS-3 system. It is included in a set of reports presenting how the KBS-3 repository is designed, produced and inspected. The set of reports is denominated *Production reports*. The Production reports and their short names used as references within the set are illustrated in Figure 1-1. The reports within the set referred to in this report and their full names are presented in Table 1-1.

This report is part of the safety report for the KBS-3 repository and repository facility, see **Repository production report**, Section 1.2. It is based on the results and review of the most recent long-term safety assessment and the current knowledge, technology and results from research and development.

1.1.2 The spent nuclear fuel to be deposited

All spent nuclear fuel from the currently approved Swedish nuclear power programme shall be deposited in the KBS-3 repository. The presented amounts and types of fuel to be deposited are based on the spent fuel already accumulated and in interim storage and a prognosis of the amounts and types of spent fuel to be generated in a scenario for the future operation of the nuclear power plants.

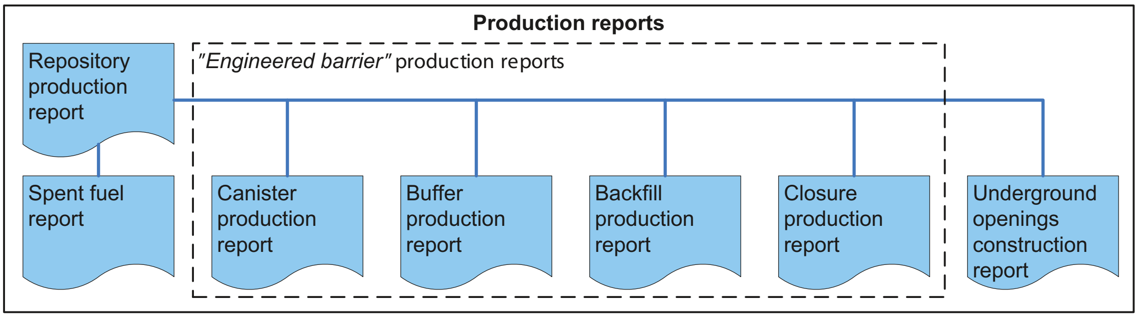


Figure 1-1. *The reports included in the set of reports describing how the KBS-3 repository is designed, produced and inspected.*

Table 1-1. The reports within the set of Production reports referred to in this report.

Full title	Short name used within the Production reports	Text in reference lists
Design and production of the KBS-3 repository	Repository production report	**Repository production report, SKB 2010.** Design and production of the KBS-3 repository. SKB TR-10-12, Svensk Kärnbränslehantering AB.
Design, production and initial state of the canister	Canister production report	**Canister production report, SKB 2010.** Design, production and initial state of the canister. SKB TR-10-14, Svensk Kärnbränslehantering AB.
Design, construction and initial state of the underground openings	Underground openings construction report	**Underground openings construction report, SKB 2010.** Design, construction and initial state of the underground openings. SKB TR-10-18, Svensk Kärnbränslehantering AB.

1.1.3 The handling of the spent nuclear fuel

The presented handling of the spent fuel is based on the assumption that there is a system, the KBS-3 system, comprising the facilities required to manage the spent nuclear fuel and finally deposit it in a KBS-3 repository. The KBS-3 system and its facilities are presented in Chapter 4 in the **Repository production report**.

SKB considers that the methods to handle the spent fuel and inspect its properties presented in this report are technically feasible. Methods for handling and inspection may, however, be further developed before the encapsulation and disposal of the fuel commences.

1.2 Purpose, objectives and delimitations

1.2.1 Purpose

The purpose of this report is to present the spent nuclear fuel to be disposed in the KBS-3 repository. The report shall provide the information on the encapsulated spent fuel required for the long-term safety report, **SR-Site,** as well as for the operational safety report, **SR-Operation**.

With this report SKB intends to present the requirements on the handling of the spent fuel for its disposal in the KBS-3 repository and how it can be handled in conformity to these requirements. The report shall present the handling and inspection and summarise the efforts that supports that the spent fuel is handled in conformity to the stated requirements.

1.2.2 Objectives

Based on the above purpose the objectives of this report are to describe:

- spent fuel types and quantities to be deposited in the KBS-3 repository,
- fuel properties and parameters of importance for the assessment of the long-term safety,
- how the fuel assemblies are handled, inspected and selected for encapsulation and deposition,
- the initial state of the spent fuel, i.e. the expected values of parameters of importance for the assessment of the long-term safety of the encapsulated spent nuclear fuel.

1.2.3 Limitations

The **Spent fuel report** is based on a reference scenario for the future operation of the nuclear power plants and also includes the spent fuel that is stored in the interim storage facility. Alternative scenarios for the operation of the nuclear power plants are not included.

The **Spent fuel report** includes data about the spent fuel that are required to assess the safety of the KBS-3 repository and repository facility. Other fuel data of interest for SKB will be documented elsewhere.

The **Spent fuel report** only includes the inspections performed within the KBS-3 system that are required to inspect parameters of importance for the assessment of the safety of the KBS-3 repository and repository facility. Inspections that are performed at the nuclear power plants, or with respect to safeguards of nuclear materials, or operational safety of the other facilities or transport system within the KBS-3 system are reported elsewhere.

1.3 Interfaces to other reports included in the safety report

The role of the production reports in the safety report is presented in Section 1.2 in the **Repository production report**. A summary of the interfaces to other reports included in the safety report is given below.

1.3.1 The safety report for the long-term safety

By providing a basic understanding of the repository performance for different time periods and by the identification of scenarios that can be shown to be especially important from the standpoint of risk, the long-term safety assessment provides feedback to the allocation of spent fuel assemblies to canisters, and the design of the engineered barriers and underground openings. The methodology used for deriving design premises from the long-term safety assessment is introduced in the **Repository production report**, Section 2.5.2. A more thorough description as well as the resulting design premises are given in the report "Design premises for a KBS-3V repository based on results from the safety assessment SR-Can and some subsequent analyses" /SKB 2009/, hereinafter referred to as **Design premises long-term safety.** These design premises constitute an input to the required handling of the spent nuclear fuel.

As stated in Section 1.2 this report provides information about the initial state of the encapsulated spent nuclear fuel required for the long-term safety assessment. This report also provides the data concerning the initial state used in the calculations included in the long-term safety assessment.

1.3.2 The safety report for the operational safety

The spent fuel line includes the selection of fuel assemblies to be encapsulated and ends in the encapsulation plant when the fuel is put in the canister. The general description of the facility and its main activities given in Chapter 5 in the Safety Report for the central interim storage and encapsulation plant, Clink, constitute input to this report.

The objectives for the operational safety and radiation protection in the final repository facility given in Chapter 3 in the safety report **SR-Operation** constitute input to this report.

The radioactivity of the encapsulated spent fuel is also of importance for the control of radioactive substances and radiation protection in the repository facility, and constitutes input to Chapters 6 and 7 in **SR-Operation**.

1.3.3 The other production reports

The **Repository production report** presents the context of the set of Production reports and their role within the safety report. It also includes definitions of some central concepts of importance for the understanding of the Production reports.

The **Repository production report** provides an overview of the KBS-3 system and the facilities required to manage and deposit the spent nuclear fuel. It also points out the properties of the spent nuclear fuel of importance for the design of the KBS-3 repository and how the design of the KBS-3 repository will impact the handling of the spent fuel.

The design premises the spent fuel imposes on the design and construction of the engineered barriers and underground openings are presented in this report. These design premises are repeated and verified in the production reports for the engineered barriers and underground openings on which the spent nuclear fuel imposes design premises.

The handling of the spent nuclear fuel and the canister production line intersect in the Clink facility. The selection and inspection of spent fuel assemblies for each canister, and the activities taking place before the canister is sealed are described in this report. The sealing of the canister and all activities taking place after the sealing are presented in the **Canister production report**.

1.4 Structure and content

The **Spent fuel report** contains following chapters.

- **Spent nuclear fuel to be deposited in the KBS-3 repository**
  Including information about fuel types and quantities and the fuel parameters of importance for the assessment of the long-term safety.
- **Requirements on the handling of spent nuclear fuel**
  Including the requirements from the KBS-3 repository and its engineered barriers on the handling and selection of fuel assemblies for encapsulation and the design premises imposed on the engineered barriers by the spent fuel.
- **The handling of the spent nuclear fuel**
  Including the activities included in the spent fuel line and the inspections and verifications of the fuel parameters of importance for the safety report.
- **The canisters to be deposited**
  The number of canisters to be deposited as a result of the planned handling and encapsulation.
- **Initial state – encapsulated spent nuclear fuel**
  Including the expected fuel parameters of importance for the assessment of the long-term safety.

In addition to this an introduction to the report is given in this chapter and in Appendix D abbreviations and branch terms used in this report are explained.

2 Spent nuclear fuel to be deposited in the KBS-3 repository

2.1 Types and sources of spent nuclear fuel

The major part of the nuclear fuel to be deposited in the KBS-3 repository consists of spent fuel from the operation of the twelve Swedish nuclear power plants. The quantities and properties of the spent fuel from the twelve Swedish nuclear power plants are based on the scenario for their operation presented in Section 2.1.1.

There are also minor quantities of other fuel types from research and the early part of the nuclear power programme to be deposited in the KBS-3 repository. These fuels are in the following referred to as *miscellaneous fuels*. A background to the sources of these fuels is presented in Section 2.1.2.

2.1.1 Reference scenario for the operation of the nuclear power plants

In this section the reference scenario used by SKB for the operation of the nuclear power plants is presented. This scenario is the basis for the reference design of the final repository and the analyses within SR-Site.

The Swedish nuclear power plants are of the type boiling light water reactors (BWR) and pressurised light water reactors (PWR). In total there are 12 reactors; two BWR at Barsebäck (B1 and B2), three BWR at Forsmark (F1, F2 and F3), one BWR (R1) and three PWR at Ringhals (R2, R3 and R4) and three BWR at Oskarshamn (O1, O2 and O3).

The maximum enrichment of the spent nuclear fuel to be deposited in the KBS-3 repository is set to 5% and the average assembly burnup is limited to 60 MWd/kg U for the uranium oxide fuel (UOX fuel) from the BWR and PWR plants and 50 MWd/kg HM[1] for the mixed oxide fuel (MOX) BWR fuel /SKB 2008/.

The B1 and B2 reactors in Barsebäck are closed down and had been in operation for approximately 24 years and 28 years, respectively, when they were taken out of operation. The operating times are set to 50 years for the reactors at Ringhals and Forsmark and 60 years for the reactors at Oskarshamn. In the reference scenario the last reactor will be taken out of operation in 2045. In the scenario for the future operation of the power plants the thermal powers of the reactors given in Table 2-1 are assumed. The table also includes the reactor powers of the closed-down reactors in Barsebäck.

Table 2-1. Thermal powers of the reactors and last year of operation for the 12 Swedish nuclear power plants as assumed in the reference scenario used by SKB. Based on information provided by the nuclear power plant operators.

Reactor	Reactor power (MW_{th})	Increases in reactor power (MW_{th}) 2009	2010	2011	2012	Last year of operation
B1	1,800					1999
B2	1,800					2005
F1	2,928		3,255			2030
F2	2,928	3,255				2031
F3	3,300			3,775		2035
O1	1,375					2032
O2	1,800				2,300	2034
O3	3,300	3,900				2045
R1	2,540					2025
R2	2,652					2025
R3	2,992	3,144				2031
R4	2,775				3,300	2033

[1] Heavy metal i.e. uranium and plutonium.

In order to estimate the number of spent fuel assemblies generated as a result of the future operation of the nuclear power plants the planned target burnup of the fuel assemblies as well as the thermal power must be known. The batch average discharge burnup for the ten remaining Swedish nuclear power plants that are assumed in the reference scenario used by SKB are presented in SKBdoc 1219727, ver 2.0. (Confidential information. Available only for the Swedish Radiation Safety Authority.) The information is based on data provided by the nuclear power plants. It is secret according to The Swedish Competition Act (2008:579) and can be shown to SSM representatives on request.

The majority of the fuel used in the reactors consists of uranium oxide fuel (UOX). From Oskarshamn, there will be minor amounts of mixed oxide fuel (MOX), the background to this is given in Section 2.1.2.

2.1.2 Miscellaneous fuels – background and sources

In the early part of the Swedish nuclear power programme some spent nuclear fuel was reprocessed. In addition to the nuclear power plants accounted for in Section 2.1.1 there has also been a reactor for heat production and a research reactor in Sweden. There are also small amounts of fuel residues from investigations at the Studsvik facility. As a result there will be some MOX fuel and other spent fuels to be deposited in the KBS-3 repository.

Some spent fuel from Ringhals and Barsebäck was sent to La Hague for reprocessing. This spent fuel was exchanged in 1986 for spent German MOX fuel in equivalent amounts of plutonium. This fuel is referred to as "Swap MOX", and is presently stored in Clab. The amounts are included in the miscellaneous fuels presented in Section 2.2.2. Some spent fuel from Oskarshamn was sent to Sellafield for reprocessing. The plutonium resulting from this reprocessing will be used to manufacture MOX fuel to be used in one of the Oskarshamn reactors. This fuel is included in the spent fuel reference scenario presented in Section 2.1.1.

There was a pressurised heavy water reactor (PHRW) in Ågesta in operation during 1963–1974. The reactor supplied a suburb in Stockholm with district heating but also small amounts of electricity. The reactor used natural uranium in the first core and in a second core some slightly enriched uranium as fuel and heavy water as coolant. Some fuel from the first core was reprocessed. The remaining spent fuel is presently stored in Clab and the amounts are included in the miscellaneous fuels presented in Section 2.2.2.

The spent metal fuel from the research reactor at the Royal Institute of Technology in Stockholm, which was in operation during 1954–1970, has been sent to Sellafield for reprocessing. The plutonium from the reprocessing will be used to manufacture MOX fuel that will be used in the BWR reactors at Oskarshamn. A minor amount of the metal fuel is oxidised and will be disposed together with the fuel residues from fuel investigations in Studsvik.

Investigations of spent fuel performed at the Studsvik laboratory result in small amounts of fuel residues. The fuel residues are fixed in different matrices such as epoxy resin, glass, brass and then put in boxes with the dimensions of a PWR fuel assembly.

2.2 Fuel quantities, burnup and age of the spent nuclear fuel

2.2.1 Spent fuel from the reference scenario for operation of the nuclear power plants

The presented fuel quantities and burnup are based on the fuel stored in the interim storage facility Clab and the reference scenario for the future operation of the power plants presented in Section 2.1.1. The total quantity of spent fuel will depend on the energy output of the reactors, which in turn depends on the operating time and power of the reactors. The energy output is provided by the fuel assemblies in the reactor core. For a given energy output the exchange rate, and number of fuel assemblies, can be kept low if the enrichment and burnup is increased. If assemblies with lower enrichment and burnup are used the number of fuel assemblies will increase.

The number of fuel assemblies discharged from the Swedish nuclear power plants stored in the interim storage facility at the end of 2007, and the estimated total number of assemblies for the reference scenario used by SKB are given in Table 2-2 together with the corresponding estimated amount of uranium or heavy metal expressed as tonnes initial weight. As previously mentioned the major part of the spent fuel assemblies from the nuclear power plants will consist of UOX fuel. Only one of the reactors in Oskarshamn will use a minor amount of MOX fuel assemblies. The BWR assemblies accounted for in Table 2-2 also include rod cassettes which consist of fuel rods that have been dismantled for various investigations and placed in fuel rod cassettes. Approximately one out of four of the PWR assemblies will contain a control rods cluster. A small amount of the assemblies, both BWR and PWR, will contain inserts such as start-up neutron sources, boron glass rods and plugs.

Since the beginning of 1970 the burnup of the nuclear fuel has increased from approximately 23 MWd/kgU up to 53 MWd/kgU. The average burnup of the spent nuclear fuel stored in the interim storage facility is about 34 and 41 MWd/kgU for BWR and PWR fuel assemblies respectively (December 2007). For the remaining operation, the burnup will increase as a result of increased power and optimisation of the operation of the reactors. If the enrichment is increased the burnup will also be increased. Alternatively, if the enrichment is not increased, the fuel assemblies must be switched more often to achieve the same energy output. The resulting average burnup for the reference scenario is 40.4 and 44.8 MWd/kgU for BWR and PWR fuel assemblies respectively /SKBdoc 1221579/. The burnup distribution for the BWR and PWR spent fuel assemblies stored in the interim storage facility, Clab (31 December 2007) and the prognosis for the reference scenario is shown in Figure 2-1.

To provide an overview of the distribution among the total quantities of BWR and PWR spent fuel to be finally deposited in the KBS-3 repository the information from Figure 2-1 is expressed in Figure 2-2 as tonnes of heavy metal.

Throughout the operation time of the nuclear reactors fuels from different suppliers have been used and the operation efficiency improved. As a result, there are a number of specific fuel types with different enrichment, burnup and detailed design. A list of different kinds of fuel used in Swedish reactors is provided in Appendix A.

In Figure 2-3 the BWR assemblies stored in Clab at the end of 2007 and the assemblies included in the reference scenario and their burnup and age are plotted for 2045, i.e. the last year of operation of the last reactor to close down. In Figure 2-4 the corresponding information about the PWR assemblies is shown. For the assemblies included in the reference scenario large red dots represent the batch average discharge burnup. The smaller red dots represent the assumed standard deviation in burnup, i.e. ±3 MWd/kgU, averaged over single fuel assemblies included in a batch. Each dot represents several assemblies. Low burnup assemblies in the reference scenario are from the last year of operation of the nuclear power plants.

Table 2-2. Spent fuel from operation of the nuclear power plants stored in the interim storage facility, Clab, and total amounts estimated for the SKB reference scenario. The information about fuel stored in Clab is based on Clab's safeguards accountancy system. The total number is based on the reference scenario presented in Section 2.1.1.

Fuel type	Number in interim storage 31 December 2007	Total number for SKB reference scenario /SKBdoc 1221567/	Total initial weight for the reference scenario (tonnes of U or HM)
BWR assemblies from operation of the NPP at Barsebäck, Oskarshamn, Forsmark and Ringhals	21,194[1,2]	47,498[3,4]	8,312[5]
PWR assemblies from operation of the NPP at Ringhals	2,552	6,016	2,791[6]

[1] The fuel channels, see Figure 2-5, have been removed from 1,520 of the BWR assemblies.

[2] Including 3 BWR MOX assemblies stored at O1.

[3] Including 83 BWR MOX assemblies from the reactors in Oskarshamn.

[4] Including rod cassettes, i.e. dismounted fuel rods placed in fuel rod cassettes.

[5] Assumed 175 kg U/HM per assembly.

[6] Assumed 464 kg U per assembly.

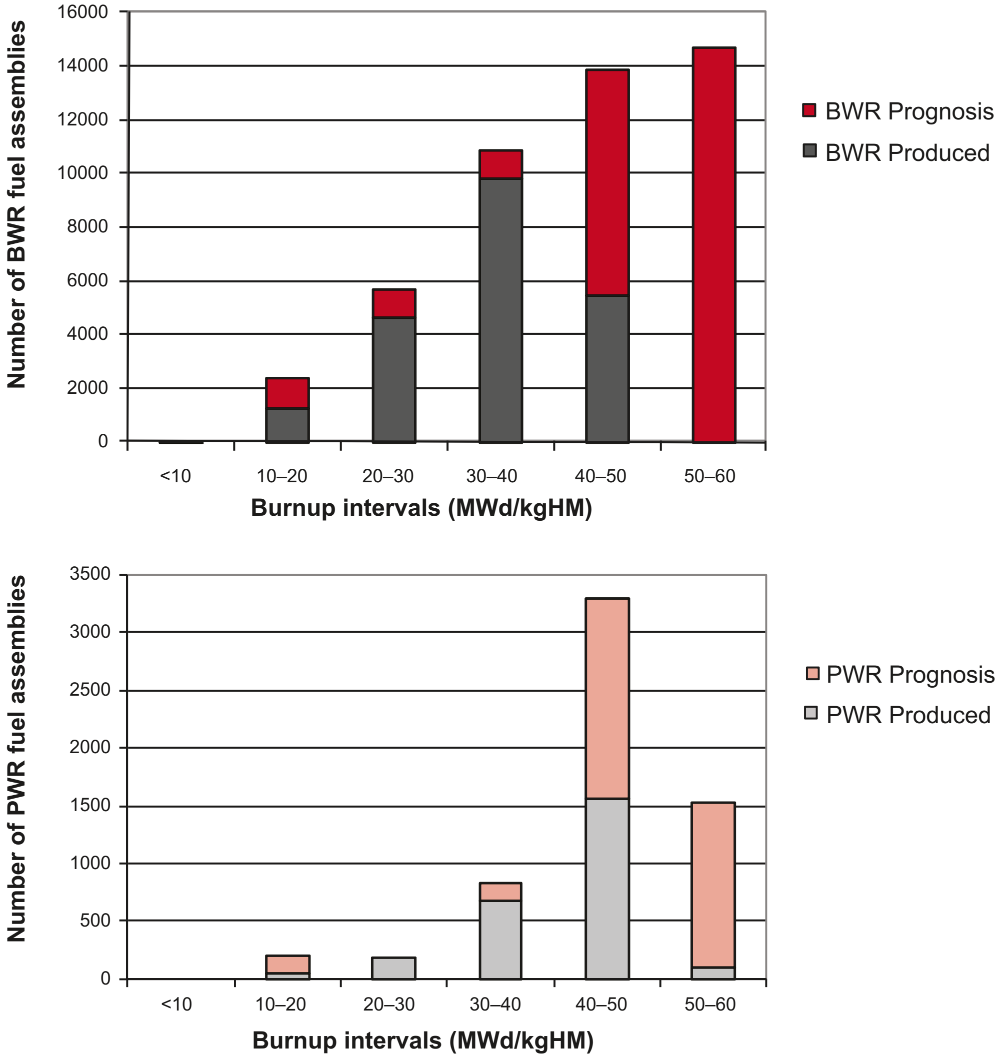


Figure 2-1. *Burnup distribution of the spent fuel assemblies stored in Clab 31 December 2007 (grey colours) and prognosis of the burnup distribution for the assemblies resulting from the future operation of the nuclear power plants (red colours) as assumed for the reference scenario used by SKB.*

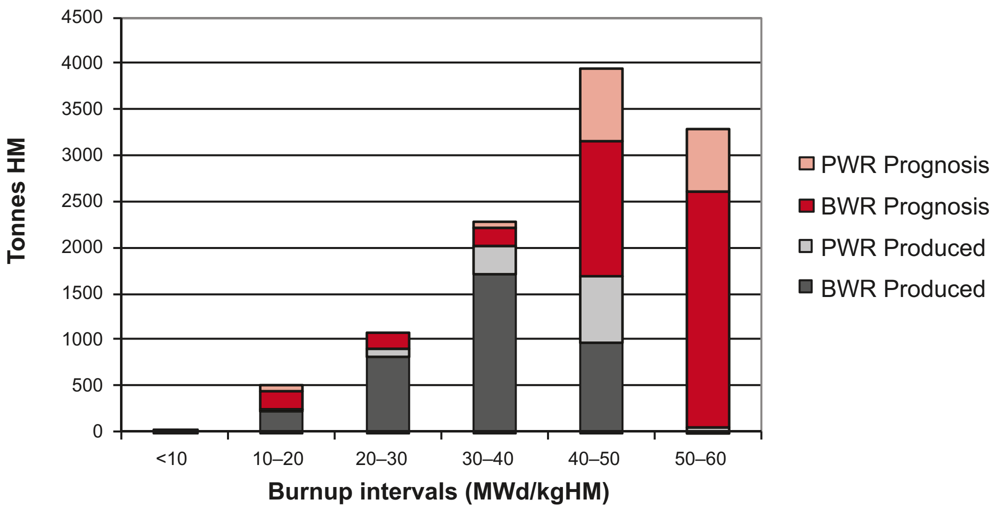


Figure 2-2. *Average burnup distribution per tonne heavy metal of the BWR and PWR fuel stored in Clab 31 December 2007 (grey colours) and prognosis of the burnup distribution for the future operation of the nuclear power plants (red colours) as assumed for the reference scenario used by SKB.*

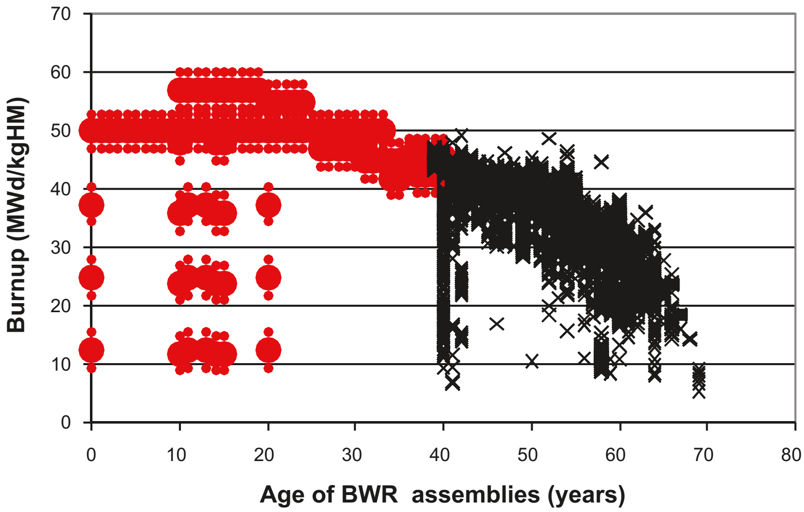


Figure 2-3. The burnup and age in 2045 of the BWR assemblies included in the SKB reference scenario. Assemblies that are currently stored in Clab are marked with black and assemblies from the future operation are marked with red. Large red dots represent batch average discharge burnup and small red dots represent standard deviation in burnup, i.e. ±3 MWd/kgU, averaged over single fuel assemblies included in a batch.

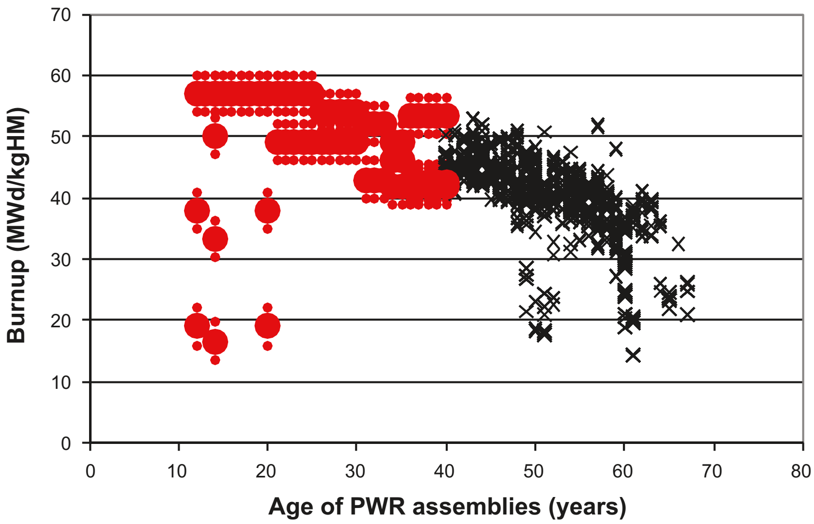


Figure 2-4. The burnup and age in 2045 of the PWR assemblies included in the reference scenario. Assemblies that are currently stored in Clab are marked with black and assemblies from the future operation are marked with red. Large red dots represent batch average discharge burnup and small red dots represent standard deviation in burnup, i.e. ±3 MWd/kgU, averaged over single fuel assemblies included in a batch.

2.2.2 Miscellaneous fuels

The amount of miscellaneous fuels and, if applicable, the number of fuel assemblies in the interim storage facility as well as the estimated total amount of uranium or heavy metal, expressed as tonnes initial weight, is presented in Table 2-3.

Table 2-3. Miscellaneous fuels stored in the interim storage facility, Clab, and total amounts estimated for the SKB reference scenario. Information about fuel stored in Clab is based on Clab's safeguards accountancy system.

Fuel type	Number in interim storage 31 December 2007	Total number for the SKB reference scenario	Total initial weight for the reference scenario (tonnes of U or HM)
Fuel assemblies from Ågesta	222 (1 unirradiated)	222 (1 unirradiated)	20
Swap MOX assemblies (BWR)	184	184	14.1
Swap MOX assemblies (PWR)	33	33	8.4
Fuel residues in special boxes from Studsvik	19	Approximately 25	3
Damaged fuel in protection boxes	0	–	–

The enrichments and burnup of the minor quantities of miscellaneous spent fuel to be deposited can be summarised as follows.

- **Ågesta PHWR fuel**
  The majority of the fuel assemblies (about 70%) contain natural uranium and the remainder are enriched to 1.35%, except one assembly that contains 2.2% U-235. The burnup ranges from 0 to 10 MWd/kg U.
- **Swap MOX fuel**
  The Swap MOX fuel has a higher transuranic content than UOX fuel. The average burnup is 16 MWd/kg HM for the BWR assemblies and 31 MWd/kg HM for the PWR assemblies.
- **Fuel residues from Studsvik**
  The fuel residues from the performed investigations at Studsvik are emplaced in special boxes with 125 kg U per box. The box has the same dimensions as a PWR fuel assembly. The enrichment varies from low up to about 20% with an average of about 3% U-235 in each box.
- **Damaged fuel**
  Fuel assemblies damaged so that it is possible that material may fall off are emplaced in protection boxes during storage. Currently there is no such fuel stored in Clab. However, there are assemblies with leaking rods in Clab. Further, some damaged fuel assemblies are currently stored at the nuclear power plants.

2.3 Fuel parameters of importance for repository design and long-term safety

As discussed in the **Repository production report**, Section 2.3, there are some properties of the spent fuel that are of importance for the design of the engineered barriers and the layout of the repository. There are also some properties that need to be known or predicted with respect to the analyses performed in the safety assessment. The design of the KBS-3 repository has in turn resulted in requirements on the selection of fuel assemblies for encapsulation in each canister and on the handling of the spent fuel before encapsulation.

In this section, the fuel parameters that need to be documented with respect to the design of the final repository or for the analyses to be performed in the long-term safety assessment are presented. Where relevant, references are given to the sections in which requirements on the handling are presented.

In addition to the parameters documented in the handling, the very low solubility of the fuel matrix is of importance for the long-term safety. The spent fuels from the approved Swedish nuclear power programme as well as the various miscellaneous fuels are in the oxide form – UOX or MOX. The fuel matrix in such fuels has very low solubility in a KBS-3 repository environment. This is an essential property in some scenarios of the long-term safety assessment.

2.3.1 Enrichment, burnup, irradiation and power history and age of the spent fuel assemblies

The radionuclide inventory is an essential input to the dose calculations in the assessment of the long-term safety. The burnup data forms the basis for calculations of the radioactivity and radionuclide inventory of the fuel assemblies. The radionuclide inventory and its distribution in the spent fuel matrix are also affected by the irradiation and power histories of the assemblies. The irradiation history will affect the amount of activation products formed in the construction materials of the fuel assemblies as a result of the neutron radiation in the reactor. The power history, which is the heat power developed per unit length of fuel rod or fuel assembly over the irradiation period in the reactor vessel, is strongly correlated to the fission gas release (FGR). The FGR reflects the part of the radionuclide inventory located at the fuel grain boundaries, fractures in the fuel pellets and other gaps within the fuel cladding. The fuel parameters of importance for the radionuclide inventory and its distribution in the fuel matrix are further discussed in Section 6.2.1.

The radioactivity is the source of the heat generation in the spent fuel. The burnup and age of the spent fuel assemblies will determine the activity content and, thus, the decay power of the spent fuel. The design of the repository and the assessment of the evolution of the barriers are based on a maximum decay power in each canister at the time of emplacement.

The radioactivity is also the source of radiation from the spent fuel. The radiation on the canister surface can cause formation of nitric acid from moist air and corrosive species from radiolysis of water. Both these processes can increase the corrosion of the copper canister. The radiation will depend on the burnup and age of the spent fuel assemblies.

Criticality depends on the amount and geometrical configuration of the fissile material and the substances and materials surrounding it. The amount of fissile material in the spent fuel depends on the fuel type, the enrichment and burnup.

Requirements on the repository design and handling with respect to enrichment, burnup and age of the fuel assemblies are presented in Sections 3.1.1 and 3.1.2.

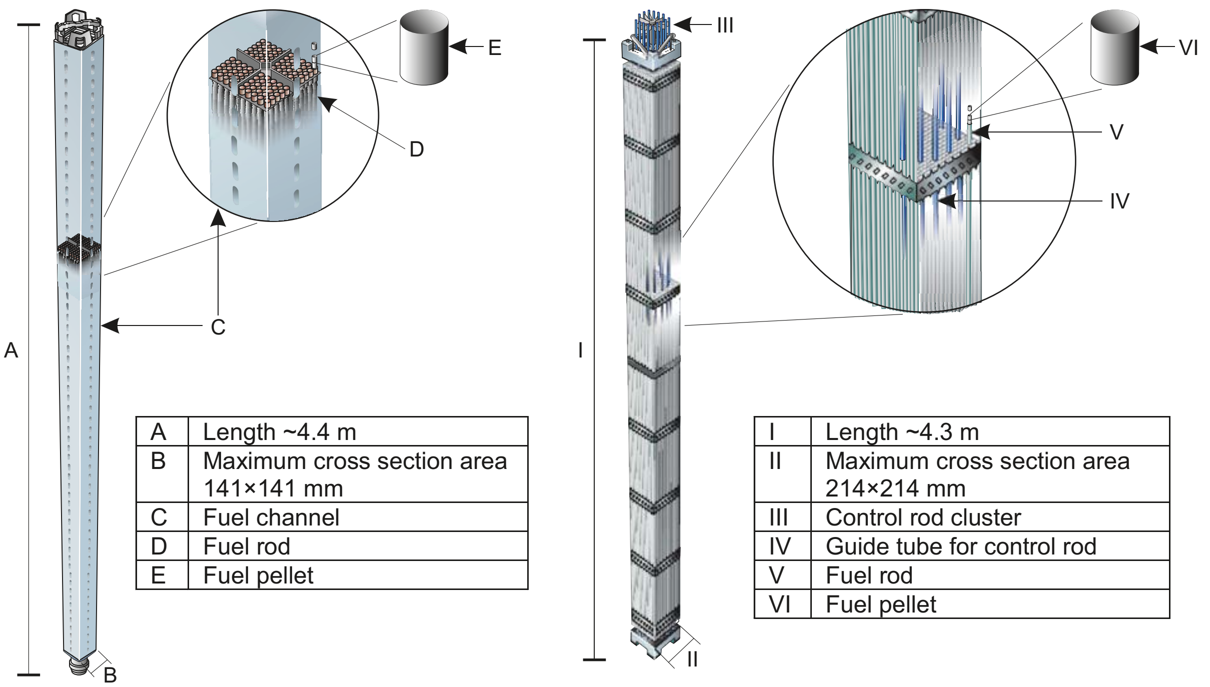
2.3.2 Dimensions and materials

The dimensions of the fuel assemblies will affect the dimensions of the canister. The BWR fuel assemblies contain about 60 up to 100 fuel rods. The fuel rods consist of zirconium alloy tubes filled with cylindrical fuel pellets. The rods are arranged in square arrays enclosed in a fuel channel. The cross-sectional area of the fuel assemblies is about 0.141×0.141 m^2 and the total length can be up to about 4.4 m. The PWR fuel assemblies contain 204 or 264 fuel rods, arranged in square arrays. The cross-sectional area is about 0.214×0.214 m^2 and the total length is about 4.3 m.

A BWR and a PWR assembly are illustrated in Figure 2-5. As previously mentioned, the detailed designs of the fuel assemblies have been altered as a result of optimisation of the operation of the nuclear power plants and utilisation of the uranium. Detailed information on dimensions for different BWR and PWR fuel types can be found in Appendix A. During the irradiation in the reactor the dimensions of the assemblies may be altered so that they deviate from the specified.

Requirements on the design of the canister and on the handling of the fuel assemblies with respect to their dimensions are given in Section 3.1.3.

The material compositions of the fuel assemblies are used as premises in the analysis of the long-term safety. An overview of the materials in typical BWR and PWR fuel assemblies is given in Table 2-4. More detailed information is provided in Appendix B.



A	Length ~4.4 m
B	Maximum cross section area 141×141 mm
C	Fuel channel
D	Fuel rod
E	Fuel pellet

I	Length ~4.3 m
II	Maximum cross section area 214×214 mm
III	Control rod cluster
IV	Guide tube for control rod
V	Fuel rod
VI	Fuel pellet

Figure 2-5. *Arbitrary BWR (left) and PWR (right) assemblies. Data based on specifications provided by the nuclear fuel suppliers.*

Table 2-4. Overview of the materials in typical BWR and PWR fuel assemblies. Data based on specifications provided by the nuclear fuel suppliers. Control rod clusters in PWR assemblies are not included (for details, see Appendix B).

Material composition	BWR Svea 96 Optima 2	PWR Areva 17×17
	Weight in 1 fuel assembly (kg)	
Fuel		
U-tot	175	464
O	23	62
Cladding material		
Zirconium alloys	49	108
Stainless steel	–	3
Fuel channel		
Zirconium alloys	32	–
Stainless steel	8	–
Other constructions (bottom and top plate, spacers etc)		
Stainless steel	5	12
Zirconium alloys	–	21
Nickel alloys	1	2

Four elements of particular interest for the assessment of the long-term safety are nitrogen (N), chlorine (Cl), nickel (Ni) and niobium (Nb). The contents of these elements in the construction materials for a PWR Areva 17×17 assembly and a BWR Svea 96 Optima 2 assembly are given in Table 2-5. The content of these elements is based on the material compositions given in Appendix B. The variability of these elements in the different kinds of BWR and PWR assemblies to be deposited has been investigated by randomly selecting a number of fuel types and comparing the amounts of construction materials in these assemblies with the amounts in Svea 96 Optima 2 and Areva 17×17 respectively. The conclusion is that the amounts will be similar for all BWR and PWR assemblies respectively.

Table 2-5. The content of N, Cl, Ni and Nb in the construction material for typical BWR and PWR fuel assemblies.

Element	Weight in 1 fuel assembly (kg)						
	BWR Svea 96 Optima 2				PWR Areva 17×17		
	Cladding	Other	Fuel channel	Total	Cladding	Other	Total
N	$2\cdot10^{-3}$	$2\cdot10^{-3}$	$5\cdot10^{-3}$	$9\cdot10^{-3}$	$5.5\cdot10^{-3}$	$5.8\cdot10^{-3}$	$1.1\cdot10^{-2}$
Cl		$6\cdot10^{-6}$	$8\cdot10^{-6}$	$1.4\cdot10^{-5}$	$3\cdot10^{-6}$	$1.4\cdot10^{-5}$	$1.7\cdot10^{-5}$
Ni	$2\cdot10^{-2}$	1.1	0.86	1.99	0.27	2.19	2.46
Nb		$8.2\cdot10^{-3}$	$8\cdot10^{-4}$	$9\cdot10^{-3}$	1.08	$9\cdot10^{-2}$	1.17

There are no requirements on the handling with respect to the material composition of the fuel assemblies.

2.3.3 Encapsulated liquids and gases

Liquids and gases that remain in the canister when it is sealed may cause corrosion of the cast iron insert. Water will lead to anaerobic corrosion of the cast iron insert. Radiolysis will form nitric acid from water and air in the canister. Nitric acid will corrode the insert and may also cause stress corrosion cracking in areas with tensile stresses.

Requirements on the handling of the spent nuclear fuel with respect to liquids and gases are presented in Section 3.1.4 Encapsulated liquids and gases.

3 Requirements on the handling of the spent fuel

In this chapter, the requirements on the handling of the spent fuel related to its final disposal in the KBS-3 repository are presented. The chapter also contains design premises imposed on the canister by the spent nuclear fuel. The presentation has been divided into two sections:

- requirements related to repository design and long-term safety,
- requirements related to the operation of the KBS-3 system.

3.1 Requirements related to repository design and long-term safety

3.1.1 Decay power

The decay power of a fuel assembly depends on its burnup, age and the mass of heavy metal (HM). The decay power of the spent nuclear fuel will influence the temperature increase in the final repository. High temperatures will impact the properties of the engineered barriers and may affect their barrier functions. High temperatures will also generate rock stresses that may cause rock fracturing. As a consequence of this, the KBS-3 repository must be designed so that a maximum allowed temperature will not be exceeded.

In **Design premises long-term safety** the following design premise is stated for the maximum allowed temperature.

- *The buffer geometry (e.g. void spaces), water content and distances between deposition holes should be selected such that the temperature in the buffer is <100°C.*

In order to determine a repository design where the temperature stays below the maximum allowed in the buffer, the total decay power in the canisters must be known. The total decay power in a canister will depend on the burnup and age of the fuel assemblies and also on the number of assemblies in the canister. The following requirement and criterion are set for the selection of fuel assemblies for encapsulation.

- Requirement on handling: *The fuel assemblies to be encapsulated in any single canister shall be selected with respect to burnup and age so that the total decay power in the canister will not result in temperatures exceeding the maximum allowed in the buffer.*
- Criterion: *The total decay power in each canister must not exceed 1,700 W.*

The background to these requirements is discussed in the **Repository production report**, Section 2.3. Note that the allowed decay power may be altered as a result of optimisation when more detailed information on burnup, operation times and conditions at repository depth are available. It may also be possible to allow higher decay power in peripheral deposition holes.

3.1.2 Criticality

Criticality must always be prevented outside the reactor vessel, and the following design requirement is set for the canister; see **Design premises long-term safety**, Section 3.1.4.

- *The spent fuel properties and geometrical arrangement in the canister should be such that criticality is avoided even if water should enter the canister.*

In the analysis of the propensity or potential for criticality of the fuel assemblies placed in the canister, the sensitivity of canister material composition and dimensions are investigated, see Section 4.4.1.

The sensitivity analyses are based on the assumption that the insert is made of nodular cast iron with an iron content of at least 90%. The iron in the insert acts as a neutron reflector. Alloying elements occurring in nodular iron that are more potent neutron reflectors than iron are silicon (Si) and carbon (C). In the analysis it was concluded that the content of these substances shall be kept below 6% (C) and 4% (Si) in order not to increase the propensity for criticality.+ Further, the propensity for criticality increases if the assemblies are placed close together. The loading curves presented in Section 4.4.1 are based on the closest possible distances based on the acceptable distances between the channel tubes for the reference design of the canister.

The following requirement and criterion are set for the selection of fuel assemblies to be encapsulated.

- Requirement on handling: *The fuel assemblies to be encapsulated shall be selected with respect to enrichment, burnup, geometrical configuration and materials in the canister so that criticality will not occur during the handling and storage, even if the canister is filled with water.*
- Criterion: *The effective multiplication factor (k_{eff}) must not exceed 0.95 including uncertainties.*

3.1.3 Dimensions and spacing devices

The dimensions of the BWR and PWR fuel assemblies, including alterations that may occur as a result of the irradiation in the nuclear reactor, shall be considered in the design of the canister inserts. Two types of canister inserts with the same length and diameter provided with channel tubes with different inner dimensions to accommodate BWR and PWR fuel assemblies, respectively, will be manufactured; see the **Canister production report**.

SKB has decided that it shall be possible to encapsulate all spent fuel from the Swedish nuclear power programme, i.e. also the Ågesta fuel, the swap MOX fuel, the Studsvik fuel residues and the special boxes containing fuel rods, in either BWR or PWR canisters, and the following design requirement is set for the canister.

- Design requirement: *The dimensions of the fuel channel tubes of the insert shall be adapted to the dimensions of the spent fuel to be deposited.*
- Design premises: *The length of the longest BWR or PWR assembly, including induced length increase. The cross section of the largest BWR and PWR fuel assemblies, including deviations due to deformations during operation.*

The measures that shall be used in the design of the channel tubes of the insert are given in Table 3-1.

Table 3-1. Design measures for the fuel channel tubes of the insert.

Detail	BWR	PWR	Comment
Longest assembly	4,441 mm		Before irradiation.
Induced length increase	14 mm		When determining the length of the longest assembly the length before irradiation and the induced length increase is considered.
Largest cross section	141×141 mm	214×214 mm	Before irradiation.
Deviations due to deformations during operation	145.5×145.5 mm	228×228 mm	Cross sections of BWR transport cask, and PWR storage canister respectively. All assemblies in Clab have been placed in these casks or canisters, i.e. these cross sections are sufficient with respect to occurring deviations due to deformations during operation.

Since the dimensions of the miscellaneous fuel types deviate from those of the BWR and PWR assemblies, boxes or spacing devices are required in order to keep them in position in the channel tubes of the canister. The different kinds of BWR and PWR assemblies may also vary in dimensions and require spacing devices. Based on this, the following requirements are put on the handling of the spent fuel.

- Requirement on handling: *The miscellaneous fuels shall be encapsulated in either BWR or PWR canisters.*
- Requirement on handling: *Devices that keep the miscellaneous fuel types in position shall be placed in the channel tubes of the canister.*
- Requirement on handling: *Devices that keep BWR and PWR assemblies in position shall, if necessary, be placed in the channel tubes of the canister.*

Detailed design premises for the spacing devices will be provided at a later stage.

3.1.4 Encapsulated liquids and gases

Nitric acid formed from radiolysis of water and air remaining in the canister when it is sealed may cause corrosion of the cast iron insert and the copper shell. The fuel assemblies are stored in water before encapsulation and in the case of a cladding leak there may be water inside the fuel rods. Consequently, in order to limit the amount of water and air, the fuel assemblies must be dried before encapsulation and the atmosphere inside the canister changed and inspected.

With respect to this the following design premise is stated for the canister; see **Design premises long-term safety** Section 3.1.5.

- Design premise: *The amount of nitric acid formed within the insert is limited by changing the atmosphere in the insert from air to > 90% argon. The maximum amount of water left in the insert is set to 600 g.*

This is the background to the following requirements on the handling of the spent nuclear fuel.

- Requirement on handling: *Before the fuel assemblies are placed in the canister they shall be dried so that it can be justified that the allowed amount of water stated as a design premise for the canister is not exceeded.*
- Criterion: *The amount of water left in anyone canister shall be less than 600 g.*
- Requirement on handling: *Before the canister is finally sealed, the atmosphere in the insert shall be changed so that acceptable chemical conditions can be ensured.*
- Criterion: *The atmosphere in canister insert shall consist of at least 90% argon.*

Equipment for drying of fuel assemblies and exchange of atmosphere shall be designed to conform to these requirements and criteria. These requirements and criteria shall also be considered in the instructions for the handling of the fuel in the encapsulation plant. The required inspections are further discussed in Chapter 4.

3.1.5 Radiation

The radiation at the canister surface may result in the formation of nitric acid and other corrosive species that may cause increased corrosion of the copper canister surface in the final repository. If the radiation is limited, these processes can be neglected; see **Design premises long-term safety,** Section 3.1.5**.** The radiation at the surface of the canister will depend on the radiation shielding provided by the insert and copper shell and the radioactivity of the encapsulated spent fuel. The radioactivity of the spent fuel will depend on the burnup and the age of the fuel assemblies. The burnup and age of the encapsulated spent fuel will, consequently for a specific design of the canister, set a limit for the radiation dose rate at the canister surface. Based on this, the following requirement and criterion is set for the handling of the spent fuel.

- Requirement on handling: *It shall be verified that the radiation dose rate on the canister surface will not exceed the level used as a premise in the assessment of the long-term safety.*
- Criterion: *The radiation dose rate at the surface of the canister must not exceed 1 Gy/h.*

This criterion shall be verified for the canister after the placement of the fuel assemblies selected for encapsulation; see Sections 4.4.1 and 4.7.2.

3.2 Requirements related to the operation of the KBS-3 system

3.2.1 Encapsulation

The encapsulation of the spent nuclear fuel shall be adapted to the nuclear power programme so that the costs and environmental impact are minimised. With respect to this, provided that the selected assemblies conform to the acceptance criteria for decay power and criticality, it is an advantage if the canisters are filled to their maximum capacity and the following objective is set for the selection of assemblies.

- Requirement on handling: *The number of canisters shall be minimised and, if possible, all assembly positions in the deposited canisters shall be filled.*

At the end of the nuclear power programme, fuel assemblies with high burnup are expected. Unless these assemblies can be combined with assemblies with low decay power, they require long interim storage times to conform to the decay power criterion if all positions in the canister are to be filled. This means that canisters that are not filled to their maximum capacity may have to be deposited to conform to the decay power criterion. The number of such canisters will, in addition to burnup, depend on when the encapsulation is initiated, on the length of the operation period and on the encapsulation rate. The facilities shall be designed for an encapsulation and deposition rate of up to 200 canisters per year.

The fuel assemblies are stored in pools with water, inside storage canisters with 16 or 25 BWR assemblies and 5 or 9 PWR assemblies in each storage canister. It is an advantage, regarding safety and radiation protection and costs, if the number of movements of the fuel assemblies can be minimised. Based on this, the following objectives are set for the selection of assemblies.

- Requirement on handling: *The number of lifts and movements of fuel assemblies shall be minimised.*
- Requirement on handling: *If possible, storage canisters shall be emptied before they are brought back from the encapsulation plant to the interim storage facility.*

3.2.2 Operational safety and radiation protection

For the operational safety assessments of the facilities and radiation protection considerations of the transports within the KBS-3 system, the radioactivity content in the canister and the radiation at the canister surface must be known. Based on this, the following requirement is set for the handling.

- Requirement on handling: *It shall be verified that the radioactivity content and radiation level on the canister surface will not exceed the contents and levels used as premises in the assessments of the operational safety.*

The criteria for conformity to this requirement as well as the verification are addressed in the safety report for the Clink facility, **SR-Operation** and the safety report for the transports of encapsulated spent fuel.

3.2.3 Control of nuclear material – Safeguards

Sweden's commitments regarding non-proliferation and the planned actions to control the management of fissile material within the KBS-3 system is presented in /SKBdoc 1172138/. After the spent fuel has been encapsulated it is no longer possible to control individual assemblies and each canister will constitute a unit for the account and control of nuclear material. The requirement on marking of the canister that follows from this is described in the **Canister production report,** Section 2.3.2**.**

4 The handling of the spent nuclear fuel

4.1 Overview

The handling of the spent nuclear fuel comprises the following main parts:

- transport and delivery of fuel assemblies,
- interim storage,
- selection of assemblies and encapsulation.

A flow chart for the spent fuel line and its stages, with references to the sections in which the different stages are presented, is given in Figure 4-1. The interim storage and encapsulation of the spent nuclear fuel will take place in Clink (Central interim storage and encapsulation plant). The Clink facility will consist of the current interim storage facility (Clab) and the encapsulation plant to be constructed in connection to Clab.

4.2 Transport and delivery of fuel assemblies

4.2.1 Activities

The spent nuclear fuel is transported to the interim storage facility in accordance with applicable regulations for transports of radioactive material.

When the transport arrives at Clab (later Clink), the transport cask is inspected and transported to a cell for cooling. When the temperature has stabilised, the transport cask is transported to the cask pool and connected to the unloading pool. The fuel assemblies are unloaded and placed in the storage canisters in which they are kept during the interim storage. If the assemblies are damaged, or suspected to be leaking, the storage canister is transported to a service pool and the assemblies are searched for damages. If necessary, damaged assemblies are put in protective boxes and placed in special storage canisters.

Each storage canister and each fuel assembly is marked with a unique identity code. When the fuel assembly is put in the storage canister, its identity is linked to the identity of the storage canister and its position in the storage canister is registered.

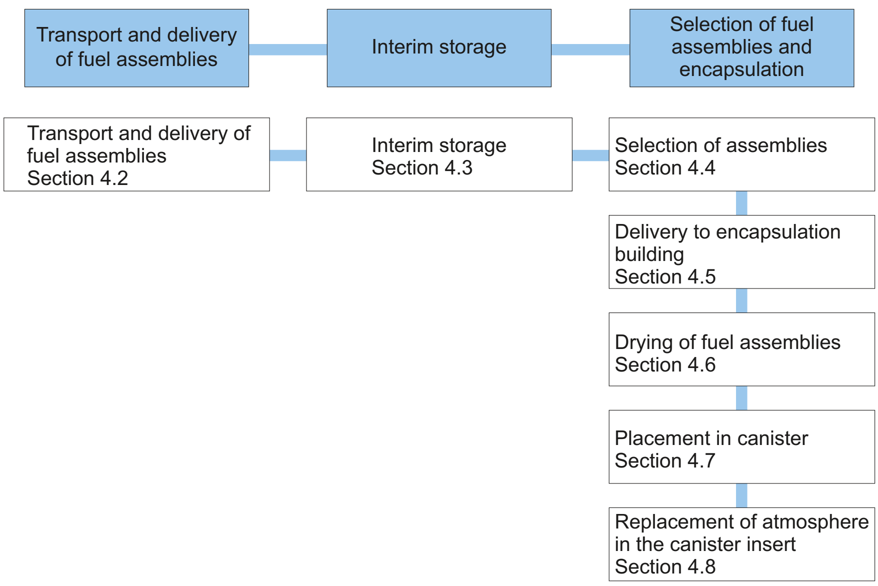


Figure 4-1. *Flow chart for the spent fuel line and its stages. A flow chart for the handling of the encapsulated spent fuel is provided in the Canister production report, Figure 6-1.*

4.2.2 Inspections

Before a transport is approved the nuclear power plants announce which fuel assemblies they intend to send and the properties of the assemblies. SKB inspects the documentation to ensure that the assemblies are allowed for transportation by the transportation system and for interim storage in Clab (later Clink). Then the transport can be performed.

When the fuel assemblies are unloaded, their identities are visually inspected. The inspection is performed with a camera and the information is documented. Properties of importance for the control of nuclear material, safeguards, are stored in a special database for this purpose. Properties of importance for the operation of SKB's facilities and long-term safety are stored in other databases containing all relevant information about the fuel assemblies. Regarding the burnup SKB intends to use data provided by the nuclear power plants. The burnup must be regularly calculated and accurately measured to achieve a reliable and efficient operation of the reactors. The registration of burnup during operation is required by the Swedish Radiation safety authority (SSM) and the data is quality assured. If required, the burnup can also be measured by γ-scanning.

4.3 Interim storage

4.3.1 Activities

When the placement and registration of assemblies in the storage canister is finished, the storage canister is transported to a fixed position with a unique identity in one of the storage pools. The assemblies are stored until they are selected for encapsulation. They are selected based on their decay power; see further Section 4.4.1.

The storage is constantly monitored and all movements of the storage canisters are registered.

4.3.2 Inspections

If the storage canisters or fuel assemblies are moved, their identities are inspected and their new positions registered. No other inspections of the fuel are required during the interim storage. If required, it is possible to measure the radiation and decay power of the assemblies in the interim storage facility. The operation of the facility, e.g. levels and temperatures of the water in the storage pools, is regularly measured.

4.4 Selection of assemblies

4.4.1 Activities

The selection of fuel assemblies for encapsulation is iteratively performed by applying the requirements presented in Section 3.1 in the following order.

- *The fuel assemblies to be encapsulated shall be selected with respect to burnup and age so that the total decay power in the canister will not result in temperatures exceeding the maximum allowed. The total decay power in each canister must not exceed 1,700 W.*
- *The number of canisters shall be minimised and, if possible, all assembly positions in the deposited canisters shall be filled.*
- *The fuel assemblies to be encapsulated shall be selected with respect to enrichment and burnup and the geometrical configuration and materials in the canister so that criticality will not occur during the handling and storage of canisters even if the canister is filled with water. The effective multiplication factor (k_{eff}) must not exceed 0.95 including uncertainties.*
- *It shall be verified that the radiation dose rate on the canister surface will not exceed the level used as a premise in the assessment of the long-term safety. The radiation dose rate at the surface of the canister must not exceed 1 Gy/h.*
- *The number of lifts and movements of fuel assemblies shall be minimised.*

- *If possible, storage canisters shall be emptied before they are brought back from the encapsulation plant to the interim storage facility.*

The selection process can be summarised as follows.

1. Compile information for the selection.
2. Preliminarily selection – based on decay power and the objective to fill all assembly positions in the canisters to be deposited.
3. Check criticality – adjust the selection in case of non-conformity to the criterion for criticality.
4. Check radiation dose rate on the canister surface – adjust the selection in case of non-conformity to the criterion for maximum allowed radiation dose rate.
5. Lifts and movements – investigate the number of lifts and movements of assemblies and storage canisters and adjust the selection if the number of lifts can be reduced and the selection still conforms to criteria for decay power and criticality.
6. Final selection – determine a selection and make a plan for transport of storage canisters and assemblies.

Compile information for the selection

The preliminary selection will be based on documented and calculated information of the total inventory of assemblies in interim storage at the time of encapsulation. In order to optimise the selection the full inventory of assemblies, i.e. both the assemblies currently in interim storage and the assemblies from the future operation, are considered in the strategy for selection. Before the selection is made, the decay power is calculated for all assemblies and loading curves to check the criticality criterion are produced.

The decay power for each assembly is calculated from its individual burnup data and the date the assembly was taken out from the reactor core. All data are provided by the nuclear power plants. A well documented and verified code will be used for the calculations. In Figure 4-2 the decay power as a function of age is illustrated for BWR assemblies with different burnup.

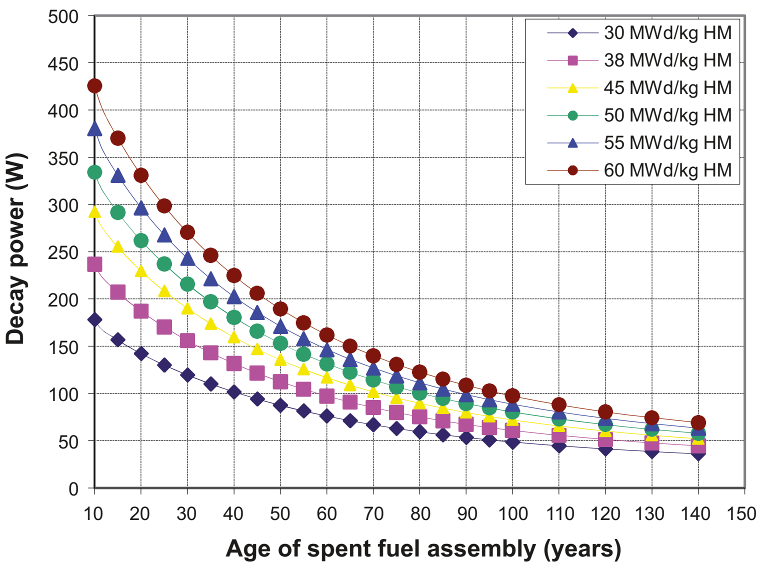


Figure 4-2. *The decay power as a function of age for BWR assemblies with different burnup. (The allowed average decay power per encapsulated assembly is 142 W for a canister with 12 identical assemblies.) /SKBdoc 1198314/.*

Criticality is checked by calculating loading curves. In those calculations, combinations of enrichments and burnups that will result in a multiplication factor (k_{eff}) of 0.95 for the encapsulated assemblies are derived /SKBdoc 1193244/. The calculations are based on typical BWR and PWR assemblies, i.e. Svea Optima 2 and Areva 17×17, respectively, and on the reference design of the canister and typical properties of bentonite and rock. Further, it is assumed that identical assemblies occupy all positions in the canister. In the calculations, a systematic investigation of uncertainties is made. All parameters with potential impact on the criticality are investigated. Parameters that can be shown to be insignificant are set to a typical value while parameters that are significant are set so as they favour criticality.

The calculated loading curves and the combinations of average burnup and enrichment for the assemblies that currently are stored in Clab are given in Figure 4-3 and Figure 4-4. Fuel assemblies with a combination burnup/enrichment that are plotted above the loading curves in Figure 4-3 and Figure 4-4 will have a k_{eff} that exceeds 0.95 and will thus not conform to the criterion for criticality.

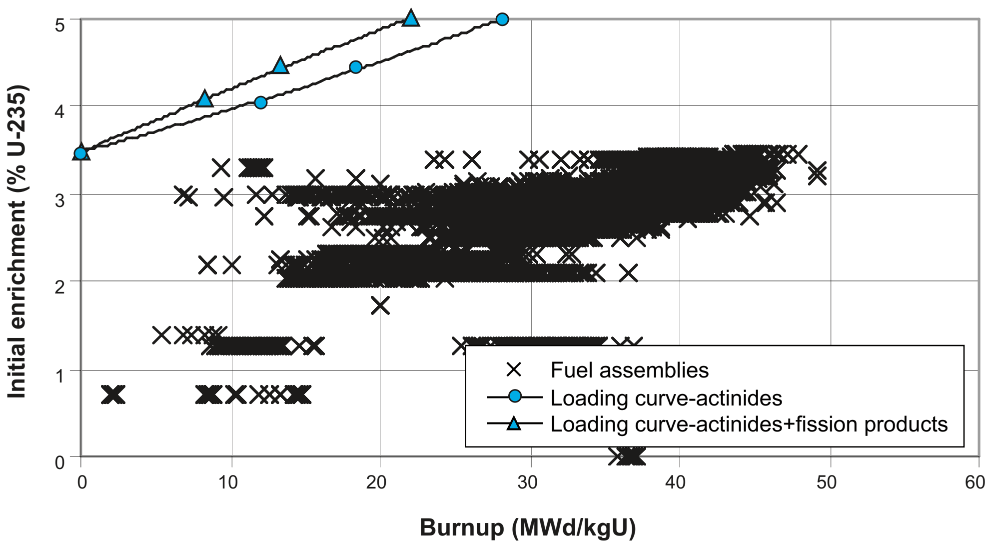


Figure 4-3. *Loading curves for BWR-canisters with 12 identical fuel assemblies and enrichment and average burnup for the assemblies currently stored in Clab /SKBdoc 1193244/.*

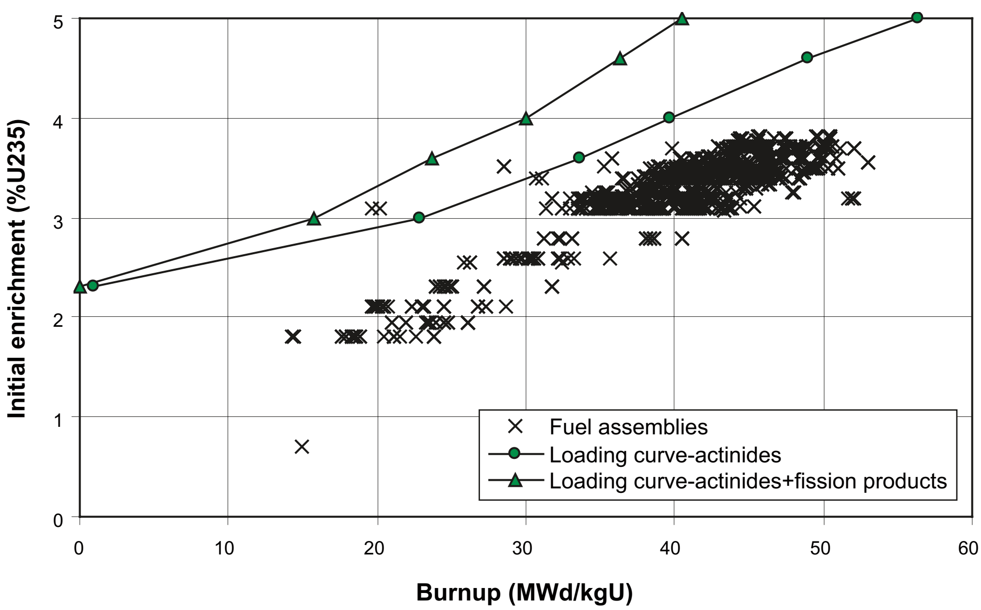


Figure 4-4. *Loading curves for PWR-canisters with 4 identical fuel assemblies and enrichment and average burnup for the assemblies currently stored in Clab /SKBdoc 1193244/.*

As can be seen from Figure 4-3 and Figure 4-4, respectively, un-irradiated BWR assemblies with enrichment higher than 3.5% and un-irradiated PWR assemblies with enrichment higher than 2.4% will be plotted above the loading curves. The enrichment for the fuel assemblies currently stored in Clab is typically between 3.6–4.2% U-235, and enrichment up to 5% U-235 is allowed. Consequently, as can be seen from the loading curves, credit must be taken for burnup in order to conform to the criticality criterion.

Preliminary selection

When the decay power is calculated for all assemblies, the preliminary selection of assemblies to be encapsulated is made. With the exception of the fuel residues in protection boxes from Studsvik (see Table 2-3), the preliminary selection is made from all the BWR and PWR assemblies actually stored in the Clink facility at the specific time, and from the miscellaneous fuels (Section 2.2.2).

The fuel residues from Studsvik are encapsulated separately since some of them contain epoxy resin that may build up pressure from gases generated by radiolysis and must not be placed together with assemblies with relatively high decay power and radioactivity.

With respect to the objective to fill all assembly positions in the canisters, the total decay power shall lie as close as possible to the maximum allowed. The assemblies are selected so that the conformity to the decay power criterion is ensured. Currently, the selection is made so that the sum of the calculated decay powers for the selected assemblies is 1,650 W or less. The limit of 1,650 W is set to cover divergences between the calculated and actual decay power so that the actual decay power will always be below 1,700 W. The divergences between calculated and actual decay power have based on comparisons between calculated and measured decay powers been estimated to be 2% /SKB 2006/.

Check of criticality

When the selection based on the decay power has been made it is checked against the criterion to avoid criticality. If $k_{eff} < 0.95$ for all individual assemblies, i.e. if their combination of enrichment and burnup, lies under the calculated loading curve, they can be encapsulated without further checks. If there are assemblies with $k_{eff} > 0.95$, the criticality is calculated for the full set of preliminary selected assemblies. In these calculations, the assembly with $k_{eff} > 0.95$ is placed in the canister in the worst position for a potential criticality. If the calculations show that k_{eff} is above 0.95 for the whole canister, that selection of assemblies is not encapsulated. Instead a new set of assemblies is selected.

There will be low burnup assemblies with $k_{eff} > 0.95$, that lies on the criticality side of the loading curve. These assemblies can be combined with high burnup assemblies that lies with equal or larger distance from the loading curve on the non-criticality side, so that the specific set of assemblies conform to the criticality criterion, $k_{eff} < 0.95$. If it is not possible to find a set of assemblies that conform to the criticality criterion, the low burnup assemblies can be encapsulated alone in a canister. Should it neither be possible to combine the low burnup assemblies with high burnup assemblies nor to encapsulate them alone to conform to the criticality criterion, the ultimate measure is to alter the geometry, i.e. to reconstruct the assembly.

Check of radiation

The maximum allowed decay power in a canister will, since the decay power is a result of the radioactivity of the spent fuel assemblies, also restrict the radiation at the canister surface. The radiation dose rate at the canister surface has been calculated for the reference design of the canister /SKBdoc 1077122/. The results show that highest radiation dose rate to material in contact with the canister is obtained for PWR canisters. The obtained dose rate is 0.18 Gy/h on limited areas of the canister tube surface. The burnup and age of the assemblies was set to 60 MWd/kgU and 30 years, i.e. an overestimation since this combination of assemblies is not allowed with respect to the decay power criterion. The average radiation dose rate from the copper tube surface is calculated to be 0.055 Gy/h. These results, which give an ample margin to the design premise (1 Gy/h), imply that all canisters conforming to the criterion for maximum allowed decay power will also conform to the design premise for maximum allowed radiation dose rate at the canister surface.

Consequently, given the reference design of the canister, **Canister production report**, Chapter 3, and the current criteria for decay power and radiation, there is no need to calculate the radiation dose rate on the surface of individual canisters.

Lifts and movements

When the preliminary selection has been made and the criticality checked, the storage canisters in which the assemblies are stored are identified. Information on the full set of assemblies in the selected storage canisters is compiled and it is investigated whether several selections for encapsulation can be made from the same storage canisters. If not, it is investigated whether a storage canister can be changed for another containing a more suitable set of assemblies. If it is anticipated that a change is needed in order to minimise the number of lifts and movements of storage canisters, the selection of assemblies will be made for several canisters at the same time.

Final selection

When all calculations and checks according to the description above have been performed, the selection is documented and a plan for the encapsulation is made.

4.4.2 Inspections

The calculated decay powers and loading curves shall be reviewed in accordance with SKB's management system. Input data as well as computer programmes and calculations are reviewed.

4.5 Delivery to the encapsulation building

4.5.1 Activities

When a selection of assemblies has been determined and a plan for the encapsulation made, the storage canisters in which the selected assemblies are stored are lifted from their positions in the storage pools and transported to the encapsulation building. The transport is performed under water, via the fuel elevator in the interim storage building and the connecting pool in the encapsulation building, to the handling pool in the encapsulation building. Up to 12 storage canisters can be placed in the handling pool. When all storage canisters with selected assemblies have been moved to the handling pool, the assemblies to be encapsulated for deposition in a specific canister are reloaded to a transfer canister used for transports within the encapsulation building. The transfer canister is loaded with the full set of assemblies to be encapsulated, i.e. either 12 BWR or 4 PWR assemblies (or fewer in unfilled canisters). The location of the assemblies and altered content in the storage canisters is registered in the database for control of fissionable material and in SKB's databases for operation of the facilities.

Damaged assemblies that are stored in protective boxes are placed with the protective box in the transfer canister and the lid is removed before the transfer canister is placed in the drying position.

4.5.2 Inspections

The identities of the storage canisters and assemblies in the storage canister are visually inspected both before they are lifted from the storage pools and when they are delivered to the handling pool. The inspection can be documented by a camera. The documented decay power, burnup, enrichment and other information of importance for the encapsulation is inspected for each assembly.

If there are uncertainties regarding the decay power, it can be measured for verification. This is performed by measuring the γ-radiation from the assembly. All measurements can be performed in the handling pool.

4.6 Drying of fuel assemblies

4.6.1 Activities

The drying of assemblies shall conform to the following requirement.

- *Before the fuel assemblies are placed in the canister they shall be dried so that it can be warranted that the allowed amount of water stated as a design premise for the canister is not exceeded.*

When the transfer canister is filled with the assemblies to be encapsulated, it is lifted from the handling pool and kept hanging so that the water can run and drip off. After that, it is transported to one of the two drying positions in the handling cell. The drying is performed by decreasing the pressure in the drying cell so that all water is vaporised. The vapour is conducted to a system for condensation and treatment of process water. Damaged assemblies that are kept in protective boxes are dried in the box.

4.6.2 Inspections

In **Design premises long term safety** it is stated: "*The maximum amount of water left in the insert is set to 600 g.*" Consequently, it shall be verified that the water content in the canister insert is 600 g or less when it is sealed. The relationship between temperature, pressure and humidity are well known and by measuring the pressure in the cell it can be verified that the fuel assemblies are sufficiently dry when they leave the cell. If water remains in the system, the pressure in the cell will rise after the drying is finished.

4.7 Placement in the canister

4.7.1 Activities

The canister in which the selected fuel assemblies will be encapsulated is connected to the handling cell. Each canister to be deposited shall have a unique identity; this identity is read and registered. A guiding frame is placed on top of the canister to protect it and guide the assemblies to their correct position (channel tube). Before the fuel assemblies are lifted from the transfer canister and placed in the canister, their identity is registered. Fuel assemblies significantly smaller than typical BWR or PWR assemblies, which may be damaged due to punches when the canister is moved or lifted, are provided with distance devices that prevent them from moving in the channel tubes in the canister. This will be necessary for the miscellaneous fuels (Ågesta, Studsvik and swap MOX). The assembly is then lifted to the channel tube in which it will be placed and lowered into the canister. With respect to the operational safety, to reduce the radiation on the canister surface, the assemblies generating more radiation, i.e. MOX assemblies or the assemblies with the largest decay power are placed in the centre of the insert in BWR canisters. When all assemblies from the transfer canister are placed in the canister, the guiding frame is removed and the identities of the assemblies are registered and associated to the identity of the canister. The channel tubes in canisters with less than twelve or four assemblies are left empty. Then the steel lid is put on the insert.

Damaged assemblies that are stored in protective boxes are lifted from the boxes to the canister. The boxes are then inspected and material that may remain in them is taken care of. If required it can be placed in a canister and encapsulated. If material should come off from the assembly when it is handled in the cell this is taken care of, and if required encapsulated. This is described in Chapter 5, of the safety report for the interim storage and encapsulation plant – Clink.

4.7.2 Inspections

Before the steel lid is placed on the insert, the identities of the assemblies in the canister are inspected and checked against the planned content, i.e. the selected assemblies. The inspection is visual and can be documented by photograph.

Before the atmosphere in the insert is changed and the canister is sealed the γ-radiation at the canister surface is measured. It shall be verified that the radiation dose rate including any contribution from neutrons does not exceed 1 Gy/h.

4.8 Replacement of atmosphere in the canister insert

4.8.1 Activities

The atmosphere in the insert is replaced by argon. The position for sealing of the insert has evacuation equipment that is connected to a valve in the steel lid of the canister insert. The pressure in the insert is then lowered by evacuation of the air and argon enters the insert. The atmosphere replacement is then repeated and the composition of the evacuated gas is investigated. If the content of argon is not above 90%, another repetition of the process is performed. After the atmosphere has been changed the weld joint surface is checked and the copper lid is placed on the canister. The canister can then be transferred to the welding station for sealing.

4.8.2 Inspections

The content of argon of the evacuated gas is inspected after each replacement of the atmosphere in the canister insert. The argon concentration of the evacuated gas shall be greater than 90%.

5 The canisters to be deposited

In this chapter the number of canisters that is required to deposit the spent fuel accounted for in Chapter 2 is assessed. The assessment is based on the requirements on the handling and selection of assemblies presented in Chapter 3 and Section 4.4.1 respectively. In addition, the encapsulation period is postulated.

5.1 Factors that will affect the number of canisters

The number of canisters to be deposited will depend on:

- the number of fuel assemblies in the canisters,
- the number of spent fuel assemblies to be encapsulated,
- the burnup of the assemblies,
- the allowed decay power in the canister,
- when the encapsulation is initiated and the encapsulation rate.

The maximum number of BWR and PWR assemblies in a canister is twelve and four, respectively.

The number and burnup of the assemblies to be deposited will depend on the thermal powers, number and type of assemblies in the reactor core and operation times of the nuclear power reactors (see Table 2-1), and on the past and planned average target burnups.

The allowed total decay power of the assemblies encapsulated in a canister will ultimately depend on the decay power that can be accepted for the temperature in the buffer not to exceed 100°C. The temperature in the buffer will depend on the thermal properties of the buffer and bedrock and on the distances between deposition holes; see **Underground openings construction report**, Section 4.2.1. The current layout of the final repository is based on a maximum decay power in a canister of 1,700 W.

The decay power of the fuel assemblies will decrease with time; consequently, the age of the assemblies at encapsulation will affect the decay power. The age of the spent fuel assemblies at any time during the encapsulation period will depend on the operation times of the nuclear power plants and on when the encapsulation and deposition is initiated and completed.

The assemblies for encapsulation shall be selected with respect to their burnup and age so that they conform to the criterion for maximum allowed total decay power in a canister.

In Figure 5-1 the relation between burnup and age of the spent fuel assemblies to reach 1,700 W in a BWR and PWR canister, respectively, is shown, assuming that all positions in the canister are occupied by assemblies with the same decay power.

From the interpolation of data in Figure 5-1, the interim storage time required for the average burnup BWR and PWR assembly to reach 1,700 W in the canister can be derived. For the spent fuel from the reference scenario for the operation of the nuclear power plants presented in Section 2.1.1 this time is 38.7 years for the average burnup BWR assembly and 37.9 years for the average burnup PWR assembly. If all positions in all canisters are to be filled and their total decay powers not exceed 1,700 W, the average interim storage time must be at least 38.7 and 37.9 years for BWR and PWR canisters respectively. Shorter interim storage time will mean that all positions cannot be filled in all canisters and longer interim storage time will result in lower total decay power in some canisters.

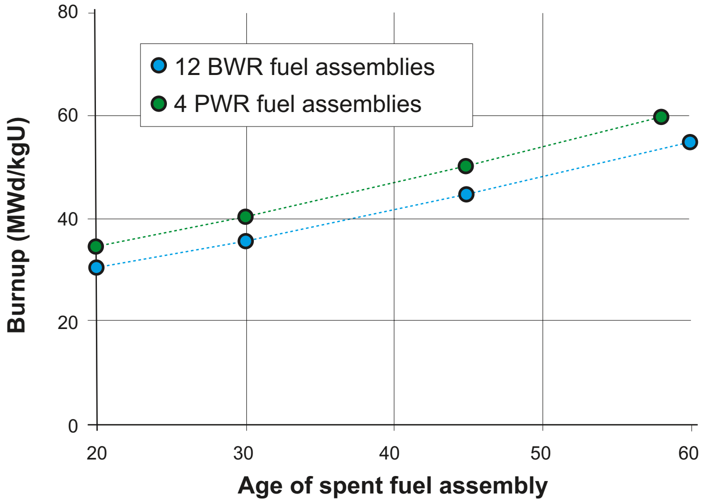


Figure 5-1. *Burnup and age of spent fuel assembly required to reach a decay power of 1,700 W in a BWR and PWR canister respectively assuming that all positions are occupied by assemblies with the same decay power.*

5.2 Simulation of the encapsulation

The encapsulation of the spent nuclear fuel to be deposited has been simulated assuming that the encapsulation is initiated in 2023 and completed 25 years after the last nuclear power plant is taken out of operation (2070) /SKBdoc 1221567/. The assemblies were selected so that the conformity to the decay power criterion was ensured, see Section 4.4.1. Further, the simulation is based on the encapsulation rate given in Table 5-1.

The assemblies stored in Clab as of December 31[th] 2007 and the assemblies included in the SKB reference scenario for the future operation were included in the simulations. With exception of the fuel residues from Studsvik, which require all together seven canisters, the miscellaneous fuels presented in Table 2-3 were also included in the simulation. The total numbers and kinds of assemblies included in the simulation are compiled in Table 5-2. The burnup and age of the BWR and PWR assemblies included in the simulation are illustrated in Figure 2-3 and Figure 2-4. In the simulations it is anticipated that no more than one MOX assembly will be encapsulated in any canister.

For each simulated year of encapsulation the inventory of assemblies in the Clink facility and their decay power was calculated. For assemblies in interim storage, documented data were used and for the remaining assemblies, the planned reactor operation information was used as input. Other data required for the calculations were based on typical BWR and PWR assemblies, i.e. Svea Optima 2 and Areva 17×17, respectively.

Table 5-1. Encapsulation rate assumed in the simulation of canisters to be deposited in the KBS-3 repository.

Year	Encapsulation rate (canisters/yr)		
	BWR	PWR	Total
2023	17	6	23
2024	58	22	80
2025–2027	87	33	120
2028–2054	109	41	150
2055–2069	73	27	100
2070	77	13	90

Table 5-2. Total numbers and kinds of spent fuel included in the simulation of the encapsulation.

Kind of spent fuel	Number of assemblies
BWR UOX From the reactors B1, B2, F1, F2, F3, O1, O2, O3 and R1 Including rod cassettes, i.e. dismounted fuel rods placed in fuel rod cassettes.	47,415
PWR UOX From the reactors R2, R3 and R4	6,016
Ågesta (encapsulated in BWR canisters)	222
BWR MOX From Oskarshamn (O1 and O3) Swap from Germany	 83 184
PWR MOX Swap from Germany	 33
Total for encapsulation in BWR canisters	47,904
Total for encapsulation in PWR canisters	6,049
Total for encapsulation	53,953

The assemblies were selected to give a combined decay power of 1,700 W at the time of disposal. That is, assemblies with a combination of burnup/age lying above the curves in Figure 5-1 were combined with assemblies with a combination burnup/age below the curves. Based on the burnup and age of the assemblies to be deposited and the assumed encapsulation period, it is not possible to fill all the assembly positions in all of the canisters if their summed up decay power shall not exceed 1,700 W. Consequently, there will be a number of canisters containing less than the maximum possible number of assemblies. The simulations were carried out to minimise the number of unfilled assembly positions in the canisters. In Figure 5-2 and Figure 5-3 the resulting number of BWR and PWR canisters to be deposited is illustrated. The canisters are divided into groups with respect to the average burnup of the encapsulated assemblies and the number of assemblies in the canisters. A detailed account of the canisters to be deposited each year is given in Table C-3 and C-4 in Appendix C.

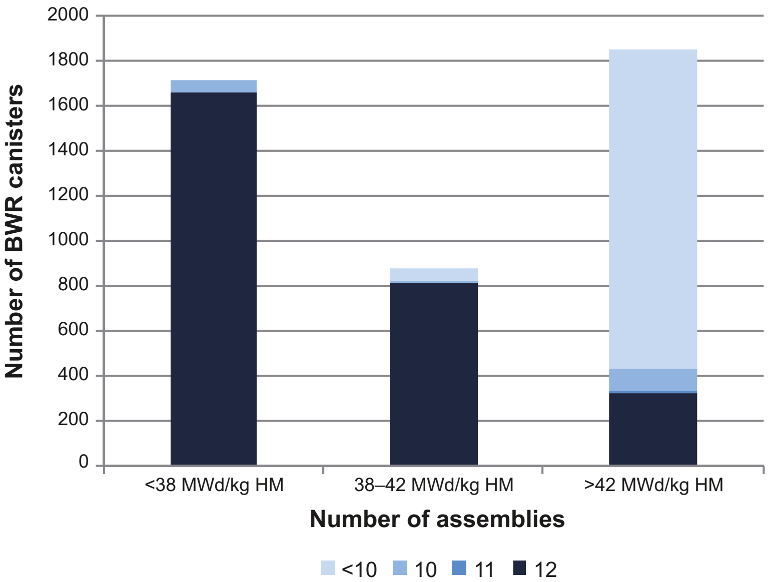


Figure 5-2. The BWR canisters to be deposited divided into groups with respect to the average burnup of the encapsulated assemblies. The number of assemblies in the canister is indicated by different colours. The total number of BWR canisters is 4,451.

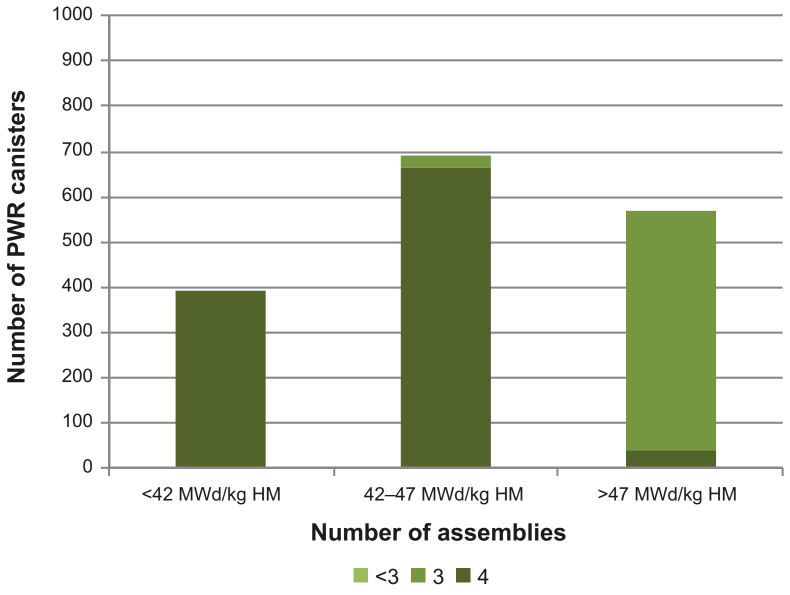


Figure 5-3. *The PWR canisters to be deposited divided into groups with respect to the average burnup of the encapsulated assemblies. The number of assemblies in the canister is indicated by different colours. The total number of PWR canisters is 1,652.*

6 Initial state – encapsulated spent nuclear fuel

6.1 Introduction

Encapsulated spent fuel is the spent nuclear fuel encapsulated for deposition in the KBS-3 repository. Gases and liquids in the cavities of the canister and fuel assemblies are considered as a part of the encapsulated spent fuel. The initial state of the encapsulated spent fuel refers to the properties of the spent fuel and the gases and liquids in the cavities of the canister when the canister is finally sealed and no further handling of the individual fuel assemblies is possible.

In this chapter the expected values of the fuel parameters given in Section 2.3 for the initial state are presented. These values depend on the properties of the spent nuclear fuel to be deposited, the selection and drying of the fuel assemblies and their emplacement in the canister, and the change of atmosphere in the canister described in Chapter 4.

The radionuclide inventory is an essential input to the long-term safety assessment. For the initial state the radionuclide inventory is presented as the total inventory in the final repository (Section 6.2.2) and the inventory for a set of categories of canisters, referred to as *type-canisters* (Sections 6.2.3 and 6.2.4). The total radionuclide inventory in the repository will ultimately depend on the total energy output from the nuclear power plants. The inventory in each canister will, in addition, depend on the number of assemblies in the canister, the criteria for selection of assemblies for encapsulation and the time for encapsulation.

There are also fuel parameters of importance for the long-term safety assessment, e.g. total decay power in a canister and content of gases and liquids in the canister, which in principal solely will depend on the selection of assemblies for encapsulation and the handling of the assemblies. These parameters are expected to be similar for all canisters.

6.2 Radionuclide inventory

6.2.1 Fuel parameters of importance for the radionuclide inventory

The burnup is the fuel parameter that has the largest impact on the radionuclide inventory of the spent fuel. The burnup reflects the amount of nuclear fissions and neutron radiation that has occurred in the assembly and, thus, the content of fission and activation products and transuranium elements. High burnup leads to higher content of fission products and, due to neutron capture, higher content of activation products and altered content of transuranium elements. The radionuclide inventory is to some extent also affected by:

- the irradiation and power history of the assemblies, i.e. how long time and to which power they have been utilised in the reactor,
- the fuel type, i.e. UOX or MOX.

When the construction materials in the fuel assemblies are exposed to neutron radiation in the reactor vessel, neutron capture will lead to formation of activation products. The amount and composition of the construction materials and the length of time the fuel assembly has been in the reactor will determine the content of activation products in the construction materials.

The power history, i.e. the power developed per length of fuel rod or fuel assembly over the irradiation period in the reactor vessel, referred to as the linear heat generation rate, is strongly correlated to the fission gas release (FGR). The FGR in turn is used to determine the part of the radionuclide inventory that is located at the fuel grain boundaries and in the gap between the fuel and the cladding /Werme et al. 2004/. This part of the inventory is referred to as the gap inventory and will in comparison to the radionuclides embedded in the fuel matrix be released very rapidly if the spent fuel pellets are exposed to vapour or water.

The type of fuel, UOX or MOX, will affect the content of transuranium elements.

The radionuclide inventory will get altered as the radioactive decay proceeds; consequently the age of the fuel assemblies will also impact the radionuclide inventory.

6.2.2 Total radionuclide inventory in the final repository

As stated in Section 6.1, the total radionuclide inventory in the final repository will ultimately depend on the total energy output from the nuclear power plants. As discussed in Section 2.2.1, the energy can be provided by several assemblies with low burnup or fewer with high burnup. The parameters determining the number of spent fuel assemblies a nuclear power plant will generate are:

- the thermal power of the nuclear power plant,
- the number and type of assemblies in the reactor core,
- the target burnup of the assemblies, in which the accessibility and utilisation of the power plant is considered,
- the operation time of the nuclear power plant.

The presented total radionuclide inventory is based on the reference scenario for the operation of the nuclear power plants presented in Section 2.1.1, which will result in the total number of BWR and PWR assemblies presented in Table 2-2. From the total number of assemblies and their summed burnup, the average burnups of a BWR and PWR assembly have been calculated to 40.4 and 44.8 MWd/kgU, respectively. The total radionuclide inventory was calculated for the average age of the fuel assemblies in 2045, i.e. the year the last nuclear power plant is closed down according to the reference scenario.

As discussed in Section 6.2.1, the burnup is the fuel parameter that has the largest impact on the radionuclide inventory of a spent fuel assembly. The content of activation products is also, to some extent, affected by the amount and composition of the construction materials and the irradiation history. The impact of the construction materials on the content of fission products and transuranium elements can be neglected. The content of transuranium elements is though affected by the irradiation history, but the burnup is the main parameter determining the inventory. Further, the relationship between the content of transuranium elements in an assembly and its burnup is not linear, but the generation of transuranium elements will be higher for the last produced MWh than for the first. Thus, calculating the total inventory from the mean burnup will result in a certain underestimation of the transuranium element inventory, also see Section 6.2.6.

The inventory of radionuclides and stable isotopes was calculated for the typical BWR and PWR assemblies presented in Section 2.3.2. The calculated total radionuclide inventory has been divided into the inventory in the fuel pellets, i.e. the fuel matrix, and the inventory in the construction materials, i.e. all other materials in the assembly. The inventory in the spent fuel matrix was calculated for the average BWR and PWR burnups with the computer program Origen-S /SKBdoc 1221579/. The inventory of activations products was estimated from calculations with the computer program IndAct and CrudAct /SKBdoc 1198314/. In the estimates of the activation product inventory, the burnup was assumed to be the same as for the fuel matrix. The resulting total inventory in the BWR and PWR assemblies of thirteen radionuclides of importance for the radiotoxicity and calculated long-term risk are given in Table 6-1. The total inventory of all radionuclides is given in Table C-2 in Appendix C.

When calculating the total radionuclide inventory in the construction materials, it was assumed that one out of four PWR assemblies included a control rod cluster (see Figure 2-5). In addition, some assemblies will include inserts, such as start-up neutron sources, boron glass rods and plugs. Since these assemblies are very few, these inserts were not included in the calculated inventory in the construction materials. Note that the fuel channels, see Figure 2-5, have been removed from 1,520 of the BWR assemblies.

The calculated average burnup of the BWR and PWR assemblies given in Table 6-1 are based on the total number of assemblies from the reference scenario for the operation of the nuclear power plants. However, the radionuclide inventory of the 83 BWR MOX assemblies used (3 assemblies), or to be used (80 assemblies), in Oskarshamn has been calculated separately assuming a burnup of 50 MWd/kgHM.

The radionuclide inventories of the miscellaneous fuels accounted for in Section 2.1.2 have not been analysed in detail. For these fuels, assumptions concerning the radionuclide inventories that will result in overestimations have been made according to Table 6-2.

Table 6-1. The total inventory in BWR and PWR assemblies of thirteen radionuclides (Bq) of importance for radiotoxicity and calculated long-term risk (alphabetic order). The inventories are calculated for the calendar year 2045.

Radio-nuclide	Radionuclide inventory (Bq)									
	BWR assemblies				PWR assemblies					
	47,637 BWR UOX[1] 40.4 Mwd/kgU		267 BWR MOX[2] 50 MWd/kgHM		6,016 PWR UOX 44.8 MWd/kgU			33 PWR MOX 34 MWd/kgHM		
	Fuel	Constr. materials/ crud	Fuel	Constr. materials/ crud	Fuel	Constr. materials/ crud	Control rods	Fuel	Constr. materials/ crud	Total activity
Am-241	1.2E+18	1.1E+11	4.8E+16	2.4E+09	4.9E+17	6.6E+10		4.5E+15	3.4E+08	1.7E+18
C-14	1.1E+14	2.9E+14	6.2E+11	2.1E+12	3.4E+13	6.9E+13	3.7E+12	9.7E+10	3.3E+11	5.1E+14
Cl-36	1.8E+12	2.8E+10	1.2E+10	2.1E+08	4.9E+11	8.0E+09	1.1E+09	1.4E+09	3.8E+07	2.3E+12
Cs-137	1.7E+19	5.2E+12	8.5E+16	2.7E+10	6.3E+18	1.1E+12	0.0E+00	1.0E+16	3.3E+09	2.3E+19
I-129	1.0E+13	0.0E+00	9.6E+10		3.8E+12			1.3E+10		1.4E+13
Nb-94	4.6E+10	3.6E+13	6.2E+08	2.6E+11	1.7E+10	8.9E+14	3.2E+11	8.8E+07	4.3E+12	9.3E+14
Pu-238	9.5E+17	9.7E+10	2.1E+16	1.3E+09	3.7E+17	6.0E+10		1.6E+15	1.7E+08	1.3E+18
Pu-239	1.0E+17	9.2E+11	2.7E+15	5.3E+09	3.8E+16	2.8E+11	0.0E+00	2.2E+14	1.5E+09	1.4E+17
Pu-240	1.9E+17	1.2E+10	6.4E+15	3.2E+08	6.1E+16	8.2E+09		5.9E+14	3.9E+07	2.5E+17
Pu-241	7.4E+18	6.4E+11	1.3E+17	6.3E+09	3.0E+18	4.0E+11		9.3E+15	7.1E+08	1.1E+19
Sr-90	1.1E+19	4.8E+12	2.8E+16	2.5E+10	4.2E+18	1.0E+12	0.0E+00	3.6E+15	3.1E+09	1.6E+19
U-234	4.0E+14	3.5E+07	3.8E+12	3.5E+05	1.9E+14	2.3E+07		3.3E+11	1.1E+05	6.0E+14
U-238	9.7E+13	7.9E+06	6.5E+12	4.3E+04	3.2E+13	4.4E+06		1.0E+11	2.5E+04	1.3E+14

[1] Includes 222 assemblies from Ågesta.

[2] Includes 184 Swap BWR MOX.

Table 6-2. Assumed radionuclide inventories in the miscellaneous fuels.

Fuel type	Number	Radionuclide inventory
Spent fuel from Ågesta	222	Set to the same as in the average burnup BWR assembly.
Swap BWR MOX fuel assemblies	184	Set to the same as for the BWR MOX assemblies from Oskarshamn.
Swap PWR MOX	33	Matrix inventory calculated for a swap MOX assembly with a burnup of 34 MWd/kgHM.
Special boxes with fuel residues from Studsvik	25	Set to the same as in the average burnup PWR assembly.

The assumptions in Table 6-2 will result in an overestimation of the radionuclide inventories of the miscellaneous fuels since they contain a smaller amount of heavy metal and have an essentially lower burnup than the average BWR and PWR assemblies and the BWR MOX assemblies from Oskarshamn. However, since the miscellaneous fuels comprise in total 464 assemblies out of about 54,000 assemblies, the impact of their divergent inventories on the total inventory can be neglected.

6.2.3 The type canister approach

At the time for the closure of the final repository when the encapsulation and deposition is finished, the burnup, irradiation and power history and age of the assemblies in each canister will be known and the radionuclide inventory can be calculated for each individual canister. However, at this stage it is not reasonable to calculate the inventory in individual canisters. Therefore, a set of type-canisters has been defined based on the assumption that the criteria for maximum allowed total decay power in a canister will restrict the possible variation in radionuclide inventory. The type-canisters shall provide a representative and adequate description of the canisters' content of fuel, its burnup and age and the resulting radionuclide inventory in each canister.

The radionuclide inventory in each canister will depend on:

- the number of assemblies in the canister,
- the burnup of the assemblies,
- the age of the assemblies when they are encapsulated.

The burnup and the number of assemblies will be the parameters of most importance for the radionuclide inventory. In a long-term perspective, the age of the fuel at deposition is of minor importance for the radionuclide inventory since the short lived nuclides of importance for the decay power will successively decay and no longer remain in the canister.

The part of the inventory located at the fuel grain boundaries and in the gap between the fuel and the cladding is correlated to the fission gas release and power history. This part of the radionuclide inventory is discussed in Section 6.3.

To illustrate the range of burnup and, thus, radionuclide inventory, a set of canisters with reasonable combinations of burnup/age of the assemblies have been selected so that the total decay power in the canisters will not exceed 1,700 W. The ages and burnups of the assemblies are based on the results from the simulation of the encapsulation presented in Section 5.2. The set of canisters and their total activity are presented in Table 6-3. The inventory in the fuel matrix of thirteen radionuclides of importance for decay power, radiotoxicity and calculated long-term risk are given in Table 6-4.

As can be seen from Table 6-3 and Table 6-4 both the total activity content and radionuclide inventory varies within the same order of magnitude. For full PWR canisters, the average and combined canisters have similar total activity and radionuclide inventory. This illustrates that the decay power criterion will restrict the variation in radionuclide content.

Table 6-3. The total activity in the fuel matrix in a set of reasonable BWR and PWR canisters with a total decay power of 1,700 W at time for encapsulation /SKBdoc 1221579/.

Canister	Number of assemblies	Burnup (MWd/kgU))	Age of assemblies (years)	Total activity (10^{16} Bq/canister)
BWR low	12	30.7	20	2.1
BWR average	12	40.4	37	1.6
BWR high a	12	47.8	48	1.4
BWR high b	12	57	60	1.2
BWR unfilled	9	47.8	32	1.6
BWR-MOX	11	37.7	43	1.4
	1	50	50	
PWR low	4	34.2	20	2.0
PWR average	4	44.8	38	1.6
PWR high	4	57	55	1.3
PWR combination a	1	57	20	1.7
	3	34.2	40	
PWR combination b	2	57	51	1.5
	1	57	20	
PWR-MOX	3	44.8	32	1.6
	1	34.8	57	

Table 6-4. The inventory in the fuel matrix of thirteen radionuclides (Bq/canister) of importance for decay power, radiotoxicity and calculated long-term risk (alphabetic order) in the selected canisters at time for encapsulation /SKBdoc 1221579/.

Radionuclide inventory in fuel matrix for BWR canisters (Bq/canister)						
Radionuclide	**BWR low**	**BWR average**	**BWR high a**	**BWR high b**	**BWR unfilled**	**BWR-MOX**
Am-241	1.84E+14	2.99E+14	3.49E+14	3.84E+14	2.34E+14	4.48E+14
C-14	2.71E+10	2.84E+10	3.01E+10	4.79E+10	2.74E+10	2.59E+10
Cl-36	3.16E+08	4.54E+08	5.76E+08	1.67E+06	4.32E+08	4.23E+08
Cs-137	4.81E+15	4.21E+15	3.80E+15	3.38E+15	4.19E+15	3.48E+15
I-129	1.87E+09	2.54E+09	3.06E+09	3.71E+09	2.30E+09	2.51E+09
Nb-94	7.83E+06	1.16E+07	1.48E+07	1.88E+07	1.11E+07	1.20E+07
Pu-238	1.46E+14	2.38E+14	3.11E+14	3.87E+14	2.66E+14	2.59E+14
Pu-239	2.49E+13	2.52E+13	2.49E+13	2.45E+13	1.87E+13	3.30E+13
Pu-240	3.62E+13	4.67E+13	5.36E+13	6.06E+13	4.01E+13	6.43E+13
Pu-241	3.28E+15	1.80E+15	1.14E+15	6.83E+14	1.92E+15	1.69E+15
Sr-90	3.51E+15	2.81E+15	2.37E+15	1.96E+15	2.68E+15	2.23E+15
U-234	9.58E+10	1.02E+11	1.15E+11	1.38E+11	7.46E+10	1.11E+11
U-238	2.46E+10	2.44E+10	2.43E+10	2.40E+10	1.82E+10	2.44E+10
Radionuclide inventory in fuel matrix for PWR canisters (Bq/canister)						
Radionuclide	**PWR low**	**PWR average**	**PWR high**	**PWR com a**	**PWR com b**	**PWR-MOX**
Am-241	1.96E+14	3.26E+14	3.93E+14	2.64E+14	2.64E+14	3.67E+14
C-14	1.64E+10	2.26E+10	3.03E+10	1.99E+10	2.28E+10	1.99E+10
Cl-36	2.38E+08	3.23E+08	4.29E+08	2.86E+08	3.22E+08	2.85E+08
Cs-137	4.73E+15	4.10E+15	3.48E+15	4.17E+15	3.85E+15	3.86E+15
I-129	1.84E+09	2.50E+09	3.27E+09	2.20E+09	2.45E+09	2.27E+09
Nb-94	7.79E+06	1.15E+07	1.61E+07	9.87E+06	1.21E+07	1.13E+07
Pu-238	1.60E+14	2.45E+14	3.53E+14	2.18E+14	2.98E+14	2.41E+14
Pu-239	2.47E+13	2.54E+13	2.53E+13	2.48E+13	1.90E+13	2.56E+13
Pu-240	3.29E+13	4.08E+13	4.96E+13	3.69E+13	3.70E+13	4.86E+13
Pu-241	3.47E+15	1.87E+15	9.22E+14	2.23E+15	1.80E+15	2.19E+15
Sr-90	3.44E+15	2.72E+15	2.10E+15	2.81E+15	2.39E+15	2.49E+15
U-234	1.25E+11	1.27E+11	1.46E+11	1.27E+11	9.75E+10	1.02E+11
U-238	2.16E+10	2.14E+10	2.12E+10	2.15E+10	1.59E+10	1.92E+10

6.2.4 The type canisters and their radionuclide inventory

Based on the results presented in Table 6-3 and Table 6-4, demonstrating that the allowed total decay power in a canister will restrict the possible variation in radionuclide inventory and the canisters to be deposited according to the results of the simulation of the encapsulation presented in Section 5.2, the following type canisters have been selected as a basis for descriptions of the radionuclide inventory in individual canisters.

- BWR I: representing the largest part of the BWR canisters and of all deposited canisters (BWR average).
- BWR II: representing the high end radionuclide inventory in BWR canisters (BWR high a).
- BWR III: representing the unfilled BWR canisters (BWR unfilled).
- BWR-MOX: representing BWR canisters containing MOX assemblies (BWR-MOX).
- PWR I: representing the largest part of the PWR canisters (PWR average).
- PWR II: representing the high end radionuclide inventory in PWR canisters (PWR high).
- PWR III: representing the unfilled PWR canisters (PWR combination b).
- PWR-MOX: representing PWR canisters containing MOX assemblies (PWR-MOX).

The complete calculated radionuclide inventories for the type canisters are provided in Appendix C. Appendix C also includes the contribution to the total decay power in the canisters for the predominant radionuclides.

The selection of type canisters is made based on the burnup of the assemblies since it is the main parameter determining the radionuclide inventory. The BWR I and PWR I canisters have been selected since they represent the average canisters resulting from the simulation of the encapsulation of the assemblies to be deposited. The high burnup BWR II and PWR II canisters have been selected since they represent the high end canisters with respect to radionuclide inventory. The BWR III and PWR III canisters have been selected to represent the unfilled canisters, which are the result of the current decay power criterion and assumed encapsulation period. Finally, the BWR- and PWR MOX canisters have been selected to represent the canisters containing MOX assemblies.

In the simulation of the encapsulation the resulting total number of canisters is 6,110 comprising 4,451 BWR canisters, 1,652 PWR canisters and 7 canisters with fuel residues from Studsvik, see Section 5.2. The total numbers of the different type canisters is illustrated in Figure 6-1 and their radionuclide inventory is presented and discussed in the next section.

The BWR I type canister

The radionuclide inventory in the BWR I type-canister is set to the inventory in the canister denominated "BWR average" in Table 6-3 and Table 6-4. The average burnup of the assemblies in this canister is 40.4 MWd/kgU and the radionuclide inventory is regarded to be representative for all canisters where 12 BWR assemblies with different burnup and ages have been combined so that their total decay power conform to the decay power criterion (see Section 4.4.1) and their average burnup lies in the interval 38–42 MWd/kgU.

The BWR I type canister also represents the radionuclide inventory in canisters with 12 assemblies with an average burnup of the assemblies less than 38 MWd/kgU. Assuming the same radionuclide inventory in the canisters with an average burnup of less than 38 MWd/kgU as in the BWR average canister will result in an overestimated but still adequate inventory.

A summary of the canisters for which the BWR I canister is considered to provide an adequate description of the radionuclide inventory is given in Table 6-5. The radionuclide inventory in the BWR I canister is accounted for in Appendix C.

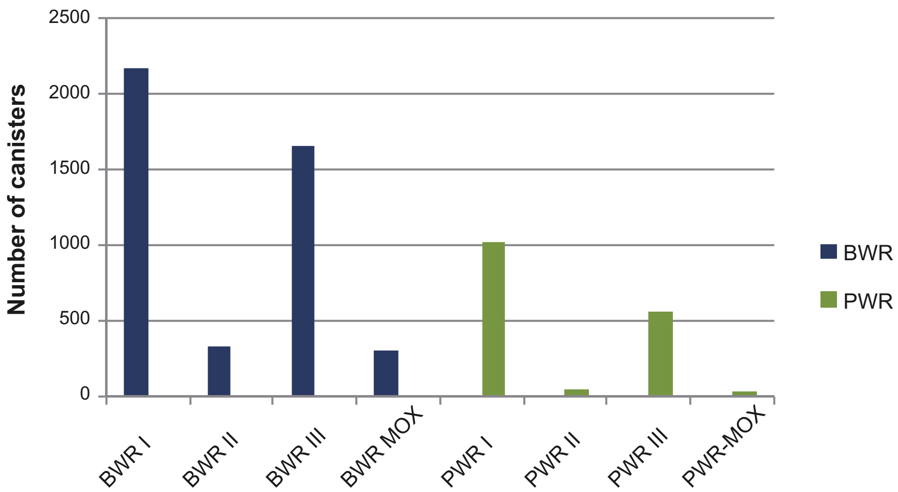


Figure 6-1. *The total numbers of the different kinds of type canisters.*

Table 6-5. The number of canisters for which the BWR I type canister provides an adequate description of the radionuclide inventory.

	Radionuclide inventory		
	Representative	Overestimated but adequate	Total
Number of canisters	547	1,661	2,208
Part of BWR canisters (4,451)	12%	37%	50%
Part of all canisters (6,110)	9%	27%	36%

The BWR II type canister

The radionuclide inventory in the BWR II type-canister is set to the inventory in the canister denominated “BWR high a” in Table 6-3 and Table 6-4. The average burnup of the assemblies in this canister is 47.8 MWd/kgU. With respect to the total inventory of assemblies to be deposited, the applied criteria for selection of assemblies and the assumed period for encapsulation and deposition, the radionuclide inventory in the BWR II canister is regarded as the high end of the BWR canisters to be deposited.

The radionuclide inventory in the BWR II canister represents all canisters where BWR assemblies with different burnup and age have been combined so that their total decay power is 1,700 W and their average burnup is at least 42 MWd/kgU. In most of these canisters, the average burnup of the assemblies will be less than 47.8 MWd/kg. The number of canisters for which the BWR II canister is considered to provide an adequate description of the radionuclide inventory is given in Table 6-6. The radionuclide inventory in the BWR II canister is accounted for in Appendix C.

Table 6-6. The number of canisters for which the BWR II type canister provides an adequate description of the radionuclide inventory.

	Representative radionuclide inventory
Number of canisters	321
Part of BWR canisters (4,451)	7%
Part of all canisters (6,110)	5%

The BWR III type canister

The radionuclide inventory in the BWR III type-canister is set to the inventory in the canister denominated “BWR unfilled” in Table 6-3 and Table 6-4. The average burnup of the assemblies in the BWR III type canister is 47.8 MWd/kgU. The BWR III canister represents all BWR canisters with 11 or fewer assemblies. The bulk of these canisters will have an average burnup higher than 42 MWd/kgU.

The content of short lived fission and activation products mainly depends on the burnup and age of the assemblies and will be similar as for the full canisters. However, the content of transuranium elements and isotopes with long half lives will mainly depend on the encapsulated mass of uranium and will, thus, be lower than in the canisters that contain 12 assemblies. The average number of assemblies in the unfilled BWR canisters is 8.7 assemblies. Seen over the full set of unfilled canisters, the radionuclide inventory in the BWR III canister is adequate and slightly overestimated. The radionuclide inventory in the canisters containing ten or eleven assemblies will be underestimated and the inventory in canisters containing fewer than nine assemblies will be overestimated but still adequate. The number of canisters containing eleven, ten, nine or less than nine assemblies is given in Table 6-7.

Table 6-7. The number of canisters containing eleven, ten, nine or less than nine assemblies for which the BWR III canisters provides an adequate description of the radionuclide inventory.

	Radionuclide inventory				Total
	Underestimated but adequate		Representative	Overestimated but adequate	
	11 assemblies	10 assemblies	9 assemblies	< 9 assemblies	
Number of canisters	10	173	738	734	1,655
Part of BWR canisters (4,451)	0% (0.2%)	4%	17%	16%	37%
Part of all canisters (6,110)	0% (0.2%)	3%	12%	12%	27%

The BWR-MOX type canister

The radionuclide inventory in the BWR-MOX type-canister is set to the inventory in the canister denominated "BWR-MOX" in Table 6-3 and Table 6-4. It is anticipated that only one MOX assembly will be encapsulated in any given canister. The burnup of the MOX assembly is set to the maximum allowed, i.e. 50 MWd/kgHM and the burnup of the remaining assemblies is 37.7 MWd/kgU. The radionuclide inventory in the BWR-MOX canister is regarded to be representative for all BWR canisters containing a MOX assembly.

The inventory is calculated for a potential MOX assembly from Oskarshamn. The swap MOX assemblies from Germany all have an essentially lower burnup and also a smaller mass of heavy metal. However, since the MOX assembly is one out of twelve, the impact on the total radionuclide content in the canister will be limited.

A summary of the canisters for which the BWR-MOX canister is considered to provide an adequate description of the radionuclide inventory is given in Table 6-8. The radionuclide inventory in the BWR-MOX canister is accounted for in Appendix C.

Table 6-8. The number of canisters for which the BWR-MOX type-canister provides an adequate description of the radionuclide inventory.

	Representative radionuclide inventory
Number of canisters	267
Part of BWR canisters (4,451)	6%
Part of all canisters (6,110)	4%

The PWR I type-canister

The radionuclide inventory in the PWR I type-canister is set to the inventory in the canister denominated "PWR average burnup" in Table 6-3 and Table 6-4. The average burnup of the assemblies in this canister is 44.8 MWd/kgU and the radionuclide inventory is regarded to be representative for all canisters where four PWR assemblies with different burnup and age have been combined so that their total decay power is 1,700 W and their average burnup lies in the interval 42–47 MWd/kgU. The PWR I type canister also represents full canisters with an average burnup of the assemblies less than 42 MWd/kgU. Assuming the same radionuclide inventory in these canisters as in the PWR average canister will result in an overestimated but still adequate inventory.

A summary of the canisters for which the PWR I canister is considered to provide an adequate description of the radionuclide inventory is given in Table 6-9. The radionuclide inventory in the PWR I canister is accounted for in Appendix C.

Table 6-9. The number of canisters for which the PWR I type canister provides an adequate description of the radionuclide inventory.

	Radionuclide inventory		
	Representative	Overestimated but adequate	Total
Number of canisters	633	391	1,024
Part of PWR canisters (1,652)	38%	24%	62%
Part of all canisters (6,110)	10%	6%	17%

The PWR II type-canister

The radionuclide inventory in the PWR II type-canister is set to the inventory in the canister denominated "PWR high burnup" in Table 6-3 and Table 6-4. The average burnup of the assemblies in this canister is 57 MWd/kgU. With respect to the total inventory of assemblies to be deposited, the applied criteria for selection of assemblies and the assumed period for encapsulation and deposition, the radionuclide inventory in the PWR II canister is regarded as the high end of the PWR canisters to be deposited.

The radionuclide inventory in the PWR II canister represents all canisters where PWR assemblies with different burnup and age have been combined so that their total decay power is 1,700 W and their average burnup is at least 45 MWd/kgU. In most of these canisters, the average burnup of the assemblies will be less than 57 MWd/kg/U. A summary of the canisters for which the PWR II canister is considered to provide an adequate description of the radionuclide inventory is given in Table 6-10. The radionuclide inventory in the PWR II canister is accounted for in Appendix C.

Table 6-10 The number of canisters for which the PWR II type canister provides an adequate description of the radionuclide inventory.

	Representative radionuclide inventory
Number of canisters	38
Part of PWR canisters (1,652)	2%
Part of all canisters (6,110)	1%

The PWR III type canister

The radionuclide inventory in the PWR III type-canister is set to the inventory in the canister denominated "PWR combination b" in Table 6-3 and Table 6-4. The PWR III canister represents all PWR canisters with three assemblies. Based on the results from the simulation of the encapsulation, there are no PWR canisters with less than three assemblies. The average burnup of the assemblies in this canister is 57 MWd/kgU. The bulk of the canisters with three assemblies will have an average burnup lower than this.

The content of short lived fission and activation products mainly depends on the burnup and age of the assemblies and will be similar as for the full canisters. The content of transuranium elements and isotopes with long half lives will mainly depend on the encapsulated mass of uranium and will, thus, be lower than in the canisters that contain four assemblies. The number of canisters containing three assemblies is given in Table 6-11.

Table 6-11 The number of canisters containing three assemblies for which the PWR III type canister provides an adequate description of the radionuclide inventory.

	Representative radionuclide inventory and total
Number of canisters	557
Part of PWR canisters (1,652)	34%
Part of all canisters (6,110)	9%

__The PWR-MOX canister__

The radionuclide inventory in the PWR-MOX type-canister is set to the inventory in the canister denominated "PWR-MOX" in Table 6-3 and Table 6-4. Each PWR-MOX canister contains one swap PWR MOX assembly. The burnup of the MOX assembly is set to 34.8 MWd/kgHM, which is the maximum burnup of the swap PWR MOX assemblies. The burnup of the remaining three assemblies is 44.8 MWd/kgU, i.e. the average PWR assembly burnup. The radionuclide inventory in the PWR-MOX canister is regarded to be representative for all PWR canisters containing a MOX assembly.

A summary of the canisters for which the PWR-MOX canister is considered to provide an adequate description of the radionuclide inventory is given in Table 6-12. The radionuclide inventory in the BWR-MOX canister is accounted for in Appendix C.

Table 6-12 The number of canisters for which the BWR-MOX type-canister provides an adequate description of the radionuclide inventory.

	Representative radionuclide inventory
Number of canisters	33
Part of PWR canisters (1,652)	2%
Part of all canisters (6,110)	1%

6.2.5 Comparison of inventory in type canisters with total inventory

To inspect whether the type canisters provide a representative and adequate description of the radionuclide inventory in the canisters, the summed up total inventory in all type-canisters has been compared with the total radionuclide inventory calculated for all assemblies in the reference scenario for the operation of the nuclear power plants. The resulting total inventories of thirteen radionuclides of importance for decay power, radiotoxicity and calculated long-term risk and the summed up inventory of all nuclides are given in Table 6-13. The table includes the summed up inventory in all the different type canisters as well as in all canisters in the repository independently of type. The full inventories are given in Appendix C.

Table 6-13 The total inventories of thirteen radionuclides of importance for decay power, radiotoxicity and calculated long-term risk in all type canisters and all canisters in the repository (calculated for the year of encapsulation) compared to the total inventory calculated for the calendar year 2045 for the reference scenario for the operation of the nuclear power plants.

Radio-nuclide	BWR I	BWR II	BWR III	BWR-MOX	PWR I	PWR II	PWR III	PWR-MOX	Total in all type canisters	Total for the reference scenario[1]
Am-241	6.6E+17	1.1E+17	3.9E+17	1.2E+17	3.3E+17	1.5E+16	1.5E+17	1.2E+16	1.8E+18	1.7E+18
C-14	2.2E+14	3.7E+13	1.5E+14	2.6E+13	7.3E+13	3.3E+12	3.6E+13	2.2E+12	5.5E+14	5.1E+14
Cl-36	1.0E+12	1.9E+11	7.3E+11	1.1E+11	3.4E+11	1.7E+10	1.8E+11	9.6E+09	2.6E+12	2.3E+12
Cs-137	9.3E+18	1.2E+18	6.9E+18	9.3E+17	4.2E+18	1.3E+17	2.1E+18	1.3E+17	2.5E+19	2.3E+19
I-129	5.6E+12	9.8E+11	3.8E+12	6.7E+11	2.6E+12	1.2E+11	1.4E+12	7.5E+10	1.5E+13	1.4E+13
Nb-94	2.0E+13	3.4E+12	1.3E+13	2.3E+12	6.1E+14	2.6E+13	2.8E+14	1.9E+13	9.8E+14	9.3E+14
Pu-238	5.3E+17	1.0E+17	4.4E+17	6.9E+16	2.5E+17	1.3E+16	1.7E+17	8.0E+15	1.6E+18	1.3E+18
Pu-239	5.6E+16	8.0E+15	3.1E+16	8.8E+15	2.6E+16	9.6E+14	1.1E+16	8.5E+14	1.4E+17	1.4E+17
Pu-240	1.0E+17	1.7E+16	6.6E+16	1.7E+16	4.2E+16	1.9E+15	2.1E+16	1.6E+15	2.7E+17	2.5E+17
Pu-241	4.0E+18	3.7E+17	3.2E+18	4.5E+17	1.9E+18	3.5E+16	1.0E+18	7.2E+16	1.1E+19	1.1E+19
Sr-90	6.2E+18	7.6E+17	4.4E+18	6.0E+17	2.8E+18	8.0E+16	1.3E+18	8.2E+16	1.6E+19	1.6E+19
U-234	2.2E+14	3.7E+13	1.2E+14	3.0E+13	1.3E+14	5.5E+12	5.4E+13	3.4E+12	6.1E+14	6.0E+14
U-238	5.4E+13	7.8E+12	3.0E+13	6.5E+12	2.2E+13	8.0E+11	8.8E+12	6.3E+11	1.3E+14	1.3E+14

[1] Detailed information in Table 6-1 and Appendix C.

The total inventory for the reference scenario for the operation of the nuclear power plants has been calculated for the age of the spent fuel in the year the last reactor will be taken out of operation, i.e. 2045. The encapsulated assemblies have been selected for encapsulation based on their decay power and will have different ages. The ages will depend on the burnup and age of the assemblies the year they were selected for encapsulation. The span of ages of the encapsulated assemblies at encapsulation will be in the range 20–50 years. The fact that different ages have been used when calculating the total inventory will impact the inventory of nuclides with half-lives of about 30 years or shorter. In the long-term time perspective only radionuclides with long half-lifes will remain in the repository. Consequently, regarding the inventory of nuclides remaining in the long-term time perspective, the ages at encapsulation will be of minor importance. The inventories of long-lived nuclides at encapsulation will mainly depend on the burnup and amount of heavy metal.

As indicated by the results given in Table 6-13, it can be concluded that the total radionuclide inventory calculated solely on the basis of the spent fuel assemblies to be deposited and the summed radionuclide inventory in the deposited canisters as calculated based on the selected type canisters are similar.

6.2.6 Uncertainties in the calculated fuel matrix radionuclide inventories

Origen-S

The computer program Origen (**O**ak **R**idge **i**sotope **gen**eration) used for calculating the radionuclide inventory in the fuel matrix is developed as part of the code package Scale (**S**tandardised **c**omputer **a**nalyses for **l**icensing **e**valuation). The program package which contains input data libraries and preparation codes as well as simulation codes is regularly developed and maintained according to quality assurance procedures approved by U.S. Nuclear Regulatory Commission and Department of Energy /SKBdoc 1198314/.

The uncertainties in the code can be estimated by comparison to measured activities. Comparisons between nuclide contents calculated with Origen and measured contents are compiled in /SKBdoc 1221579/. The results are given in Table 6-14 the agreement is good and within the uncertainty interval of the measured data.

Uncertainties in data

The uncertainties in the calculated inventories are mainly related to the used input data. The calculated inventories are based on four kinds of input:

- decay constants,
- fission product yield,
- when calculating dose rates conversion factors to kerma and
- cross-section libraries for neutron and gamma reactions.

The decay constants, fission product yield and conversion factors to kerma are regarded as well known or of minor importance for the uncertainties of the calculated inventories /Håkansson 2000/.

Table 6-14 Comparison between nuclide contents calculated with Origen and measured contents.

Fuel type	Nuclide	Measeured/calculated mean inventories
PWR	U and Pu	1.01
	Other actinides	1.11
	Fission products	1.01
BWR	U and Pu	1.02
	Other actinides	0.94
	Fission products	0.93

The used cross-section library will impact the results. The used libraries are regarded as representative for modern fuel types /SKBdoc 1198314/. Further, the input data is compiled as detailed and similar to the actual conditions as possible. Axial differentiated enrichment, power density and burnup are considered. The power history is differentiated for the cycles in the reactor with declining power in the assemblies and utilisation periods with intervening decay intervals to simulate the revision periods with interruption in the operation of the power plants. The same input are used in /SKBdoc 1198314/ and /SKBdoc 1222975/.

To estimate the uncertainties in the radionuclide inventory a parameter study was performed /SKBdoc 1198314/. In the study the enrichment, power density and cross-section library was altered. In addition, for PWR the content of Gd (gadolinium) in the assemblies has been altered. In /SKBdoc 1077122/ the radionuclide inventories calculated by Origen-S are considered as realistic and well reflecting the real inventories, also for high burnups. The content of fission products, e.g. Cs-137, is primarily determined by the burnup and therefore the uncertainties in the calculations are considered small, typically ±5%. The production of transuranium elements is more sensitive to the studied parameters and the uncertainties are estimated to ±20%. However, the content of transuranium elements is strongly correlated to the burnup, i.e. alterations of burnup affects the content more than alterations of the studied parameters. In /SKBdoc 1198314/ it was concluded that the uncertainties can be handled by selecting conservative burnups.

Based on the discussions in /SKBdoc 1198314/ the radionuclide inventories in the fuel matrix presented in this report can be regarded as representative for the presented scenarios for the operation of the power plants and encapsulation. The total radionuclide inventory in the final repository has been calculated for assemblies with mean burnup. Since the number of assemblies with lower and higher burnup respectively is similar this will yield a reasonable estimation of the total inventory. However, the inventory of transuranium elements is to some extent underestimated. This is due to that the relationship between the content of transuranium elements in an assembly and its burnup is not linear, but the generation of transuranium elements will be higher for the last produced MWh than for the first.

The inventory in individual canisters has been calculated based on the burnup of the assemblies, also in this case the presented radionuclide inventories can be regarded as representative for the presented scenarios. Following the recommendation in /SKBdoc 1198314/ conservative inventories can be yielded by assuming only high burnup canisters in the repository.

6.2.7 Uncertainties in the calculated radionuclide inventories in construction materials and crud

IndAct and CrudAct

In the computer program IndAct the material composition, radiated mass and neutron fluxes are specified for each part of the fuel assembly. The computer program CrudAct uses the same basis for activation as IndAct. This results in differentiated and correct calculations of the activation. The codes have been reviewed and used for licensing purposes. For CrudAct measusrements have been used for validation of the code. /SKBdoc 1198314/ regards the calculated radionuclide inventories as realistic.

Uncertainties in data

The input to IndAct and CrudAct comprises general radio physical data, material data and reactor specific data. Both for the IndAct and CrudAct calculations realistic data from the Oskarshamn-3 and Ringhals-3 reactors have been used for BWR and PWR assemblies respectively /SKBdoc 1198314/.

The content of activation products will depend on the material composition of the components. The content of Co-60 is of special interest since it is the dominant radiation source for the activation products. The induced activity in control rods will depend on the time the rods have been inserted in the reactor core. The calculations are based on normal use of the control rods. The amount of crud is based on measurements, and is considered to be realistic. However, the amount of crud will vary between reactors. In /SKBdoc 1198314/ the uncertainty in the calculated induced activity in the construction materials and construction materials released to the crud have been estimated to a factor of two, and multiplication with two to obtain conservative values is recommended.

The content of transuraniums in the crud is continuously inspected during the operation and can be regarded as a mean value for the Swedish reactors /SKBdoc 1198314/. The content of transuraniums in the crud depends on the occurrence of damages on the fuel assemblies. There are examples of damages that have caused an increase of the content of transuraniums in the crud of a factor of ten. Consequently, there are a few assemblies with up to ten times more transuraniums in the crud than the contents stated in this report.

6.3 Fission gas release and gap inventory

6.3.1 Fission gas release in the spent fuel assemblies

As stated in Section 6.2.1 and discussed in /Werme et al. 2004/, the fraction of the radionuclides in the fuel/cladding gap is considered to be proportional to the fission gas release (FGR). The FGR is strongly correlated to the linear heat generation rate, which in turn will depend on the thermal power of the nuclear reactor, the number of assemblies and configuration of the assemblies in the reactor core and on how the fuel assemblies are utilised during operation. High fission gas release is to be avoided because it may cause high pressure inside the fuel rod that may cause damages on the rods. Further, in case of a severe reactor accident, the amount of gases that potentially can escape from the reactor should be kept low.

The nuclear fuel suppliers and power companies have developed fuel performance codes that, among other fuel data, can be used to calculate the fission gas release. The codes have been developed to support a safe and efficient operation of the nuclear reactors. The correlations between calculated and measured fuel data have been investigated and the codes are validated for fuel rod performance and licensing analysis. Such codes have been used to calculate the fission gas release for a set of BWR and PWR reactor operational cases. The BWR cases are presented in /Oldberg 2009/ and the PWR cases in /Nordström 2009/. The purpose of the calculations was to investigate the fission gas release after the planned increases in thermal powers and burnup. The results confirm the strong correlation between the linear heat generation and the FGR. If the linear heat generation exceeds a threshold value, the FGR will increase significantly.

The calculated average fission gas releases for the operational cases in Table 6-15 are given in Figure 6-2 for the BWR cases and Figure 6-3 for the PWR cases. The average fission gas release has been calculated per batch of fuel assemblies after each irradiation cycle in the reactor. Uncertainties and complete data are given in /Oldberg 2009/ for the BWR cases and in /Nordström 2009/ for the PWR cases.

Table 6-15. Operational cases for which the FGR have been calculated.

Type of reactor	Reactor	Operational case	Thermal power (MW)	Batch average discharege burnup (MWd/kgU)
BWR	KKL[1]	Case 1	3,600	59.8
BWR	O3	Case 2	3,292	58.0
BWR	O1	Case 3	1,375	60.4
BWR	O3	Case 4	3,292	44.3
BWR	O3	Case 5	3,900	43.9
BWR	O1	Case 6	1,375	44.9
PWR	R2	Equilibrium	2,652	59.4
PWR	R2	Cycle 33	2,652	48.4
PWR	R4	Equilibrium	3,292	59.8
PWR	R4	Cycle 26	2,775	52.4

[1] Kernkraftwerk Leibstadt in Germany, a reactor with high average discharge burnups.

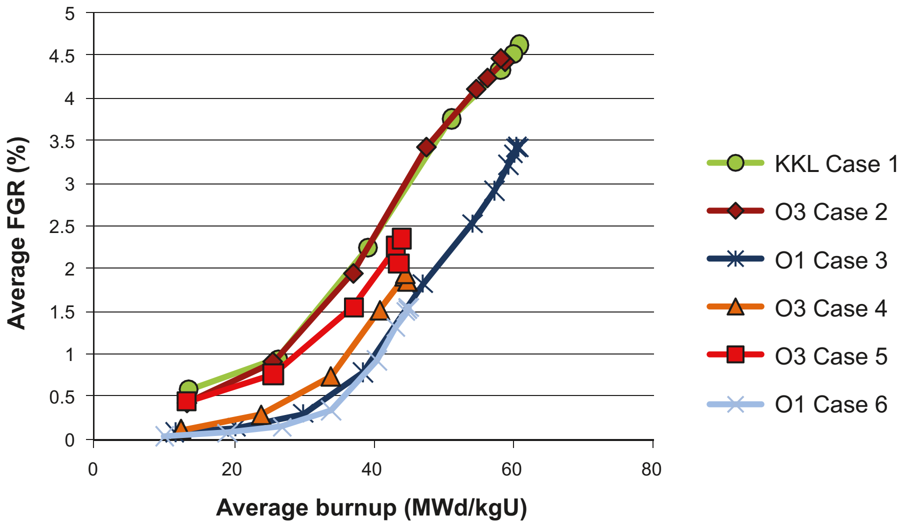


Figure 6-2. *Calculated average fission gas release at the end of each cycle for the BWR cases.*

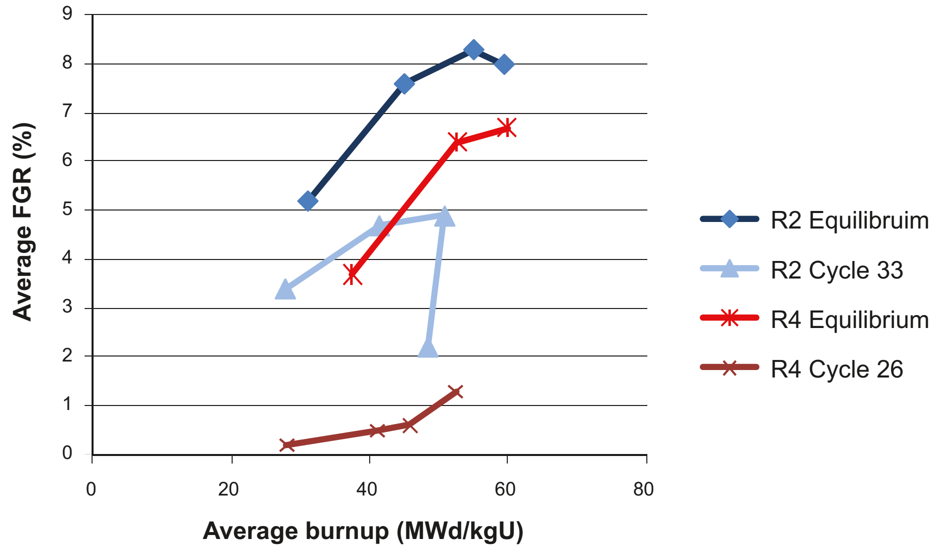


Figure 6-3. *Calculated average fission gas release at the end of each cycle for the PWR cases. The drop in burnup and FGR for R2 (Ringhals 2) Cycle 33 is explained by that only the low burnup assemblies were loaded in the last cycle.*

In /SKBdoc 1222975/ the relations illustrated in Figure 6-2 and Figure 6-3 are used to extrapolate reactor-specific relations between average burnup and FGR. In the interpolation, the numbers of assemblies in the reactors and their thermal powers have been considered. The relations are based on the assumption that the FGR is correlated to the linear heat generation. The interpolated relations between burnup and FGR have then been used to estimate the FGR of the spent fuel assemblies included in the reference scenario for the operation of the nuclear power plants.

From the average burnup of each assembly, the reactor it has been used in, and whether it was used before or after the increase in power, the extrapolated reactor-specific relations were used to estimate the FGR in each individual assembly. The resulting average FGR for all BWR assemblies is 1.9% with a standard deviation of ±1.13%. The number of BWR assemblies in different FGR intervals is illustrated in Figure 6-4. The resulting average FGR for all PWR assemblies is 4.3% with a standard deviation of ±3.11%. The number of PWR assemblies in different FGR intervals is illustrated in Figure 6-5.

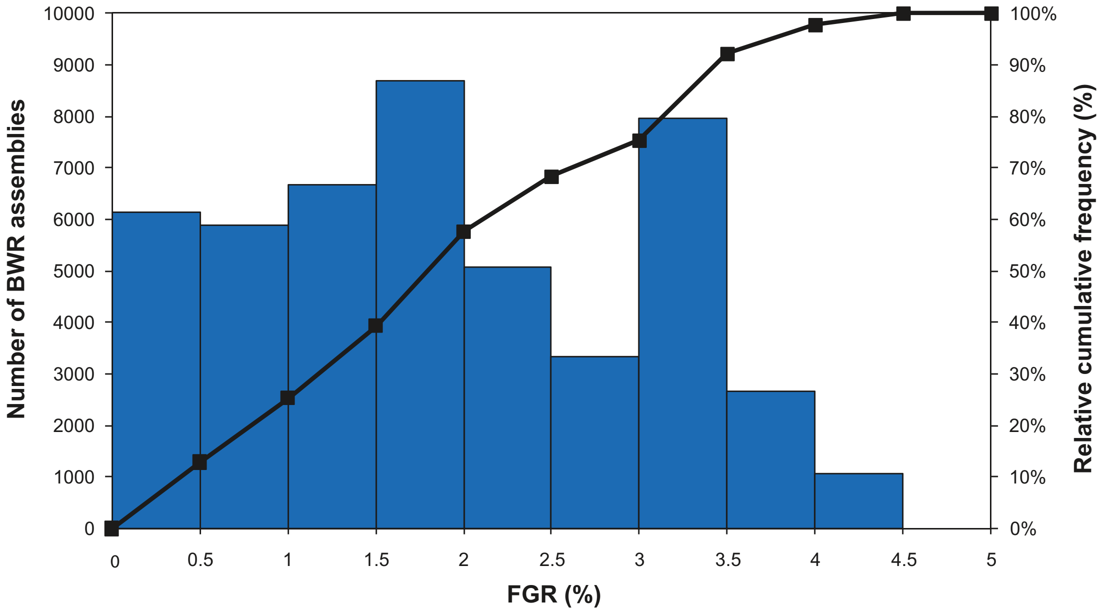


Figure 6-4. *The number of BWR assemblies in different FGR intervals and relative cumulative frequency of FGR in BWR assemblies.*

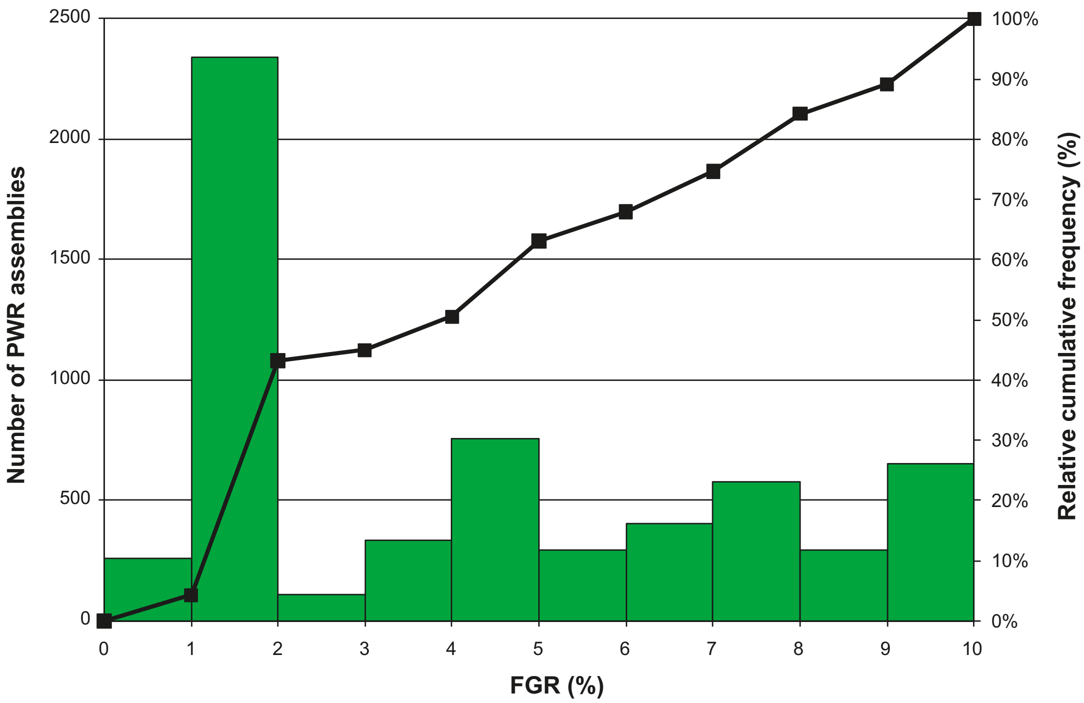


Figure 6-5. *The number of PWR assemblies in different FGR intervals and relative cumulative frequency of FGR in PWR assemblies.*

6.3.2 The fission gas release in the type canisters

The fission gas release of the assemblies in the different type canisters has been estimated based on the average burnup of the assemblies in the canisters. However, the FGR in an assembly is strongly correlated to the linear power, and the linear power can vary between assemblies with similar burnup depending on in which reactor and to which power the assembly has been utilised. With respect to this, the following approach was applied. First, the assemblies were sorted based on their burnup. Then the FGR in the different type canisters was calculated as the average FGR of all the BWR or PWR assemblies in an interval of ±5 MWd/kgU from the average burnup in each type canister. The resulting estimated FGR for the assemblies in the type canisters is given in Table 6-16.

Note that the high burnup assemblies with potentially high FGR must be combined with assemblies with lower burnup in order for the decay power of the encapsulated assemblies to conform to the criterion for acceptable decay power in a canister.

For the PWR-MOX canister the FGR was not estimated since the information required to estimate reactor-specific relations between burnup and FGR was not available for the German reactors from which the PWR-MOX assemblies originate. With respect to the low burnup of the MOX assembly and an average burnup close to that of the PWR I canister, the FGR in these canisters can be assumed to be similar or less than in the PWR I type canister.

6.4 Decay power

The decay power will be calculated for all assemblies before they are selected for encapsulation. A margin is added to the calculated decay power to ensure that the actual decay power conform to the criteria 1,700 W. Based on comparisons between calculated and measured decay powers, the uncertainty in calculated decay power is estimated to 2%, and the current selection of assemblies is made so that the total calculated decay power of the assemblies in a canister does not exceed 1,650 W. The decay power of each assembly, if required, can be measured in conjunction with the delivery to the encapsulation part of the Clink facility.

At the initial state, the decay power is expected not to exceed 1,700 W in any canister. In most of the canisters, the decay power is expected to be slightly less than 1,700 W.

6.5 Encapsulated gases and liquids

The content of gases and liquids depends on the result of the drying and gas exchange. At this stage of development, it is not possible to provide detailed information on the actual content of gases or liquids. However, the technique for drying is well known from countries applying dry storage, and the drying process presented in Section 4.6 is used in many of these countries. Change of atmosphere in a tight container is also a well-known technique. In summary, there is no reason to believe that the water content in the canisters will exceed the 600 g given as a premise or that the argon content will be below the acceptable level of at least 90%.

Table 6-16. Estimated FGR of the assemblies in the different type canisters.

BWR canisters		PWR canisters	
Type canister	FGR (%)	Type canister	FGR (%)
BWR I	1.3	PWR I	2.9
BWR II	2.1	PWR II	7.8
BWR III	2.1	PWR III	7.8
BWR-MOX	2.3	PWR-MOX	(2.9)

6.6 Radiation at the canister surface

The radiation dose rate at the canister surface has been calculated in /SKBdoc 1077122/. The highest obtained radiation dose rate was 0.18 Gy/h on limited areas of the canister. These levels were obtained for PWR canisters containing four assemblies with a burnup of 60 MWd/kgU and an age of 30 years. The total decay power of four such assemblies will exceed the allowed total decay power in a canister. Since the decay power as well as the radiation are related to the radioactivity of the assemblies, it can be concluded that the radiation dose rate on the canister surface will be well below the acceptable 1.0 Gy/h as long as the fuel assemblies selected for encapsulation conform to the decay power criterion.

6.7 Criticality

The assemblies must not under any circumstances be encapsulated if the criticality criteria cannot be met. Loading curves for check of criticality have been calculated for both BWR and PWR canisters. When calculating the loading curves, all uncertainties regarding the propensity for criticality have been systematically investigated according to principles that are generally applied for all handling of nuclear fuel /SKBdoc 1193244/. Before encapsulation, each individual assembly is checked against the loading curve. If an individual assembly does not conform to the acceptance criteria in this check, an inspection of the specific set of assemblies selected for encapsulation will be made. If it can not be shown that the selected combination of assemblies conforms to the criticality criteria, a new selection will be made. If it is not possible to combine an individual assembly with other assemblies to conform to the criticality criterion it can be encapsulated alone in a canister. If this is not sufficient, the ultimate measure will be to alter the geometry, i.e. to reconstruct the assembly.

6.8 Dimensions and other parameters of interest for the safety assessment

A number of suppliers of nuclear fuels have been and will be used by the different nuclear power plants. The detailed design of the assemblies can vary between suppliers, see Section 2.3.2. The fuel assemblies used in the calculations of decay power, radionuclide inventory, criticality and fission gas release are given in Appendix A which also includes detailed information about geometries and materials of these kinds of assemblies.

7 References

SKB's (Svensk Kärnbränslehantering AB) publications can be found at www.skb.se/publications.
References to SKB's unpublished documents are listed separately at the end of the reference list.
Unpublished documents will be submitted upon request to document@skb.se.

Canister production report, SKB 2010. Design, production and initial state of the canister. SKB TR-10-14, Svensk Kärnbränslehantering AB.

Design premises long-term safety, SKB 2009. Design premises for a KBS-3V repository based on results from the safety assessment SR-Can and some subsequent analyses. SKB TR-09-22, Svensk Kärnbränslehantering AB.

Repository production report, SKB 2010. Design and production of the KBS-3 repository. SKB TR-10-12, Svensk Kärnbränslehantering AB.

Underground openings construction report, SKB 2010. Design, construction and initial state of the underground openings. SKB TR-10-18, Svensk Kärnbränslehantering AB.

Håkansson R, 2000. Beräkning av nuklidinnehåll, resteffekt, aktivitet samt doshastighet för utbränt kärnbränsle. SKB R-99-74, Svensk Kärnbränslehantering AB (in Swedish).

Nordström E, 2009. Fission gas release data for Ringhals PWRs. SKB TR-09-26, Svensk Kärnbränslehantering AB.

Oldberg K, 2009. Distribution of fission gas release in 10×10 fuel. SKB TR-09-25, Svensk Kärnbränslehantering AB.

SKB, 2006. Measurements of decay heat in spent nuclear fuel at the Swedish interim storage facility, Clab. SKB R-05-62, Svensk Kärnbränslehantering AB.

SKB, 2009. Plan 2008. Costs starting in 2010 for the radioactive residual products from nuclear power. Basis for fees and guarantees in 2010 and 2011. SKB TR-09-23, Svensk Kärnbränslehantering AB.

SKB, 2009. Design premises for a KBS-3V repository based on results from the safety assessment SR-Can and some subsequent analyses. SKB TR-09-22, Svensk Kärnbränslehantering AB.

Werme L, Johnson L H, Oversby V, King F, Spahiu K, Grambow B, Shoesmith D W, 2004. Spent fuel performance under repository conditions: A model for use in SR-Can. SKB TR-04-19, Svensk Kärnbränslehantering AB.

Unpublished documents

SKBdoc id, version	Title	Issuer, year
1077122, 2.0	Strålskärmsberäkningar för kopparkapslar innehållande BWR, MOX och PWR bränsleelement	SKB, 2009
1172138, 1.0	Kontroll av kärnämnen inom KBS-3-systemet	SKB, 2009
1193244, 4.0	Criticality safety calculations of disposal canisters	SKB, 2010
1198314, 1.0	Källstyrkor för bränsleelement under driftskede för Clink, slutförvarsanläggning och slutförvar	SKB, 2009
1221567, 2.0	Simulering av fyllning av kapslar för slutförvaring av utbränt kärnbränsle	SKB, 2010
1221579, 2.0	Aktivitetsinnehåll i kapslar för slutförvar	SKB, 2010
1222975, 2.0	Beräkning av fissionsgasfrigörelse för bränslet i slutförvaret	SKB, 2010

Appendix A

Fuel types

Different types of fuel assemblies have been used and will be used in the operation of the Swedish nuclear power plants. Information on the number of fuel assemblies of representative fuel types stored in the pools in Clab and in the storage pools at the reactors as well as the fuel in the reactor cores at the end of December 2008 is given in Table A-1. There are minor variations of the different fuel types, all these variations are not included in Table A-1. However, the table contains a set of fuel types representative for the current inventory of spent fuel assembles. The BWR and PWR fuel assemblies used in the calculations of the radionuclide inventory, criticality, decay power and fission gas release are given in Table A-2. In Table A-3 and A-4 main parameters for fuel assemblies representative for Swedish fuel are given.

Table A-1. Types of fuel assemblies used in Swedish reactors until the end of 2008.

Fuel assembly type		No of fuel assemblies[1]	Comment
BWR	AA8×8	8,720	
	Exxon 8×8	1,040	
	KWU 8×8-2	120	
	ANF 9×9-5	620	
	KWU 9×9-5	730	
	KWU 9×9-Q	380	
	Atrium 9A, 9B	190	
	Atrium 10B (incl XM,XP)	3,380	More will come
	GE11S	110	
	GE12S	480	
	Svea 64 (incl. 64-1)	4,660	
	Svea 100	3,140	
	Svea 96 (incl. 96S/L)	1,970	
	Svea 96 Optima	480	
	GE14	750	More will come
	Svea 96 Optima 2	1,200	More will come
	Svea 96 Optima 3		More will come
BWR MOX	AA 8×8 (MOX O1)	3	
PWR	W15×15	370	
	Areva 15×15	170	
	KWU 15×15	640	
	W 17×17	520	
	F 17×17	890	
	AA 17×17	170	
	F 17×17 AFA3G (incl. variants)	270	
	17×17 HTP (incl. X5,M5)	330	More will come

[1] To be noted the numbers are rounded.

Table A-2. Types of fuel assemblies used in calculations for this report.

Fuel assembly type		Calculations
BWR	Svea 96 Optima 2	Decay power Radionuclide inventory Fission gas release
	Svea 96 Optima 3	Criticality
PWR	Areva 15×15 AGORA-5A	Fission gas release
	17×17 Areva AGORA-7H. (AGORA-7H is not used in Swedish PWR:s	Fission gas release
	15×15 AFA3G F 17×17	Criticality calculations Decay power Radionuclide inventory

Table A-3. Main parameters for BWR fuel assemblies /SKBdoc 1193244/.

Fuel type	AA 8x8	Exxon 8x8	KWU 8x8-2	ANF 9x9-5	KWU 9x9-5	KWU 9x9-Q	Atrium 9A	Atrium 10B	Atrium 10 XM	Atrium 10 MOX	Atrium 10 XP	GE11 S	GE12 S	GE14	GNF 2	Svea 64	Svea 100	Svea 100	Svea 96	Svea 96	Svea 96	Svea 96 Optima	Svea 96 Optima 2	Svea 96 Optima 3
No of fuel rods	63	63	62	76	76	72	72	91	91	91	91	74	91	92	92	63	100	100	96	96	96	96	96	96
No water rods	1	1	2	5	5	9	9	9	9	9	9	17	9	8	8	1	0	0	4	4				
UO2 density (g/cc)	10.4	10.5	10.5	10.4	10.5	10.5	10.6	10.6	10.8	10.8	10.5	10.6	10.7	10.4	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.6	10.6
Fuel rod pitch (mm)	16.3	16.3	16.3	14.5	14.5	14.5	14.5	13	13	13	13	14.4	13	13	13	15.8	12.4	12.7	12.4	12.7	12.7	12.7	12.8	12.8
Fuel rod outer diameter (mm)	12.3	12.3	12.3	10.6	11	11	11	10.1	10.3	10.3	10.3	11.2	10.3	10.3	10.3	12.3	9.62	9.62	9.62	9.62	9.62	10.3/9.62	9.84	9.84
Fuel rod inner diameter (mm)	10.7	10.7	10.7	9.11	9.66	9.66	9.67	8.86	9.04	9.04	9.1	9.76	8.98	8.94	9.06	10.7	8.36	8.36	8.36	8.36	8.36	8.94/ 8.36	8.63	8.63
Cladding thickness (mm)	0.8	0.84	0.82	0.74	0.67	0.67	0.67	0.6	0.62	0.62	0.59	0.71	0.64	0.66	0.6	0.8	0.63	0.63	0.63	0.63	0.63			
Cladding material	Zr2	Zr2	Zr2	Zr2	Zr2	Zr2	Zr2	Zr2	Zr2	Zr2	Zr2	Zr2	Zr2	Zr2	Zr2	Zr2	Zr2	Zr2	Zr2	Zr2	Zr2			
Pellet diameter (mm)	10.4	10.3	10.4	9.49	9.5	9.5	9.5	8.67	8.87	8.87	8.87	9.55	8.81	8.76	8.88	10.4	8.19	8.19	8.19	8.19	8.19	8.77/8.19	8.48	8.48
Water rod outer diameter (mm)	12.3	12.3	15	14	13.2	20.9x 20.9	37.5x 37.5	35.15x 35.15				24.9	24.9	24.9	24.9	12.3								
Water rod inner diameter (mm)	10.7	10.7	13.4	13.4	11.6							23.4	23.4	23.4	23.4	11.1								
Cladding thickness (mm)	0.8	0.84	0.8	0.3	0.78	0.72	0.73	0.73				0.76	0.76	0.76	0.76	0.6								
Compartment outer measures (mm)	139	139	139	139	139	139	139	139	139	139	139	137	137	137	138	140	140	140	140	140	140	140	140	140
Compartment inner measures (mm)	134	134	134	134	134	134	134	134	134	134	134	134	134	134	134	137	137	137	137	137	137	137	137	137
Compartment wall thickness (mm)	2.3	2.3	2.3	2.3	2.3	2.3	2.3	2.3	2.3	2.3	2.3	1.65	1.65	1.65	1.65	1.1	1.1	1.1	1.1	1.1	1.4	1.1	1.1	1.4
Compartment material	Zr	Zr	Zr	Zr	Zr	Zr	Zr	Zr	Zr	Zr	Zr	Zr	Zr	Zr	Zr	Zr	Zr	Zr	Zr	Zr	Zr	Zr	Zr	Zr
Central cross inner width (mm)																29.7	29.7	29.7	29.7	29.6	29.6	29.6	29.6	29.6
Central cross wall thickness (mm)																0.8	0.8	0.8	0.8	0.8	0.8	0.8	0.8	0.8
Central compartment outer measures (mm)									33.4x 33.4	33.4x 33.4	33.4x 33.4													
Central compartment outer measures (mm)																								
Central compartment wall thickness									0.8	0.8	0.8													
Central compartment material									Zr	Zr	Zr													
Active fuel length (mm)	3712					3680		3712					3712	3680	3680	3712								

Table A-4. Main parameters for PWR fuel assemblies /SKBdoc 1193244/.

Fuel type	W15x15	KWU15x15	F15x15AFA3G	15x15AGORA	W17x17	AA17x17	F17x17	S17x17HTP	17x17 HTP X5	17x17 HTP M5	17x17 HTP X5	17x17 AFA3G
No of fuel rods	204	204	204	204	264	264	264	264	264	264	264	264
Fuel rod pitch (mm)	14.3	14.3	14.3	14.3	12.6	12.6	12.6	12.6	12.6	12.6	12.6	12.6
Fuel rod outer diameter (mm)	10.72	10.75	10.72	10.77	9.5	9.5	9.5	9.55	9.55	9.5	9.5	9.5
Fuel rod inner diameter (mm)	9.48	9.3	9.484	9.505	8.36	8.36	8.36	8.33	8.33	8.364	8.35	8.355
Cladding thickness (mm)	0.62	0.725	0.618	0.6325	0.57	0.57	0.57	0.61	0.61	0.568	0.575	0.5725
Pellet diameter (mm)	9.29	9.11	9.294	9.33	8.19	8.19	8.19	8.17	8.165	8.192	8.192	8.192
Cladding material	Zr4	Zr4	M5	Zr4	Zr2	Zr4	Zr4	Zr4	Zr4	M5	M5	Zr4
Active fuel length (mm)	3658	3658	3658	3658	3658	3658	3658	3658	3658	3658	3658	3658
UO2 density (g/cc)*	10.7	10.7	10.7	10.7	10.7	10.7	10.7	10.7	10.7	10.7	10.7	10.7
No of guide tubes	20	20	20	20	24	24	24	24	24	24	24	24
Guide tube material	Zr4	Zr4	M5	M5	Zr4	Zr4	Zr4	PCAm	PCAm	PCAm	PCAm	Zr4
Guide tube outer diameter (mm)	13.87	13.86	14.1	14.1	12.09	12.24	12.05	12.24	12.45	12.45	12.24	12.45
Guide tube inner diameter (mm)	13.01	13	13.05	13.05	11.05	11.44	11.25	11.3	11.45	11.45	11.3	11.45
Guide tube cladding thickness (mm)	0.43	0.43	0.525	0.525	0.52	0.4	0.4	0.47	0.5	0.5	0.47	0.5
No of instrument tubes	1	1	1		1	1	1	1	1	1	1	1
Instrument tube material	Zr4	Zr4	M5	M5	Zr4	Zr4	Zr4	PCAm	PCAm	PCAm	PCAm	Zr4
Instrument tube outer diameter (mm)	13.87	13.89	14.1	14.1	12.24	12.24	12.05	12.24	12.24	12.45	12.24	12.45
Instrument tube inner diameter (mm)	13.01	13.03	13.05	13.05	11.428	11.428	11.25	11.3	11.3	11.45	11.3	11.45
Instrument tube cladding thickness (mm)	0.43	0.43	0.525	0.525	0.406	0.406	0.4	0.47	0.47	0.5	0.47	0.5

* The UO2-density 10.7g/cc is used for all fuel types. This density is higher than the fabricated values.

Material specification for representative fuel types for BWR and PWR

Table B-1. Fuel matrix compostion of unirradiated fuel in one fuel assembly /SKBdoc 1221579/.

Material data	**PWR** **Areva 17×17**	**BWR** **Svea 96 Optima 2**
Enrichment	4.0 (% U-235)	3.6 (% U-235)
Weight in 1 fuel assembly (kg)		
Fuel composition		
U-tot	464	175
U-234[1]	0.19	0.05
U-235[1]	18.6	6.3
U-236[1]	0.1	0.04
U-238[1]	445.2	168.6
O	62	23

[1] In the fuel matrix a content of 0.03% U234 + 0.02% U236 for BWR and 0.04% U234 + 0.02% U236 for PWR is assumed.

Material data	**Atrium 10B** **O3 MOX**
Enrichment	4.6 (% Pu_{fiss}) + 0.2 (% U-235)
Weight in 1 fuel assembly (kg)	
Fuel composition	
HM-tot	174
U-235	0.3
U-238	162.3
Pu-238	0.1
Pu-239	7.2
Pu-240	2.9
Pu-241	0.7
Pu-242	0.5
O	23

Table B-2. Impurities in the fuel matrix /SKBdoc 1221579/.

Element	Assumed in calculations (ppm)	Representative values for fuel matrices[1] (ppm)	
Ag	0.05	<0.05	
Al	6	3–6	
B	0.05	<0.05	
Bi	0.5	<0.5	
Ca	3	<3	1/3 above LRV
Cd	0.233	average 0.233	min 0.2 max 0.6
Co	0.5	<0.5	
Cr	1	<1	10% above LRV
Cu	0.5	average 0.5	min 0.2 max 7
F	2	<2	20% above LRV
Fe	5	<5	20% above LRV
In	0.3	<0.3	
Li	0.05	<0.05	
Mg	1	<1	
Mn	2	<2	
Mo	5	<5	
N[2]	14	–	
Ni	–	<1	
Pb	0.6	<0.6	20% above LRV
Si	10	<10	
Sn	0.8	0.6–0.8	
Ti	10	<10	
V	0.3	<0.3	
Zn	25	<25	
Dy	10	<10	
Eu	0.02	<0.02	
Gd	0.06	<0.06	
Sm	0.04	<0.04	
C	8.4	average 8.4	min 3 max 28
Cl	2	2	
Ni	5	5	
W	0.2	0.2	
(LRV_Lowest reported value)			

[1] Personal communication Westinghouse.
[2] Assumed in accordance with /SKBdoc 1198314/.

Table B-3. Components and material composition for components in a BWR fuel assembly with fuel channel /SKBdoc 1198314/.

Component	Fuel assembly						Fuel channel	
	Cladding	Spacer	Springs, part length rod	Springs, ordinary rod	Top plate with handle	Bottom plate	Channel	Channel bottom piece
Material	Zry-2	X-750	AISI 302	AISI 302	AISI 304	AISI 304L	Zry-2	AISI 304L
Weight (kg)	49.3	0.86	0.088	1.26	2.6	0.96	31.5	8.4
Material composition (%)								
C	0.015	0.03	0.07	0.07	0.04	0.02	0.015	0.02
N	0.004	0.01	0.04	0.04	0.04	0.04	0.004	0.04
O	0.14	0.01	0.01	0.01	0.01	0.01	0.14	0.01
Al	0.005	0.7	0.002	0.002	0.002	0.002	0.005	0.002
Si	0.01	0.3	0.6	0.6	0.6	0.6	0.01	0.6
P		0.005	0.02	0.02	0.02	0.02		0.02
S		0.005	0.015	0.015	0.015	0.015		0.015
Cl		0.0001	0.0001	0.0001	0.0001	0.0001		0.0001
Ti	0.004	2.5	0.01	0.01	0.01	0.01	0.004	0.01
V		0.01	0.001	0.001	0.001	0.001		0.001
Cr	0.1	16	18.5	18.5	18.5	18.5	0.1	18.5
Mn	0.003	0.2	1.3	1.3	1.3	1.3	0.003	1.3
Fe	0.15	7.00	69.05	69.05	69.08	69.10	0.15	69.1
Co	0.0001	0.01	0.03	0.03	0.03	0.03	0.0001	0.03
Ni	0.05	71.99	10.00	10.00	10.00	10.00	0.05	10
Cu	0.003	0.1	0.1	0.1	0.1	0.1	0.003	0.1
As		0.05	0.01	0.01	0.01	0.01		0.01
Nb		0.9	0.01	0.01	0.01	0.01		0.01
Mo	0.0005	0.05	0.2	0.2	0.2	0.2	0.0005	0.2
Sn	1.5	0.01	0.01	0.01	0.01	0.01	1.5	0.01
Sb		0.005	0.001	0.001	0.001	0.001		0.001
Ta		0.1	0.01	0.01	0.01	0.01		0.01
W	0.005	0.01	0.01	0.01	0.01	0.01	0.005	0.01
Hf	0.01						0.01	
Zr[1]	98.0						98.0	
Th	0.00002						0.00002	
U	0.00015						0.00015	

[1] The mass given for Zr is the total mass in the fuel assembly, the water cross is included.

Table B-4. Components and material composition for components in a PWR fuel assembly /SKBdoc 1198314/.

Component	Top nozzle SS	Top nozzle Zr	Bottom nozzle SS	Spacer Zr	Spacer inconel	Guide thimble Zr	Guide thimble SS	Cladding Zr	Cladding[1] SS
Material	304L_1	Inc718_1	304L_2	Zry4_3	Inc718_3	Zry4_4	316L_4	M5_5	302_5
Weight (kg)	6.5	1.1	4.8	7.2	0.75	14.2	0.7	108.1	3
Material Composition (%)									
Li	0.00001		0.00001				0.00001		0.00001
C	0.02	0.03	0.02	0.015	0.03	0.015	0.025	0.015	0.12
N	0.04	0.01	0.04	0.004	0.01	0.004	0.04	0.004	0.04
O	0.01	0.01	0.01	0.14	0.01	0.14	0.01	0.14	0.01
Na	0.001		0.001				0.001		0.001
Al	0.002	0.5	0.002	0.005	0.5	0.005	0.002	0.005	0.002
Si	0.6	0.3	0.6	0.01	0.3	0.01	0.6	0.01	0.6
P	0.02	0.005	0.02		0.005		0.02		0.02
S	0.015	0.005	0.015		0.005		0.015		0.015
Cl	0.0001	0.0001	0.0001		0.0001		0.0001		0.0001
Ca	0.002		0.002				0.002		0.002
Ti	0.01	0.9	0.01	0.004	0.9	0.004	0.01	0.004	0.01
V	0.001	0.01	0.001		0.01		0.001		0.001
Cr	18.5	19	18.5	0.1	19	0.1	17		18.5
Mn	1.3	0.3	1.3	0.003	0.3	0.003	1.3		1.3
Fe	69.05	21.05	69.05	0.22	21.05	0.22	66.25		69.95
Co	0.03	0.05	0.03	0.0001	0.05	0.0001	0.03	0.00001	0.03
Ni	10	52.50	10	0.004	52.5	0.004	12		9
Cu	0.1	0.1	0.1	0.003	0.1	0.003	0.1		0.1
Zn	0.01		0.01				0.01		0.01
As	0.01	0.05	0.01		0.05		0.01		0.01
Se	0.004		0.004				0.004		0.004
Nb	0.03	4.59	0.03	0.01	4.59	0.01	0.03	1	0.03
Mo	0.2	0.05	0.2	0.0005	0.05	0.0005	2.5		0.2
Ag	0.0001		0.0001				0.0001		0.0001
Sn	0.01	0.01	0.01	1.5	0.01	1.5	0.01		0.01
Sb	0.001	0.005	0.001		0.005		0.001		0.001
Ce	0.01		0.01				0.01		0.01
Ta	0.01	0.51	0.01		0.51		0.01		0.01
W	0.01	0.01	0.01	0.005	0.01	0.005	0.01	0.005	0.01
Hf				0.006		0.006		0.01	
Zr	0.001			97.97		97.97		98.81	
Th				0.00002		0.00002		0.00002	
U				0.00015		0.00015		0.00015	

[1] The stainless steel in the springs in the fission gas plenum.

Table B-5. Material composition for control rod clusters in a PWR fuel assembly /SKBdoc 1198314/.

Component	Absorber pins	Absorber pins	Top piece
Material	304_1	AgInCd_1	304_2
Weight (kg)	12.5	51.4	3.8
Material composition (%)			
Li	0.00001		0.00001
C	0.07		0.07
N	0.04		0.04
O	0.01		0.01
Na	0.001		0.001
Al	0.002		0.002
Si	0.6		0.6
P	0.02		0.02
S	0.015		0.015
Cl	0.0001		0.0001
Ca	0.002		0.002
Ti	0.01		0.01
V	0.001		0.001
Cr	18.5		18.5
Mn	1.3		1.3
Fe	68.9		68.9
Co	0.1		0.1
Ni	10		10
Cu	0.1		0.1
Zn	0.01		0.01
As	0.01		0.01
Se	0.004		0.004
Nb	0.03		0.03
Mo	0.2		0.2
Ag	0.0001	80	0.0001
Cd		5	
In		15	
Sn	0.01		0.01
Sb	0.001		0.001
Ce	0.01		0.01
Ta	0.01		0.01
W	0.01		0.01
Zr	0.001		0.001

Appendix C

Radionuclide inventory

List of tables

The different ages of the fuels in the type canisters depend on the different decay times required to reach an acceptable total decay power in the canisters for different burnups.

Table C-1. Radionuclide information – halflives used in Origen-S calculations /SKBdoc 1221579/ and calculated specific activities given as Bq/g and Bq/mol.

Nuclide	Half-life (years)	Specific activity (Bq/g)	Specific activity (Bq/mol)	Nuclide	Half-life (years)	Specific activity (Bq/g)	Specific activity (Bq/mol)
Ac-227	21.8	2.67E+12	6.07E+14	Pa-231	3.28E+04	1.75E+09	4.04E+11
Ag-108 m	127.0	9.650E+11	1.042E+14	Pa-233	27.0 d	7.68E+14	1.79E+17
Am-241	432.7	1.269E+11	3.059E+13	Pa-234 m	1.2 m	2.5E+19	5.8E+21
Am-242	16.0 h	2.99E+16	7.25E+18	Pb-210	22.3	2.83E+12	5.94E+14
Am-242m	141.0	3.879E+11	9.387E+13	Pd-107	6.50E+06	1.90E+07	2.04E+09
Am-243	7.37E+03	7.39E+09	1.80E+12	Pu-238	87.7	6.34E+11	1.51E+14
C-14	5.73E+03	1.65E+11	2.31E+12	Pu-239	2.41E+04	2.30E+09	5.49E+11
Cd-113m	14.1	8.31E+12	9.39E+14	Pu-240	6.56E+03	8.41E+09	2.02E+12
Cl-36	3.01E+05	1.22E+09	4.40E+10	Pu-241	14.3	3.84E+12	9.26E+14
Cm-242	162.9 d	1.23E+14	2.97E+16	Pu-242	3.73E+05	1.47E+08	3.55E+10
Cm-243	28.5	1.91E+12	4.64E+14	Ra-226	1.60E+03	3.66E+10	8.27E+12
Cm-244	18.1	3.00E+12	7.31E+14	Se-79	2.95E+05	5.68E+08	4.49E+10
Cm-245	8.50E+03	6.36E+09	1.56E+12	Sm-151	90.0	9.74E+11	1.47E+14
Cm-246	4.73E+03	1.14E+10	2.80E+12	Sn-121m	55.0	1.99E+12	2.41E+14
Cs-135	2.30E+06	4.26E+07	5.75E+09	Sn-126	1.00E+05	1.05E+09	1.32E+11
Cs-137	30.0	3.22E+12	4.41E+14	Sr-90	28.1	5.23E+12	4.71E+14
Eu-152	13.3	6.55E+12	9.95E+14	Tc-99	2.11E+05	6.34E+08	6.27E+10
H-3	12.3	3.59E+14	1.08E+15	Th-229	7.88E+03	7.33E+09	1.68E+12
Ho-166m	1.20E+03	6.64E+10	1.10E+13	Th-230	7.54E+04	7.63E+08	1.76E+11
I-129	1.57E+07	6.54E+06	8.43E+08	Th-232	1.40E+10	4.08E+03	9.45E+05
Mo-93	3.50E+03	4.07E+10	3.78E+12	Th-234	24.1 d	8.57E+14	2.00E+17
Nb-93m	16.1	8.84E+12	8.22E+14	U-233	1.59E+05	3.57E+08	8.32E+10
Nb-94	2.03E+04	6.94E+09	6.52E+11	U-234	2.46E+05	2.30E+08	5.38E+10
Ni-59	7.50E+04	2.99E+09	1.76E+11	U-235	7.04E+08	8.00E+04	1.88E+07
Ni-63	100.1	2.099E+12	1.322E+14	U-236	2.34E+07	2.40E+06	5.66E+08
Np-237	2.14E+06	2.61E+07	6.19E+09	U-237	6.8 d	3.0E+15	7.1E+17
Np-238	50.8 h	9.59E+15	2.28E+18	U-238	4.47E+09	1.24E+04	2.96E+06
Np-239	56.5 h	8.59E+15	2.05E+18	Z-r93	1.53E+06	9.30E+07	8.65E+09

m) minutes, h) hours, d) days.

Table C-2. Total radionuclide inventory based on the number of fuel assemblies in the SKB reference scenario. Inventory for fuel matrix based on information in /SKBdoc 1221579/ and construction material, Crud and PWR control rod estimated from data given in /SKBdoc 1198314/. The inventories are calculated for year 2045.

Radio-nuclide	Radionuclide inventory (Bq)									
	47,637 BWR[1] assemblies 40.4 Mwd/kgU		6,016 PWR assemblies 44.8 MWd/kgU			267 BWR_{MOX}[2] assemblies 50 MWd/kgHM		33 PWR_{MOX} assemblies 34 MWd/kgHM		
	Fuel	Constr. materials /crud	Fuel	Constr. materials /crud	Control rods	Fuel	Constr. materials /crud	Fuel	Constr. materials /crud	Total activity
Ac-227	6.24E+09	5.09E+02	2.43E+09	2.71E+02		4.49E+06	3.60E+00	1.49E+06	1.83E+00	8.67E+09
Ag-108m	1.99E+12	1.55E+12	4.82E+11	4.17E+11	6.86E+16	2.08E+09	1.25E+10	2.00E+09	2.07E+09	6.86E+16
Am-241	1.18E+18	1.06E+11	4.86E+17	6.59E+10		4.77E+16	2.37E+09	4.52E+15	3.36E+08	1.72E+18
Am-242	2.97E+15	5.04E+08	9.12E+14	2.13E+08		4.31E+14	2.00E+07	1.53E+13	7.59E+05	4.33E+15
Am-242m	2.98E+15	5.07E+08	9.16E+14	2.14E+08		4.33E+14	2.01E+07	1.54E+13	7.62E+05	4.34E+15
Am-243	9.50E+15	1.15E+09	4.01E+15	6.52E+08		4.54E+14	3.43E+07	5.08E+13	2.15E+06	1.40E+16
C-14	1.13E+14	2.87E+14	3.39E+13	6.90E+13	3.75E+12	1.93E+12	2.11E+12	9.66E+10	3.25E+11	5.10E+14
Cd-113m	2.89E+13	0.00E+00	1.14E+13	0.00E+00	4.53E+15	9.29E+11	0.00E+00	2.79E+10	0.00E+00	4.57E+15
Cl-36	1.80E+12	2.82E+10	4.86E+11	8.03E+09	1.12E+09	3.14E+10	2.07E+08	1.40E+09	3.80E+07	2.34E+12
Cm-242	2.45E+15	4.17E+08	7.54E+14	1.76E+08		3.57E+14	1.66E+07	1.27E+13	6.28E+05	3.58E+15
Cm-243	3.41E+15	2.98E+08	1.27E+15	1.75E+08		1.89E+14	7.97E+06	8.70E+12	3.78E+05	4.88E+15
Cm-244	3.45E+17	4.96E+10	1.45E+17	3.00E+10		1.84E+16	1.47E+09	1.09E+15	3.71E+07	5.09E+17
Cm-245	1.27E+14	2.14E+07	5.79E+13	1.52E+07		3.30E+13	2.12E+06	1.28E+12	3.53E+04	2.19E+14
Cm-246	2.57E+13	5.53E+06	1.09E+13	3.95E+06		4.54E+12	6.62E+05	2.19E+11	8.26E+03	4.14E+13
Cs-135	1.77E+14	0.00E+00	5.57E+13			2.90E+12		2.29E+11		2.36E+14
Cs-137	1.70E+19	5.17E+12	6.32E+18	1.11E+12	0.00E+00	8.49E+16	2.72E+10	9.99E+15	3.30E+09	2.34E+19
Eu-152	3.90E+14	0.00E+00	1.10E+14	0.00E+00	0.00E+00	4.76E+12	0.00E+00	2.71E+11	0.00E+00	5.05E+14
H-3	2.83E+16	0.00E+00	1.01E+16	2.15E+13	1.56E+12	9.72E+13	0.00E+00	8.94E+12	3.33E+10	3.85E+16
Ho-166m	3.24E+13	0.00E+00	1.26E+13	0.00E+00	0.00E+00	5.49E+11	0.00E+00	3.84E+10	0.00E+00	4.53E+13
I-129	1.01E+13	0.00E+00	3.76E+12			9.63E+10		1.29E+10		1.39E+13
Mo-93	2.58E+11	1.18E+12	1.24E+11	1.48E+12	1.83E+11	4.19E+09	8.70E+09	5.21E+08	7.03E+09	3.24E+12
Nb-93m	5.35E+14	9.46E+15	1.98E+14	2.39E+17	5.32E+13	2.82E+12	3.79E+13	4.16E+11	4.88E+14	2.50E+17
Nb-94	4.62E+10	3.57E+13	1.73E+10	8.92E+14	3.21E+11	1.65E+10	2.58E+11	8.85E+07	4.29E+12	9.32E+14
Ni-59	9.77E+11	1.44E+15	2.62E+11	2.58E+14	2.45E+13	1.60E+12	1.01E+13	8.07E+08	1.25E+12	1.73E+15
Ni-63	1.03E+14	1.53E+17	2.68E+13	2.76E+16	2.16E+15	6.17E+08	1.02E+15	6.95E+10	1.14E+14	1.85E+17
Np-237	1.25E+14	1.10E+07	5.16E+13	7.25E+06		8.30E+11	4.76E+04	1.03E+11	3.10E+04	1.78E+14
Np-238	1.34E+13	2.28E+06	4.12E+12	9.63E+05		1.95E+12	9.06E+04	6.94E+10	3.43E+03	1.96E+13
Np-239	9.50E+15	1.15E+09	4.01E+15	6.52E+08	0.00E+00	4.54E+14	3.43E+07	5.08E+13	2.15E+06	1.40E+16
Pa-231	1.05E+10	8.44E+02	4.07E+09	4.66E+02		7.06E+06	4.49E+00	2.66E+06	2.79E+00	1.46E+10
Pa-233	1.25E+14	1.10E+07	5.16E+13	7.25E+06		8.30E+11	4.76E+04	1.03E+11	3.10E+04	1.78E+14
Pa-234m	9.69E+13	7.93E+06	3.22E+13	4.42E+06		5.15E+11	4.35E+04	1.04E+11	2.46E+04	1.30E+14
Pb-210	3.13E+08	2.77E+01	1.52E+08	1.72E+01		2.33E+06	4.28E–01	2.85E+05	2.62E–01	4.67E+08
Pd-107	4.27E+13	0.00E+00	1.61E+13			6.42E+11		8.39E+10		5.95E+13
Pu-238	9.50E+17	9.68E+10	3.72E+17	6.01E+10		2.06E+16	1.33E+09	1.57E+15	1.70E+08	1.34E+18
Pu-239	1.00E+17	9.22E+11	3.82E+16	2.79E+11	0.00E+00	2.67E+15	5.28E+09	2.18E+14	1.52E+09	1.41E+17
Pu-240	1.85E+17	1.23E+10	6.14E+16	8.23E+09		6.35E+15	3.23E+08	5.94E+14	3.92E+07	2.54E+17
Pu-241	7.39E+18	6.39E+11	2.97E+18	3.96E+11		1.34E+17	6.32E+09	9.32E+15	7.08E+08	1.05E+19
Pu-242	7.88E+14	7.77E+07	3.15E+14	4.41E+07		2.68E+13	2.23E+06	3.43E+12	1.76E+05	1.13E+15
Ra-226	1.03E+09	8.90E+01	4.98E+08	5.64E+01		7.53E+06	1.18E+00	8.48E+05	6.50E–01	1.53E+09
Se-79	2.70E+13	0.00E+00	1.00E+13	4.88E+09	1.76E+09	1.56E+11	0.00E+00	2.23E+10	2.31E+07	3.73E+13
Sm-151	9.85E+16	0.00E+00	3.98E+16	0.00E+00	0.00E+00	1.87E+15	0.00E+00	1.40E+14	0.00E+00	1.40E+17
Sn-121m	3.06E+15	5.95E+15	1.14E+15	1.97E+14	4.87E+10	2.72E+13	3.52E+13	3.45E+12	7.14E+11	1.04E+16
Sn-126	1.97E+14	0.00E+00	7.36E+13			2.04E+12		2.73E+11		2.73E+14
Sr-90	1.13E+19	4.85E+12	4.20E+18	1.04E+12	0.00E+00	2.77E+16	2.53E+10	3.59E+15	3.06E+09	1.56E+19
Tc-99	4.98E+15	2.03E+11	1.84E+15	2.21E+11	2.90E+10	3.21E+13	1.47E+09	4.64E+12	1.08E+09	6.86E+15
Th-229	1.08E+08	1.22E+01	4.09E+07	6.50E+00		6.63E+05	4.87E–02	4.67E+04	3.40E–02	1.50E+08
Th-230	1.26E+11	1.08E+04	6.10E+10	6.97E+03		9.67E+08	1.20E+02	9.67E+07	5.37E+01	1.88E+11
Th-232	1.89E+05	1.48E–02	7.01E+04	9.41E–03		1.00E+02	9.54E–06	4.71E+01	6.20E–05	2.59E+05
Th-234	9.69E+13	7.93E+06	3.22E+13	4.42E+06		5.15E+11	4.35E+04	1.04E+11	2.46E+04	1.30E+14
U-233	2.47E+10	1.00E+10	1.09E+10	2.55E+09	0.00E+00	1.15E+08	5.80E+07	1.71E+07	1.38E+07	4.84E+10
U-234	4.03E+14	3.45E+07	1.90E+14	2.26E+07		3.76E+12	3.53E+05	3.32E+11	1.14E+05	5.97E+14
U-235	4.79E+12	3.63E+05	1.82E+12	2.45E+05		3.70E+09	2.05E+02	2.06E+09	1.40E+03	6.61E+12
U-236	9.88E+13	7.47E+06	3.67E+13	4.83E+06		4.33E+10	3.75E+03	1.68E+10	2.13E+04	1.35E+14
U-237	1.77E+14	1.53E+07	7.11E+13	9.47E+06		3.21E+12	1.51E+05	2.23E+11	1.69E+04	2.51E+14
U-238	9.69E+13	7.93E+06	3.22E+13	4.42E+06		5.15E+11	4.35E+04	1.04E+11	2.46E+04	1.30E+14
Zr-93	6.61E+14	1.02E+14	2.45E+14	2.62E+13	2.50E+06	3.15E+12	7.49E+11	4.52E+11	1.24E+11	1.04E+15

[1] Includes 222 assemblies from Ågesta.

[2] Includes 184 Swap BWR MOX.

BWR:

- UO2-values from /SKBdoc 1221579, Table 14/.
- Inventory for construction material from /SKBdoc 1198314/ by linear interpolation between 38 and 60 MWd/kg U to find inventory at 40.4 MWd/kg U.
  (Data from appendix: folder 'IndAct-Ett element-rev3', files '88_Ind-B38-000.xls' and '89_Ind-B60-000.xls'.)
- Inventory for Crud from /SKBdoc 1198314/ by linear interpolation between 38 and 60 MWd/kg U to find inventory at 40.4 MWd/kg U.
  (Data from appendix: folder 'CrudAct-Ett element-rev3', files '95_Crud-B38-000.xls' and '96_Crud-B60-000.xls'.)
- Average age for BWR assemblies 36.3 years and BWRmox assemblies 50 years.

PWR:

- UO2-values from /SKBdoc 1221579, Table 13/.
- Inventory for construction material from /SKBdoc 1198314/ by linear interpolation between 30 and 60 MWd/kg U to find inventory at 44.8 MWd/kg U.
  Assumption: At any given burnup or age the construction material data is the same for PWR and PWRMOX.
  (Data from appendix: folder 'IndAct-Ett element-rev3', files '90_Ind-P30-000.xls' and '91_Ind-P60-000.xls'.)
- Inventory for Crud from /SKBdoc 1198314/ by linear interpolation between 30 and 60 MWd/kg U to find inventory at 44.8 MWd/kg U.
  Assumption: At any given burnup or age the crud data is the same for PWR and PWRMOX.
  (Data from appendix: folder 'CrudAct-Ett element-rev3', files '97_Crud-P30-000.xls' and '98_Crud-P60-000.xls'.)
- Inventory for control rods from /SKBdoc 1179234, appendix: folder 'Styrstavar-rev3', file '110_PWR-ss.xls'/.
- Average age for PWR assemblies 36.9 years and PWRmox assemblies 57 years.

Table C-3. Simulation of BWR canister encapsulation rate during year 2023–2070 /SKBdoc 1221567/.

Year	Canisters/year			BWR														
	BWR	PWR	Tot	<38 MWd					38–42 MWd					>42 MWd				
No of fuel assemblies				12	11	10	9	<9	12	11	10	9	<9	12	11	10	9	<9
2023	17	6	23	17														
2024	58	22	80	58														
2025	87	33	120	87														
2026	87	33	120	87														
2027	87	33	120	87														
2028	109	41	150	109														
2029	109	41	150	109														
2030	109	41	150	109														
2031	109	41	150	109														
2032	109	41	150	109														
2033	109	41	150	109														
2034	109	41	150	109														
2035	109	41	150	109														
2036	109	41	150	103					6									
2037	109	41	150	92					17									
2038	109	41	150	8					101									
2039	109	41	150	1					108									
2040	109	41	150						109									
2041	109	41	150						109									
2042	109	41	150						109									
2043	109	41	150						109									
2044	109	41	150						91					18				
2045	109	41	150	40					25					44				
2046	109	41	150						16					93				
2047	109	41	150						13					96				
2048	109	41	150						1					70	10	28		
2049	109	41	150								12					78	19	
2050	109	41	150	38		19											52	
2051	109	41	150	38		17					1	1					52	
2052	109	41	150	25		1						13					70	
2053	109	41	150														51	58
2054	109	41	150	25								14					3	67
2055	73	27	100	40		17											12	4
2056	73	27	100														41	32
2057	73	27	100														25	48
2058	73	27	100														11	62
2059	73	27	100															73
2060	73	27	100														16	57
2061	73	27	100															73
2062	73	27	100														26	47
2063	73	27	100														39	34
2064	73	27	100														31	42
2065	73	27	100	43								24					6	
2066	73	27	100														73	
2067	73	27	100														72	1
2068	73	27	100														12	61
2069	73	27	100														15	58
2070	77	13	90														60	17
Total	4,451	1,652	6,103	1,661	0	54	0	0	814[1]	0	13	52	0	321	10	106	686	734

[1] Including 267 BWR MOX.

BWR I + BWR MOX	2475	BWR II	321	BWR III	1655

Table C-4. Simulation of PWR canister encapsulation rate during year 2023–2070 /SKBdoc 1221567/.

Year	Canisters/year			PWR								
	BWR	PWR	Tot	<42 MWd			42–47 MWd			>47 MWd		
No of fuel assemblies				4	3	<3	4	3	<3	4	3	<3
2023	17	6	23	6								
2024	58	22	80	22								
2025	87	33	120	33								
2026	87	33	120	33								
2027	87	33	120	33								
2028	109	41	150	41								
2029	109	41	150	39			2					
2030	109	41	150	37			4					
2031	109	41	150	32			9					
2032	109	41	150	25			16					
2033	109	41	150	22			19					
2034	109	41	150	10			31					
2035	109	41	150	4			37					
2036	109	41	150	1			40					
2037	109	41	150	1			40					
2038	109	41	150	2			39					
2039	109	41	150				41					
2040	109	41	150				41					
2041	109	41	150				41					
2042	109	41	150				41					
2043	109	41	150				39			2		
2044	109	41	150				40			1		
2045	109	41	150				31			10		
2046	109	41	150				31			10		
2047	109	41	150				37			4		
2048	109	41	150				31			10		
2049	109	41	150				31			1	9	
2050	109	41	150								41	
2051	109	41	150	32							9	
2052	109	41	150								41	
2053	109	41	150	18				23				
2054	109	41	150					2			39	
2055	73	27	100								27	
2056	73	27	100								27	
2057	73	27	100								27	
2058	73	27	100								27	
2059	73	27	100								27	
2060	73	27	100								27	
2061	73	27	100								27	
2062	73	27	100								27	
2063	73	27	100								27	
2064	73	27	100								27	
2065	73	27	100								27	
2066	73	27	100								27	
2067	73	27	100								27	
2068	73	27	100				1				26	
2069	73	27	100				12				15	
2070	77	13	90				12				1	
Total	4,451	1,652	6,103	391	0	0	666[1]	25	0	38	532	0

[1] Including 33 PWRMOX.

PWR I + PWRMOX	1057	PWR II	38	PWR III	557

Table C-5. Total radionuclide inventory estimated from the inventory in the type canisters at encapsulation (different times for the different type canisters).

Radionuclide inventory (Bq)									
Type	**BWR I**	**BWR II**	**BWR III**	**BWR-MOX**	**PWR I**	**PWR II**	**PWR III**	**PWR-MOX**	**TOTAL**
Number of canisters	**2,208**	**321**	**1,655**	**267**	**1,024**	**38**	**557**	**33**	**6,103**
Ac-227	3.52E+09	6.47E+08	1.87E+09	4.48E+08	1.69E+09	8.52E+07	7.55E+08	3.69E+07	9.06E+09
Ag-108m	1.96E+12	3.38E+11	1.32E+12	2.19E+11	4.66E+16	1.68E+15	2.61E+16	1.52E+15	7.59E+16
Am-241	6.61E+17	1.12E+17	3.87E+17	1.20E+17	3.34E+17	1.49E+16	1.47E+17	1.21E+16	1.79E+18
Am-242	1.64E+15	2.90E+14	1.22E+15	6.20E+14	6.17E+14	2.81E+13	3.32E+14	3.07E+13	4.78E+15
Am-242m	1.65E+15	2.92E+14	1.22E+15	6.23E+14	6.20E+14	2.82E+13	3.33E+14	3.09E+13	4.80E+15
Am-243	5.28E+15	1.22E+15	4.71E+15	9.30E+14	2.73E+15	1.93E+14	2.12E+15	1.17E+14	1.73E+16
C-14	2.22E+14	3.72E+13	1.52E+14	2.55E+13	7.26E+13	3.28E+12	3.65E+13	2.20E+12	5.52E+14
Cd-113m	1.55E+13	1.42E+12	1.25E+13	2.17E+12	2.93E+15	4.71E+13	3.83E+15	1.29E+14	6.98E+15
Cl-36	1.02E+12	1.88E+11	7.26E+11	1.15E+11	3.37E+11	1.66E+10	1.82E+11	9.60E+09	2.59E+12
Cm-242	1.36E+15	2.40E+14	1.01E+15	5.13E+14	5.11E+14	2.32E+13	2.75E+14	2.54E+13	3.95E+15
Cm-243	1.86E+15	3.11E+14	1.81E+15	3.34E+14	8.40E+14	3.81E+13	6.32E+14	3.24E+13	5.86E+15
Cm-244	1.87E+17	3.51E+16	2.57E+17	3.07E+16	9.44E+16	4.86E+15	1.09E+17	4.00E+15	7.22E+17
Cm-245	7.06E+13	2.24E+13	8.69E+13	3.85E+13	3.94E+13	4.51E+12	4.96E+13	2.23E+12	3.14E+14
Cm-246	1.43E+13	6.16E+12	2.39E+13	5.53E+12	7.45E+12	1.30E+12	1.43E+13	3.99E+11	7.33E+13
Cs-135	9.83E+13	1.71E+13	6.61E+13	1.36E+13	3.79E+13	1.82E+12	2.01E+13	1.15E+12	2.56E+14
Cs-137	9.30E+18	1.22E+18	6.93E+18	9.29E+17	4.19E+18	1.32E+17	2.14E+18	1.27E+17	2.50E+19
Eu-152	2.09E+14	1.90E+13	1.75E+14	2.47E+13	7.07E+13	1.25E+12	3.92E+13	2.65E+12	5.42E+14
H-3	1.51E+16	1.42E+15	1.38E+16	1.23E+15	6.48E+15	1.22E+14	4.30E+15	2.33E+14	4.27E+16
Ho-166m	1.80E+13	3.44E+12	1.33E+13	1.99E+12	8.62E+12	4.34E+11	4.82E+12	2.47E+11	5.09E+13
I-129	5.61E+12	9.83E+11	3.80E+12	6.71E+11	2.56E+12	1.24E+11	1.37E+12	7.48E+10	1.52E+13
Mo-93	7.99E+11	1.39E+11	5.40E+11	9.24E+10	1.22E+12	5.21E+10	5.92E+11	3.80E+10	3.47E+12
Nb-93m	5.41E+15	5.94E+14	4.41E+15	4.94E+14	1.56E+17	3.27E+15	8.07E+16	4.43E+15	2.55E+17
Nb-94	1.99E+13	3.39E+12	1.31E+13	2.32E+12	6.07E+14	2.59E+13	2.85E+14	1.90E+13	9.76E+14
Ni-59	7.99E+14	1.34E+14	5.17E+14	9.38E+13	1.92E+14	8.07E+12	9.10E+13	6.04E+12	1.84E+15
Ni-63	8.50E+16	1.35E+16	5.85E+16	9.49E+15	2.02E+16	7.66E+14	9.65E+15	6.19E+14	1.98E+17
Np-237	6.99E+13	1.24E+13	4.56E+13	8.15E+12	3.52E+13	1.72E+12	1.82E+13	9.38E+11	1.92E+14
Np-238	7.43E+12	1.31E+12	5.51E+12	2.80E+12	2.79E+12	1.27E+11	1.50E+12	1.39E+11	2.16E+13
Np-239	5.28E+15	1.22E+15	4.71E+15	9.30E+14	2.73E+15	1.93E+14	2.12E+15	1.17E+14	1.73E+16
Pa-231	5.90E+09	9.21E+08	3.21E+09	7.20E+08	2.80E+09	1.14E+08	1.16E+09	6.64E+07	1.49E+10
Pa-233	6.99E+13	1.24E+13	4.56E+13	8.15E+12	3.52E+13	1.72E+12	1.82E+13	9.38E+11	1.92E+14
Pa-234m	5.39E+13	7.79E+12	3.01E+13	6.51E+12	2.19E+13	8.05E+11	8.84E+12	6.35E+11	1.31E+14
Pb-210	1.83E+08	5.10E+07	6.61E+07	3.25E+07	1.12E+08	1.00E+07	6.31E+07	1.96E+06	5.20E+08
Pd-107	2.37E+13	4.48E+12	1.73E+13	3.00E+12	1.10E+13	5.92E+11	6.51E+12	3.50E+11	6.70E+13
Pu-238	5.25E+17	9.97E+16	4.40E+17	6.92E+16	2.51E+17	1.34E+16	1.66E+17	7.95E+15	1.57E+18
Pu-239	5.56E+16	8.00E+15	3.10E+16	8.82E+15	2.60E+16	9.62E+14	1.06E+16	8.46E+14	1.42E+17
Pu-240	1.03E+17	1.72E+16	6.63E+16	1.72E+16	4.18E+16	1.88E+15	2.06E+16	1.60E+15	2.70E+17
Pu-241	3.97E+18	3.66E+17	3.17E+18	4.50E+17	1.92E+18	3.51E+16	1.00E+18	7.22E+16	1.10E+19
Pu-242	4.38E+14	8.97E+13	3.47E+14	6.85E+13	2.15E+14	1.25E+13	1.38E+14	8.61E+12	1.32E+15
Ra-226	5.93E+08	1.41E+08	2.36E+08	9.65E+07	3.59E+08	2.58E+07	1.75E+08	6.95E+06	1.63E+09
Se-79	1.50E+13	2.53E+12	9.78E+12	1.72E+12	6.84E+12	3.12E+11	3.43E+12	1.88E+11	3.98E+13
Sm-151	5.45E+16	7.53E+15	3.31E+16	7.47E+15	2.69E+16	9.45E+14	1.17E+16	8.21E+14	1.43E+17
Sn-121m	4.97E+15	7.35E+14	3.55E+15	5.39E+14	9.00E+14	3.51E+13	4.71E+14	2.75E+13	1.12E+16
Sn-126	1.10E+14	1.96E+13	7.59E+13	1.32E+13	5.01E+13	2.50E+12	2.75E+13	1.49E+12	3.00E+14
Sr-90	6.20E+18	7.60E+17	4.43E+18	5.95E+17	2.78E+18	7.96E+16	1.33E+18	8.22E+16	1.63E+19
Tc-99	2.77E+15	4.61E+14	1.78E+15	3.22E+14	1.25E+15	5.62E+13	6.18E+14	3.50E+13	7.30E+15
Th-229	6.10E+07	1.67E+07	5.07E+07	7.56E+06	2.85E+07	2.42E+06	2.19E+07	6.41E+05	1.89E+08
Th-230	7.18E+10	1.38E+10	3.29E+10	1.04E+10	4.28E+10	2.29E+09	1.81E+10	9.53E+08	1.93E+11
Th-232	1.07E+05	2.12E+04	5.53E+04	1.33E+04	4.91E+04	2.75E+03	2.28E+04	1.05E+03	2.72E+05
Th-234	5.39E+13	7.79E+12	3.01E+13	6.51E+12	2.19E+13	8.05E+11	8.84E+12	6.35E+11	1.31E+14
U-233	1.95E+10	3.72E+09	1.10E+10	2.43E+09	9.33E+09	5.21E+08	4.61E+09	2.33E+08	5.13E+10
U-234	2.25E+14	3.69E+13	1.23E+14	2.95E+13	1.30E+14	5.55E+12	5.43E+13	3.37E+12	6.08E+14
U-235	2.66E+12	2.61E+11	1.01E+12	3.42E+11	1.24E+12	2.61E+10	2.87E+11	3.20E+10	5.86E+12
U-236	5.49E+13	8.35E+12	3.22E+13	5.99E+12	2.50E+13	9.71E+11	1.07E+13	6.20E+11	1.39E+14
U-237	9.50E+13	8.76E+12	7.59E+13	1.08E+13	4.59E+13	8.39E+11	2.40E+13	1.73E+12	2.63E+14
U-238	5.39E+13	7.79E+12	3.01E+13	6.51E+12	2.19E+13	8.05E+11	8.84E+12	6.35E+11	1.31E+14
Zr-93	4.24E+14	7.10E+13	2.74E+14	4.83E+13	1.85E+14	8.27E+12	9.10E+13	5.04E+12	1.11E+15

Table C-6. Radionuclide inventory in BWR I type canister (12 assemblies, burnup 40.4 MWd/kgHM, age 37 years. Inventory for fuel matrix based on information in /SKBdoc 1221579/ and construction material, Crud and PWR control rod estimated from data given in /SKBdoc 1198314/.

Radionuclide	BWR I – Radionuclide inventory (Bq/canister)			
	UO_2-matrix	Constr. mtrl.	Crud	Total
Ac-227	1.60E+06		1.30E–01	1.60E+06
Ag-108m	4.99E+08	0.00E+00	3.90E+08	8.89E+08
Am-241	2.99E+14		2.70E+07	2.99E+14
Am-242	7.45E+11		1.27E+05	7.45E+11
Am-242m	7.48E+11		1.27E+05	7.48E+11
Am-243	2.39E+12		2.90E+05	2.39E+12
C-14	2.84E+10	7.23E+10		1.01E+11
Cd-113m	7.04E+09	0.00E+00		7.04E+09
Cl-36	4.54E+08	7.10E+06		4.62E+08
Cm-242	6.16E+11		1.05E+05	6.16E+11
Cm-243	8.44E+11		7.39E+04	8.44E+11
Cm-244	8.45E+13		1.22E+07	8.45E+13
Cm-245	3.20E+10		5.38E+03	3.20E+10
Cm-246	6.46E+09		1.39E+03	6.46E+09
Cs-135	4.45E+10			4.45E+10
Cs-137	4.21E+15	1.28E+09		4.21E+15
Eu-152	9.45E+10	0.00E+00		9.45E+10
H-3	6.85E+12	0.00E+00		6.85E+12
Ho-166m	8.16E+09	0.00E+00		8.16E+09
I-129	2.54E+09			2.54E+09
Mo-93	6.49E+07	2.92E+08	5.39E+06	3.62E+08
Nb-93m	1.36E+11	2.26E+12	5.23E+10	2.45E+12
Nb-94	1.16E+07	8.79E+09	2.00E+08	9.00E+09
Ni-59	2.46E+08	3.57E+11	5.05E+09	3.62E+11
Ni-63	2.58E+10	3.79E+13	5.40E+11	3.85E+13
Np-237	3.17E+10		2.77E+03	3.17E+10
Np-238	3.37E+09		5.72E+02	3.37E+09
Np-239	2.39E+12	0.00E+00	2.90E+05	2.39E+12
Pa-231	2.67E+06		2.14E–01	2.67E+06
Pa-233	3.17E+10		2.77E+03	3.17E+10
Pa-234m	2.44E+10		2.00E+03	2.44E+10
Pb-210	8.30E+04		7.34E–03	8.30E+04
Pd-107	1.08E+10			1.08E+10
Pu-238	2.38E+14		2.43E+07	2.38E+14
Pu-239	2.52E+13	2.30E+08	1.96E+06	2.52E+13
Pu-240	4.67E+13		3.10E+06	4.67E+13
Pu-241	1.80E+15		1.56E+08	1.80E+15
Pu-242	1.98E+11		1.96E+04	1.98E+11
Ra-226	2.69E+05		2.33E–02	2.69E+05
Se-79	6.81E+09	0.00E+00		6.81E+09
Sm-151	2.47E+13	0.00E+00		2.47E+13
Sn-121m	7.65E+11	1.48E+12	1.21E+08	2.25E+12
Sn-126	4.96E+10			4.96E+10
Sr-90	2.81E+15	1.20E+09		2.81E+15
Tc-99	1.26E+12	5.03E+07	8.41E+05	1.26E+12
Th-229	2.76E+04		3.11E–03	2.76E+04
Th-230	3.25E+07		2.77E+00	3.25E+07
Th-232	4.84E+01		3.80E–06	4.84E+01
Th-234	2.44E+10		2.00E+03	2.44E+10
U-233	6.31E+06	2.53E+06	5.45E–01	8.84E+06
U-234	1.02E+11		8.74E+03	1.02E+11
U-235	1.21E+09		9.13E+01	1.21E+09
U-236	2.49E+10		1.88E+03	2.49E+10
U–237	4.30E+10		3.73E+03	4.30E+10
U-238	2.44E+10		2.00E+03	2.44E+10
Zr-93	1.66E+11	2.58E+10	2.18E+06	1.92E+11

2,208 BWR I canisters:

- UO_2-values from /SKBdoc 1221579, Table 14/.
- Inventory for construction material from /SKBdoc 1198314/ by linear interpolation between 38 and 60 MWd/kg U to find inventory at 40.4 MWd/kg U.
  (Data from appendix: folder 'IndAct-Ett element-rev3', files '88_Ind-B38-000.xls' and '89_Ind-B60-000.xls'.)
- Inventory for Crud from /SKBdoc 1198314/ by linear interpolation between 38 and 60 MWd/kg U to find inventory at 40.4 MWd/kg U.
  (Data from appendix: folder 'CrudAct-Ett element-rev3', files '95_Crud-B38-000.xls' and '96_Crud-B60-000.xls'.)

Table C-7. Radionuclide inventory in PWR I type canister (4 assemblies, burnup 44.8 MWd/kgHM, age 38 years). Inventory for fuel matrix based on information in /SKBdoc 1221579/ and construction material and Crud estimated from data given in /SKBdoc 1198314/.

Radionuclide	PWR I – Radionuclide inventory (Bq/canister)				
	UO_2-matrix	Constr. mtrl.	Crud	Control rod	Total
Ac-227	1.65E+06		1.85E–01		1.65E+06
Ag-108m	3.24E+08	2.22E+08	5.53E+07	4.55E+13	4.55E+13
Am-241	3.26E+14		4.42E+07		3.26E+14
Am-242	6.03E+11		1.41E+05		6.03E+11
Am-242m	6.06E+11		1.42E+05		6.06E+11
Am-243	2.67E+12		4.33E+05		2.67E+12
C-14	2.26E+10	4.59E+10		2.49E+09	7.09E+10
Cd-113m	7.16E+09	0.00E+00		2.86E+12	2.86E+12
Cl-36	3.23E+08	5.34E+06		7.47E+05	3.29E+08
Cm-242	4.99E+11		1.17E+05		4.99E+11
Cm-243	8.20E+11		1.13E+05		8.20E+11
Cm-244	9.22E+13		1.92E+07		9.22E+13
Cm-245	3.85E+10		1.01E+04		3.85E+10
Cm-246	7.27E+09		2.63E+03		7.27E+09
Cs-135	3.71E+10				3.71E+10
Cs-137	4.10E+15	7.18E+08		0.00E+00	4.10E+15
Eu-152	6.90E+10	0.00E+00		0.00E+00	6.90E+10
H-3	6.32E+12	1.35E+10		9.76E+08	6.33E+12
Ho-166m	8.42E+09	0.00E+00		0.00E+00	8.42E+09
I-129	2.50E+09				2.50E+09
Mo-93	8.22E+07	9.86E+08	6.78E+04	1.22E+08	1.19E+09
Nb-93m	1.33E+11	1.52E+14	1.09E+08	3.38E+10	1.52E+14
Nb-94	1.15E+07	5.93E+11	4.14E+05	2.14E+08	5.93E+11
Ni-59	1.74E+08	1.71E+11	3.18E+08	1.63E+10	1.88E+11
Ni-63	1.80E+10	1.82E+13	3.06E+10	1.43E+12	1.97E+13
Np-237	3.44E+10		4.84E+03		3.44E+10
Np-238	2.73E+09		6.37E+02		2.73E+09
Np-239	2.67E+12	0.00E+00	4.33E+05	0.00E+00	2.67E+12
Pa-231	2.74E+06		3.14E–01		2.74E+06
Pa-233	3.44E+10		4.84E+03		3.44E+10
Pa-234m	2.14E+10		2.94E+03		2.14E+10
Pb-210	1.09E+05		1.23E–02		1.09E+05
Pd-107	1.07E+10				1.07E+10
Pu-238	2.45E+14		3.97E+07		2.45E+14
Pu-239	2.54E+13	1.82E+08	3.47E+06	0.00E+00	2.54E+13
Pu-240	4.08E+13		5.47E+06		4.08E+13
Pu-241	1.87E+15		2.50E+08		1.87E+15
Pu-242	2.10E+11		2.93E+04		2.10E+11
Ra-226	3.51E+05		3.96E–02		3.51E+05
Se-79	6.68E+09	3.24E+06		1.17E+06	6.68E+09
Sm-151	2.62E+13	0.00E+00		0.00E+00	2.62E+13
Sn-121m	7.50E+11	1.29E+11	3.55E+05	3.19E+07	8.79E+11
Sn-126	4.90E+10				4.90E+10
Sr-90	2.72E+15	6.73E+08		0.00E+00	2.72E+15
Tc-99	1.23E+12	1.47E+08	1.16E+04	1.93E+07	1.23E+12
Th-229	2.79E+04		4.41E–03		2.79E+04
Th-230	4.18E+07		4.78E+00		4.18E+07
Th-232	4.79E+01		6.42E–06		4.79E+01
Th-234	2.14E+10		2.94E+03		2.14E+10
U-233	7.41E+06	1.70E+06	9.99E–01	0.00E+00	9.11E+06
U-234	1.27E+11		1.52E+04		1.27E+11
U-235	1.21E+09		1.63E+02		1.21E+09
U-236	2.44E+10		3.21E+03		2.44E+10
U-237	4.48E+10		5.99E+03		4.48E+10
U-238	2.14E+10		2.94E+03		2.14E+10
Zr-93	1.63E+11	1.74E+10	1.31E+04	1.66E+03	1.80E+11

1,024 PWR I canisters:

- UO2-values from /SKBdoc 1221579, Table 13/.
- Inventory for construction material from /SKBdoc 1198314/ by linear interpolation between 30 and 60 MWd/kg U to find inventory at 44.8 MWd/kg U.
  (Data from appendix: folder 'IndAct-Ett element-rev3', files '90_Ind-P30-000.xls' and '91_Ind-P60-000.xls'.)
- Inventory for Crud from /SKBdoc 1198314/ by linear interpolation between 30 and 60 MWd/kg U to find inventory at 44.8 MWd/kg U.
  (Data from appendix: folder 'CrudAct-Ett element-rev3', files '97_Crud-P30-000.xls' and '98_Crud-P60-000.xls'.)
- Inventory for control rods from /SKBdoc 1179234, appendix: folder 'Styrstavar-rev3', file '110_PWR-ss.xls'/.

Table C-8. Radionuclide inventory in BWR II type canister (12 assemblies, burnup 47.8 MWd/kgHM, age 48 years). Inventory for fuel matrix based on information in /SKBdoc 1221579/ and construction material and Crud estimated from data given in /SKBdoc 1198314/.

Radionuclide	BWR II – Radionuclide inventory (Bq/canister)			
	UO_2-matrix	Constr. mtrl.	Crud	Total
Ac-227	2.02E+06		1.84E–01	2.02E+06
Ag-108m	5.58E+08	0.00E+00	4.96E+08	1.05E+09
Am-241	3.49E+14		3.12E+07	3.49E+14
Am-242	9.04E+11		1.50E+05	9.04E+11
Am-242m	9.08E+11		1.51E+05	9.08E+11
Am-243	3.79E+12		3.91E+05	3.79E+12
C-14	3.01E+10	8.59E+10		1.16E+11
Cd-113m	4.43E+09	0.00E+00		4.43E+09
Cl-36	5.76E+08	8.43E+06		5.84E+08
Cm-242	7.48E+11		1.24E+05	7.48E+11
Cm-243	9.70E+11		7.60E+04	9.70E+11
Cm-244	1.09E+14		1.30E+07	1.09E+14
Cm-245	6.99E+10		1.02E+04	6.99E+10
Cm-246	1.92E+10		3.08E+03	1.92E+10
Cs-135	5.33E+10			5.33E+10
Cs-137	3.80E+15	1.16E+09		3.80E+15
Eu-152	5.93E+10	0.00E+00		5.93E+10
H-3	4.42E+12	0.00E+00		4.42E+12
Ho-166m	1.07E+10	0.00E+00		1.07E+10
I-129	3.06E+09			3.06E+09
Mo-93	8.01E+07	3.46E+08	7.79E+06	4.34E+08
Nb-93m	1.69E+11	1.63E+12	4.59E+10	1.85E+12
Nb-94	1.48E+07	1.03E+10	2.84E+08	1.06E+10
Ni-59	3.04E+08	4.09E+11	7.03E+09	4.16E+11
Ni-63	3.03E+10	4.12E+13	7.14E+11	4.20E+13
Np-237	3.85E+10		3.53E+03	3.85E+10
Np-238	4.09E+09		6.77E+02	4.09E+09
Np-239	3.79E+12	0.00E+00	3.91E+05	3.79E+12
Pa-231	2.87E+06		2.66E–01	2.87E+06
Pa-233	3.85E+10		3.53E+03	3.85E+10
Pa-234m	2.43E+10		1.98E+03	2.43E+10
Pb-210	1.59E+05		1.48E–02	1.59E+05
Pd-107	1.40E+10			1.40E+10
Pu-238	3.11E+14		3.16E+07	3.11E+14
Pu-239	2.49E+13	2.31E+08	2.05E+06	2.49E+13
Pu-240	5.36E+13		3.38E+06	5.36E+13
Pu-241	1.14E+15		9.81E+07	1.14E+15
Pu-242	2.79E+11		2.30E+04	2.79E+11
Ra-226	4.38E+05		4.04E–02	4.38E+05
Se-79	7.88E+09	0.00E+00		7.88E+09
Sm-151	2.35E+13	0.00E+00		2.35E+13
Sn-121m	8.09E+11	1.48E+12	1.46E+08	2.29E+12
Sn-126	6.11E+10			6.11E+10
Sr-90	2.37E+15	1.08E+09		2.37E+15
Tc-99	1.44E+12	5.89E+07	1.18E+06	1.44E+12
Th-229	5.20E+04		5.96E–03	5.20E+04
Th-230	4.30E+07		3.94E+00	4.30E+07
Th-232	6.59E+01		5.69E–06	6.59E+01
Th-234	2.43E+10		1.98E+03	2.43E+10
U-233	9.01E+06	2.56E+06	8.31E–01	1.16E+07
U-234	1.15E+11		1.07E+04	1.15E+11
U-235	8.15E+08		8.81E+01	8.15E+08
U-236	2.60E+10		2.18E+03	2.60E+10
U-237	2.73E+10		2.35E+03	2.73E+10
U-238	2.43E+10		1.98E+03	2.43E+10
Zr-93	1.91E+11	3.05E+10	3.15E+06	2.21E+11

321 BWR II canisters:

- UO2-values from /SKBdoc 1221579, Table 14/.
- Inventory for construction material from /SKBdoc 1198314/ by linear interpolation between 38 and 60 MWd/kg U to find inventory at 47.8 MWd/kg U.
  (Data from appendix: folder 'IndAct-Ett element-rev3', files '88_Ind-B38-000.xls' and '89_Ind-B60-000.xls'.)
- Inventory for Crud from /SKBdoc 1198314/ by linear interpolation between 38 and 60 MWd/kg U to find inventory at 47.8 MWd/kg U.
  (Data from appendix: folder 'CrudAct-Ett element-rev3', files '95_Crud-B38-000.xls' and '96_Crud-B60-000.xls'.)

Table C-9. Radionuclide inventory in BWR III type canister (9 assemblies, burnup 47.8 MWd/kgHM, age 32 years). Inventory for fuel matrix based on information in /SKBdoc 1221579/ and construction material and Crud estimated from data given in /SKBdoc 1198314/.

Radionuclide	BWR III – Radionuclide inventory (Bq/canister)			
	UO_2-matrix	Constr. mtrl.	Crud	Total
Ac-227	1.13E+06		1.02E–01	1.13E+06
Ag-108m	4.18E+08	0.00E+00	3.82E+08	8.00E+08
Am-241	2.34E+14		2.10E+07	2.34E+14
Am-242	7.36E+11		1.22E+05	7.36E+11
Am-242m	7.39E+11		1.23E+05	7.39E+11
Am-243	2.85E+12		2.94E+05	2.85E+12
C-14	2.74E+10	6.46E+10		9.20E+10
Cd-113m	7.55E+09	0.00E+00		7.55E+09
Cl-36	4.32E+08	6.32E+06		4.38E+08
Cm-242	6.09E+11		1.01E+05	6.09E+11
Cm-243	1.09E+12		8.56E+04	1.09E+12
Cm-244	1.55E+14		1.85E+07	1.55E+14
Cm-245	5.25E+10		7.69E+03	5.25E+10
Cm-246	1.44E+10		2.32E+03	1.44E+10
Cs-135	4.00E+10			4.00E+10
Cs-137	4.19E+15	1.28E+09		4.19E+15
Eu-152	1.06E+11	0.00E+00		1.06E+11
H-3	8.36E+12	0.00E+00		8.36E+12
Ho-166m	8.05E+09	0.00E+00		8.05E+09
I-129	2.30E+09			2.30E+09
Mo-93	6.01E+07	2.60E+08	5.87E+06	3.26E+08
Nb-93m	1.10E+11	2.48E+12	6.98E+10	2.66E+12
Nb-94	1.11E+07	7.70E+09	2.13E+08	7.92E+09
Ni-59	2.28E+08	3.07E+11	5.28E+09	3.12E+11
Ni-63	2.26E+10	3.47E+13	6.02E+11	3.54E+13
Np-237	2.75E+10		2.52E+03	2.75E+10
Np-238	3.33E+09		5.51E+02	3.33E+09
Np-239	2.85E+12	0.00E+00	2.94E+05	2.85E+12
Pa-231	1.94E+06		1.77E–01	1.94E+06
Pa-233	2.75E+10		2.52E+03	2.75E+10
Pa-234m	1.82E+10		1.48E+03	1.82E+10
Pb-210	4.00E+04		3.81E–03	4.00E+04
Pd-107	1.05E+10			1.05E+10
Pu-238	2.66E+14		2.70E+07	2.66E+14
Pu-239	1.87E+13	1.74E+08	1.54E+06	1.87E+13
Pu-240	4.01E+13		2.51E+06	4.01E+13
Pu-241	1.92E+15		1.65E+08	1.92E+15
Pu-242	2.10E+11		1.72E+04	2.10E+11
Ra-226	1.43E+05		1.33E–02	1.43E+05
Se-79	5.91E+09	0.00E+00		5.91E+09
Sm-151	2.00E+13	0.00E+00		2.00E+13
Sn-121m	7.49E+11	1.40E+12	1.38E+08	2.15E+12
Sn-126	4.58E+10			4.58E+10
Sr-90	2.68E+15	1.34E+08		2.68E+15
Tc-99	1.08E+12	4.42E+07	8.88E+05	1.08E+12
Th-229	3.06E+04		3.70E–03	3.06E+04
Th-230	1.99E+07		1.82E+00	1.99E+07
Th-232	3.34E+01		2.92E–06	3.34E+01
Th-234	1.82E+10		1.48E+03	1.82E+10
U-233	4.71E+06	1.92E+06	4.36E–01	6.63E+06
U-234	7.46E+10		6.83E+03	7.46E+10
U-235	6.11E+08		6.60E+01	6.11E+08
U-236	1.95E+10		1.63E+03	1.95E+10
U-237	4.59E+10		3.94E+03	4.59E+10
U-238	1.82E+10		1.48E+03	1.82E+10
Zr-93	1.43E+11	2.29E+10	2.36E+06	1.66E+11

1,655 BWR III canisters:

- UO2-values from /SKBdoc 1221579, Table 14/.
- Inventory for construction material from /SKBdoc 1198314/ by linear interpolation between 38 and 60 MWd/kg U to find inventory at 47.8 MWd/kg U.
  (Data from appendix: folder 'IndAct-Ett element-rev3', files '88_Ind-B38-000.xls' and '89_Ind-B60-000.xls'.)
- Inventory for Crud from /SKBdoc 1198314/ by linear interpolation between 38 and 60 MWd/kg U to find inventory at 47.8 MWd/kg U.
  (Data from appendix: folder 'CrudAct-Ett element-rev3', files '95_Crud-B38-000.xls' and '96_Crud-B60-000.xls'.)

Table C-10. Radionuclide inventory in BWR MOX type canister (11 BWR assemblies, burnup 37.7 MWd/kgHM, age 43 years and 1 MOX assembly, burnup 50 MWd/kgHM, age 50 years). Inventory for fuel matrix based on information in /SKBdoc 1221579/ and construction material and Crud estimated from data given in /SKBdoc 1198314/.

Radionuclide	BWR_{MOX} – Radionuclide inventory (Bq/canister)			
	UO_2-matrix	Constr. mtrl.	Crud	Total
Ac-227	1.68E+06		1.38E–01	1.68E+06
Ag-108m	4.55E+08	0.00E+00	3.64E+08	8.19E+08
Am-241	4.48E+14		3.36E+07	4.48E+14
Am-242	2.32E+12		1.78E+05	2.32E+12
Am-242m	2.33E+12		1.79E+05	2.33E+12
Am-243	3.48E+12		3.60E+05	3.48E+12
C-14	2.59E+10	6.96E+10		9.55E+10
Cd-113m	8.13E+09	0.00E+00		8.13E+09
Cl-36	4.23E+08	6.84E+06		4.30E+08
Cm-242	1.92E+12		1.47E+05	1.92E+12
Cm-243	1.25E+12		8.15E+04	1.25E+12
Cm-244	1.15E+14		1.25E+07	1.15E+14
Cm-245	1.44E+11		1.14E+04	1.44E+11
Cm-246	2.07E+10		3.23E+03	2.07E+10
Cs-135	5.08E+10			5.08E+10
Cs-137	3.48E+15	1.07E+09		3.48E+15
Eu-152	9.26E+10	0.00E+00		9.26E+10
H-3	4.59E+12	0.00E+00		4.59E+12
Ho-166m	7.44E+09	0.00E+00		7.44E+09
I-129	2.51E+09			2.51E+09
Mo-93	6.06E+07	2.81E+08	4.91E+06	3.46E+08
Nb-93m	1.34E+11	1.68E+12	3.64E+10	1.85E+12
Nb-94	1.20E+07	8.50E+09	1.84E+08	8.70E+09
Ni-59	2.30E+08	3.46E+11	4.66E+09	3.51E+11
Ni-63	2.29E+10	3.51E+13	4.72E+11	3.56E+13
Np-237	3.05E+10		2.55E+03	3.05E+10
Np-238	1.05E+10		8.03E+02	1.05E+10
Np-239	3.48E+12	0.00E+00	3.60E+05	3.48E+12
Pa-231	2.70E+06		2.13E–01	2.70E+06
Pa-233	3.05E+10		2.55E+03	3.05E+10
Pa-234m	2.44E+10		2.00E+03	2.44E+10
Pb-210	1.22E+05		1.11E–02	1.22E+05
Pd-107	1.12E+10			1.12E+10
Pu-238	2.59E+14		2.31E+07	2.59E+14
Pu-239	3.30E+13	2.30E+08	2.12E+06	3.30E+13
Pu-240	6.43E+13		3.96E+06	6.43E+13
Pu-241	1.69E+15		1.29E+08	1.69E+15
Pu-242	2.57E+11		2.51E+04	2.57E+11
Ra-226	3.62E+05		3.21E–02	3.62E+05
Se-79	6.46E+09	0.00E+00		6.46E+09
Sm-151	2.80E+13	0.00E+00		2.80E+13
Sn-121m	7.02E+11	1.32E+12	1.01E+08	2.02E+12
Sn-126	4.94E+10			4.94E+10
Sr-90	2.23E+15	9.98E+08		2.23E+15
Tc-99	1.21E+12	4.86E+07	7.74E+05	1.21E+12
Th-229	2.83E+04		2.60E–03	2.83E+04
Th-230	3.90E+07		3.35E+00	3.90E+07
Th-232	4.99E+01		3.74E–06	4.99E+01
Th-234	2.44E+10		2.00E+03	2.44E+10
U-233	6.57E+06	2.52E+06	5.55E–01	9.09E+06
U-234	1.11E+11		9.35E+03	1.11E+11
U-235	1.28E+09		8.56E+01	1.28E+09
U-236	2.24E+10		1.64E+03	2.24E+10
U-237	4.04E+10		3.09E+03	4.04E+10
U-238	2.44E+10		2.00E+03	2.44E+10
Zr-93	1.56E+11	2.48E+10	2.00E+06	1.81E+11

267 BWRMOX canisters:

- UO2-values from /SKBdoc 1221579, Table 14/.
- Inventory for construction material from /SKBdoc 1198314/ at 38 MWd/kg U for 11 BWR assemblies and 50 MWd/kg U for 1 MOX assembly.
  (Data from appendix: folder 'IndAct-Ett element-rev3', files '88_Ind-B38-000.xls' and '93_Ind-M50-000.xls'.)
- Inventory for Crud from /SKBdoc 1198314/ at 38 MWd/kg U for 11 BWR assemblies and 50 MWd/kg U for 1 MOX assembly.
  (Data from appendix: folder 'CrudAct-Ett element-rev3', files '95_Crud-B38-000.xls' and '100_Crud-M50-000.xls'.)

Table C-11. Radionuclide inventory in PWR II type canister (4 assemblies, burnup 57 MWd/kgHM, age 55 years Inventory for fuel matrix based on information in /SKBdoc 1221579/ and construction material and Crud estimated from data given in /SKBdoc 1198314/.

Radionuclide	PWR II – Radionuclide inventory (Bq/canister)				
	UO_2-matrix	Constr. mtrl.	Crud	Control rod	Total
Ac-227	2.24E+06		2.97E–01		2.24E+06
Ag-108m	3.54E+08	2.40E+08	5.02E+07	4.43E+13	4.43E+13
Am-241	3.93E+14		5.62E+07		3.93E+14
Am-242	7.39E+11		1.74E+05		7.39E+11
Am-242m	7.42E+11		1.75E+05		7.42E+11
Am-243	5.07E+12		6.42E+05		5.07E+12
C-14	3.03E+10	5.35E+10		2.49E+09	8.64E+10
Cd-113m	3.62E+09	0.00E+00		1.24E+12	1.24E+12
Cl-36	4.29E+08	6.24E+06		7.47E+05	4.36E+08
Cm-242	6.11E+11		1.44E+05		6.11E+11
Cm-243	1.00E+12		1.08E+05		1.00E+12
Cm-244	1.28E+14		1.63E+07		1.28E+14
Cm-245	1.19E+11		1.71E+04		1.19E+11
Cm-246	3.42E+10		4.60E+03		3.42E+10
Cs-135	4.80E+10				4.80E+10
Cs-137	3.48E+15	5.65E+08		0.00E+00	3.48E+15
Eu-152	3.28E+10	0.00E+00		0.00E+00	3.28E+10
H-3	3.21E+12	5.99E+09		3.74E+08	3.22E+12
Ho-166m	1.14E+10	0.00E+00		0.00E+00	1.14E+10
I-129	3.27E+09				3.27E+09
Mo-93	1.09E+08	1.14E+09	7.13E+04	1.21E+08	1.37E+09
Nb-93m	1.81E+11	8.59E+13	5.62E+07	1.65E+10	8.61E+13
Nb-94	1.61E+07	6.81E+11	4.36E+05	2.13E+08	6.81E+11
Ni-59	2.23E+08	1.95E+11	3.37E+08	1.63E+10	2.12E+11
Ni-63	2.13E+10	1.88E+13	2.90E+10	1.27E+12	2.02E+13
Np-237	4.53E+10		6.75E+03		4.53E+10
Np-238	3.34E+09		7.88E+02		3.34E+09
Np-239	5.07E+12	0.00E+00	6.42E+05	0.00E+00	5.07E+12
Pa-231	2.99E+06		4.23E–01		2.99E+06
Pa-233	4.53E+10		6.75E+03		4.53E+10
Pa-234m	2.12E+10		2.89E+03		2.12E+10
Pb-210	2.63E+05		3.69E–02		2.63E+05
Pd-107	1.56E+10				1.56E+10
Pu-238	3.53E+14		5.15E+07		3.53E+14
Pu-239	2.53E+13	1.83E+08	3.74E+06	0.00E+00	2.53E+13
Pu-240	4.96E+13		6.38E+06		4.96E+13
Pu-241	9.22E+14		1.29E+08		9.22E+14
Pu-242	3.30E+11		3.92E+04		3.30E+11
Ra-226	6.78E+05		9.53E–02		6.78E+05
Se-79	8.21E+09	3.78E+06		1.17E+06	8.22E+09
Sm-151	2.49E+13	0.00E+00		0.00E+00	2.49E+13
Sn-121m	8.06E+11	1.18E+11	2.96E+05	2.52E+07	9.23E+11
Sn-126	6.59E+10				6.59E+10
Sr-90	2.10E+15	5.23E+08		0.00E+00	2.10E+15
Tc-99	1.48E+12	1.66E+08	1.22E+04	1.93E+07	1.48E+12
Th-229	6.38E+04		8.86E–03		6.38E+04
Th-230	6.03E+07		8.53E+00		6.03E+07
Th-232	7.24E+01		1.14E–05		7.24E+01
Th-234	2.12E+10		2.89E+03		2.12E+10
U-233	1.20E+07	1.73E+06	1.81E+00	0.00E+00	1.37E+07
U-234	1.46E+11		2.08E+04		1.46E+11
U-235	6.88E+08		1.54E+02		6.88E+08
U-236	2.56E+10		3.99E+03		2.56E+10
U-237	2.21E+10		3.08E+03		2.21E+10
U-238	2.12E+10		2.89E+03		2.12E+10
Zr-93	1.97E+11	2.03E+10	1.38E+04	1.66E+03	2.18E+11

38 PWR II canisters:

- UO2-values from /SKBdoc 1221579, Table 13/.
- Inventory for construction material from /SKBdoc 1198314/ by linear interpolation between 30 and 60 MWd/kg U to find inventory at 57 MWd/kg U.
  (Data from appendix: folder 'IndAct-Ett element-rev3', files '90_Ind-P30-000.xls' and '91_Ind-P60-000.xls'.)
- Inventory for Crud from /SKBdoc 1198314/ by linear interpolation between 30 and 60 MWd/kg U to find inventory at 57 MWd/kg U.
  (Data from appendix: folder 'CrudAct-Ett element-rev3', files '97_Crud-P30-000.xls' and '98_Crud-P60-000.xls'.)
- Inventory for control rods from /SKBdoc 1179234, appendix: folder 'Styrstavar-rev3', file '110_PWR-ss.xls').

Table C-12. Radionuclide inventory in PWR III type canister (2 assemblies, burnup 57 MWd/kgHM, age 51 years and 1 assembly burnup 57 MWd/kgHM, age 20 years). Inventory for fuel matrix based on information in /SKBdoc 1221579/ and construction material and Crud estimated from data given in /SKBdoc 1198314/.

Radionuclide	PWR III – Radionuclide inventory (Bq/canister)				
	UO_2-matrix	Constr. mtrl.	Crud	Control rod	Total
Ac-227	1.36E+06		1.75E–01		1.36E+06
Ag-108m	2.88E+08	1.85E+08	3.85E+07	4.69E+13	4.69E+13
Am-241	2.64E+14		3.79E+07		2.64E+14
Am-242	5.96E+11		1.41E+05		5.96E+11
Am-242m	5.99E+11		1.41E+05		5.99E+11
Am-243	3.81E+12		4.82E+05		3.81E+12
C-14	2.28E+10	4.02E+10		2.50E+09	6.55E+10
Cd-113m	7.22E+09	0.00E+00		6.88E+12	6.88E+12
Cl-36	3.22E+08	4.68E+06		7.47E+05	3.27E+08
Cm-242	4.93E+11		1.16E+05		4.93E+11
Cm-243	1.14E+12		1.22E+05		1.14E+12
Cm-244	1.96E+14		2.49E+07		1.96E+14
Cm-245	8.91E+10		1.29E+04		8.91E+10
Cm-246	2.57E+10		3.46E+03		2.57E+10
Cs-135	3.60E+10				3.60E+10
Cs-137	3.85E+15	6.26E+08		0.00E+00	3.85E+15
Eu-152	7.04E+10	0.00E+00		0.00E+00	7.04E+10
H-3	7.71E+12	1.44E+10		2.66E+09	7.73E+12
Ho-166m	8.65E+09	0.00E+00		0.00E+00	8.65E+09
I-129	2.45E+09				2.45E+09
Mo-93	8.20E+07	8.58E+08	5.36E+04	1.22E+08	1.06E+09
Nb-93m	1.19E+11	1.45E+14	9.47E+07	7.20E+10	1.45E+14
Nb-94	1.21E+07	5.11E+11	3.27E+05	2.14E+08	5.11E+11
Ni-59	1.68E+08	1.47E+11	2.53E+08	1.63E+10	1.63E+11
Ni-63	1.78E+10	1.57E+13	2.41E+10	1.62E+12	1.73E+13
Np-237	3.28E+10		4.89E+03		3.28E+10
Np-238	2.69E+09		6.36E+02		2.69E+09
Np-239	3.81E+12	0.00E+00	4.82E+05	0.00E+00	3.81E+12
Pa-231	2.09E+06		2.82E–01		2.09E+06
Pa-233	3.28E+10		4.89E+03		3.28E+10
Pa-234m	1.59E+10		2.17E+03		1.59E+10
Pb-210	1.13E+05		1.59E–02		1.13E+05
Pd-107	1.17E+10				1.17E+10
Pu-238	2.98E+14		4.35E+07		2.98E+14
Pu-239	1.90E+13	1.38E+08	2.80E+06	0.00E+00	1.90E+13
Pu-240	3.70E+13		4.76E+06		3.70E+13
Pu-241	1.80E+15		2.52E+08		1.80E+15
Pu-242	2.47E+11		2.94E+04		2.47E+11
Ra-226	3.15E+05		4.42E–02		3.15E+05
Se-79	6.16E+09	2.83E+06		1.17E+06	6.16E+09
Sm-151	2.09E+13	0.00E+00		0.00E+00	2.09E+13
Sn-121m	7.36E+11	1.10E+11	2.76E+05	4.09E+07	8.46E+11
Sn-126	4.94E+10				4.94E+10
Sr-90	2.39E+15	5.88E+08		0.00E+00	2.39E+15
Tc-99	1.11E+12	1.25E+08	9.14E+03	1.93E+07	1.11E+12
Th-229	3.93E+04		5.34E–03		3.93E+04
Th-230	3.25E+07		4.58E+00		3.25E+07
Th-232	4.09E+01		6.42E–06		4.09E+01
Th-234	1.59E+10		2.17E+03		1.59E+10
U-233	6.98E+06	1.29E+06	1.05E+00	0.00E+00	8.27E+06
U-234	9.75E+10		1.39E+04		9.75E+10
U-235	5.16E+08		1.16E+02		5.16E+08
U-236	1.91E+10		2.99E+03		1.91E+10
U-237	4.31E+10		6.02E+03		4.31E+10
U-238	1.59E+10		2.17E+03		1.59E+10
Zr-93	1.48E+11	1.52E+10	1.04E+04	1.66E+03	1.63E+11

557 PWR III canisters:

- UO2-values from /SKBdoc 1221579, Table 13/.
- Inventory for construction material from /SKBdoc 1198314/ by linear interpolation between 30 and 60 MWd/kg U to find inventory at 57 MWd/kg U.
  (Data from appendix: folder 'IndAct-Ett element-rev3', files '90_Ind-P30-000.xls' and '91_Ind-P60-000.xls'.)
- Inventory for Crud from /SKBdoc 1198314/ by linear interpolation between 30 and 60 MWd/kg U to find inventory at 57 MWd/kg U.
  (Data from appendix: folder 'CrudAct-Ett element-rev3', files '97_Crud-P30-000.xls' and '98_Crud-P60-000.xls'.)
- Inventory for control rods from /SKBdoc 1179234, appendix: folder 'Styrstavar-rev3', file '110_PWR-ss.xls'/.

Table C-13. Radionuclide inventory in PWR MOX type canister (3 PWR assemblies, burnup 44.8 MWd/kgHM, age 32 years and 1 MOX assembly burnup 34.8 MWd/kgHM, age 57 years). Inventory for fuel matrix based on information in /SKBdoc 1221579/ and construction material and Crud estimated from data given in /SKBdoc 1198314/.

Radionuclide	PWR_{MOX} – Radionuclide inventory (Bq/canister)				
	UO_2-matrix	Constr. mtrl.	Crud	Control rod	Total
Ac-227	1.12E+06		1.91E–01		1.12E+06
Ag-108m	3.14E+08	2.15E+08	5.57E+07	4.60E+13	4.60E+13
Am-241	3.67E+14		4.31E+07		3.67E+14
Am-242	9.32E+11		1.29E+05		9.32E+11
Am-242m	9.36E+11		1.30E+05		9.36E+11
Am-243	3.54E+12		3.90E+05		3.54E+12
C-14	1.99E+10	4.43E+10		2.49E+09	6.66E+10
Cd-113m	8.18E+09	0.00E+00		3.89E+12	3.90E+12
Cl-36	2.85E+08	5.16E+06		7.47E+05	2.91E+08
Cm-242	7.70E+11		1.07E+05		7.70E+11
Cm-243	9.82E+11		9.86E+04		9.82E+11
Cm-244	1.21E+14		1.61E+07		1.21E+14
Cm-245	6.77E+10		8.63E+03		6.77E+10
Cm-246	1.21E+10		2.22E+03		1.21E+10
Cs-135	3.47E+10				3.47E+10
Cs-137	3.86E+15	6.52E+08		0.00E+00	3.86E+15
Eu-152	8.02E+10	0.00E+00		0.00E+00	8.02E+10
H-3	7.04E+12	1.17E+10		1.39E+09	7.06E+12
Ho-166m	7.50E+09	0.00E+00		0.00E+00	7.50E+09
I-129	2.27E+09				2.27E+09
Mo-93	7.76E+07	9.52E+08	6.70E+04	1.22E+08	1.15E+09
Nb-93m	1.06E+11	1.34E+14	9.75E+07	4.41E+10	1.34E+14
Nb-94	1.13E+07	5.75E+11	4.10E+05	2.14E+08	5.75E+11
Ni-59	1.55E+08	1.66E+11	3.14E+08	1.63E+10	1.83E+11
Ni-63	1.64E+10	1.72E+13	2.95E+10	1.49E+12	1.88E+13
Np-237	2.84E+10		4.55E+03		2.84E+10
Np-238	4.21E+09		5.84E+02		4.21E+09
Np-239	3.54E+12	0.00E+00	3.90E+05	0.00E+00	3.54E+12
Pa-231	2.01E+06		3.17E–01		2.01E+06
Pa-233	2.84E+10		4.55E+03		2.84E+10
Pa-234m	1.92E+10		2.95E+03		1.92E+10
Pb-210	5.95E+04		1.65E–02		5.95E+04
Pd-107	1.06E+10				1.06E+10
Pu-238	2.41E+14		3.51E+07		2.41E+14
Pu-239	2.56E+13	1.82E+08	3.42E+06	0.00E+00	2.56E+13
Pu-240	4.86E+13		5.29E+06		4.86E+13
Pu-241	2.19E+15		2.19E+08		2.19E+15
Pu-242	2.61E+11		2.73E+04		2.61E+11
Ra-226	2.11E+05		4.78E–02		2.11E+05
Se-79	5.68E+09	3.13E+06		1.17E+06	5.69E+09
Sm-151	2.49E+13	0.00E+00		0.00E+00	2.49E+13
Sn-121m	7.13E+11	1.20E+11	3.35E+05	3.48E+07	8.33E+11
Sn-126	4.50E+10				4.50E+10
Sr-90	2.49E+15	6.10E+08		0.00E+00	2.49E+15
Tc-99	1.06E+12	1.43E+08	1.14E+04	1.93E+07	1.06E+12
Th-229	1.94E+04		4.27E–03		1.94E+04
Th-230	2.89E+07		5.10E+00		2.89E+07
Th-232	3.17E+01		6.57E–06		3.17E+01
Th-234	1.92E+10		2.95E+03		1.92E+10
U-233	5.38E+06	1.69E+06	9.90E–01	0.00E+00	7.07E+06
U-234	1.02E+11		1.48E+04		1.02E+11
U-235	9.69E+08		1.65E+02		9.69E+08
U-236	1.88E+10		3.05E+03		1.88E+10
U-237	5.24E+10		5.24E+03		5.24E+10
U-238	1.92E+10		2.95E+03		1.92E+10
Zr-93	1.36E+11	1.68E+10	1.30E+04	1.66E+03	1.53E+11

33 PWRMOX canisters:

- UO2-values from /SKBdoc 1221579, Table 13/.
- Inventory for construction material from /SKBdoc 1198314/ by linear interpolation between 30 and 60 MWd/kg U to find inventory at 44.8 MWd/kg U.
  Assumption: At any given burnup or age the construction material data is the same for PWR and PWRMOX.
  (Data from appendix: folder 'IndAct-Ett element-rev3', files '90_Ind-P30-000.xls' and '91_Ind-P60-000.xls'.)
- Inventory for Crud from /SKBdoc 1198314/ by linear interpolation between 30 and 60 MWd/kg U to find inventory at 44.8 MWd/kg U.
  Assumption: At any given burnup or age the crud data is the same for PWR and PWRMOX.
  (Data from appendix: folder 'CrudAct-Ett element-rev3', files '97_Crud-P30-000.xls' and '98_Crud-P60-000.xls'.)
- Inventory for control rods from /SKBdoc 1179234, appendix: folder 'Styrstavar-rev3', file '110_PWR-ss.xls'/.

Table C-14. Contribution from predominant radionuclides to the total decay power at time for encapsulation /SKBdoc 1221579/.

Radionuclide	Decay power (W)							
	BWR I	BWR II	BWR III	BWR MOX	PWR I	PWR II	PWR III	PWR MOX
Am- 241	270	315	211	404	294	355	238	331
Am- 243	2	3	2	3	2	4	3	3
Ba- 137m	423	381	420	349	411	349	386	387
Cm-242	1	1	1	2		1		1
Cm-243	1	1	1	1	1	1	1	1
Cm-244	80	103	147	108	87	121	185	114
Co-60								1
Cs-134							2	
Cs-137	127	114	126	105	123	105	116	116
Eu-154	8	4	11	5	7	3	13	10
Kr-85	3	2	4	2	3	1	4	4
Pu-238	213	278	238	232	220	316	267	216
Pu-239	21	21	16	28	21	21	16	22
Pu-240	39	45	34	54	34	42	31	41
Pu-241	2	1	2	1	2	1	2	2
Sr-90	88	74	84	70	85	66	75	78
Y-90	420	355	401	334	407	314	358	373

Table C-15. Contribution from different isotopes in the fuel matrix to the total mass in one BWR I and one PWR I type canister at the time for encapsulation (for isotope masses greater than or equal to 1 g/canister in either BWR I or PWR I type canister).

Isotope	BWR I Mass(g)	PWR I Mass(g)
Ag-109	188	189
Al-27	13	11
Am-241	2,358	2,569
Am-242m	2	2
Am-243	324	361
Ba-134	449	455
Ba-135	1	<1
Ba-136	78	71
Ba-137	1,919	1,900
Ba-138	3,326	3,268
Br-81	53	52
C-12	17	15
Ca-40	6	5
Cd-110	102	102
Cd-111	58	59
Cd-112	29	28
Cd-114	32	32
Cd-116	12	12
Ce-140	3,210	3,143
Ce-142	2,858	2,809
Cl-35	3	3
Cm-244	28	31
Cm-245	5	6
Cr-52	2	2
Cs-133	2,867	2,802
Cs-135	1,045	870
Cs-137	1,309	1,272
Dy-160	7	<1
Dy-161	6	<1
Dy-162	6	2
Dy-163	8	5
Dy-164	<1	2
Er-166	3	4
Eu-151	8	9
Eu-153	300	304
Eu-154	3	3
F-19	4	4
Fe-56	10	9
Gd-152	7	<1
Gd-154	132	67
Gd-155	15	15
Gd-156	1,697	233
Gd-158	1,830	52
Gd-160	953	3
He-4	11	11
Ho-165	9	7
I-127	117	115
I-129	389	383
In-115	4	4
Kr-82	2	2
Kr-83	100	98
Kr-84	299	291

Isotope	BWR I Mass(g)	PWR I Mass(g)
Kr-85	6	5
Kr-86	461	451
La-139	3,109	3,056
Mg-24	2	2
Mn-55	4	3
Mo-100	2,398	2,356
Mo-95	1,914	1,880
Mo-96	117	113
Mo-97	2,062	2,023
Mo-98	2,103	2,063
N-14	29	26
Nd-142	62	59
Nd-143	1,926	1,912
Nd-144	3,474	3,390
Nd-145	1,684	1,647
Nd-146	1,818	1,787
Nd-148	936	922
Nd-150	456	448
Ni-58	7	6
Ni-60	3	3
Np-237	1,214	1,320
Pd-104	648	625
Pd-105	1,018	988
Pd-106	945	960
Pd-107	565	564
Pd-108	367	368
Pd-110	120	121
Pr-141	2,850	2,799
Pu-238	375	387
Pu-239	10,972	11,052
Pu-240	5,563	4,860
Pu-241	470	490
Pu-242	1,356	1,432
Rb-85	304	297
Rb-87	622	608
Rh-103	1,148	1,142
Ru-100	305	311
Ru-101	1,987	1,954
Ru-102	2,051	2,017
Ru-104	1,428	1,412
Sb-121	9	9
Sb-123	12	12
Se-77	2	2
Se-78	6	6
Se-79	12	12
Se-80	32	31
Se-82	84	83
Si-28	19	17
Sm-147	596	560
Sm-148	369	342
Sm-149	7	9
Sm-150	756	768
Sm-151	25	27
Sm-152	317	304
Sm-154	92	92

Isotope	BWR I Mass(g)	PWR I Mass(g)
Sn-116	7	8
Sn-117	11	11
Sn-118	9	9
Sn-119	9	9
Sn-120	9	9
Sn-122	12	12
Sn-124	20	20
Sn-126	47	47
Sr-86	1	<1
Sr-88	858	839
Sr-90	538	520
Tb-159	57	6
Tc-99	1,982	1,935
Te-125	27	27
Te-126	2	1
Te-128	221	218
Te-130	958	942
Ti-46	2	<1
Ti-47	2	<1
Ti-48	15	13
U-234	443	552
U-235	15,084	15,116
U-236	10,397	10,184
U-238	1,963,200	1,722,000
Xe-128	9	9
Xe-130	19	19
Xe-131	1,029	992
Xe-132	2,878	2,856
Xe-134	3,904	3,840
Xe-136	5,815	5,884
Y-89	1,146	1,121
Zn-64	25	22
Zn-66	15	13
Zn-67	2	2
Zn-68	10	9
Zr-90	878	864
Zr-91	1,506	1,474
Zr-92	1,638	1,604
Zr-93	1,790	1,754
Zr-94	1,958	1,919
Zr-96	2,051	2,011

Appendix D

Glossary of abbreviations and branch terms

BWR	Boiling water reactor
Clab	Central interim storage facility
Clink	Central interim storage facility and encapsulation plant
Crud	Deposits on the surface of the fuel cladding
Crudact	Computer program for calculation of radionuclide inventory in crud
FGR	Fission gas release
HM	Heavy metal
Indact	Computer program for calculation of induced (neutron activation) radionuclide inventory
K_{eff}	Effective multiplication factor, shall be less than 0.95 to avoid criticality
MOX	Mixed oxide
Origen-S	Computer program for calculation of radionuclide inventory in irradiated fuel
PWR	Pressurised water reactor
transfer canister	Canister used to transfer spent fuel assemblies between handling pool in the interim storage part and encapsulation part of Clink
storage canister	Canister used for interim storage of spent fuel assemblies
UOX	Uranium oxide