

Technical report

Feasibility of KBS-3 spent fuel disposal concept for Norwegian spent fuel

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ABSTRACT

The feasibility of using the KBS-3 spent fuel disposal concept in disposing of Norwegian spent fuel is investigated in this report. Norway has operated several research reactors, and some of the fuel used in these reactors contains metallic uranium. Metallic uranium is unstable chemically compared to uranium oxide in commercial spent nuclear fuel, which typically is disposed of with the KBS-3 concept in Sweden and Finland. In this report, its disposal without any additional treatment is studied. The properties of metallic uranium and the related features, events and processes in the disposal facility are reviewed.

The number of KBS-3 canisters required to dispose the Norwegian spent fuel is preliminarily estimated. Various encapsulation options are investigated and their engineering requirement estimated for future use. Knowledge gaps and future work required for the use of the KBS-3 disposal concept are evaluated. The conclusion of this study is that the disposal of metallic uranium is possible with the KBS-3 concept, although uncertainties remain especially in the design of the canister internals. Preliminarily, 25 to 37 KBS-3 disposal canisters with varying lengths and internal designs are probably required for disposal.

Keywords: KBS-3, spent fuel disposal, metallic uranium

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1 Introduction

1.1 Aim of the study

Norwegian Nuclear Decommissioning (NND, Norsk nukleær dekommisjonering) is responsible for the storage, handling and disposal of radioactive waste in Norway. The Institute for Energy Technology (IFE, Institutt for energiteknikk) has operated several research reactors in Norway, all of which are now shut down.

The first Norwegian reactor, JEEP I, operated in Kjeller between 1951 and 1967 (Mattson, 2010). In 1959, the Halden Boiling Water Reactor (HBWR) was built in Halden, and JEEP II in Kjeller in 1966 (Mattson, 2010). The reactors were used for research purposes, and in the HBWR various experimental nuclear fuels have been tested over the years. In 2018, IFE decided to shut down and decommission the HBWR, and in 2019, the same decision was made for the JEEP II reactor. All three reactors have produced spent nuclear fuel (SNF) that must be disposed of safely. In this report, the spent fuel, which is currently in the care of IFE, is nevertheless referred to as NND spent fuel. Part of this fuel is in metallic form, which differs considerably in its material properties from commercial SNF. NND is exploring various options for the treatment of this spent fuel, for example reprocessing of the fuel, stabilization of the metallic uranium, disposal in its present form in a deep geological repository or a borehole repository.

In Sweden and Finland, commercial SNF made from uranium oxide fuel pellets in zirconium alloy cladding is to be disposed of with the KBS-3 repository concept. The KBS-3 concept was developed by SKB AB in Sweden), and is based on a system of multiple natural and engineered barriers for containing radioactive materials in the disposed waste. A feature of the ceramic uranium dioxide fuel, the slow dissolution rate on contact with groundwater, acts like a barrier which strongly limits releases of radioactive nuclides. The next release barrier is the disposal canister, which is composed of a corrosion-resistant copper overpack and a cast iron insert which holds the fuel assemblies in place and brings mechanical strength to the canister. The further release barriers are the bentonite buffer around the canister and the bentonite backfill of the disposal tunnel. The natural release barrier is the granitic bedrock, of which there is 400 to 500 meters above the disposal facility (Posiva Oy, 2020). In Figure 1.1-1, the KBS-3 concept is schematically shown. A long-term safety case for the construction licence of the disposal facility in Finland was presented by Posiva in 2012 and will be updated for the operation licence application in 2022.

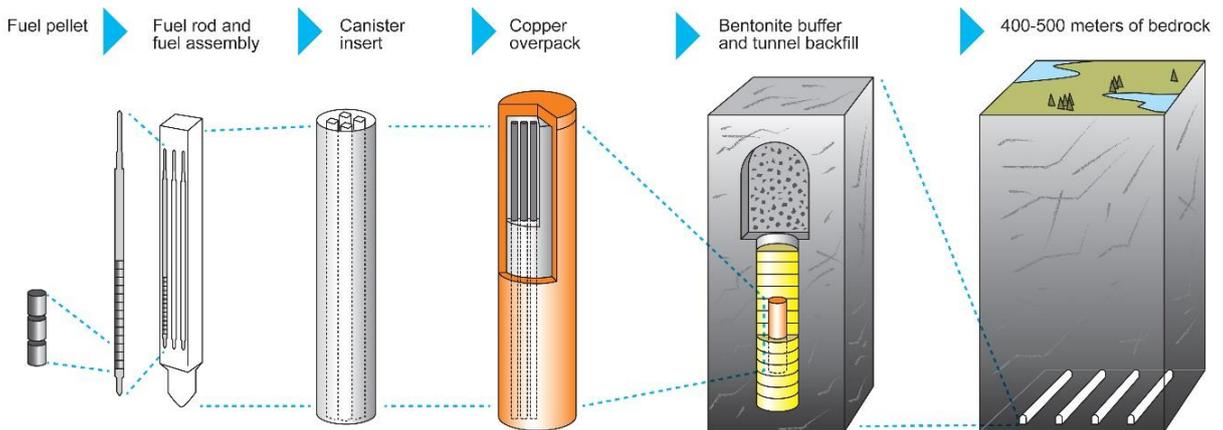


Figure 1.1-1. Schematic representation of the KBS-3 final repository concept (Posiva Oy, 2020).

In this study, the possibility for the deposition of all NND spent fuel without any further treatment in a KBS-3 type deep geological repository is considered. The standard copper containers of the KBS-3 concept are used, but variations in the internal structure of the canister are investigated.

The unstable nature of the metallic uranium in contact with groundwater is especially concentrated upon. For example, the metallic uranium generates hydrogen gas through reaction with water, oxidizes and

expands, and may release radioactivity from the fuel faster than conventional oxide fuel. In the rest of this introduction, similar projects in other countries are briefly introduced, and the relevant Norwegian and international legislation presented. The waste inventory is presented in detail in section 2. Section 3 discusses in detail the effects of the change of material form on the safety case of the disposal facility. In section 4, the feasibility of using KBS-3 canisters for the disposal of NND spent fuel and possible required changes to the internal structure of the canisters are presented. Due to the instability of metallic uranium, different release barriers may be required compared to the disposal of uranium oxide spent fuel. Such release barriers are discussed in section 5. Finally, section 6 presents knowledge gaps and future work identified in this work for the use of the KBS-3 disposal concept with metallic uranium, and section 7 summarizes the findings of this report.

1.2 Other metallic fuel disposal projects

1.2.1 US

US DOE has a large inventory of metallic uranium nuclear fuels from various reactors. The largest stockpile of metallic fuel is from the plutonium production reactor, named as N, operated at the Hanford Site in Washington (US NWTRB, 2017). The N reactor fuel consists of cylindrical fuel elements where metallic uranium is encased in zirconium cladding. There is a total of 2096 tHM (metric tons of heavy metal) of this type of fuel to be disposed of (US NWTRB, 2017).

US DOE long planned a disposal facility at Yucca Mountain, where the US DOE spent fuel (DSNF) would be disposed of with commercial spent nuclear fuel. DSNF was planned to be disposed of in stainless steel canisters, and the host rock was dry volcanic tuff. Little water would be in contact with the waste form in the Yucca Mountain repository. The US safety assessment report for a repository hosting exotic fuel types is currently the most comprehensive such report.

1.2.2 The Netherlands

The Netherlands hosts or hosted three experimental reactors: the Low Flux Reactor (LFR, shut down in 2010), the High Flux Reactor (HFR, operational) and Hoger Ondewijs Reactor (HOR, operational) (Deismann et al., 2016). The fuel from these reactors is aluminium-clad uranium aluminide-aluminum (UAlx-Al) dispersion fuel (LFR, HFR and HOR) or uranium silicide-aluminum (U₃Si₂-Al) dispersion fuel (HOR). The fuel elements are of various geometrical forms, typically concentric plates.

The fuel is planned to be disposed of with the OPERA disposal concept: fuel elements are placed in stainless steel canisters and these canisters disposed of in a boom clay formation in the Netherlands (Deismann et al., 2016). The OPERA disposal concept is based on a supercontainer containing concrete, concrete support structures of the disposal tunnels and a concrete backfill, and so the water in contact with the fuel is alkaline and contains dissolved cementitious materials (Deismann et al., 2016).

1.2.3 UK

The UK has started to review options for the disposal of their metallic fuels, which consist of spent fuel from Magnox reactors, breeder materials of the Dounreay fast reactor and small amounts (compared to the former) of metallic research reactor fuels (Swift et al., 2018). The initial plan is to reprocess all of the Magnox and Dounreay fast reactor fuels, but their disposal without reprocessing is considered as a contingency option (Swift et al., 2018).

The Magnox fuel elements consist of cylindrical fuel rods in round assemblies. The fuel rods contain uranium metal within a magnesium alloy cladding. The Dounreay fast breeder reactor breeder material is uranium metal encased in stainless steel, and several such rods are contained in one breeder element. Depending on assumptions about reprocessing up to 2500 tHM of Magnox fuel is expected to be disposed, whereas 44 tHM of Dounreay fast breeder reactor breeder material is expected. In the Dounreay fast breeder reactor fuel, a stainless steel cladding encapsulates the fuel, which is more durable compared to the aluminium cladding of the NND fuels. Various uranium-aluminum fuels have been used in the research reactors, and their total mass is less than 1 tHM. (Swift et al., 2018)

Several disposal concepts are being investigated, and one of these is a concept where a copper or carbon steel canister is used with bentonite backfill in high-strength host rock (Swift et al., 2018). This is the most relevant scenario for comparison with the KBS-3 concept.

1.3 Relevant regulations

1.3.1 Norwegian regulations

In Norwegian law, Forskrift om gjenvinning og behandling av avfall (avfallsforskriften, here referred to as the Waste Regulations) , and the Lov om vern mot forurensninger og om avfall (forurensningsloven, here referred to as Pollution Control Act) especially through the Forskrift om forurensningslovens anvendelse på radioaktiv forurensning og radioaktivt avfall (here referred to as Regulation on the application of Pollution Control Act on radioactive waste) are related to the regulation of radioactive waste management in Norway.

The Waste Regulations and the Regulation on the application of the Pollution Control Act on radioactive waste both classify radioactive waste into categories. Radioactive waste (Norwegian: radioaktivt avfall) is specified as waste (as specified in the section § 27 of the Pollution Control Act) that contains amounts of radioactive material above limits specified in Attachment I Section a of the Regulation on the application of the Pollution Control act on radioactive waste. In addition, radioactive waste that must be deposited (Norwegian: deponeringspliktig radioaktivt avfall), is specified as waste that exceeds the specific and total annually generated radioactivities specified in Attachment I Section b) of said regulation. Spent fuel falls into the latter category.

The first paragraph of § 16-4 of the Waste Regulation on proper storage of radioactive waste specifies that

“Radioaktivt avfall skal håndteres forsvarlig. Alle som oppbevarer, transporterer eller håndterer radioaktivt avfall, skal treffe nødvendige tiltak for å unngå fare for forurensning eller skade på mennesker eller dyr.”

or loosely translated,

“Radioactive waste must be handled properly. Anyone who stores, transports or handles radioactive waste must take the necessary measures to avoid the risk of contamination or injury to humans or animals.”

This objective must obviously be fulfilled by any disposal system.

The second paragraph of § 16-4 of the Waste Regulation specifies that

“Radioaktivt avfall skal ikke blandes sammen med annet avfall og ulike typer radioaktivt avfall skal ikke sammenblandes dersom dette kan medføre fare for forurensning eller skape problemer for den videre håndteringen av avfallet.”

or loosely translated,

“Radioactive waste should not be mixed with other waste and different types of radioactive waste should not be mixed if this could cause a risk of contamination or create problems for the further handling of the waste.”

The requirement for not mixing different types of radioactive waste may be relevant to the problem of disposing the various spent fuel types in NND spent fuel. Mixing of waste types is not allowed if this causes a risk of contamination, which in the repository should not be the case anyhow, as the objective of the repository is to contain the radioactive material. This requirement might be interpreted also to require that mixing of waste does not negatively affect the performance of the repository.

For example, in the Yucca Mountain final disposal facility safety assessment, where various very different types of fuels were planned to be disposed of in the facility, the mixing of waste types was explicitly treated

in the safety assessment. The interactions between codisposed waste and the heterogeneity of the waste inventory were treated as features of the disposal system.

1.3.2 IAEA safety standards

IAEA Specific Safety Requirements SSR-5 (IAEA, 2011) in requirement 8 specifies standards for the containment of radioactive waste. The effect of the waste form is mentioned in several paragraphs detailing requirement 8. SSR-5 specifies in paragraph 3.39 that

*“The containment **may be provided** by the characteristics of the waste form and the packaging and by the characteristics of other engineered components of the disposal system and the host environment and geological formation.”*

In addition, paragraph 3.40 states that

*“**The containment of the radionuclides in the waste form and the packaging** over a defined period has to ensure that the majority of shorter lived radionuclides decay in situ. For low level waste, such periods would be of the order of several hundred years; for high level waste the period would be several thousands of years. For high level waste, it also has to be ensured that any migration of radionuclides outside the disposal system would occur only after the heat produced by radioactive decay has substantially decreased.”*

Furthermore, paragraph 3.42 specifies that

*“Containment is most important for more highly concentrated radioactive waste, such as intermediate level waste and vitrified waste from fuel reprocessing, or for spent nuclear fuel. **Attention also has to be given to the durability of the waste form.** The most highly concentrated waste has to be emplaced in a containment configuration that is designed to retain its integrity for a long enough period of time to enable most of the shorter lived radionuclides to decay and for the associated generation of heat to decrease substantially.”*

Containment in the waste form and the packaging is required by paragraph 3.40, which may be interpreted to either require containment in both the waste form and the packaging, or in the waste form and the packaging as a whole. The latter interpretation would allow for offsetting a less stable waste form with a more durable packaging, which the former interpretation does not allow. The former, more conservative interpretation prohibits disposal of metallic uranium fuel. Paragraph 3.42 is more relaxed, as the durability of the waste form is not required but clearly must be taken into account in any performance analysis of the disposal system, and paragraph 3.39 allows for containment to be provided by the waste form, the packaging or any other engineered and natural barriers of the disposal system.

Requirement 7 of SSR-5 on multiple safety functions specifies in paragraph 3.35 the engineered and physical barriers and safety functions:

“The engineered and physical barriers that make up the disposal system are physical entities, such as the waste form, the packaging, the backfill, and the host environment and geological formation. A safety function may be provided by means of a physical or chemical property or process that contributes to containment and isolation, such as: impermeability to water; limited corrosion, dissolution, leach rate and solubility; retention of radionuclides; and retardation of radionuclide migration.”

Paragraph 3.38 of requirement 7 requires that the safety case justifies the functions of all features and safety functions:

*“The safety case has to explain and justify the functions performed by each physical element and other features. It also has to identify the time periods over which physical components and other features are expected to perform their various safety functions, and also **the alternative or additional safety functions that are available if a physical element does not fully perform or another safety function is not fulfilled.**”*

In the case of metallic fuel the safety function of slow dissolution rate of the waste form, as in the safety case for commercial SNF in KBS-3 type repository, is not fulfilled, but it could be compensated with some

other safety functions. In section 5 of this report, safety functions applicable for metallic uranium fuel are proposed. Due to the instability of metallic uranium fuel, some other safety functions are required if the containment by the waste form and the packaging does not last for at least several thousands of years.

2 Waste inventory

The largest part of NND spent fuel is metallic uranium, and uranium oxide and various exotic fuels have smaller inventories. Table 2-1 shows the main classes of various fuel types and their inventories.

In the following analysis, the metallic uranium in aluminium cladding is concentrated on due to its large inventory and the difference in behavior compared to commercial SNF. Especially the HBWR 2nd to 5th charge fuel and HBWR booster fuel are very similar to commercial SNF, albeit with a higher enrichment in some cases. This is partially the case also with HBWR experimental fuels.

JEEP II fuel is a borderline case, as the fuel is uranium oxide, but the cladding is aluminium. As the cladding is not considered as a release barrier for radioactivity in the disposal of commercial SNF, but the slow release from uranium oxide is a feature of the disposal system, the JEEP II fuel can be thought to require little changes relative to the safety case for disposal of zirconium-clad uranium oxide.

Table 2-1. Description of the various fuel types of NND spent fuel. Most data from Bennett (2020). *: Bennett (2020a), **: calculated based on other values in the table.

Reactor	JEEP I	HBWR 1 st charge	JEEP II	HBWR 2 nd to 4 th charge	HBWR 5 th charge	HBWR Booster	HBWR experimental
Fuel	U metal	U metal	UO ₂	UO ₂	UO ₂	UO ₂	UO ₂ , MOX, ThO ₂
²³⁵ U enrichment (%)	0.72	0.72	3.5	≤ 10	≤ 10 (mostly 6)	≤ 20	≤ 20, >90 (HEU-Th)
Cladding material	Al	Al	Al	Zircaloy	Zircaloy	Zircaloy	Various
Uranium mass per rod (kg)	19	22	1	0.6 – 0.9	0.4 – 0.9	0.2 – 0.9	0.1 – 0.9
Burn-up (MWd/kgU)	≤ 1	≤ 0.021	≤ 15	≤ 79.4	≤ 79.4	≤ 79.4	≤ 102.1
Rod length (m)	2.4	2.8	1.5	~ 1.8	1.1	≤ 1.1	≤ 1.1
Rod diameter (mm)	25	40	15	12.25 – 14.3	12.25	6.25 – 9.5	6.25 – 14.3
Number of rods (ca.)	170	300	1500	700	4500	2000	
Assemblies or single rods*	Assemblies	Rods	Assemblies	Assemblies	Mostly rods	Rods and assemblies	Mostly rods
Assembly length (m)	2.8	2.8	1.5	2nd charge: 2.83, 3rd & 4th charge: 3.66	1.1	1.2	Max. 1.2
Assembly diameter (mm)	70	40	90	≤ 70	≤ 70	≤ 70	≤ 70
Number of rods per assembly	2*	1*	11*	7*			
Number of assemblies (ca.)	85**	300**	136**	100**			
Mass (kg)	3000	7000.0	1500	3600		1400	
Mass per assembly (kg)	35.29	23.3	11.03	4.2 - 6.3	0.4 – 0.9	0.2 – 0.9	0.1 – 0.9
Current storage condition	Dry	Dry	Dry	Dry / wet	Dry / wet	Dry / wet	Dry / wet

3 Effects on features, events and processes

3.1 Description of analysis

In the following, the features, events and processes (FEPs), preliminarily judged most important, in the NND spent fuel waste form and the KBS-3 canister are described. The effects of the differences of the NND spent fuel waste form relative to the KBS-3 concept are judged for each FEP. Any biological effects are left out of scope of this review.

The following sections correspond to sections of the OECD/NEA IFEP classification (Capouet et al., 2019).

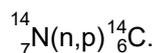
3.2 Features of the waste form

3.2.1 Contaminant inventory and preliminary release estimate

3.2.1.1 Radionuclide content

The radionuclide inventory in the NND waste contains less fission and activation products than commercial SNF due to the lower burnup of the NND waste. The enrichment of the NND waste is lower than that of average commercial SNF, as the uranium in most of the fuel is natural uranium. More higher actinides may have been produced in this kind of fuel relative to higher enrichment fuel with the same burnup. Activation products in the metallic parts of the assemblies may also contribute to the total radionuclide inventory. A very preliminary estimate of the effect of the radionuclide inventory is performed in the following, but detailed inventory calculations are necessary. This very preliminary estimate is used in comparison with Finnish release limits to estimate its effect on release, but the results are subject to change when the exact radionuclide inventory is determined.

Especially the amount of ^{14}C is very dependent on the nitrogen content of the materials to be disposed of, due to the following nuclear reaction:



Nitrogen can be present in the uranium fuel, but also in various metallic parts of the assemblies. All of these may contribute to the overall ^{14}C inventory. Fuel samples containing nitride fuel or materials otherwise high in nitrogen are a special case, which have to be analysed separately when determining the inventory.

In addition, ^{226}Ra is produced from the ^{238}U decay chain in the uranium fuel. ^{226}Ra is an important nuclide as it may be transported quickly from the repository to the biosphere (Poteri et al., 2014), and ^{226}Ra is a very radiotoxic nuclide (IAEA, 2014). In the process of ^{235}U enrichment, ^{234}U , a daughter isotope of ^{238}U , is also enriched and its content is higher than the equilibrium amount. As the half-life of ^{234}U is much shorter (1/18000th) than that of ^{238}U , higher initial amount of ^{234}U leads to higher amounts of nuclides of the ^{238}U decay chain that are lighter than ^{234}U . This leads to higher ^{226}Ra amount in the long term in the repository with enriched fuel. However, this effect is not present in natural, unenriched uranium, which is a positive factor regarding the disposal of the NND metallic fuel types.

Anttila (2005) calculated the activity inventories for various Finnish spent fuel types. A collection of this data is shown in Appendix A. The inventory is scaled with the maximum burnup of each fuel type, and the resulting approximate activity inventories for the some relevant nuclides, ^{14}C , ^{129}I and ^{135}Cs , are shown in Table 3.2-1. In addition, the activity inventory in the metallic fuel types, HBWR 1st charge and JEEP I, are shown separately, along with those of the mostly oxide fuel types. It is evident that most of the inventory is contained in the non-metallic fuel types. Due to the instability of the metallic fuel, the inventory in the metallic fuels is examined in more detail.

The inventories are compared to Finnish regulatory limits on activity release rates to the living environment from the disposal facility (STUK, 2018). The derivation of these limits is discussed in more detail by Ruokola (2002). An indicative minimum of an allowable duration of a release is given by the quotient of the activity inventory and the release rate limit. The comparison of the minimum durations for the three nuclides is shown in Table 3.2-2. Without surpassing the aforementioned annual release limits, the metallic fuel inventory from Table 3.2-1 could be released very quickly at a constant release rate. However, in reality, for example a small hole in the weld of the copper canister limits the release rate effectively, and the actual durations of releases are orders of magnitude longer.

Table 3.2-1. Approximate and very preliminary activity inventories in the NND spent fuel types and sums for metallic and non-metallic fuels.

Fuel type	Burnup (MWd/kgU)	Total mass (kg)	Approximate activity inventory (GBq)		
			¹⁴ C	¹²⁹ I	¹³⁵ Cs
HBWR 1st charge (metallic)	0.021	7000.0	0.14	0.004	0.08
JEEP I (metallic)	1.0	3000.0	2.89	0.09	1.69
JEEP II	15.0	1500.0	21.65	0.69	12.64
HBWR 2nd to 5th charge	79.4	3600.0	275.05	8.72	160.57
HBWR booster and experimental	102.1	1400.0	137.54	4.36	80.30
Sum of metallic fuels		10000.0	3.0	0.1	1.8
Sum of (mostly) oxide fuels		6500.0	434.2	13.8	253.5

Table 3.2-2. Preliminary indicative minimum durations for release. The release durations from a canister will be orders of magnitude longer.

	Nuclide		
	¹⁴ C	¹²⁹ I	¹³⁵ Cs
Inventory in metallic fuels (GBq)	3.0	0.1	1.8
Release rate limit (GBq/a)	0.3	0.1	0.3
Minimum required duration of release (a)	10	1	6

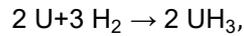
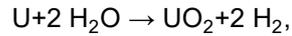
3.2.1.2 Chemical content

The cladding in several NND waste types is aluminium, which is not present in the commercial SNF being disposed of according to the KBS-3 concept. Presence of aluminium may affect the chemical processes within the disposal canister, and investigations on the in-canister processes performed for the standard KBS-3 method should be repeated accounting for the presence of aluminium or at least more closely investigated.

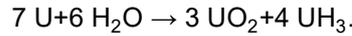
3.2.2 Waste form characteristics and properties: Pyrophoricity of metallic fuel

Metallic uranium with a high surface area, e.g. with a small particle size, is pyrophoric, so in the presence of oxygen a self-sustaining exothermic chain reaction may begin with this kind of material. However, the spent metallic fuel is in the form of bulk fuel rods, which have a low surface area to volume ratio. Also, little oxygen is present in the canister and it may be thought to be consumed by reaction with the metals within the canister. When the waste form is in contact with groundwater, water is present in excess compared to oxygen. Reactions with water are therefore judged to be more relevant than those with oxygen.

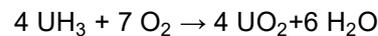
During oxidation with water, uranium hydride is formed in addition to the oxidation products of uranium:



which yields the total reaction



This reaction is important as the resulting uranium hydride UH_3 is more pyrophoric than uranium metal. Uranium hydride is more important in transport and handling operations performed on the fuel. Again, in the repository, little oxygen is available for reaction and there is no risk for a self-sustaining exothermic chain reaction of the pyrophoric materials. The reaction of uranium hydride with oxygen proceeds as follows:



which has a Gibbs energy of reaction (based on data from Grenthe et al. (1992)) at 25 °C of -5547 kJ/mol. Reaction of uranium hydride with water to form uranium oxide would proceed as follows:



but has a Gibbs energy of reaction at 25 °C of -772 kJ/mol. This means that the reaction with oxygen is thermodynamically much more favoured, but uranium hydride also reacts with water. It is notable that in this reaction, gas is generated from solid materials. This means that in a high pressure environment, the reaction equilibrium shifts towards the solid starting materials.

3.3 Waste package processes

3.3.1 Thermal processes

The design decay heat power limit for the KBS-3 concept is 1700 W per a BWR canister of certain dimensions, as reported by SKB (SKB, 2010). For a Loviisa-type canister, the decay heat power limit is lower, 1370 W (Raiko, 2012). The lower limit is used in the following for conservativeness.

The temperature rise is affected by the thermal conductivity of the bentonite buffer and heat transfer properties of the surrounding rock. The temperature of the bentonite buffer used in the KBS-3 concept must not exceed 100 °C (Ikonen et al., 2018). Above this temperature, the properties of bentonite could degrade. In practice, the temperature limit of 95 °C is used to account for various uncertainties (Ikonen et al., 2018). In the case of metallic fuel, it may be necessary to also consider chemical heat production. Both radiogenic and chemical heat production are discussed in the following.

3.3.1.1 Radiogenic heat production and transfer

Fission product amount is approximately linearly proportional to burnup (Baurens, 2014), so decay heat generation from fission products is similar to uranium oxide fuel with similar burnup. However, the amount of actinides may be slightly greater due to the higher atomic density of uranium in the metallic fuel due to increased neutron absorption to ^{238}U . Before 50 years of cooling time, various fission products such as cesium, barium and yttrium isotopes dominate the heat generation in spent fuel (Juutilainen & Häkkinen, 2019). Only at longer cooling times, various actinides become the dominant heat generating nuclides (Juutilainen & Häkkinen, 2019), but decay heat production relevant to temperature rise is mostly a problem at shorter cooling times.

The decay heat powers of fuels are estimated in the following based on publicly available data on commercial SNF in Anttila (2005). Due to the smaller accumulated burnup, relative to commercial SNF, of the NND spent fuel, the radiogenic heat production rate of the waste is lower than that of commercial

SNF disposed according to the KBS-3 concept. Although part of the fuel is metallic, burnup is the determining factor on decay heat power, so the commercial oxide fuel data can be used to obtain an order of magnitude estimate. As can be seen later, the low burnup in the metallic fuels yields a very small decay heat power for these assemblies.

In Appendix A, the decay heat powers per uranium mass unit for Finnish commercial SNF types are shown for cooling times of 30 and 50 years, respectively. The maximum value at cooling time of 30 years is 1530 W/tU, and at 50 years 1070 W/tU. The higher value from the shorter cooling time is used in the calculating the decay heat power per uranium mass for each burnup of the NND spent fuel types, shown in Table 3.3-1. Table 3.3-2 shows the decay heat powers in canisters with varying numbers of assemblies within a canister, as well as the maximum number of assemblies that could be placed in a canister to respect the decay heat limit. As a conservative estimate, the Loviisa-type canister decay heat limit of 1370 W is used (Raiko et al., 2012).

The results in Table 3.3-2 show that decay heat is not a limitation in placing the fuel rods in canisters, as the highest calculated number of assemblies that could be placed is 98 of the HBWR booster and experimental fuel types.

Table 3.3-1. Decay heat power estimates for the NND spent fuel types at cooling times of 30 a.

Fuel type	Burnup (MWd/kgU)	Estimated decay heat power (W/tU)
HBWR 1st charge	0.021	0.5
JEEP I	1.0	25.5
JEEP II	15.0	382.5
HBWR 2nd to 5th charge	79.4	2024.7
HBWR booster and experimental	102.1	2603.6

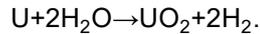
Table 3.3-2. Estimated decay heat power of canisters with different numbers of NND spent fuel assemblies and the maximum number of assemblies in a canister respecting the 1370 W decay heat power limit.

Fuel type	Uranium mass of assembly (kg)	Decay heat power in canister by number of assemblies in canister (W)				Maximum number of assemblies in canister
		12 pc.	24 pc.	50 pc.	100 pc.	
HBWR 1st charge	22.0	0.1	0.3	0.6	1.2	116289
JEEP I	38.0	11.6	23.3	48.5	96.9	1414
JEEP II	11.0	50.5	101.0	210.4	420.8	326
HBWR 2nd to 5th charge	6.3	153.1	306.1	637.8	1275.6	107
HBWR booster and experimental	6.3	196.8	393.7	820.1	1640.2	84

3.3.1.2 Chemical heat production and transfer

Due to the metallic, reactive uranium, chemical heat production rate in the NND waste is higher than of commercial SNF disposed by means of the KBS-3 concept. RWM (2016) preliminarily assessed the chemical heat power and at a certain temperature they judged that the chemical heat production is negligible compared to radiogenic heat production. This analysis is revisited in the following.

The reaction of water and uranium metal proceeds as follows, assuming an uranium oxide end product:



The corresponding enthalpy of reaction, based on data from the OECD/NEA TDB project (Grenthe, 1992) is

$$\Delta H_{r,ox} = \sum \Delta H_{f,products} - \sum \Delta H_{f,reactants} = -513.34 \text{ kJ/mol}.$$

The heat production rate from the oxidation reaction was calculated for two temperatures: 60 °C, as this was used by RWM (2016), and 104.9 °C, the highest temperature in a homogenized spent fuel-cast iron insert as calculated by Ikonen et al. (2018). The heat production rates at these temperatures, using the reaction rate from section 3.4.1, were 1.17 and 20.15 W/m², respectively. The chemical heat production in the metallic fuel assemblies was calculated with the initial surface area of the fuels using these estimates for chemical heat production rate, and the resulting chemical heat production rates of the fuel assemblies are shown in Table 3.3-3. Using the same temperature as RWM (2016), the chemical heat production is not significant. However, due to the exponential relationship of the reaction rate and temperature, the heat production at higher temperatures is significant.

The highest temperature in the canister is attained within 10-15 years after emplacement in the KBS-3 type repository (Ikonen et al., 2018). Therefore the highest temperature at the fuel surface, here estimated as 104.9 °C, occurs at a time when the disposal canister has not been breached and there is little water available for reaction with the uranium. The chemical heat production rate is therefore highly dependent on the canister breach time and volume of water flowing into the canister. In some scenarios, an initial penetrating defect of the canister is assumed, where the chemical heat power at this higher temperature is relevant (Nykyri et al., 2008). However, the initial penetrating defect allows only for a low water inflow rate, which is shown to lead to an insignificant chemical heat power in the fuel.

The maximum chemical heat power is calculated by the following equation:

$$P = -kA\Delta H,$$

where P is the chemical heat power in W, k the molar reaction rate in mol/m²·s, A the initial fuel rod surface area of the assembly in m² and ΔH the enthalpy of reaction of the oxidation reaction in J/mol. The negative sign is due to the fact that the heat released by the exothermic reaction is -ΔH.

The chemical heat power limited by water inflow rate is calculated from the water inflow rate and assuming that all the water reacts. The following formula can be used to obtain the chemical heat production rate:

$$P = -\frac{Q_w \rho_w \Delta H}{Mv},$$

where Q_w is the water flow rate in cm³/s, ρ_w is the density of water in g/cm³, M the molar mass of water in g/mol and v the stoichiometric number of water molecules consumed by a mole of uranium in the oxidation reaction. With a water flow rate of 4 cm³/a or 1.27·10⁻⁷ cm³/s, the above formula leads to a maximum chemical heat production rate of 2 mW, which is insignificant. The water flow rate is discussed at the end of Section 3.3.2.

In Table 3.3-3, the chemical heat powers at different temperatures and the decay heat in a single assembly of both metallic fuel types are shown. The maximum, very pessimistic, heat generation rate from both fuel types is also calculated. In this calculation, continuous unlimited availability of water is assumed. At higher temperature, this is most probably not the case, as discussed in the following sections, as an initial penetrating defect is quite small and water inflow is limited by the defect size and water inflow rate. The chemical heat production rate limited by water inflow, also shown in Table 3.3-3, is insignificant even at the time of highest temperature in the fuel. At the time of other types of canister failure, the temperature has decreased so that even the chemical heat production rate limited by the molar reaction rate coefficient does not limit the number of assemblies in the canister considerably.

In Table 3.3-3, the maximum number of assemblies fitting the decay heat power limit of 1370 W per canister (based on the Loviisa canister) was also calculated based on the aforementioned very conservative chemical heat power data. Especially with the HBWR 1st charge fuel, the pessimistic estimate of highest temperature would limit the maximum number of assemblies to only 96 assemblies.

Table 3.3-3. The chemical heat production in the metallic fuels compared to decay heat estimated from Tables 3.3-1 and 3.3-2.

	HBWR 1st charge	JEEP I
Surface area (m ²)	0.709	0.379
Decay heat power (W)	0.0083	0.97
Chemical heat power at 60 °C (W)	0.83	0.44
Chemical heat power at 104 °C (W)	14.28	7.64
Water inflow-limited chemical heat power (W)	0.002	
Maximum decay and chemical heat power (W)	14.29	8.60
Maximum number of assemblies in canister	96	159

3.3.2 Hydraulic processes: saturation/desaturation

Low water inflow rates may limit the processes related to interaction of water with the metallic fuel, such as hydrogen production and chemical heat generation, but also radionuclide transfer from the canister. Water inflow is limited by the size of the defect in the canister copper overpack and the low hydraulic conductivity of the bentonite buffer. The TURVA2012 safety case of Posiva includes three different phenomena for canister breach: an initial penetrating defect, failure due to corrosion, and failure due to rock shear. In the rock shear event, the canister may be completely cut open. However, a rock shear event is only theorized to occur with higher probability during and after the end phases of the next ice age, which occurs earliest in several tens of thousands of years in the future. As changes from the standard design of the copper overpack should not be necessary, the data from literature regarding the behavior of the copper overpack can be used as is.

The TURVA2012 safety case of Posiva assumes in the reference case that 1 out of 4500 canisters has an initial penetrating defect (Posiva Oy, 2012b). The initial penetrating defect is assumed to be a hole of 1 mm in diameter (Posiva Oy, 2012b), which could be left undetected from non-destructive examination of the canister (Holmberg & Kuusela, 2011).

In TURVA2012, the initial penetrating defect is thought to provide resistance to radionuclide transport (Posiva Oy, 2012b). The transport resistance due to diffusive mass flow rate out of the canister is calculated based on the material that is assumed to fill the defect, which may be, for example, water, corrosion products or bentonite or a combination.

After corrosion failure of the canister, the canister is thought not to provide containment of the radioactivity (Posiva Oy, 2012).

The reference case of TURVA2012 assumes that it takes at least 1000 years for water to fill the canister through the initial penetrating defect and form a transport pathway from inside the canister to the buffer (Posiva Oy, 2012b). However, this is a very cautious assumption, as even the filling of the gap between the cast iron insert and copper overpack is initially estimated to take 3700 to 19000 years (Posiva Oy, 2013). This is based on an estimate in a scoping calculation for the water flow rate into the canister in TURVA2012 (Posiva Oy, 2013). The estimate is based on the following equation:

$$Q=2\pi aK\frac{P_h-P}{\rho_w g} ,$$

where Q is the volumetric water flow rate through the hole (m³/s), a the radius of the hole (m), K the hydraulic conductivity of saturated bentonite at the mouth of the hole (m/s), P the pressure within the

canister (Pa), P_h the hydrostatic pressure at repository depth (Pa), ρ_w the density of water (1000 kg/m³) and g the gravitational acceleration (9.81 m/s²). The hole diameter, as before, is assumed to be 1 mm, pressure within the canister as 100 kPa and hydrostatic pressure plus atmospheric pressure as 4.2 MPa. Three values are estimated for the hydraulic conductivity of saturated bentonite from the range of estimated values: $2 \cdot 10^{-14}$, $5 \cdot 10^{-14}$ and $1 \cdot 10^{-13}$ m/s. The above equation then gives three estimated flow rates of 0.2, 2.0 and 4.0 cm³/a.

3.3.3 Mechanical processes: Material volume changes

When uranium metal oxidizes, the volume of the material increases. The volume increase may have to be taken into account in the design of the canister insert, but at this stage the effects of this phenomenon are uncertain and has to be investigated later in more detail. For conservativeness of the canister amount estimate, the maximum volume requirement is calculated and used to estimate the number of canisters.

There are several possibilities how to treat the oxidation volume increase. One option is to include void space in the canister to account for the increase in material volume, which leads to a large void space initially. The effect of a large void space in the canister on radionuclide release is uncertain. Another is to minimize the void space so that after some expansion of the corroding metals the corrosion products would fill the void space to stop further ingress of water.

The effect of an initial large void space is uncertain, as well as the mechanical effects in the expansion of the corroding metals for a smaller void space. In the following, the largest material volume increase is estimated for canister amount calculations, as this gives an upper limit for required canisters. In reality, the void volume for uranium volume increase to take place is probably lower.

Delegard and Smith (2008) characterized a sludge formed from metallic uranium with XRD. The sludge was formed from metallic fuel from the Hanford N reactor which was stored in concrete pools in storage canisters. In the storage pool, the water is oxygenated so uranium oxidizes to a greater extent than in anoxic water in a disposal facility. The estimate of the phase composition is therefore conservative regarding volume change, as generally a more oxidized species has a larger molar volume than a less oxidized species.

They identified several phases from the sludge, which are listed in Table 3.3-4. For each phase, the Pilling-Bedworth ratio was calculated. The Pilling-Bedworth ratio is the ratio of the volume of the unit cell of oxide relative to the volume of the unit cell of the metal. The Pilling-Bedworth ratio is defined as

$$R_{P-B} = \frac{M_{ox}\rho_m}{nM_m\rho_{ox}}$$

where M_i are the molar masses of oxide and metal, respectively, ρ_i are the densities of the oxide and metal and n is the number of metal atoms in one molecule of the oxide.

Similar treatment can be performed with aluminum. Aluminum is assumed to form aluminum oxide, Al₂O₃. The material properties and the resulting Pilling-Bedworth ratio for aluminum are also given in Table 3.3-4.

Table 3.3-4. Volume change for oxidation of uranium to different oxidation products. *: Volume reduction relative to UO₂ (McEachern & Taylor, 1997), so not calculated, **: source: (Knovel, 2020), ***: source: (Webmineral, 2020).

Common name	Chemical formula	Molar mass (g/mol)	Number of uranium atoms in oxide	Density (g/cm ³)	Pilling-Bedworth ratio
Uranium metal	U	238	1	19.1	
Uranium trihydride	UH ₃	241	1	11.1 **	1.74
Uranium dioxide	UO ₂	270	1	10.97 **	1.98
Tetrauranium nonaoxide	U ₄ O ₉	1096	4		*
Triuranium heptoxide	U ₃ O ₇	826	3		*
Triuranium octaoxide	U ₃ O ₈	842	3	8.38 **	2.69
Schoepites	UO ₃ ·2.25H ₂ O	326.5	1	4.8 ***	5.46
Studtite	UO ₄ ·4H ₂ O	374	1	3.64 ***	8.25
Metastudtite	UO ₄ ·2H ₂ O	338	1	4.67 ***	5.81
Aluminum metal	Al	27		2.7 **	
Aluminum oxide	Al ₂ O ₃	102	2	3.99 **	1.28

Based on this data and the inventory data in section 2, the volume requirement for the oxidized fuel assemblies can be calculated. The total fuel rod volume is first calculated from the dimensions. In the case of a JEEP I assembly, which contains two rods, the volume of two cylinders is calculated, and one cylinder for the HBWR 1st charge. The uranium volume calculated from the uranium mass per rod. Assuming everything else in the rod is aluminum, the aluminum volume is calculated from the uranium and total volumes.

The oxidized volume requirement is calculated by multiplying the uranium and aluminum volumes with suitable Pilling-Bedworth ratios and summing them together. The most conservative Pilling-Bedworth ratio, from oxidation to studtite, is used for uranium. This is a conservative approach, as it is not probable that a single compound is produced on oxidation as evident from the XRD data in Delegard and Smith (Delegard & Smith, 2008). The oxidized diameter requirement calculation is shown in Table 3.3-5.

Table 3.3-5. Calculation of the oxidized volume requirement for the metallic fuels.

	JEEP I	HBWR 1st charge
Rod length (m)	2.4	2.8
Rod diameter (mm)	25	40
Fuel rod volume (m ³)	0.001178	0.003519
Uranium mass per rod (kg)	19	22
Uranium volume in a rod (m ³)	0.000995	0.001152
Aluminum volume in a rod (m ³)	0.000183	0.002367
Oxidized volume requirement of assembly (m ³)	0.016874	0.012526
Nominal assembly diameter (mm)	70	40
Oxidized assembly diameter requirement (mm)	94.6	75.5

3.3.4 Chemical processes

3.3.4.1 Evolution of redox conditions and migration of chemical species

A substantial amount of hydrogen is formed by reaction of water with the metallic uranium fuel. This may change the redox conditions in the failed canister towards more reducing conditions. In reducing conditions, solubilities are typically lower than in oxidizing conditions. In addition, the distribution coefficients, K_d , are typically larger in reducing conditions and radionuclides are better retained by different materials.

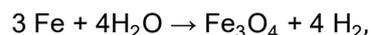
3.3.4.2 Alteration (including gas production)

The alteration of the waste package covers also gas production processes, such as hydrogen generation. On contact with water, the uranium metal reacts with water, or oxygen, if present. Water may come into contact with the fuel only through water carryover with the fuel during encapsulation, or inflow through a defect in the canister. The upper limit of water within a single canister is assumed as 600 grams (SKB, 2004). It is assumed in this work that the fuel is dried to an extent that satisfies this requirement.

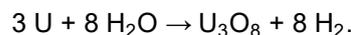
Oxygen above 5 ppm has a slight inhibiting effect on the reaction of water with uranium metal (Hilton, 2000), but in the repository conditions, anoxic conditions are present and the higher reaction rate is probable. Swift et al. (Swift et al., 2019) judged that gas generation is potentially limited by water inflow. Hydrogen gas production can be calculated from the corrosion rate, treated in more detail in section 3.4.1, or from the water inflow, depending on which is the more limiting factor. Carpenter et al. (2019) suggest that

“The impact of gas generation could initially be assessed by comparing the rate and total volume of gas generated by the wasteform with the rate and total volume generated by standard components of the GDF near field, for example standard waste containers.”

In a standard waste container, the most hydrogen generation occurs from the corrosion of the cast iron insert. Magnetite, Fe_3O_4 , has been found from archaeological studies to be the most common corrosion product of cast iron. The corrosion of iron to magnetite produces hydrogen gas through the following reaction:



In the case of uranium oxidation, oxidation to U_3O_8 is considered. The reaction proceeds as follows:



In the following, the potential amount of gas generated from a single KBS-3 canister cast iron insert is compared to the potential amount of gas generated from the metallic fuel proposed to be disposed of in a single canister. In Section 4, the maximum proposed number of metallic fuel assemblies in a canister is 66 HBWR 1st loading assemblies. This amounts to 1452 kg of uranium in a single canister. By comparison, the mass of iron in a BWR type canister is 10 600 kg (Raiko, 2012).

Table 3.3-6 shows this preliminary calculation, which results in that uranium generates possibly only 6.4 % of the hydrogen compared to possible generation from the cast iron insert. Even though a mole of uranium produces eight moles of hydrogen, double the amount produced by iron, the mass of uranium is lower and molar mass higher compared to iron.

Table 3.3-6. Comparison of hydrogen generation from standard BWR cast iron insert to that from metallic uranium in a single canister.

	kg	kmol
Amount of U	1452	6.1
Amount of Fe	10600	189.8
H ₂ generated from U	0.2	16.3
H ₂ generated from Fe	0.25	253.1
H ₂ (U)/H ₂ (Fe)	6.4 %	

As discussed in the section on hydraulic effects, the canister internal pressure has an effect on the water flow rate into the canister. Increase in the internal pressure of the canister decreases the rate of inflow of water through the defect. With the increased hydrogen production from metallic fuel, one could be lead to think that the canister internal pressure would rise faster than that of a canister containing commercial SNF. However, according to analyses by Bond et al. (1996), the canister internal pressure only rises above the canister outside pressure if the reaction between water and canister internal materials is slow enough that water remains unreacted at the time of canister pressure reaching the outside pressure. As the pressure difference is zero, water inflow stops and no additional water may come into contact with the metallic fuel (except through diffusion). Therefore the higher reaction rate between metallic uranium and water actually limits the impact of gas generation, as it is improbable that unreacted water remains within the canister when the pressure difference drops to zero.

However, as the reaction rate is highly dependent on temperature, this should be confirmed using more accurate decay heat data and estimates on canister breach. The low burnup of the metallic fuels and subsequent low decay heat leads to lower temperatures. However, in any case, the corrosion rate of the metallic fuel is higher than that of oxide fuel.

3.3.5 Radiological processes

3.3.5.1 Radiolysis

Due to the lower burnup and lower radioactivity of the fuel, radiolysis effects are smaller compared to commercial SNF. In addition, as hydrogen is possibly produced faster through reaction with metallic uranium than the cast iron insert, as in the standard KBS-3 concept, there is a larger amount of hydrogen present within the canister which has a limiting effect on radiolytic oxidation of the fuel (Ollila, 2011).

3.3.5.2 Helium production

The inventories and half-lives of alpha-active nuclides determine the helium production in the repository. Due to lower burnup in the metallic fuel, the radionuclide inventory is smaller and helium production rate is lower than in commercial SNF.

3.3.5.3 Criticality

The cast iron insert for commercial SNFs is, among other things, designed to minimize the possibility of criticality in the repository. The metallic NND waste contains primarily natural uranium and only small amounts of fissile actinides are produced due to low burnups. Criticality with ground water is not possible with only natural uranium, so criticality is not an issue for the metallic fuel types.

The TURVA2012 safety case incorrectly assumes that the worst case scenario for criticality safety for enriched oxide fuel is the hypothetical scenario where spent fuel material is homogeneously dispersed with the groundwater leaked into the canister (Posiva Oy, 2012). However, this is not the case, as separating fuel and moderator reduces resonance absorption, which significantly increases reactivity. Heterogeneous geometries where uranium and water are separated and in a suitable geometry would be

the most limiting case, which can occur, for example, if the fuel concentrates in a certain part of the disposal canister.

The oxide fuels in NND spent fuel contain also uranium with a higher than usual enrichment, up to 20 %. For these fuels, detailed criticality safety analyses are required to ensure subcriticality in the repository. Subcriticality is typically defined cautiously with a k_{eff} lower than 0.95 (Posiva Oy, 2012).

3.4 Contaminant release from waste form

3.4.1 Liquid-mediated release

The dissolution rate of metallic uranium is orders of magnitude higher than that of ceramic uranium oxide fuel.

The reaction rate of uranium with water is dependent on the oxygen content of the water (Hilton, 2000). Above 5 ppm concentration of oxygen, oxygen has a slight inhibiting effect on the reaction, while in anoxic conditions the oxidation reaction is faster. In a geological disposal facility the water is assumed to be anoxic, so the anoxic reaction rate is used in the following.

Hilton reports the following corrosion rate in water for anoxic conditions:

$$k = 5.03 \cdot 10^9 \cdot e^{\frac{E_a}{RT}}$$

The rate is based on data at a temperature range of 20 to 300 °C, and the rate is reported in units of $\text{mg}\cdot\text{cm}^{-2}\cdot\text{h}^{-1}$. E_a is the activation energy, reported as $-66.4 \text{ kJ}\cdot\text{mol}^{-1}$. In original source, the corrosion rate pre-exponential factor A is $5.03\cdot 10^9 \text{ mg}\cdot\text{cm}^{-2}\cdot\text{h}^{-1}$, which in molar units yields $2.11\cdot 10^8 \text{ mol}\cdot\text{m}^{-2}\cdot\text{h}^{-1}$. R is the gas constant and T the absolute temperature in Kelvin scale.

As the reaction rate increases with increasing temperature, the most conservative reaction rate is obtained with the highest temperature. Ikonen et al. (2018) report a highest temperature for a homogenized cast iron insert (homogenized from cast iron insert and fuel assembly) of 104.3 °C. In Table 3.4-1, the corrosion rate at this temperature is compared to value at 60 °C both to a value calculated in the same manner and to a value reported by RWM (2016).

Table 3.4-1. Corrosion rate calculated by the anoxic corrosion rate by Hilton (Hilton, 2000) compared to the estimate from RWM (RWM, 2016).

Corrosion rate					
T (°C)	$\text{mg}\cdot\text{cm}^{-2}\cdot\text{h}^{-1}$	$\text{mol}\cdot\text{m}^{-2}\cdot\text{s}^{-1}$	Chemical production rate ($\text{W}\cdot\text{m}^{-2}$)	heat rate	Comment
60	0.1	$1.00\cdot 10^{-6}$	0.5		From (RWM, 2016)
60	0.19	$2.27\cdot 10^{-6}$	1.17		Calculated
104.9	3.36	$3.93\cdot 10^{-5}$	20.15		Calculated

The instant release fraction (IRF) is the fraction of radionuclides released relatively instantly from the fuel on contact with groundwater, and typically is thought to consist of two inventories: the gas gap inventory and the grain boundary inventory (Johnson et al., 2012). The rest of the radionuclide inventory is contained in the uranium oxide matrix, which dissolves much more slowly. The IRF is considered in TURVA2012 as the release of the labile fraction.

However, in the case of metallic fuel, which dissolves much more rapidly compared to oxide fuel, this distinction may not be as useful. In any case, a pessimistic estimate for the instant release fraction should be assumed. Analyses performed for UK RWM have assumed an instant release fraction of 100 % for metallic fuel (Karney & Thetford, 2011; White et al., 2013).

Galvanic corrosion between cast iron and copper is not thought to be a significant process in the disposal facility (Posiva Oy, 2012). However, the potential difference between cast iron and uranium metal is larger than that between cast iron and copper. This issue might have to be investigated, but it is probably not an issue.

3.4.2 Gas-mediated release

When enough gas is generated in the canister, the gas phase may separate and rise to the top of the canister (Posiva, 2012). The radionuclides that may occur in gaseous form, for example ^{14}C , can then partition into the gas phase. If there is an overpressure of gas in the canister, the gas can flow out of the canister into the bentonite buffer. In this case the associated radionuclides are released.

Gas migration through the bentonite has been studied for long, and experimental studies have shown that gas passes through the bentonite only if the gas pressure exceeds slightly the hydrostatic pressure and the swelling pressure of bentonite (Vieno & Nordman, 1999; Nykyri et al., 2008). The required overpressure is reported to be 1 MPa by Vieno & Nordman (1999), that is, 1 MPa over the bentonite swelling pressure and hydrostatic pressure at repository depth. After an initial breakthrough of gas, the breakthrough pressure has been found to be lower than in undisturbed bentonite (Vieno & Nordman, 1999).

As was discussed in relation to gas generation, a high reaction rate of water with uranium may ensure that hydrogen is continually generated in the canister. When the canister pressure reaches the pressure of the surroundings and inflow of water is stopped, no residual water is present in the canister. In this case, significant overpressure does not occur within the canister. However, the reaction rate is highly dependent on the temperature, and the later water inflow into the canister starts, the more likely it is that reaction rate is low. In this case liquid water may be present in the canister as pressure within the canister equalizes with the surroundings. In this case the generated hydrogen can push gas or groundwater out of the canister.

4 Applicability of as-fabricated KBS-3 canisters

4.1 Dimensions of as-fabricated canisters

In order to benefit from the established manufacturing process of the standard KBS-3 canisters, it is best to keep the canister design as close as possible to the existing design. Especially the outer diameter and thickness of the copper overpack should be the same. The standard design is shown in Figure 4.1-1. However, in order to accommodate the various fuel types encountered in NND spent fuel, the cast iron insert and canister length need to be redesigned. In the following, options with the standard SKB/Posiva insert designs and custom NND insert designs are compared.

The inner length of the Loviisa canisters is the shortest of Posiva designs at 3245 mm, only approximately 400 mm longer than the JEEP I and HBWR assemblies at 2800 mm. Some of the HBWR 2nd to 4th loading fuels are 3660 mm long, and would only fit in BWR type canisters with an inner length of 4450 mm.

The Loviisa VVER-440 fuel assemblies are hexagonal, and insert openings are round. The insert has an outer diameter of 949 mm. The inner width of the insert openings is 173.7 mm (Raiko, 2012), whereas the corner-to-corner diameter of the fuel assembly is 169.7 mm¹. The diametral gap between the cast iron insert and the fuel assembly is thus 4 mm.

In the Loviisa insert, the insert openings are spaced 36.2 mm apart, of which 20 mm is the profile on the surface of the openings (profile thickness is 10 mm) and 16.2 mm the actual cast iron insert. The opening spacing has been selected for both strength and subcriticality (Raiko, 2012). Neck thickness (the closest approach of insert opening and insert surface) is 45.6 mm. Neck thickness value has been selected for strength of the insert (Raiko, 2012).

For the Olkiluoto 1-2 insert, the insert openings are rectangular with rounded corners. The outer diameter is the same, and the inner width of the insert openings is 160 mm. In both the Loviisa and Olkiluoto 1,2 inserts the number of openings is 12.

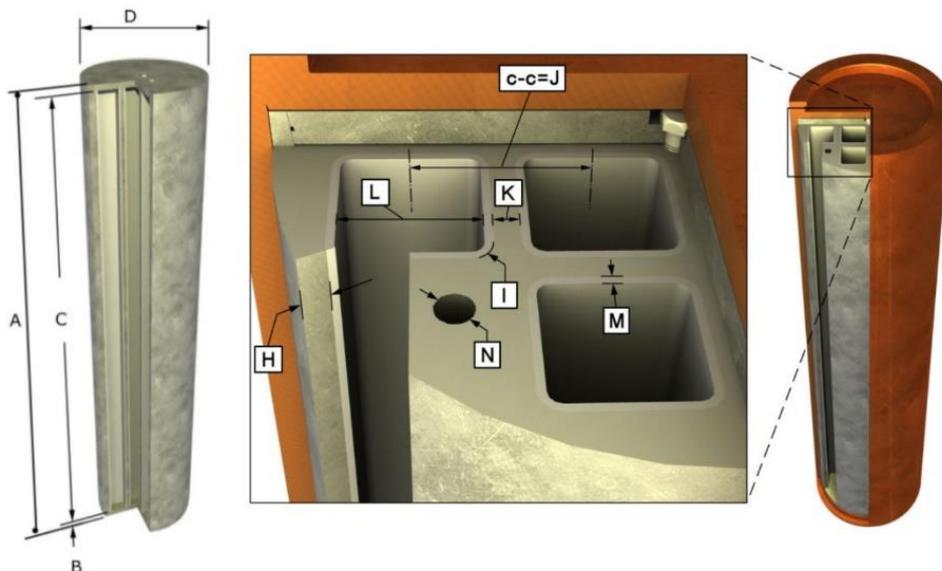


Figure 4.1-1. The different dimensions of the canister insert. C: inner height, H: neck thickness, K: insert spacing, L: inner width of opening, M: profile thickness. Figure from Raiko et al. (2010).

¹ Assembly pitch of 147 mm (Kuopanportti & Lahtinen, 2018) was used as the side-to-side diameter, which multiplied by $\frac{2}{\sqrt{3}}$, the ratio of the circumradius to the inradius of the hexagon, yields 169.7mm.

Another issue to be accounted for is the volume increase on insert oxidation. Empty space may have to be left in the canister for this volume increase. This is taken into account with data from section 3.3.3. However, this is a conservative assumption, as the effect of the material volume increase should be studied in more detail. Due to this conservatism, the actual number of canisters required may be lower than the values reported below.

Most rods from HBWR 5th loading have been dismantled from their assemblies, and the experimental fuel rods are typically single fuel rods. A support structure should be designed for these fuel rods so that they could be treated as assembly-like units containing several rods per unit. In the following, a support structure holding 7 fuel rods with a diameter of 70 mm is assumed.

There are seven different combinations of assembly diameter and assembly height, as shown in Table 4.1-1. However, as the initial data does not separate between the number of HBWR 2nd loading fuel (with an assembly height of 2830 mm) and 3rd and 4th loading (with an assembly height of 3660 mm), these are grouped together and the larger height used.

*Table 4.1-1. Fuel data for canister insert design. *: initial data does not specify the amount of each length of fuel assembly, so all fuels of this type are treated as having the larger length of 3660 mm.*

Fuel group	Fuel type	Assembly or unit diameter (mm)	Assembly or unit height (mm)	Number of assemblies or units
1	JEEP I, metallic	70	2800	85
2	HBWR 1st loading, metallic	40	2800	300
3	JEEP II	90	1500	136
4	HBWR 2nd to 4th loading	70	2830, 3660 *	100
5	HBWR 5th loading	70	1100	643
6	HBWR booster/experimental	70	1200	286

4.2 Canister dimensioning

4.2.1 Insert with standard openings

Using an insert with standard openings may be either difficult or costly, although little engineering is required to manufacture these canisters for NND spent fuel. The insert openings are larger than the NND spent fuel assemblies, so they fit into the standard canister insert openings. Two options are possible: either one assembly is inserted into a single opening or several assemblies are to be inserted in a single opening. In the case where only one NND spent fuel assembly is inserted in one opening, the number of canisters becomes very large, as is shown in the following.

Another option to decrease the number of canisters is to separate the canister into two sections as shown in figure 4.2-1 as the "layered" type, so that effectively two assemblies fit into one opening. In this case, two separate sections of the cast iron insert could be fabricated. The insert bottom is closed, which separates the two sections of the canister. The upper insert bottom would have to hold the weight of the assemblies in the upper section of the canister. Each of these options are investigated in the following.

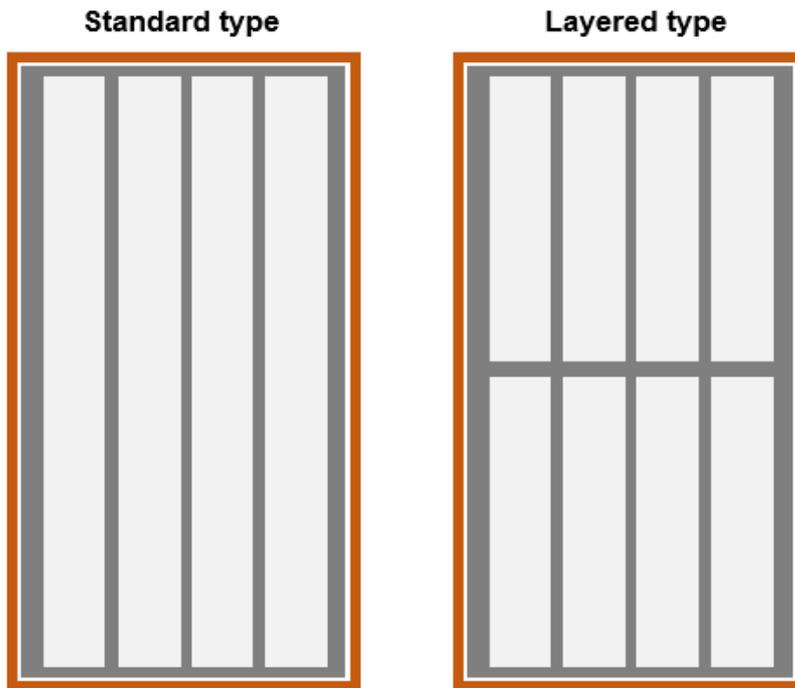


Figure 4.2-1. Standard and “layered” type canisters, schematic view from the side.

Inserting several assemblies as is in a single opening has uncertainties in the encapsulation phase, as there is no support structure keeping each assembly in place in the opening. In the other hand, a custom support structure could be designed to hold several assemblies in place.

In the options A and B of the following analysis, a single assembly is inserted in a single opening. In option B, the “layered” canisters are used for shorter fuel assemblies. In option C multiple assemblies are placed in each opening of the as-fabricated cast iron inserts, and “layered” canisters are used for shorter fuel assemblies.

The Loviisa 1-2 and Olkiluoto 1-2 cast iron inserts contain both 12 openings for fuel assemblies, but the Olkiluoto canister is longer. For all but the longer HBWR 2nd to 4th loading fuel the Loviisa canister is of sufficient length. Some dimensions of the canister and the number of assemblies in both types of canisters are shown in Table 4.2-1.

Table 4.2-1. Parameters of the Loviisa 1-2 and Olkiluoto 1-2 canisters.

Canister type	Assemblies	Inner height (mm)	Outer height (mm)	Outer diameter (mm)
LO1-2	12	3245	3552	1050
OL1-2	12	4450	4752	1050

4.2.1.1 Option A

Option A is to use only as-fabricated canisters and inserts. In this case, the maximum number of assemblies in a single canister is 12. This results in a relatively large number of canisters required for disposal. However, in this case the volume expansion due to uranium oxidation is less of an issue, as there is more volume available within the canister. Table 4.2-2 shows the calculated required number of canisters for this disposal option.

Table 4.2-2. Required number of canisters for option A.

Fuel group	Canister type	Number of assemblies or units	Required number of canisters
1, 2, 3, 5, 6	LO1,2	1450	121
4	OL1,2	100	8
all		1550	129

4.2.1.2 Option B

In option A, much of the length of the canister, especially in the case of HBWR booster, HBWR experimental and JEEP II fuel is empty. If a separator plate is used so that the space within the canister is divided into two length sections, the number of canisters can be decreased substantially. Table 4.2-3 shows the required number of canisters in the case where HBWR booster and experimental fuel, HBWR 5th loading as well as JEEP II fuel is stored in layered canisters.

Table 4.2-3. Required number of canisters for option B.

Fuel group	Canister type	Number of assemblies or units	Required number of canisters
1, 2	LO1/2	385	32
3, 5, 6	LO1/2 layered	1065	44
4	OL1/2	100	8
all		1550	85

4.2.1.3 Option C

In option C, several assemblies are placed in each insert opening. As discussed above, this may result in technical difficulty at the encapsulation stage. However, this dramatically decreases the required number of canisters.

The Olkiluoto 1-2 canister has rectangular openings, which fit more assemblies per opening, so they are used in this option, apart from the JEEP II spent fuel. The JEEP II spent fuel has a diameter of 90mm, and only one assembly fits in the insert opening.

The required number of canisters for this case are shown in Table 4.2-4.

Table 4.2-4. Required number of canisters for option C.

Fuel group	Canister type	Number of assemblies or units	Number of assemblies per opening	Maximum number of assemblies in canister	Required number of canisters
1, 4	OL1-2	185	4	48	4
2	OL1-2	300	9	108	3
3	LO1-2	136	1	12	11
5, 6	OL1-2 layered	929	4	96	10
all		1550			28

4.2.1.4 Oxidation volume increase in options A, B and C

The cast iron insert design is set in options A through C. However, the void space in each canister is adjusted to accommodate volume increase due to oxidation due to the uncertainties related to this effect in order to yield a conservative number of canisters. As the canister inner length is longer than that of the spent fuel and the openings are quite large, the internal volume is enough to accommodate this volume increase. However, the excess void space may also have negative effects, such as allowing more water flow within the canister.

In Table 4.2-5, the ratios of the oxidized metallic fuel volume to the total volume of the insert opening are compared in options A through C for both metallic fuels. The volume increase can be accommodated in all options. However, as can be seen from the table, the packing efficiency is quite low for options A and B, although somewhat better for option C.

Table 4.2-5. Ratios of the oxidized metallic fuel volume to the total volume of the insert opening.

Fuel	Option		
	A	B	C
JEEP I	0.219	0.219	0.550
HBWR 1st charge	0.163	0.163	0.409

4.2.1.5 Decay heat power limit

A maximum of 108 assemblies are placed in a single canister in options A through C. Therefore, the maximum number of assemblies taking into account the decay heat power limit, shown in Table 3.3-2, is not exceeded.

In option C, 108 HBWR 1st charge assemblies are placed in a single canister. This exceeds the limit placed by the pessimistic decay and chemical heat estimate of 84 assemblies shown in Table 3.3-3. However, as the assumptions used in the calculation of the chemical heat power are overly pessimistic, more detailed calculations would most probably show that this amount of assemblies can be fitted in a single canister.

4.2.2 Insert with custom openings

For the custom insert design, the Loviisa 1-2 canister design as reported by Raiko (Raiko, 2012) is used as the reference design. The insert openings in the Loviisa canister design are also round, and the NND spent fuel assemblies have a round shape. Some design values of the Loviisa cast iron insert are used as minimum values for the custom design, which should ensure the strength of the canister as well as subcriticality.

For the JEEP I, JEEP II and HBWR 1st charge fuels this preliminary dimensioning is sufficient, as the JEEP I and HBWR 1st charge fuel is natural uranium and the JEEP II fuel has an enrichment typical of commercial SNF. The rather low burnup of the JEEP II fuel calls for a closer look at criticality. In addition, due to the higher enrichments encountered in some of the HBWR 2nd to 4th charge fuels, HBWR booster fuel and the experimental fuels, detailed criticality safety analyses are required to ensure subcriticality in the canister. The high burnups of these fuels indicate that criticality should not be a serious problem, apart from fuels with the highest enrichments.

The maximum number of openings in the insert can be determined by fitting as many circles as possible in the circular cross section of the insert. Figure 4.2-2 shows this procedure and related dimensions. To respect the neck thickness, the available area of the cross section for placing insert openings is smaller than the total area. Two neck thicknesses must be deducted from the outer diameter to obtain the diameter of the available area, which is found to be 857.8 mm. This diameter is referred to in the following as the effective insert diameter.

Circles taking into account the required void area for placement of the spent fuel in the opening and the cast iron insert solid materials between to insert openings then have to be placed within the available area. This is depicted in Figure 4.2-2, and the relevant parameters and their relationships are also shown in Table 4.2-6. The void area accounts for assembly diameter, oxidation volume increase reserve and gap between assembly and profile surface (2 mm), and in Figure 4.2-2 its radius is signified by parameter c . The solid materials account for the shortest distance between profiles (thickness or K in Figure 4.1-1) and the profile thickness (M in Figure 4.1-1). The distance between centerpoints of two such circles covers then two insert opening radii, two profile thicknesses and one shortest distance between profiles. In Figure 4.2-2, the parameter d signifies one half of the solid materials between two centerpoints: half the shortest distance between profiles and one profile thickness, or using notation from Figure 4.1-1, $d = M + K/2 = 18.1$ mm. Packing these circles in the most efficient manner preserves the shortest distance between profiles and profile thickness. These circles are referred to as placement circles in the following discussion.

Above, the effective insert diameter was calculated simply by deducting the neck thickness. If the placement circles were placed in this available area, the neck thickness would increase since the circles cover also solid material between insert openings: the profile thickness and shortest distance between profiles. The radius of the available area for placement of such circles must therefore be increased by taking into account the solid materials. However, according to Raiko (2012), the profile thickness is already taken into account by the neck thickness, so only the shortest distance between profiles must be taken into account here. Using notation from Figure 4.1-1, the radius a in Figure 4.2-2 is related to the dimensions of the cast iron insert as $a = D/2 - H + K/2$. The parameter a is then 437 mm, and the effective insert diameter, $2a$, is 874 mm. Parameter b is simply found as $D/2 - a = H - K/2$.

Table 4.2-6. Relationships between the parameters in the circle packing problem. *: Calculated based on dimensions of a VVER-440 assembly.

	Value from Raiko (2012) (mm)	Figure 4-1	Figure 4-3
Insert opening diameter		L	
Assembly diameter		Included in L	Included in c
Shortest distance between profiles	16.2	K	Included in d
Profile thickness	10	M	Included in d
Oxidation volume increase reserve		-	Included in c
Gap between profile and assembly	4*	Included in L	Included in c
Neck thickness	45.6	H	$b + d - M$
Insert outer radius	949	D	$a + b$

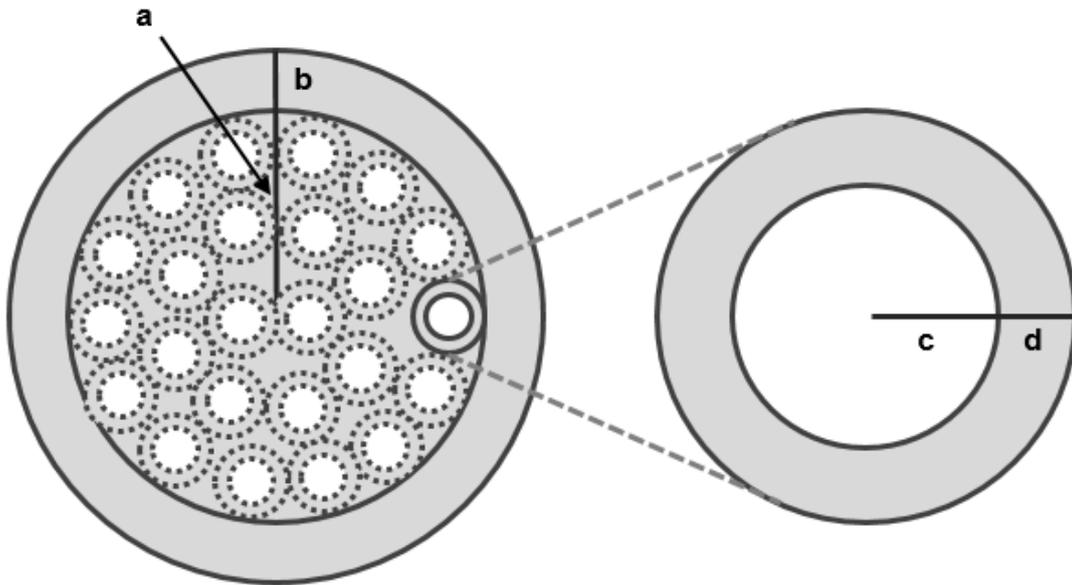


Figure 4.2-2. The parameters of the circle packing problem used to optimize the cast iron insert design. Grey color refers to solid material, and white color to void space. Circles with radius $c+d$ are fitted within a circle with radius a , and $a+b$ is equal to the radius of the cast iron insert.

The corresponding numbers of placement circles within the effective insert diameter are shown in Table 4.2-7, which yields the optimal number of openings within a single insert. The optimal number is obtained from the solutions of the circle packing problem collected by Specht (Specht, 2014).

There are three assembly diameters in the NND spent fuel, 40, 70 and 90 mm. However, the 40 mm assemblies (HBWR 1st loading) can be oxidized and may occupy a volume of a cylinder with diameter up to 75.5 mm. The JEEP I assembly with initial diameter of 70 mm, can increase in diameter up to 94.6 mm. Five different insert design options are preliminary envisioned due to there being five different possible assembly diameters. In the following, the canister type is specified by the combination of assembly diameter and height, e.g. 90/1500 for fuel group 3. The placement of the circles in each case shown in table 4.2-7 are presented in Appendix B.

The canister types holding metallic fuels are listed below with a range of assembly diameters due to the uncertainty related to the material volume change. For example, for the 40 mm assembly, the oxidized diameter is 75.5mm, so the canister type is listed as 40-76/2800. The smallest assembly diameter corresponds to the metallic fuel as is, and the highest assembly diameter to the diameter taking into account the highest material volume change. The actual value of canisters is probably in between these values, especially depending on possibility of preparatory encapsulation, as for example a stainless steel capsule over the fuel assembly would take up some space. For each of the options D through F a range of number of canisters is reported.

Table 4.2-7. Optimal number of openings of each size in the insert. *: Assembly diameter marked with asterisk corresponds to the diameter of the oxidized assembly calculated in section 3.3.3.

Assembly diameter (mm)	Placement diameter (mm)	Ratio of effective insert diameter to placement diameter	Optimal number of placement circles
94.6 *	134.8	6.48	32
90	130.2	6.71	35
75.5 *	115.7	7.55	44
70	110.2	7.93	49
40	80.2	10.89	96

4.2.2.1 Option D

Option D is to manufacture different height canisters for each of the groups in Table 4.1-1, and four different canister insert designs for each assembly diameter. This results in the number of canisters specified in Table 4.2-8.

Table 4.2-8. Required number and type of canisters for option D.

Fuel group	Canister type	Number of assemblies or units	Maximum number of assemblies in canister	Required number of canisters
1	70-95/2800	85	32-49	2-3
2	40-76/2800	300	44-96	4-7
3	90/1500	136	35	4
4	70/3660	100	49	3
5	70/1100	643	49	14
6	70/1200	286	49	6
all		1550		33-37

4.2.2.2 Option E

Option E is to place the HBWR 1st loading fuel in a canister with larger insert openings, such as 95/2800 canister used also for the JEEP I fuel.

Additionally, similar canister type can be used for storing HBWR 5th loading and HBWR booster and experimental fuels. Due to the large height of the canister, these fuel types could be stored in the canister in two layers. In this case, a separator plate should be inserted between the two layers. The required numbers of different types of canisters in this option are shown in Table 4.2-9.

Table 4.2-9. Required number and type of canisters for option E.

Fuel groups	Canister type	Number of assemblies or units	Maximum number of assemblies in canister	Required number of canisters
4	70/3660	100	49	3
1, 2	70-95/2800	385	32-49	8-13
5, 6	70/2800 layered	929	98	10
3	90/1500	136	35	4
all		1550		25-30

4.2.2.3 Option F

Option F is to use a single cast iron insert type for all fuel types. In this case the diameter must be the largest and the height of the capsule the largest, so the canister type is 95/3660. Additionally, the HBWR 5th loading, booster and experimental fuels are disposed in a layered canister. The number of canisters in this case are shown in Table 4.2-10.

Table 4.2-10. Required number and type of canisters for option F.

Fuel groups	Canister type	Number of assemblies or units	Maximum number assemblies in canister	Required number of canisters
1, 2, 3, 4	90-95/3660	621	32-35	18-20
5, 6	90-95/3660 layered	929	64-70	14-15
all		1550		32-35

4.2.2.4 Decay and chemical heat power limit

A maximum of 98 assemblies are placed in a single canister in options D through F. Therefore, the maximum number of assemblies taking into account the decay heat power limit, shown in Table 3.3-2, is not exceeded.

Taking into account the decay and pessimistic chemical heat limit, the maximum number of HBWR 1st charge assemblies in a canister is 96. However, due to conservatism in the chemical heat power calculation, this maximum number is most likely greater and should be confirmed by more detailed calculations.

Regarding JEEP I fuel, even the pessimistic chemical and decay heat power limits are also satisfied.

4.3 Additional engineering requirement

The proposed options differ in the amount of engineering required to either manufacture the canister or the placement of the spent fuel at the encapsulation plant. In Table 4.3-1, the options are compared by the required number of canisters and the level of engineering required by the option. The comment summarizes the specifics of each option.

Based on Table 4.3-1, options C, D, E or F are deemed to be optimal with regards to number of canisters, canister and encapsulation engineering requirement and encapsulation difficulty.

In standard canisters, little additional engineering is necessary. However, to decrease the amount of canisters, a layered canister is proposed, where the two cast iron insert sections are separated by the bottom of the upper cast iron insert. The two-part cast iron insert of course increases the difficulty of assembly at the encapsulation plant, but not as much as the placement of several assemblies in a single opening, for example. In option C, as multiple assemblies are placed in one insert opening, the difficulty in encapsulation is judged to be very high.

For both the custom and standard cast iron insert designs, all analyses performed for the standard designs with commercial SNF must be repeated. With more than one custom design, there is a large number of analyses (for example, mechanical and criticality safety) to be performed. The note above on layered canister applies to the custom designs as well.

Table 4.3-1. Comparison of different canister design options.

Option	Required number of canisters	Canister engineering requirement	Encapsulation difficulty or engineering requirement	Comment
A	129	Low	Low	Standard canister
B	85	Moderate	Moderate	Standard insert, layered canister
C	28	Moderate	Very high	Standard insert, layered canister, multiple assemblies in one opening
D	33-37	Very high	Low	Multiple custom insert designs
E	25-30	Very high	Moderate	Multiple custom insert designs, layered canister
F	32-35	High	Moderate	One custom insert design, layered canister

5 Applicability of additional barriers

5.1 Encapsulation of the waste form

Preparatory encapsulation of the waste form before placing the waste within the canister serves to provide an additional release barrier for radioactivity. A release barrier could counteract the instability of the metallic uranium. However, this release barrier can be efficient in practice only in the case of an initial canister defect or corrosion failure of the copper overpack. In a rock shear event, the whole canister can be cut open, which might also break the additional encapsulation of the waste form.

For example, stainless steel could be used as a corrosion-resistant encapsulation material, in which the metallic fuel assemblies would be placed. A stainless steel internal canister would have to withstand pressure from generated hydrogen, as the spent fuel material contains carryover water which reacts with the uranium over time. In this case, the outside pressure of the internal canister would be the internal pressure of the KBS-3 canister, which is lower than the KBS-3 canister outside pressure.

As the required canister amounts were estimated based on conservative material volume change, it is probable that the additional volume requirement for the internal stainless steel canister is covered by this conservative assumption.

5.2 Chemical or mechanical barriers within the canister

Chemical or mechanical barriers within the canister, e.g. filling the canister with some material with beneficial effects on radionuclide release or decrease of corrosion rates, could be inserted into the canister at the encapsulation plant. However, this requires changes at the standard encapsulation plant for the KBS-3 concept. Due to very small number of canisters also some extra operations could be feasible. The volume expansion of metallic uranium on contact with water might have to be accounted for in the amount of filler material used per canister.

6 Knowledge gaps and future work

As part of this work, knowledge gaps and future work were identified related to disposal of the NND spent fuel by the KBS-3 concept. Some of the knowledge gaps are more general, related to data most probably required by other disposal methods as well, and others more specific, regarding the KBS-3 concept specifically.

More detailed decay heat and activity inventory calculations should be performed for the NND spent fuel types. Thermal calculations have to be made to confirm that given temperature requirements are fulfilled. Simple analytical calculations can be made first and satisfy the needs of concept development. Later more detailed and comprehensive calculations are used for the safety case. Also the decay heat power limit applicable for any new canister design should be set. In the analyses performed in this report, the decay heat power limit of the Loviisa-type canister in the Posiva safety case was used as an estimate. Further analyses should be performed to confirm an applicably decay heat limit for new cast iron insert designs.

In order to accurately estimate the volume increase of the metallic fuel, the phases formed from oxidation of the fuel in relevant groundwater conditions should be well characterized. As the volume increase influences the number of assemblies that can fit within a single disposal canister, this may decrease the number of required canisters somewhat as the estimates in this report were based on conservative assumptions of the formed phases.

Mechanical analyses should be performed on the custom cast iron insert designs, if used, to confirm their mechanical strength. In addition, criticality safety analyses for these designs are most likely required especially for the fuel types with higher ^{235}U enrichment or low burnups.

As there is a significantly greater number of openings in the custom cast iron inserts, the applicability of non-destructive testing methods must be investigated. Each insert must be subjected to non-destructive testing, for example by eddy current and X-ray methods, before placement in the canister. A more complex geometry of the insert may complicate the use of such methods.

Dissolution rate of metallic uranium in relevant groundwater conditions should be investigated. Effect of the presence of cast iron and in the case of additional barriers, stainless steel or other materials should be investigated.

A holder for individual fuel rods should be designed. Single rods cannot be as simply treated at the encapsulation plant, so the encapsulation difficulty is greatly reduced by a holder which can easily be manipulated with the equipment present at the encapsulation plant. In the canister amount calculations of this work, the holder is assumed to have an outer diameter of 70 mm and hold 7 individual fuel rods each.

If an internal canister for additional encapsulation is chosen as an additional release barrier, the canister should be designed to withstand predicted pressure loads from hydrogen generation both from inside and outside and to be easily manipulated at the encapsulation plant. Damaged fuel increases the difficulty of its manipulation, so such an internal canister might be required for damaged fuel, metallic or not, as well.

The treatment of hydrogen generation in this report was quite simplistic, and serves only as an initial estimate. A more detailed assessment for the hydrogen generation as performed by Bond et al. (1997) should be performed with the inclusion of the effect of the water-uranium reaction on hydrogen generation and the time taken to increase canister internal pressure to the level of hydrostatic pressure and bentonite swelling pressure. Bond et al. report the complete model, which could be implemented with suitable properties of metallic uranium taken into account.

7 Summary and conclusions

In this report, the various aspects of disposing NND spent fuel types with the KBS-3 concept are presented. One of the most important aspects is the nature of most of the fuel, which is metallic uranium. Metallic uranium corrodes faster than the uranium oxide fuel being disposed of with the KBS-3 concept in Sweden and Finland.

There is regulatory uncertainty regarding the disposal of metallic uranium fuel. A conservative interpretation of IAEA Specific Safety Requirements prohibits disposal of metallic uranium fuel because of its chemical instability. A more liberal interpretation might allow disposal concepts to compensate for the absence of chemical stability in the waste form by adding other safety functions.

The main conclusion of this report is that, subject to approval in principle for the disposal of chemically unstable fuel, disposal of metallic uranium fuel is seems possible with the KBS-3 method, even though various issues remain to be resolved. The uncertainties related to decay heat and radionuclide amount should in the first place be resolved by detailed analyses. A conservative estimate for required number of KBS-3 canisters is also provided for various options, where 25 to 37 KBS-3 canisters are required to dispose of all the NND spent fuel. The geometry of the fuel assemblies is the most limiting factor in placing NND spent fuel in standard KBS-3 canisters.

The decay heat power and radionuclide inventory of the NND spent fuel was preliminarily estimated from literature data on commercial SNF. This estimate confirms that decay heat most likely does not limit the number of assemblies or rods placed in a KBS-3 canister. The preliminary estimate also confirms that the radionuclide inventory in the metallic fuels is very small, which in the larger picture counteracts the instability of the metallic uranium. Even pessimistically speaking, all the inventory from the metallic fuel could be released in a period of the order of tens of years, and not surpassing annual activity release limits that are in place in Finland for the final disposal facility.

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Appendix A. Decay heat power and activity inventory of commercial SNF

The data in the following tables has been collected from Anttila (2005). Void related to BWR fuel means the void fraction (fraction of gas bubbles) used in the calculation. Tables A-1 and A-2 show the decay heat power at different burnups at a cooling time of 30 and 50 years, respectively. Table A-3 shows the maximum activity inventory from any cooling time (from 0 to 1000000 years) for burnup of 40 MWd/kgU.

Table A-1. Decay heat power per uranium mass unit (W/kgU) for various Finnish fuel types at a cooling time of 30 years (Anttila (2005)).

Type	Enrichment (%)	Discharge burnup (MWd/kgU)		
		40	50	60
BWR (void 0%)	3.8	826.0	1050.0	1300.0
BWR (void 40%)	3.8	857.0	1100.0	1370.0
BWR (void 80%)	3.8	900.0	1170.0	1470.0
BWR (void 0%)	4.2	825.0	1050.0	1290.0
BWR (void 40%)	4.2	854.0	1100.0	1360.0
BWR (void 80%)	4.2	895.0	1160.0	1450.0
VVER-440 TVEL	4.2	927.0	1210.0	1520.0
EPR	3.6	929.0	1210.0	1530.0
EPR	4	924.0	1200.0	1520.0

Table A-2. Decay heat power per uranium mass (W/kgU) for various Finnish fuel types at a cooling time of 50 years (Anttila, 2005)

Type	Enrichment (%)	Discharge burnup		
		40 (MWd/kgU)	50	60
BWR (void 0%)	3.8	576.0	723.0	875.0
BWR (void 40%)	3.8	605.0	767.0	933.0
BWR (void 80%)	3.8	643.0	825.0	1010.0
BWR (void 0%)	4.2	574.0	722.0	874.0
BWR (void 40%)	4.2	603.0	765.0	931.0
BWR (void 80%)	4.2	639.0	820.0	1010.0
VVER-440 TVEL	4.2	676.0	870.0	1070.0
EPR	3.6	674.0	866.0	1070.0
EPR	4	671.0	863.0	1060.0

Table A-3. Maximum activity inventory per uranium mass for various Finnish fuel types with burnup of 40 MWd/kgU at any cooling time up to a million years.

Type	Enrichment (%)	Inventory (GBq/tU)		
		¹⁴ C	¹²⁹ I	¹³⁵ Cs
BWR (void 0%)	3.8	30.73	1.140	18.180
BWR (void 40%)	3.8	30.26	1.153	19.920
BWR (void 80%)	3.8	31.07	1.184	22.470
BWR (void 0%)	4.2	27.87	1.115	19.680
BWR (void 40%)	4.2	27.80	1.137	21.480
BWR (void 80%)	4.2			
VVER-440 TVEL	4.00	38.49	1.195	20.580
EPR	3.6	20.37	1.220	20.730
EPR	4.0	19.14	1.194	22.250
Maximum		38.49	1.220	22.470

Appendix B. Custom cast iron designs

The following figures show the optimal packings of each custom insert type as shown by Specht (2014). The coordinates of each circle can be obtained from Specht (2014).

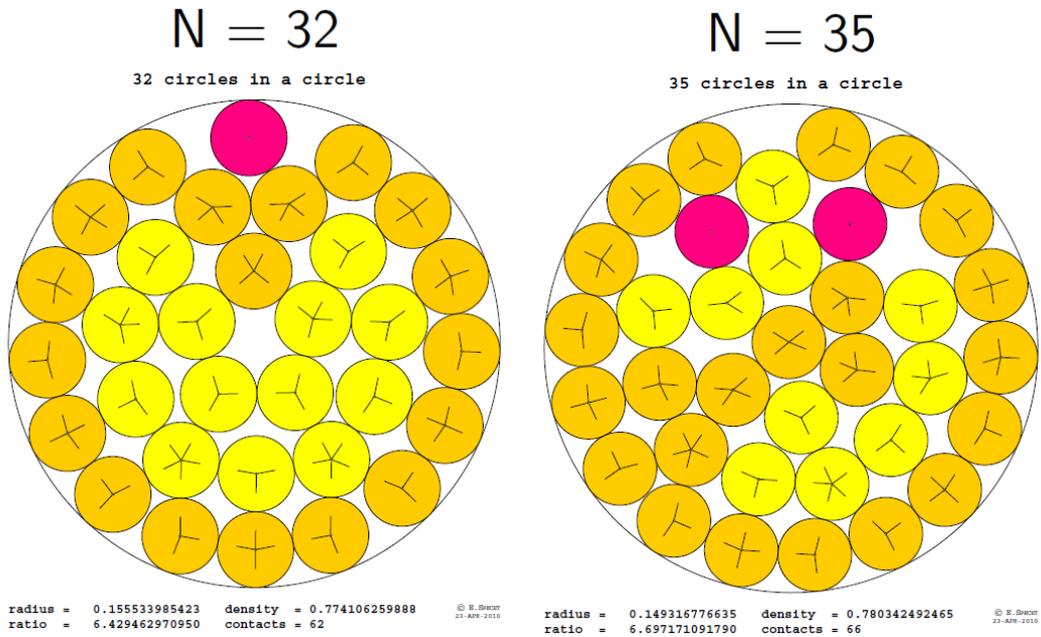


Figure A.1. Optimal packing of 32 and 35 circles. Figure from Specht (2014).

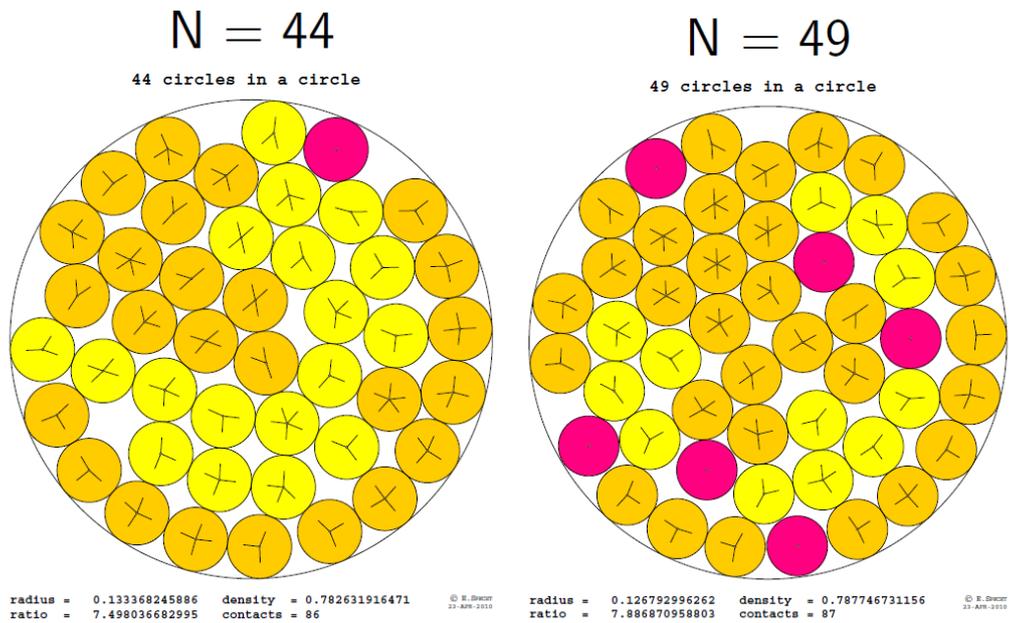


Figure A.2. Optimal packing of 44 and 49 circles. Figure from Specht (2014).

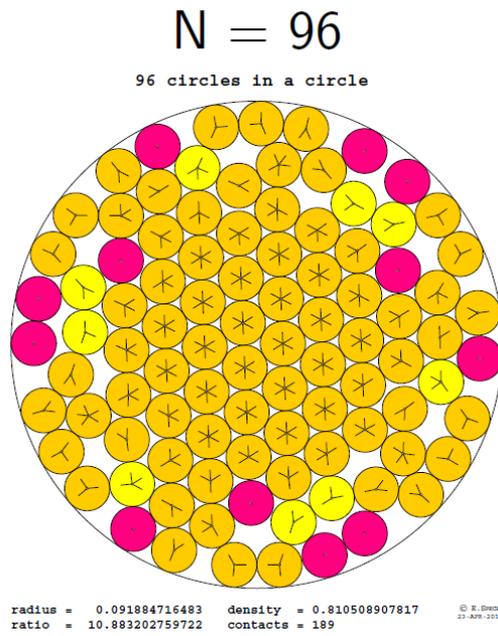


Figure A.3. Optimal packing of 96 circles. Figure from Specht (2014).



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