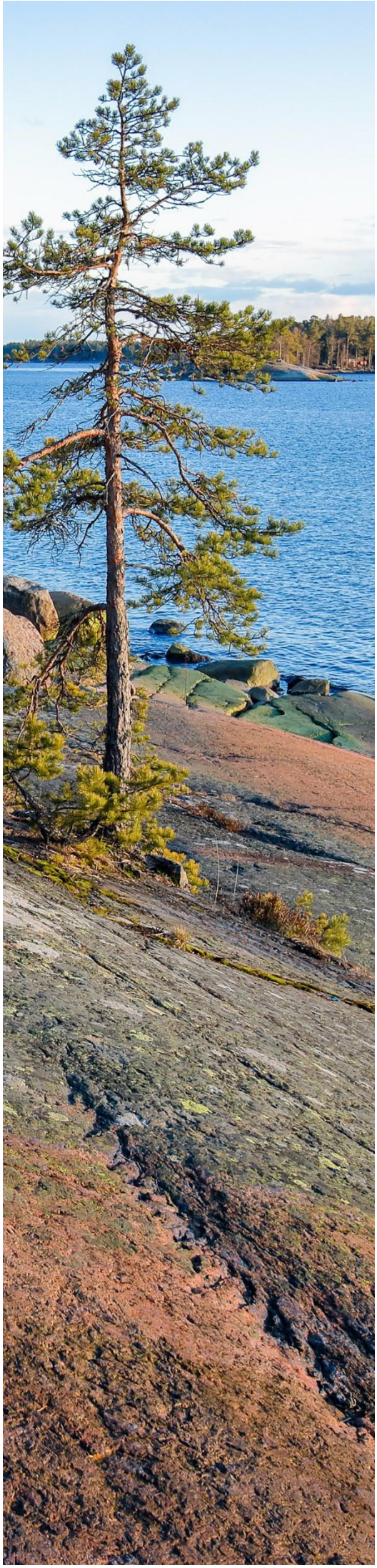

Technical report

Applicability of KBS-3 concept for vitrified waste disposal

Authors:

Pauli Juutilainen
Aimo Hautojärvi
Taina Karvonen
Suvi Karvonen

Espoo, November, 2020

AINS GROUP VTT

AINS - Civil Engineering
Tel. +358 207 911 888, www.ains.fi
VAT-number FI01135548

Project name and order number (AINS): Technical Assistance “National Facility” 872131

Report title: Applicability of KBS-3 concept for vitrified waste disposal

Document reference (AINS): ID 85402

Version: Final, 26.11.2020

Date: 26.11.2020

Additional information:

Author Juutilainen, P., VTT Hautojärvi, A., AINS Karvonen, T., AINS Karvonen, S., VTT	Approved by Ikonen, A., AINS	Accepted by the Client Bohmer, N., NND
Date 26.11.2020	date 26.11.2020	date 30.11.2020
Reviewed by Karvonen, S., VTT	Reviewed by	Reviewed by
Date 26.11.2020	date	date

Abstract

Norwegian Nuclear Decommissioning (NND, Norsk nukleær dekommisjonering) is considering spent fuel reprocessing as one of the options for the spent fuel management. In this option, vitrified reprocessing waste would have to be disposed of in the National Facility. In this study, the feasibility of the deposition of vitrified spent fuel waste in a KBS-3 type deep geological repository is considered. The standard copper containers and other engineered barriers of the KBS-3 concept are assumed to be used as such, but required changes in the internal structure of the canister (insert) are investigated and the safety of the system considered in comparison to standard KBS-3 concept. Knowledge gaps and future research needs relevant to this options are also investigated.

Keywords
Nuclear waste management, vitrified waste, KBS-3, spent fuel, radioactive waste, disposal, repository, deep geological repository, Norwegian National Facility, copper canister

AINS GROUP

TABLE OF CONTENTS

1 Introduction

1.1 Aim of the study

The Institute for Energy Technology (IFE, Institutt for energiteknikk) has operated several research reactors in Norway, all of which are now shut down. Norwegian Nuclear Decommissioning (NND, Norsk nukleær dekommisjonering) is government agency which within the next few years is intended to take over responsibility for the storage, handling and disposal of radioactive waste in Norway.

Spent fuel reprocessing is being considered as one of the options for the spent fuel management. In this scenario, the spent fuel would be shipped to the La Hague reprocessing plant in France. The applicable fissile material would be used for new mixed-oxide (MOX) fuel and sold in the market, but the vitrified reprocessing waste mostly consisting of the fission and activation products would be returned to Norway. Thereafter, NND would be responsible for the disposal of the returned vitrified waste.

The spent fuel originates from Norwegian reactors in Kjeller and Halden, no longer in operation. The reactors were used for research purposes. In 2018, IFE decided to shut down and decommission the HBWR, and in 2019, the same decision was made for the JEEP II reactor. JEEP I was in operation from 1951 to 1967. All three reactors have produced spent nuclear fuel (SNF) that must be disposed of safely. Part of this fuel is in metallic form, which has problematic qualities for final deposition, such as poor chemical stability. Therefore, NND is exploring different options for the treatment of this spent fuel, one of which is reprocessing, resulting in the vitrified waste that is the object of this report.

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 isolating and containing radioactive materials in a repository in hard rock. The SNF is placed into a canister composed of a mechanically strong cast iron insert and a corrosion resistant copper overpack. The canister is then surrounded by a bentonite buffer between the canister and the rock. In addition to these engineered barriers, the host rock acts as natural release barrier with 400 to 500 meters of rock above the disposal facility (Posiva Oy, 2020). InFigure 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.

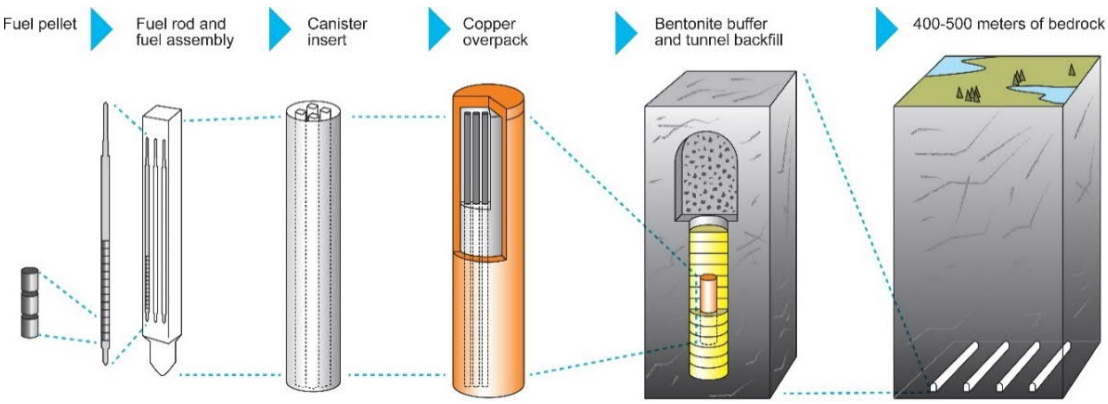

Figure 1.1-1 . Schematic representation of the KBS-3 final repository concept (Posiva Oy, 2020).

In this study, the feasibility of the deposition of vitrified spent fuel waste in a KBS-3 type deep geological repository is considered. The standard copper containers and other engineered barriers of the KBS-3 concept are assumed to be used as such, but required changes in the internal structure of the canister (insert) are investigated and the safety of the system considered in comparison to standard KBS-3 concept.

1.2 Vitrified waste management in a few countries

1.2.1 France

Most of the country's high-level waste has been or is anticipated to become vitrified. In addition to the power reactors, such waste destined to vitrification has arisen from various experimental facilities. Waste has also been vitrified at various reprocessing facilities over several decades, which has caused that the specification details of waste glass packages differ somewhat. The current concept is to emplace one or more Waste Vitrification Plant (WVP) canisters into a carbon steel container. The number of WVP canisters that can be accommodated in a single disposal canister depends on the length of each WVP canister, since multiple canisters may be positioned axially with respect to each other. (Andra, 2016a)

The disposal canister is further separated from the surrounding rock by a microtunnel casing, with the current reference material being another type of carbon steel. The details of the material compositions for the package and the casing are to be defined. The canister is separated from the casing with the help of ceramic pads. Such an arrangement has been designed to meet the reversibility requirement set by the French legislation. (Andra 2016a, Andra 2016b)

The repository is being prepared into Callovo-Oxfordian clay rock. However, granite host rock has also been studied in the past as a potential repository environment, a report on which was published in 2005. No specific site was analysed in the study, but the generic features of granite as a repository host rock was investigated and the work somewhat extensively relied on the granite studies elsewhere, such as in the Nordic countries, Switzerland and Canada. Particularly related to the disposal package concept, Andra concluded to propose a thick steel container for vitrified waste packages, but follow the KBS-3 concept for spent fuel assemblies, in case any assemblies end up being routed to final disposal (Andra, 2005).

1.2.2 Switzerland

Part of the spent fuel generated in the nuclear power reactors of Switzerland has been reprocessed, whilst some of the spent fuel assemblies are to be disposed of as such. Therefore, final disposal solutions are required for both vitrified waste and spent fuel assemblies. The final disposal site will be selected among three final candidates. However, the Opalinus Clay has already been selected for the host rock type. Zürcher Weinland is one of the candidates, for which an extensive feasibility analyses have been performed (Nagra, 2002). As far as the vitrified waste is considered, each stainless steel canister containing the waste would be packed into a carbon steel canister which is surrounded by bentonite backfill. The concept thus deviates from the Nordic KBS-3 concept with respect to both container material and host bedrock. However, whilst the carbon steel is the current reference design, the final decision on the overpack material is still to be done and e.g. a copper overpack canister may still be selected (Leupin et al., 2016).

The suitability of a crystalline basement as the final disposal environment has been studied in the Kristallin-I project, the synthesis of which was published as early as in 1994 (Nagra, 1994). At the time of these studies, it was assumed that high-level waste would exist only in the form of vitrified waste. The engineered barrier system was the same as it was later for a repository in clay host rock. The report concluded that the concept was promising as a whole, but recognized that several details would have to be further investigated. Considering the

engineered barriers, any future findings were seen as more like opportunities to reduce unnecessary conservatism rather than proving the concept unsuitable.

1.2.3 UK

The fuel irradiated at Magnox reactors has been or will for the most part bereprocessed, in addition to which reprocessing strategy has been applied to a considerable amount of the nation's Advanced Gas-cooled Reactor (AGR) fuel. Thus, vitrified waste is waiting for final disposal into a deep repository,. The selection process for the deep repository site is going on, so the properties of the host rock are naturally not known yet. The repository can be constructed into a high-strength rock, such as granite, lower-strength sedimentary rock or evaporates - or into a combination of these. A copper canister with cast iron insert, i.e. a canister closely similar to that in KBS-3V concept has been used as an illustrative model for analyses of disposal into high-strength rock. Also the vitrified waste has been drafted to be contained in this kind of a canister (RWM, 2017). For example, such a container has been applied as a model configuration in the latest criticality safety assessment for the UK spent fuel, even though the vitrified waste is not considered in the report (Hanlon et al., 2020).

1.2.4. Sweden

Sweden is currently not looking to deposit vitrified waste in its repositories, but this option was investigated in 1970's. These findings can be of some interest if NND chooses the vitrified waste option, since they deal with the KBS-1 design that precedes the current KBS-3 design and the environmental conditions (crystalline rock) are the same as in Norwegian case. The report series (SKB, 1977) expects that the safety assessment is favourable and dose limits would not be exceeded. However, as the study is over 40 years old, the science and technology of the past few decades have made it somewhat outdated and its results cannot be directly converted to modern day safety argumentation.

1.3 Relevant regulations

1.3.1 Norwegian regulations

Norwegian law does not currently provide specific regulations regarding disposal of radioactive waste, nor does it differentiate between spent fuel and vitrified reprocessing waste. A brief explanation of Norwegian regulations was given in the Loukusa et al. (2020) report and the most important findings are quoted also here:

> *"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."*

and

> *"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."*

These requirements are not seen to provide any special requirements for the reprocessed waste deposition. In general, the Norwegian legislation can be expected to develop in detail over the coming years.

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 engineered barriers, including the waste form and packaging, shall be designed, and the host environment shall be selected, so as to provide containment of the radionuclides associated with the waste. Containment shall be provided until radioactive decay has significantly reduced the hazard posed by the waste. In addition, in the case of heat generating waste, containment shall be provided while the waste is still producing heat energy in amounts that could adversely affect the performance of the disposal system."

It is noted in the SSR-5 document that containment is most important for highly concentrated radioactive waste, such as vitrified waste from fuel reprocessing. Such 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. This containment capability of the waste package has to be demonstrated by means of safety assessment to be appropriate for the waste type and the overall disposal system.

The SSR-5 document also lists more detailed requirements such as the requirement to demonstrate that possible releases of small amounts of gaseous radionuclides or other highly mobile nuclides have to be demonstrated to be acceptable.

In comparison to Feasibility of KBS-3 spent fuel disposal concept for Norwegian spent fuel (Loukusa et al., 2020) where the chemical stability of the metallic waste form was problematic, in the case of vitrified waste this issue is of less concern. The vitrification process encapsulates the final waste residues in a glass matrix with stable chemical properties, which helps to achieve the very low release rates of radionuclides from the vitrified materials over long time periods. Thus, when it comes to following international regulations, the vitrified waste form studied in this report is not seen to conflict with any specific requirements or require instant in-depth analyses of longevity issues.

2 Waste inventory

, The reprocessing waste from spent fuel reprocessing at the La Hague site of Orano is vitrified into borosilicate glass and returned to the customer in the standard canisters. The waste inventory in the glass is composed of the generic output resulting from the vitrification process whose input consists of all spent fuel shipments to the reprocessing. That is, the returned nuclides are not necessarily the same as those sent to the reprocessing, but the amount of returned waste is determined such that it is radiologically equivalent to the amount of spent fuel sent for reprocessing. The reprocessing organisation guarantees the limits not to be exceeded for the radiologically important parameters for each individual waste container. Such parameters for return containers from Orano are listed in Table 2-1. The Orano canister is called Universal Canister (UC) and in case of vitrified waste it is denoted as UC-V.

The vitrified waste contains fission and activation products, in addition to which trace amounts of uranium and plutonium exist. Some volatile active components, however, are partly released from the waste flow in the reprocessing and do not completely end up in the vitrified waste. Notable nuclides belonging to this group are C-14 and I-129 that have been identified to be among the nuclides causing the highest dose risk because they could migrate from a deep repository out to the biosphere without significant retention in the EBS system or geosphere. It has proved to be a somewhat challenging task to determine the nuclide-wise contents of an individual canister, which is rather extensively analysed in a German report (Meleshyn, 2012).

The total weight of a fully-packed canister is 500 kg, of which the canister itself weighs 100 kg and the waste matrix 400 kg. The canister is made of stainless steel with 5 mm thick wall. The outer dimensions of the canister are:

- Diameter: 440 mm
- Length: 1345 mm
- Volume: 0.183 m^3

It can be noted that there are some discrepancies in the reported dimensions of the canisters produced at La Hague. Nagra and OPERA apply values of 430 mm for the diameter and 1335 mm for length (Patel et al., 2012; Verhoef et al, 2016). The same dimensions are presented also in a CEA monograph (CEA, 2009) on nuclear waste conditioning, from which **Error! Reference source not found.** depicting the waste glass canister has been taken.

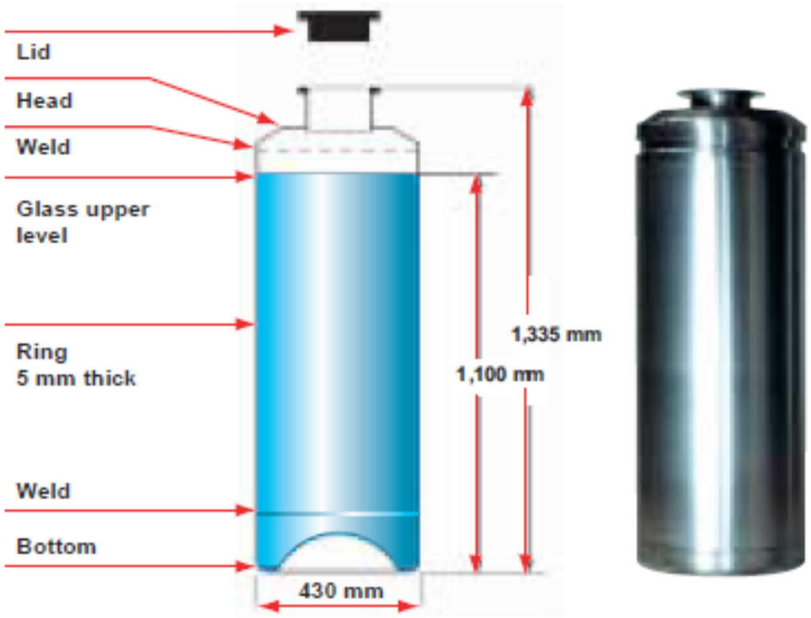

Figure 2-1.. The vitrified waste canister packed at Orano La Hague site. Figure from (CEA, 2009)

At the moment of producing the waste package, the decay heat power of a single canister may be up to 4230 W, with the average at 2950 W. After a 40-year cooling period the average heat power decreases below 600 W (McGinnes, 2002). The minimum storing time at the reprocessing facility is five years, during which the heat power decreases to approximately 1500 W (Meleshyn & Noseck, 2012).

The gamma and neutron radiation dose rates outside the canister are so high that remote handling is required. In a Nagra report (Patel et al., 2012), a 140 mm thick cast iron overpack has been calculated to bring the surface dose rate below 1 Sv/h. At the moment of production, the dose rate on the glass canister surface can be up to 3530 Sv/h. The dose rate naturally decreases over the interim storage period before final disposal, but it is still high enough to require remote-controlled encapsulation in a hot-cell.

Table 2-1 Guaranteed radiological characteristics of the vitrified waste container returned from Orano reprocessing site. Metal particles refer to platinoids such as Pd, Ru and Rh that are not soluble in the glass matrix.

Parameter	Limit (per canister)
Uranium mass	< 4500 g
Plutonium mass	< 110 g
Cs-137 activity	< 6600 TBq
Sr-90 activity	< 4625 TBq
Oxides (FP + Zr + actinides) + metal particles	4.2 - 18.5 wt-%
Decay heat	< 2 kW

3 Applicability of as-fabricated KBS-3 canisters

3.1 Dimensions of as-fabricated canisters

The suitability of the KBS-3 canisters to accommodate the vitrified waste canisters from Orano is considered in the present chapter, corresponding to the earlier analyses of IFE's metallic fuel (Loukusa et al., 2020). 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 - applied tothe BWR fuel from Olkiluoto-1 and -2 - is shown inFigure 3.1-1. Variations to the standard design are required in Finland, since the fuel assemblies from the VVER-440 units (Loviisa-1 and -2) and the EPR under construction (Olkiluoto-3) all have different dimensions.

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 mm[1]. 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. The EPR insert has similar openings, but with inner width of 235 mm, opening spacing 125 mm and neck thickness roughly 50 mm. In both the Loviisa and Olkiluoto 1-2 inserts the number of openings is 12, whilst only four openings fit in the EPR insert.

In terms of cross-sectional dimensions, the Orano WVP canister, with its 430-mm-diameter, thus significantly differs from the fuel assemblies for which the KBS-3 insert layout variants have been designed. Therefore, a fully customised insert must be designed for the waste glass canisters. With respect to the axial direction, the inner length of the shortest KBS-3 version designed for the VVER-440 assemblies is 3245 mm. That would allow two Orano canisters to be emplaced axially with 575 mm thick middle buffer. Alternatively, the 4900 mm long EPR canister could accommodate three glass canisters with 447.5 mm thick middle buffers. The canister length of 1335 mm is assumed for these dimensions. Assuming the length of 1345 mm, the buffer thickness would be 555 mm and 432.5 mm, respectively (see Chapter 2).

[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.

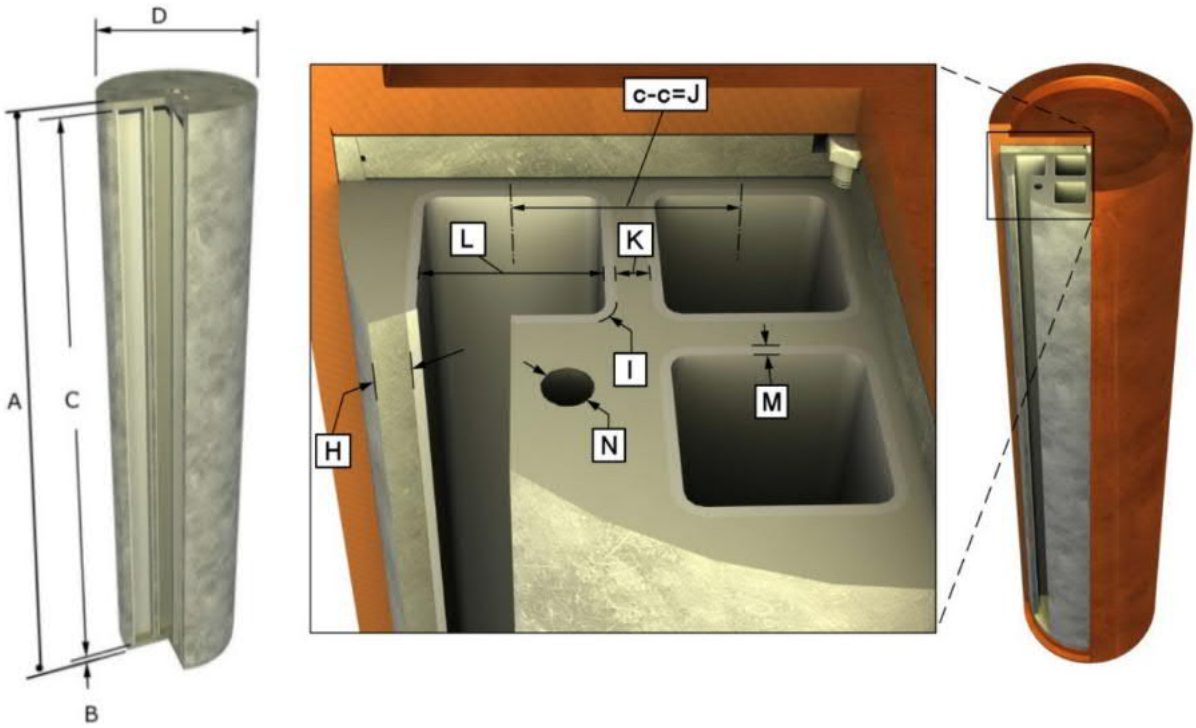

Figure 3.1-1. The different dimensions of the BWR canister insert. C: inner height, H: neck thickness, K: insert spacing, L: inner width of opening, M: profile thickness. Figure from Raiko et al. (2010).

4 Applicability of KBS-3 concept

4.1 Dimensioning

As demonstrated in the previous chapter, none of the standard inserts designed for the KBS-3 disposal canisters for the VVER-440, Nordic BWR and EPR fuel assemblies would be directly or with minor modifications usable for the Orano vitrified waste canisters, but a completely new type of an insert must be designed. The most likely configuration to qualify would be that with one WVP canister in a KBS-3 canister, or two or three canisters located axially with respect to each other. Two or three canisters adjacently into a KBS-3 canister are illustrated in Figure 4.1-1., showing that such configurations would be impossible. Mechanical strength requirements prevent these concepts.

Emplacing one WVP canister per KBS-3 canister radially allows a roughly 25 cm thick layer of cast iron between the WVP canister and the copper overpack. The configuration would be very close to e.g. the Swiss and French concepts regarding the insert. As discussed in the previous chapter, it could be feasible to emplace axially more than one glass canister into a single KBS-3 canister. The copper overpack option has been considered also both in the French and Swiss repository programmes.

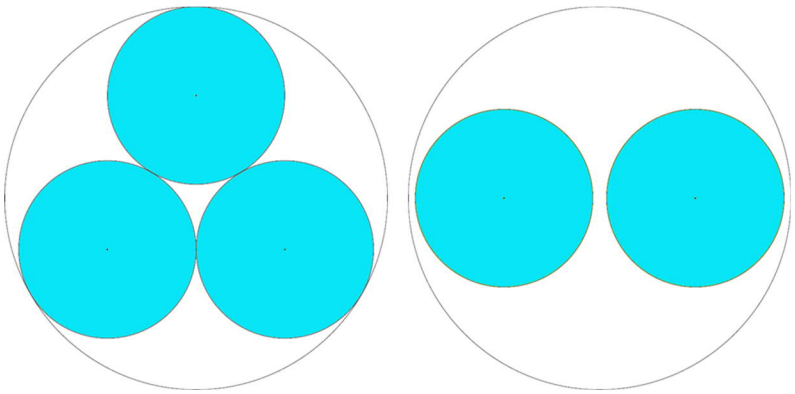

Figure 4.1-1. Illustrations for configurations with more than one waste glass canister adjacently emplaced into a KBS-3 canister. The large circle represents the inner surface of the copper overpack and the blue circles the WVP canisters. There would not be enough space for an insert to ensure adequate mechanical strength in these cases.

4.2 Mechanical integrity

To fulfill the main principles of safe disposal of radioactive waste one important component is to provide containment of the radionuclides associated with the waste. “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” as stated in Requirement 8, para. 3.39, of the IAEA publication SSR-5 (IAEA 2011).

In designing the packages, it is essential to ensure adequate mechanical strength regarding the conditions in the planned disposal system, host environment and geological formation. In the deep geological disposal systems to be applied in Sweden and Finland in hard rock, the mechanical conditions are governed by the hydrostatic pressure and swelling pressure of the bentonite buffer separating the canister from the rock. The canister is composed of a leak-tight copper shell and of a load-bearing nodular cast iron insert. The copper overpack shall provide the corrosion resistance required in the postulated repository and the iron insert shall provide the mechanical strength required.

In their Canister Production Line 2012 report (Posiva 2012), Posiva describes the requirements on canister mechanical strength as:

- The canister insert shall be designed to bear the hydrostatic pressure from groundwater and from swelling of bentonite.
- The canister insert shall be designed to bear the hydrostatic load caused by glaciation.
- The canister insert shall be designed to bear unevenly distributed swelling loads.
- The canister insert shall be designed to bear the loads from the postulated rock shear displacements in the deposition hole.

External mechanical loads

Under current environmental conditions, the hydrostatic pressure and bentonite swelling pressure will cause the mechanical external load on canisters in the long term. In Finland the nominal depth is 420 m and the groundwater table is somewhat below the ground level and is brackish, i.e. not very saline. The groundwater hydrostatic pressure can be estimated as 4.1 MPa. The swelling pressure of the buffer bentonite depends on the final saturated density. For technical reasons and to fulfill the requirements set for the buffer, a density of 1950 – 2050 kg/m^3 is chosen. This means a swelling pressure of 2-10 MPa for sodium bentonite. If Na is changed to Ca during long times, a higher pressure up to 15 MPa could be formed.

An additional pressure would be developing during glaciation periods if thick ice cover is formed. An ice layer of 2.5 km would add to the load a pressure of 25 MPa (lower density of ice than that of water is not taken into account). The total isostatic pressure would then be 44 MPa. This requirement for increased strength during glacial periods would be applicable after a very long time (probably several tens of thousands of years), when the next glaciation would appear. Depending on the inventories of very long-lived radionuclides the need of this requirement should be assessed separately.

Insert properties and wall thickness

The KBS-3 canisters for disposal of spent fuel are designed for a number of fuel assemblies and their dimensions. The positions for the assemblies are square or round holes depending on the assembly form. The structure of the insert is similar to a honeycomb and affects the strength properties in an essential way.

Figure 4.2-1. A full-scale demonstration of a BWR canister (Posiva Oy)

The thinnest distance between the outer radius of the insert and a hole is from 33 mm to 50 mm depending on the fuel type.

To make a preliminary assessment of the required thickness from the mechanical strength point of view, theoretical considerations regarding buckling of a thick-walled tube under external pressure can be applied. In literature various empirical approaches can be found and also design standards are presented by organisations like the American Society of Mechanical Engineers (ASME) and the foundation Det Norske Veritas (DNV) fulfilling its purpose through the ownership of the DNV GL group (DNV 2020).

In a recent paper Xinhu Zhang and Guang Pan (2020) presented various aspects of collapse responses of thick-walled subsea pipelines. For the present preliminary assessment, it is not necessary to consider imperfections in the forms of the ovality and the thickness eccentricity. They state that their proposed formula can provide an accurate prediction for the collapse pressure of thick-walled subsea pipelines and that the error is stable. One of their conclusions is also that the standard DNVGL-ST-F101 (DNVGL 2017) underestimates the collapse pressure of a subsea pipeline when the DTR (diameter to thickness ratio) is less than 20 (see Figure 21 in X.Zhang and G.Pan, 2020).

Here the older version of the standard DNV-OS-F101 (DNV 2012) is used, however, and is basically given in form of a cubic equation. The formula for the collapse pressure is not changed in the new standard mentioned above. Mechanical strength for a wall thickness of 100 mm was calculated. To check the result, the procedure given in the ASME (2007) was followed. The results are for the DNV (2012) and the ASME (2007) standards 42.4 MPa and 43.6 MPa, respectively, taking a Design Factor (for safety marginal) of 1.667 into account in both results. The assumed material properties were the following:

Young's modulus	E = 166 Gpa
Poisson's ratio	v = 0.32
Yield strength	σ_y = 345 Mpa

The outer diameter of the insert was taken as 0.95 m in the calculation as an upwards slightly rounded value (instead of 949 mm) and for the ASME case the unsupported length as L = 4 m. The DTR has then for the 0.1 m thickness the value of 9.5. With such a low value the critical pressure for buckling is usually not given accurately by standards mentioned above but are estimated to be too low (i.e. conservative regarding safety).

An example of calculated critical pressure as a function of insert thickness according to the DNV (2012) standard and the material properties given above is shown in Figure 4.2-1.

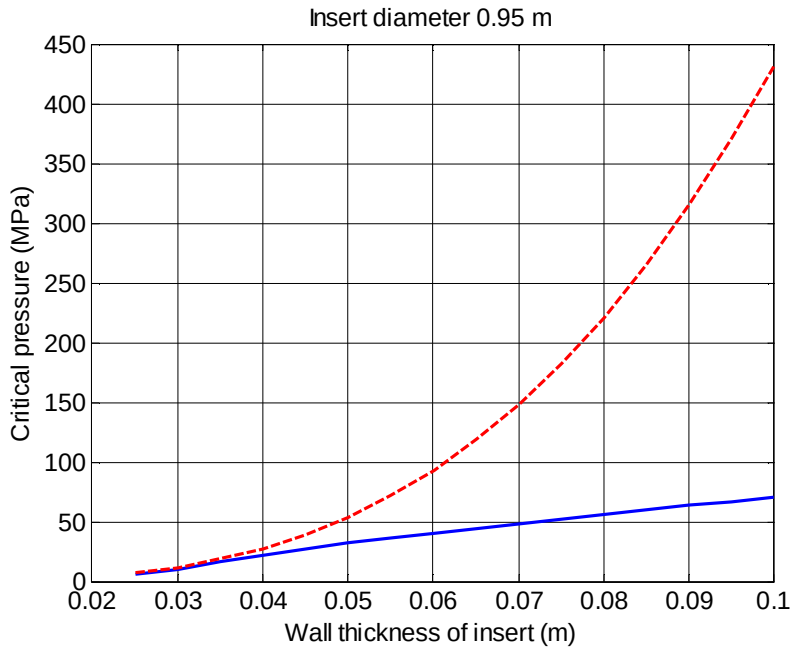

Figure 4.2-1. Critical pressure according to the DNV (2012) standard and measures and properties given in the text. The blue line represents the plastic buckling and the dashed red line for comparison the elastic buckling assuming that material properties would allow all the way elastic behaviour. This is unrealistic for larger thicknesses or small DTR values, though.

For external pressures on thin-walled tubes buckling can take place at stresses below the elastic limit or yield strength of the material. For thin-walled tubes a more simple assessment can be made by using an analytic formula as discussed in text books and also by Zhan and Pan (2020, Eq.1). The curve in Figure 4.2-1 called as elastic buckling represents the case when material properties are assumed to be elastic. In thick-walled tubes, however, stresses increase towards the inner radius and buckling takes place only after material properties are plastic. The simple analytic form can be used for DTR ratios from 35 upwards meaning a wall thickness of about 27 mm when the diameter is 950 mm. The elastic buckling curve is shown just for comparison to DNV (2020) results at thinnest wall thicknesses and deviations at thicker ones.

Insert geometry

The vitrified waste packages delivered from the reprocessing plant are stainless steel Universal Canisters having the same standardized shape and dimensions:

- maximal outer diameter: 440 mm
- maximal total length: 1345 mm

This means that a KBS-3 type insert can hold radially only one Universal Canister for vitrified waste (UC-V) positioned in a central hole but allows in the other hand for an adequate thickness for the insert tube wall. There is space up to a wall thickness of 253 mm depending on the gap and tolerance between the insert inner wall and the UC-V outer diameter. In case of about 100 mm for the needed thickness of the insert regarding mechanical strength there is “too much” space for the UC-V. It may not be optimal for various reasons to design just a full bulk insert with an inner diameter of 442 mm. At least one reason is the weight of the insert and another possible one related to the casting process (cf. the small round holes in the BWR insert).

The full bulk insert for two UC-V canisters would weigh about 18.1 ton (without any middle support between UC-V canisters). Mass of the copper of the BWR canister is 7.3 ton and waste 1.5 ton which adds up to a total of 26.9 ton. The total mass of a BWR canister with 12 assemblies of spent fuel is 24.5 ton. The masses for the VVER canister are 13.4 ton, 5.6 ton and 1.0 ton waste, respectively. This adds up to a total of 20.0 ton whereas the VVER canister with 12 assemblies weighs 18.8 ton. The masses for the EPR canister are 20.0 ton, 8.0 ton and 1.5 ton waste, respectively. This adds up to a total of 29.5 ton whereas the EPR canister with 4 assemblies weighs 29.0 ton as given by Raiko (2012).

Any design has to be checked anyway for mechanical, thermal and chemical properties and processes. Extra holes could e.g. cause more corrosion and gas production and reduce heat transfer somewhat through the insert. The possibility of using a steel tube in the casting process similarly to the BWR square holes to form the central hole for the UC-V should also be studied. A steel tube could be helpful in the casting process. It may also form a smooth surface without the hole to be machined and provide a good chemical interface for the UC-V canister.

4.3 Corrosion and chemical stability

Basis for chemical conditions

As introduced already in the section dealing with mechanical requirements the integrity of spent fuel or high level waste disposal canisters is a key question in assuring safe disposal. Chemical processes are in addition to mechanical ones main issues to potentially endanger the containment principle. Minimal corrosion rates of waste packages provide a long-term shield to prevent radionuclide releases before the activities have decayed to marginal levels. A complementary barrier is the waste form itself. In case of spent fuel the extremely low dissolving rate of UO2 matrix reduces congruent release rates of radionuclides. But also in case of vitrified waste the slowly dissolving glass matrix provides a significant limitation of radionuclides presumably over hundreds of thousands of years. The degradation of the canisters and waste matrix depend, however, on chemical conditions in the and nearby the repository.

Copper stability

Studies on potential corrosion reactions of copper have been ongoing for many decades in many countries like Sweden, Finland, Canada, Japan, and Switzerland. The main conclusions are that despite some differences in analyzed conditions and approaches to assess general corrosion and pitting the predicted canister lifetimes exceed 10^6 years in all cases discussed in King et al. (2012).

Copper corrosion conditions vary during the various phases of the canister lifetime. These phases are described in the Canister Production Line report (Posiva 2012) and it is stated that:

"During the early part of the unsaturated transient, there is the possibility for localised corrosion because of non-uniform wetting of the surface. Localised corrosion is possible during this period due to the non-permanent separation of anodic and cathodic processes, leading to a general roughening of the surface. The conditions during the unsaturated phase are the most aggressive for the copper canister because of the presence of residual oxygen and the potential for localised corrosion and stress corrosion cracking."

This period is, however, very short and corrosion depths are in submillimetre range. In the long term in saturated, anoxic and reducing conditions the prevalent mode is general corrosion due to sulphide. A summary of the phases and results is presented in Raiko (2012, Table 21) and reproduced below (the ref. King et al. 2011b in the title of Table 4.3.1 corresponds to King et al. (2012)) in the present report).

Table 4.3.1. Estimated corrosion depths during the evolution of the canister in a repository in Finnish/Swedish conditions. Based on King et al (2011b).

Time period	Corrosion depth	Comments
Encapsulation and pre-disposal	<1 µm	Oxidation in atmospheric conditions and formation of a thin oxide film.
Handling and operation	negligible	Potential for cold work and other plastic deformation have been considered but have been shown not to cause any long-term effect. Galvanic corrosion from residues of steel during this phase has been considered and neglected (see Section 8.6.2).
Repository in unsaturated, oxic conditions	0.1 to 1 mm	The main corrosion modes are general corrosion due to the presence of residual oxygen and localized corrosion due to the uneven swelling of the buffer. The corrosion depth give is the total depth due to general and localized corrosion throughout the entire evolution period (it includes also the contribution from general corrosion during the saturated conditions). The corrosion depth range is from the mass balance calculation estimate of 840 micrometres for the residual oxygen calculation plus 50 micrometres due to surface roughening (see Section 8.6.3). The potential for SCC is at maximum in the first years after deposition but then disappears because of the lack of the necessary conditions for SCC (Section 8.6.6).
Repository in saturated, anoxic and reducing conditions	1 mm after 10^6 years	General corrosion due to sulphide is the main corrosion mode (see Section 8.6.8). SCC not included because of limited tensile stress, absence of SCC agents, insufficiently positive E_{CORR}, and inhibitive effects of Cl^- in anoxic and reducing conditions.
Overall evolution (10^6 years)	2 mm max over 10^6 years	See Section 8.6.8.

Further details are presented by King et al. (2012). It is worth noticing also Section 9.2 New and continuing areas of further research in that report to get an idea how the conclusion of long-lasting integrity of the copper canister can be strengthened.

The table above is applicable mainly to a reference scenario in which the safety functions of the barriers act as planned. In such situations groundwater flow rates in the vicinity of deposition holes are low and the bentonite buffer prevents advective mass transfer within the deposition hole. However, some variant scenarios with moderately increased flow rates or higher than assumed sulphide concentrations (> 3 mg/L) do not result in copper shell deterioration. Even decrease of bentonite density leading to advective

transport in a deposition hole can cause corrosion failure of just a few canisters (in the Finnish case out of 4500 ones) in a million year time period. Canister corrosion issues for the safety case are handled in the performance assessment report by Posiva (2012a).

Glass corrosion

The issue of glass corrosion rates is actual first after a loss of integrity of the foreseen canister as the waste package. Safety analyses include often, however, also cases that have very low likelihood in practice. This is done to study individual barriers of a system separately and as “what if” cases. The corrosion rates of glasses are an interesting topic in this sense but in any case also in very long times. The shorter the expected lifetime of a waste package, the more important the glass corrosion behaviour is.

Curti (2003) reports updated dissolution rates for the two Swiss reference glasses resulting from reprocessed high-level radioactive waste. The update builds upon experimental and theoretical work performed in the last decade at Paul Scherrer Institut, Villigen PSI and in other European countries.

According to the report the corrosion rate of borosilicate glasses depends on the dissolved silica concentration in the bulk solution. This is then related to the transport problem of silica away from the vicinity of the corroding glass. A protective gel layer can also limit transport of silica and reactive species through such a layer.

The corrosion rate is significantly higher at higher temperatures. It can be expected that contact of glass with water takes place so late that canister temperatures are decreased near to original environmental levels. In cases of possible initial small penetrating defects of the copper shell it has been estimated in the Finnish programme that filling the canister interior void volume takes at least thousand years. The decay heat power of vitrified waste decreases also faster than that of spent fuel and thus the effect of higher corrosion rates at higher temperatures is not significant.

Some studies show that sorption of silica on clays is weaker than earlier was observed (smaller distribution coefficients). This would keep dissolved silica concentrations higher and thus reduce corrosion rates. An unfavorable effect, on the contrary, would be if silicates could precipitate. This would be more important in case of Al-rich glasses, which promote formation of aluminosilicates. It is much less pronounced in the case of the Al-poor borosilicate glass formulations like those used for the vitrification of Swiss high-level waste Curti (2003).

As a last point of his brief review of processes and parameters known to affect glass corrosion rates and an appreciation of the related uncertainties Curti (ibid.) presents the effect of Fe corrosion products. He points out that a factor of 5 increase in corrosion rates observed in some experiments when using magnetite powder is not conclusive because of very low surface to volume ratio (S/V) in those experiments and suggests experiments at higher S/V ratios and longer reaction times.

After presenting the experimental work and results at PSI, reference rates for safety assessment calculations are given in Table 4.3.2.

Table 4.3.2 Updated corrosion rates for safety analysis calculations.

	BNFL glass (MW)	COGEMA glass (SON68)
R_∞ [g m^{-2} d^{-1}]	1.5 x 10^{-3}	2 x 10^{-4}
R_∞ [kg m^{-2} a^{-1}]	5.5 x 10^{-4}	7.3 x 10^{-5}

This can be illustrated by the following example. If 1 m^2 of glass in the UC-V canister would be available for corrosion and fracturing increases the water contact area by a factor of 50, then the rate for the Orano

(COGEMA) glass would be 3.65e-3 kg/a and it would take 137 000 years to corrode 500 kg (assuming the rate being constant throughout).

The processes affecting the glass corrosion are discussed by Grambow (2006) in more detail. He points out especially the processes that decrease the initially somewhat high corrosion rates in long term to much lower values. In the last section of the article a summary regarding the safety assessment is given:

"Are Long-Term Predictions of Glass Corrosion and Radionuclide Release Credible?

Radionuclide release rates from glass were recently predicted for a generic near field (glass + clay barrier) of a repository in clay formations. Data concerning the dissolution rates of the glass, the diffusion of glass constituents and radionuclides in the compact near-field environment in the vicinity of the glass at low water/solid ratios, the adsorption of radionuclides onto clay minerals, and the solubility limit of these radionuclides in the water-filled canister void volume were taken into account. Expressed in units of released fractions of the radionuclide inventory in a full-sized canister, the rates are as follows: for Am, 10^{-14}/y; for Pu, 10^{-10}/y; for Cs and I, 5×10^{-6}/y (Grambow and Giffaut 2006). Only the release of Cs and of the small quantity of potentially vitrified iodine from the glass is controlled by the stability of the glass matrix, and only for these potentially mobile elements are the uncertainties in the long-term glass dissolution mechanism of importance in the safety assessment. The release of the actinides and Tc is controlled by their very low solubility limits under reducing geochemical conditions and their strong retention on clay materials (for one radioactive actinide or Tc atom being dissolved in the clay pore water, 1000 to 100,000 atoms are immobilized or adsorbed onto clay). Variations in glass dissolution rate by factors of about 1000 have no significant effect on the actinide and Tc release rates, since the soluble fraction at the glass/near-field boundary does not change. This prediction is considered to be very "robust" for European repository concepts in clay formations. Thus, the actinides Pu, Am, and Np are expected to decay in proximity to the glass, without ever migrating to the biosphere."

What was here said for the clay formations applies also to the case of crystalline rock repositories because of the rather big amounts of clay materials surrounding the canisters. Regarding uncertainties Grambow (2006) states that:

"*The largest uncertainties in long-term glass performance evaluations are probably associated with detailed coupling of the glass dissolution mechanism to the chemical reactions and transport processes in the near field. Coupled models are under development, but an assessment of the reliability of these models would be premature at present.****"***

Uncertainties are, however, compensated for in safety assessments by conservative approaches and data. For vitrified waste data it means that instead of lifetimes of millions of years, but being somewhat uncertain, lifetimes in safety assessments are typically estimated conservatively to be hundreds of thousands of years.

Comparing directly dissolution rates of different waste matrices gives one aspect regarding the waste forms among many. The commonly existing waste forms in general are uranium oxide, metallic uranium and vitrified waste. Uranium oxide is known to have a very low solubility and to dissolve very slowly, especially in anoxic reducing conditions. Because there might be options to proceed in the Norwegian waste disposal programme in a way where either metallic uranium or vitrified waste would be part of the waste form, an intercomparison of dissolution rates of these forms as such is of some interest.

In a report RWM (2016) in UK present some values on metallic uranium dissolution rates based on experimental data. The reaction of metallic uranium is relatively fast and is expected to be even faster under anoxic conditions. A rate range of 0.2 – 20 g m^{-2} d^{-1} is given in the report which gives 0.73 kg m^{-2} a^{-1} for the (logarithmic) middle value of 2 g m^{-2} d^{-1}. Comparing to the Table 4-1 of Curti (2003) It is seen that the dissolution rate of metallic uranium is 10 000 times faster than that of the COGEMA glass. Metallic uranium is expected to be transformed into UO_2. If the solubility limit of uranium is reached, i.e. uranium is not carried quickly away, the corrosion products may precipitate and reduce release of co-precipitating nuclides. Highly-soluble fission products may escape anyhow.

4.4 Heat production

In addition to the geometrical fitting, decay heat production is a significant issue to consider. Assuming slightly conservative heat production of 600 W/WVP canister, the resultant linear (axial) heating power would be ~4.5 W/cm without the end plugs of the KBS-3 canister. As a comparison to heating limits per canister set for the fuel assemblies, the 1700-W-limit for the BWR canister would denote 3.8 W/cm axially. Such a comparison suggests that the heat transfer analyses performed for the Finnish repository (e.g. Ikonen et al., 2018) are applicable for vitrified waste too. Considering the particular Norwegian final disposal repository, the calculations need to be specifically performed for the selected site, since the thermal conductivity, as an example, may differ from that of the Finnish repository site.

The objective of the heat transfer calculations has been to determine the maximum density (center to center distance) for the waste packages in the repository such that the bentonite temperature would never exceed 100°C. The repository dimensioning is not a similar problem for the Norwegian repository due to the small amount of waste, but the chemical properties of bentonite as a barrier require that the temperature limit is obeyed.

Considering the specific features of the waste form, the glass temperature must be kept below 610°C in order to prevent devitrification (Meleshyn & Noseck, 2012). Rather than for disposal, the temperature limit can be restrictive in interim storage and transportation. For the long-term durability of the glass, its temperature should decrease below 50°C before the first interaction with water. The requirement is one of the factors defining the minimum time that the integrity of the overpack canister should be maintained.

Canister options The vitrified waste canisters UC-V are rather short (1.345 m) compared to lengths of spent fuel assemblies The length of the BWR assembly is 4.127 m and inside length is 4.450 m, which would allow three UC-V canisters to be put in the BWR canister. The waste mass would be only 1500 kg instead of 12 assemblies 292 – 331 kg each making a total weight around 3600 kg. Correspondingly the VVER assembly length is 3.217 m and inside length is 3.245 m allowing two UC-V canisters to be put in the VVER canister. The waste masses are 1000 kg and 2600 kg for 12 assemblies, respectively. The maximum heat production sum of the three UC_V canisters at disposal should be limited to 1700 W and to 1370 W for the two UC-V canisters. If the heat production rate of the vitrified waste in the UC-V canisters would be higher than given above, enough cooling time would be needed to decrease the heat power to accepted levels. These maximum powers can be increased somewhat if distances between canisters are increased. A thermal optimisation has to be made when thermal properties of the bedrock are known.

Table 4.4-1. UC-V canister adaptation into BWR- and VVER-type canistersxxx

	BWR-type fuel/canister	VVER-type fuel/canister
Assembly length (m) Inside length (m) Number of UC-V canisters / length (m)	4.127 4.450 3 / 4.035	3.217 3.245 2 / 2.690
Heat power limit (W)	1700	1370
Weight of 12 assemblies (kg) Weight of all UC-V canisters (kg)	3600 1500	2600 1000
Total weight with 12 assemblies (ton) Total weight with number of UC-V canisters (ton)	24.5 26.9*)	18.8 20.0*)

*) Weight with full bulk insert without middle supports

4.5 Criticality

Criticality safety is a minor concern with vitrified waste, since the fissile content of the spent fuel is mostly removed along with the uranium and plutonium extracted in reprocessing, leaving only minor processing losses of U and Pu to the vitrified waste. Additionally, minor actinides are incorporated to the waste glass with minor fissile nuclide concentration. The high neutron capture cross-section of the isotope B-10 does also help in reducing the reactivity. The effective multiplication factor (k_{eff}) is used as the measure of criticality in the studied system. A simplified definition for k_{eff} can be presented such that it is the ratio of

the neutron population of one generation to the neutron population of the preceding generation. Thus, any k_{eff} < 1 denotes a subcritical system, or dying fission chain reaction, which must be ensured for all storage and transport configurations.One of the methodologies to demonstrate the sufficient subcriticality is presented in the documentation for the French Cigeo facility by Andra (Andra, 2016a). For each glass canister a homogeneous mixture of pure Pu-239 and CH_2 is assumed with the plutonium mass of 200 g. As a sample calculation, for an infinite lattice of disposal canisters containing such canisters in contact with each other, the k_{eff} was calculated to be 0.92. For the k_{eff} evaluation in the repository conditions, an infinite line of horizontally emplaced disposal canister (each containing two glass canisters) was assumed and k_{eff} = 0.91 was obtained.

The above-mentioned examples of calculated k_{eff} are well below the regulatory limit of 0.95 in normal conditions. As presented in Chapter 2, Orano guarantees the total plutonium mass to be less than 110 g per canister including all the Pu isotopes present in the spent fuel, so the assumption of 200 g Pu-239 can be considered very conservative.

4.6 Applicability of the tools, vehicles and other machinery intended to be used at Onkalo

This chapter presents a theoretical scenario in which the Norwegian spent nuclear fuel is reprocessed. Currently, no decision has been made regarding whether to reprocess the fuel or not.

In the case of reprocessing the high-level waste and receiving vitrified waste form, the decreased waste volume can be fitted in eight Orano canisters (Ikonen et al. 2020). For long-term disposal, the Orano canister will be installed into a copper shell, similar to be used e.g. in Sweden and Finland. As there would be gaps between Orano canister and copper shell, a special insert will be fitted in the copper shell before the Orano canister is lowered in it. The insert will also provide mechanical strength for the canister as in the case of spent nuclear fuel canister. The dimensions of Orano canister, copper shell and gaps between them are given in Chapters 2 and 3.

The operations of any nuclear material are under strict legislations and controls. For transport, IAEA has compiled a Specific Safety Requirements document (IAEA 2018), which will be followed in addition to other binding legislation and guides of participating stakeholders.

The preliminary operation phases of the vitrified fuel transport, encapsulation and disposal are illustrated in Figure 4.6-1, with explanations after the figure.

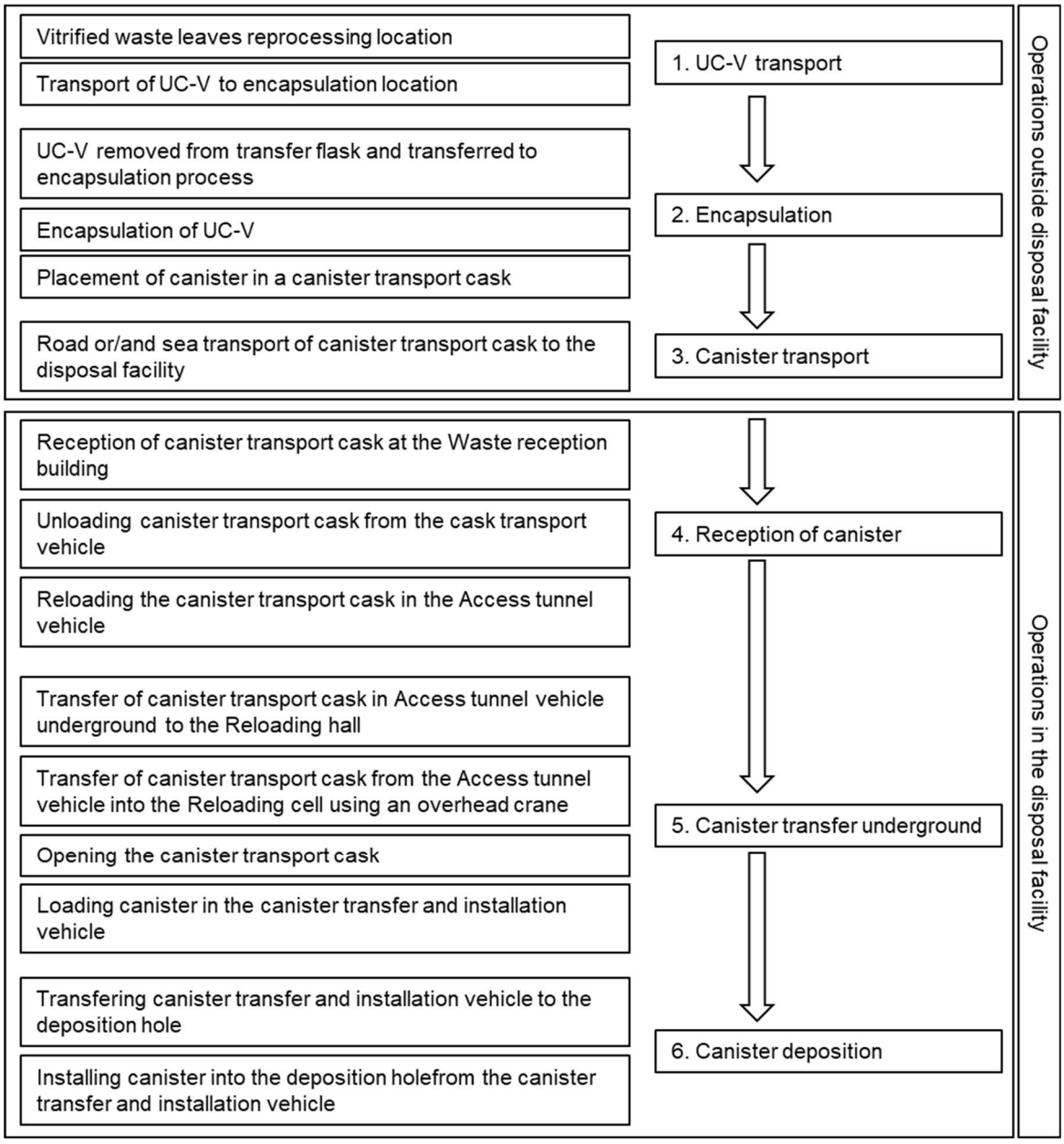

Figure 4.6-1. Preliminary operation phases for vitrified waste transport, encapsulation and disposal.

At phase 1, the Universal Canister with vitrified fuel (UC-V) is transported by standard means from Orano facility. It is assumed that the encapsulation of Orano canisters does not occur at the disposal facility but in another location. Here it is assumed it will be an encapsulation plant for spent nuclear fuel, for radiation safety and quality control purposes. Suitable locations are in e.g. Sweden and Finland.

At phase 2. the UC-V is transferred from the transport flask to encapsulation. This phase will need additional plans at the encapsulation plant, as size and function of the UC-V flask differs from the spent nuclear fuel transport casks. However, the procedure can be operated with some changes. The canister with the UC-V in the copper shell shall then be transferred to the canister welding, machining, and non-destructive testing.

At phase 3, after the encapsulation, the copper canister will be loaded into a transport cask. The cask can be similar to what will be utilised by SKB in transferring their SNF canisters from encapsulation at Clink to disposal at the Swedish repository. Schematic illustration of the SKB canister transport cask on a vehicle platform is in Figure 4.6-2. The vehicle will have shock absorbers. The casks shall comply with international standards, which include being able to withstand a drop of nine metres on to a totally

unyielding surface, more than 30 minutes in a fire at over 800 °C, and external pressure corresponding to immersion at a depth of 200 metres under water without any leakage occurring (IAEA 2018). The canister transfer cask will be loaded on transfer vehicle like that which will be used for fuel transports in different countries (e.g. Sweden or Finland). It is a platform truck with safety measures for keeping the canister cask in place during transport and a weather guard to keep the cask clean of environmental debris. The canister cask transport vehicle is expected to move with safety convoy from encapsulation location to the disposal facility location at moderate speed via planned route. Transport can be by land or also by sea, in which case the canister cask transport vehicle is transported by boat closer to the disposal facility.

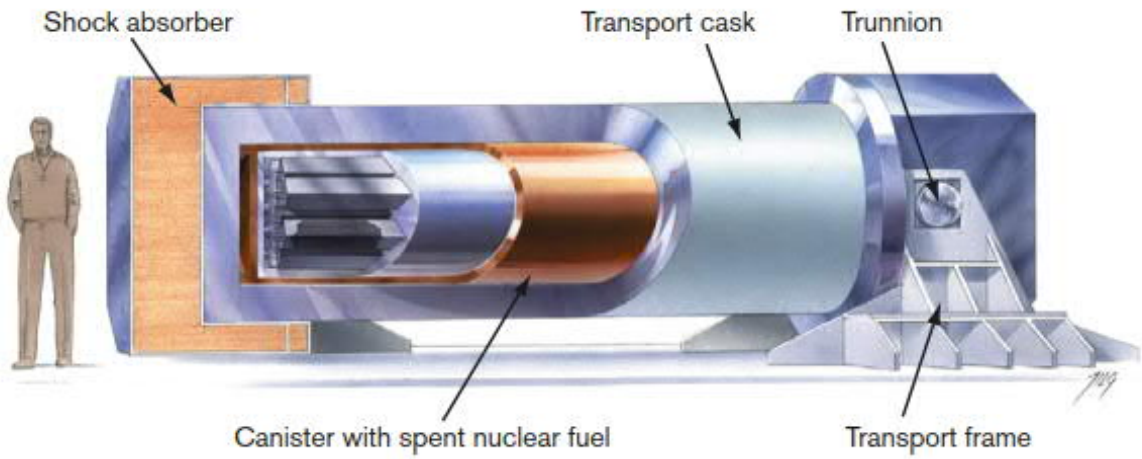

Figure 4.6-2. Schematic view of a Swedish canister in a transport cask. Trunnions in both ends of the cask are secured to the transport frame shown at the right end of the cask. (SKB 2010.)

At phase 4. the canister transport cask is driven to the waste reception building at the disposal facility. The weather guard is removed, and cask is cleaned, if necessary. It is then detached from the platform and lifted to vertical position and to the storage location ready for loading to the access tunnel vehicle. From the reception hall (and possible storage phase) the canister transport cask is lifted by an overhead crane and lowered on the access tunnel vehicle. It is secured in place and radiation is measured. The access tunnel vehicle is driven down into the Reloading hall.

The access tunnel vehicle is a more compact version of a platform truck and can be, for example, similar as will be used by SKB at Forsmark. It runs by diesel fuel and top speed is about 10 km/h. The difference to canister cask transport vehicle is that the latter is more robust and suitable for traffic on normal roads. It is meant for long transports whereas the access tunnel vehicle is limited to short, local distances. The compact size of the access tunnel vehicle is suitable for underground operations.

At phase 5. the cask is transferred into the reloading hall. In the reloading hall, canister cask is detached from the access tunnel vehicle and an overhead crane lifts it into a reloading cell. The lid of the canister cask is detached and transferred off the cask. The canister transfer and installation vehicle is driven to the reloading cell and canister is lifted from the cell into the radiation shield of the canister transfer and installation vehicle. The safety measures on cables are redundant. When the canister is lifted the radiation shield is turned to horizontal position for transferring the canister to the deposition hole.

At phase 6. the canister is transferred to disposal. At the deposition hole, the canister transfer and installation vehicle turns the radiation shield to vertical position and canister is lowered into the deposition hole with the vehicles wire cables. Prototype of one canister transfer and installation vehicle is in Figure 4.6-3.

Figure 4.6-3. Prototype of canister transfer and installation vehicle by Posiva in Finland (Posiva 2016).

The difference between disposal of spent nuclear fuel and disposal of processed and vitrified waste is in these operation phases most prominent at the encapsulation phase. All transport and transfer vehicles can be same for both options, but at the encapsulation, methods are different for lifting the UC-V into the canister than for placing the fuel in it. For other process steps, the effect is mainly that the canister is lighter with vitrified waste than what it would be with the unprocessed spent nuclear fuel. The radiation level is also lower.

In the case of vitrified waste there are transport phases of first the spent nuclear fuel to processing in France and then returning the vitrified waste from there. There is a safety related feature in these transports because HLW transports receive high public attention, which can lead to e.g. demonstrations and can attract opposition by anti-nuclear campaigners. Such transports have, however, been carried out on several occasions from France to e.g. Japan, Germany and Belgium. The technical process is feasible, and the risks can be mitigated by planning and good communication between stakeholders and communities.

As for production of the repository and installation of buffer, backfill and closure structures, there is no difference between the case of vitrified waste canisters or spent nuclear fuel (Ikonen et al. 2020). . Also, the number of canisters is smaller with vitrified waste. Thus, the repository could be even more compact.

5 Knowledge gaps and future work

5.1 Uncertainties

The following describes some uncertainties and discrepancies discovered in various documents studied as a part of this literature study; mainly with nuclide composition of the vitrified waste and the dimensions of the Orano canister.

Nuclide inventory (especially C-14)

The composition of significant nuclides in terms of long-term safety in the waste glass has been somewhat a matter of uncertainty and partly discrepancies. Among others, the German GRS has evaluated and estimated the concentrations of such nuclides (Meleshyn & Noseck, 2012). Calculations were required, as Orano (then-Areva NP) has not reported the nuclide compositions for some important nuclides with respect to the long-term safety considerations. The missing data was derived based on the information on average initial fresh fuel data and discharge burnup of the reprocessed fuel, known or estimated releases of volatile elements in the process and other nuclide-specific yield and loss mechanisms affecting its inventory in the waste. The results were compared to a Nagra study (Nagra, 2008) and an earlier GRS study (Peiffer, 2011). In the former, similar methodology has been used, but the results deviated rather considerably e.g. for C-14 and Mo-93. Considering, Cl-36 or I-129, the earlier GRS study had not taken into account any processing losses at La Hague. Nagra did not estimate Cl-36 at all and obtained much larger range for the estimated I-129 inventory. Therefore, some uncertainty regarding these nuclides exists and since the issue has long term safety implications, some investigation will be needed.

Orano canister dimensions

In the literature review done as part of this report, a slight discrepancy was detected with the outer dimensions of the Orano vitrified waste canisters in different sources. Multiple documents available online consistently suggest dimensions that are 10 mm less in both length and outer diameter than the document sent by Orano to NND. It is unlikely that such a minor deviation is going to have any significant effect on the applicability analyses of the KBS-3 concept for the waste packaging, but this issue should be clarified before any design efforts for e.g. custom insert is pursued.

5.2 Alternative solutions

The French feasibility study on high-level nuclear waste final disposal in granite host rock (Andra, 2005) recommended the further R&D for the disposal canister to be based on the KBS-3 concept for spent fuel, but for vitrified waste a very thick steel container was proposed as the reference design. Low-carbon unalloyed steel has been the reference design for the repository in clay that is the selected host rock. However, since the disposal of high-level waste is not going to be started before 2075, other material options are kept open. The container thickness is assumed to be 53 - 65 mm. It is recognised that a thicker wall would increase its mechanical strength, corrosion resistance and radiation shielding, but the constraints related to the production process tend to limit the maximum thickness. The container is going to be surrounded by a metallic casing separating the container from the bedrock and backfill, so the canister thickness may not be directly comparable to other types of disposal cells. (Andra, 2016b)

In Switzerland, a steel overpack canister with 25 cm thick walls has been used as the reference design at the earlier phase of site feasibility studies, such as the one published in 2002 (Nagra, 2002). As in France, however, no final choice has been done, but various alternatives have been studied. For the engineered barrier system, several various configurations for the whole repository have been studied, but the most deviant ones compared to the original reference layout were concluded not worth further investigation (Leunin et al., 2016). Considering the disposal package materials, it still seems that a steel overpack, or more accurately, plain-carbon steel would be the reference material to proceed with, even

though cast iron with possibly a copper overpack has been studied as one of the alternative solutions (Nagra, 2016).

The above-mentioned studies in France and Switzerland mainly focus on a disposal canister that contain one or axially more waste glass canisters. In UK, as an example, multiple repository concepts are being studied along with the choice of glass canister overpack. Additionally, concepts without any overpack for the glass canisters have been proposed (Harvey, 2012). Such deviant concepts compared to e.g. the Finnish concept are justified by the smaller material requirements, more simplified repository construction process and other practical issues. They are certainly significant issues in the British context with extensive inventory of waste to be managed, but not necessarily in Norway given the relatively small amount of waste. Otherwise, the whole disposal concept in the UK is open as mentioned in Section 1.2.3. A canister similar to KBS-3 is one of the relevant options, as well as the granite as a host rock, so the R&D in UK is definitely worth attention (RWM, 2016).

5.3 Research needs

The main difference between the standard KBS-3 system and the vitrified waste system is the need for a custom-made insert. This insert will have to be designed according to requirements, manufactured and tested to prove its mechanical capabilities. Each of these steps can represent a significant workload for the National Facility project, and incur some costs.

Another issue deriving from the custom insert is changes needed to the encapsulation process. Instead of fuel assemblies, the encapsulation system is now emplacing the Orano canister in the custom insert. This is not necessarily more difficult to accomplish than the standard process (and probably less so, as radiation levels will be lower), but nevertheless is something that a KBS-3 encapsulation facility equipment couldn't do without modification. While a foreign encapsulation facility is not necessarily available/ needed for this process, the fact remains that the encapsulation of the Orano canister into a KBS canister require R&D work, customization, and specialised machinery. It should be noted, however, that similar changes would be required for direct disposal of Norwegian spent fuel since its dimensions (as both individual rods and assemblies) differ considerably from those of commercial reactor fuels.

While not a technical issue as such, the transportation logistics and juridical issues related to transporting spent nuclear fuel internationally will also require some effort in planning and execution. The amount of Norwegian SNF is small, but international law and regulations must nevertheless be followed. It also has to be taken into account that issues with public opinion are possible, and that not all communities wish to see nuclear waste transported through their territory. Ensuring the safety of the transport will also require specialised know-how, planning and transport casks and equipment.

Finally, if this option is chosen as the solution to be used in the National Facility, a safety case will have to be made for the entire system. Some parts of it can be derived and tailored from the Finnish and Swedish KBS-3 examples, but others will have to be made specifically for the Norwegian repository. Issues dealing with fuel dissolution rates and radionuclide release rates cannot be directly copied when the waste form and inventory is different, and some modelling efforts will be required to study the expected releases in the long time span the waste remains radiotoxic. However, this is true for also the other options NND is considering (metallic or oxidized spent fuel) and isn't expected to be significantly more difficult for this option. Mechanical integrity of the canister in different scenarios will also need to be analysed and demonstrated.

6 Summary and conclusions

This report is a short, preliminary investigation of the feasibility of disposing the vitrified waste resulting from the reprocessing of Norwegian spent nuclear fuel using the standard KBS-3 repository design where possible. The main changes needed in the repository design are seen to relate to the interior design of the copper canister; namely the cast iron insert that normally holds the fuel assemblies is replaced with a cylindrical custom insert or modules of it. The report briefly investigates mechanical integrity and requirements for such an insert, as well chemical stability issues related to the vitrified waste form.

No critical issues were identified at this stage that would prevent the use of KBS-3 design to be used with vitrified waste. Other, more cost-effective options, may be available and the safety of the KBS-3 system using vitrified waste has to be further studied and demonstrated with a safety case, but it seems likely that this is a viable option. Some further research is needed on insert design and encapsulation, and some inconsistencies in starting data have to be investigated, however.

7 References

Andra (2005). Dossier 2005 Granite Synthesis - Assets of granite formations for deep geological disposal. Agence nationale pour la gestion des dêchets radioactifs. June 2005

Andra (2016a). Safety Options Report - Operating Part. CG-TE-D-NTE-AMOA-SR1-0000-15-0060. July 2016

Andra (2016b). Safety Options Report - Post-Closure Part. CG-TE-D-NTE-AMOA-SR2-0000-15-0062. July 2016

ASME (2007). 2007 ASME Boiler & Pressure Vessel Code, Section VIII –Division 2 – Alternative Rules (Code 4.4.5 Cylindrical Shell p. 4-48).

CEA (2009). Nuclear Waste Conditioning, A Nuclear Energy Division Monograph. CEA Saclay and Groupe Moniteur, Paris 2009.

Curti, E. (2003). Glass Dissolution Parameters: Update for Entsorgungsnachweis. Technical Report 02-21. Nagra, Wettingen, Switzerland, June 2003

DNV (2020) https://www.detnorskeveritas.com/

DNV (2012). Offshore Standard DNV-OS-F101, Submarine Pipeline Systems, August 2012.

DNVGL (2017). Submarine Pipelines Systems. DNVGL-ST-F101, October 2017.

Grambow, B. (2006). Nuclear Waste Glasses - How Durable?. Elements, 2 (6): 357–364.

Grambow B, Giffaut (2006). Coupling of chemical processes in the near field. Materials Research Society Symposium Proceedings 932: 55-66

Hanlon, D., Lonsdale, S., Mason, R., Putley, D., & Thallon, A. (2020) Criticality Safety for the Disposal of Spent Fuel in UK Disposal Containers. NDA Report no. RWM/Contr/20/016. Radioactive Waste Management Limited (RWM), Didcot, UK, June 2020.

Harvey, E.J. et al. (2012) Geological Disposal Concept Options for Vitrified HLW. Galson Sciences Ltd.

IAEA. (2011). Disposal of radioactive waste. Safety Standards Series, SSR-5. IAEA, Vienna, Austria.

IAEA 2018. Regulations for the Safe Transport of Radioactive Material 2018 Edition. IAEA Safety Standards for protecting people and the environment. Specific Safety Requirements No. SSR-6 (Rev. 1). International Atomic Energy Agency: Vienna, Austria.

Ikonen, K., Kuutti, J., & Raiko, H. (2018). Thermal Dimensioning for the Olkiluoto Repository − 2018 Update. Posiva Working Report 2018-26. Posiva Oy, Eurajoki, Finland.

Ikonen, A., Engelhardt, J., Fischer, T., Gardemeister, A., Karvonen, S., Keto, P., Rasilainen, K., Saanio, T. & Wanne, T. 2020 Concept Description for Norwegian National Disposal Facility for Radioactive Waste. AINS Technical Report. Espoo, June 2020.

King et al. (2012). An Update of the State-of-the-art Report on the Corrosion of Copper Under Expected Conditions in a Deep Geologic Repository. POSIVA 2011-01, Posiva Oy, Eurajoki, Finland.

Loukusa, H., Nordman, H., Karvonen, S. (2020). Feasibility of KBS-3 spent fuel disposal concept for Norwegian spent fuel. May 2020

Leupin, O.X., Smith, P., Marshall, P., Johnson, L., Savage, D., Cloet, V., Schneider, J., & Senger, R. (2016). High-level waste repository-induced effects, Nagra Technical Report 14-13. Nagra, Wettingen, Switzerland, October 2016

McGinnes, D.F. (2002). Model Radioactive Waste Inventory for Reprocessing Waste and Spent Fuel, Nagra Technical Report 01-01. Nagra, Wettingen, Switzerland, December 2002.

Meleshyn, A., Noseck, U. (2012). Radionuclide Inventory of Vitrified Waste after Spent Nuclear Fuel Reprocessing at La Hague - Basic Issues and Current State in Germany, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, GRS-294

Nagra (2002). Project Opalinus Clay, Safety Report, Demonstration of disposal feasibility for spent fuel, vitrified high-level waste (Entsorgunsnachweis), Technical Report 02-05. Nagra, Wettingen, Switzerland, December 2002

Nagra (2016). ENSI-Nachforderung zum Indikator “Tiefenlage im Hinblick auf bautechnische Machbarkeit” in SGT Etappe 2, Prüfung der Lager- und Barrierenkonzepte, Arbeitsbericht NAB 16-42. Nagra, Wettingen, Switzerland, July 2016 (in German)

Patel, R. et al. (2012). Canister Design Concepts for Disposal of Spent Fuel and High Level Waste, Nagra Technical Report 12-06. Nagra, Wettingen, Switzerland, October 2012.

Posiva (2012). Canister Production Line 2012 - Design, Production and Initial State of the Canister. POSIVA 2012-16. Eurajoki, Finland, Posiva Oy.

Posiva (2012a). Safety Case for the Disposal of Spent Nuclear Fuel at Olkiluoto - Performance Assessment 2012. POSIVA 2012-04. Eurajoki, Finland, Posiva Oy

Posiva (2016). Presentation by Tiina Jalonen in Nuclear Science and Technology Symposium 3rd of November 2016.

Posiva (2020). Multiple barriers helps to ensure that no nuclear waste will be released to living nature. Webpage: http://www.posiva.fi/en/final_disposal/basics_of_the_final_disposal#.XmdraxRS-Uk. Accessed 6.11.2020.

Raiko, H. (2005). Disposal Canister for Spent Nuclear Fuel – Design Report. POSIVA 2005-02. Posiva Oy, Eurajoki, Finland.

Raiko H., Sandström R., Rydén H., Johansson M. (2010) Design analysis report for the canister. Technical report TR-10-28, Swedish Nuclear Fuel and Waste Management Co., Stockholm, Sweden.

Raiko, H. (2012). Canister Design 2012. POSIVA 2012-13, Posiva Oy, Eurajoki, Finland.

RWM (2016). Geological Disposal - Waste Package Evolution Status Report. NDA Report no. DSSC/451/01. Radioactive Waste Management Limited (RWM), Didcot, UK, December 2016.

RWM (2017). Geological Disposal - Concept Status Report. NDA/RWM/155. Radioactive Waste Management Limited (RWM), Didcot, UK, October 2017.

SKB 1977. Handling of Spent Nuclear Fuel and Final Storage of Vitrified High Level Reprocessing Waste. Sweden 1977, report series available in parts (accessed at 6.11.2020):

- https://www.skb.se/publikation/13748/KBS721webb.pdf
- https://www.skb.se/publikation/4377/722%20KBS-1%20Eng%201.pdf
- https://www.skb.se/publikation/1678130/722%20KBS-1%20Eng%20Suppl.pdf
- https://www.skb.se/publikation/1678128/722%20KBS-1%20Eng%204.pdf
- https://www.skb.se/publikation/1678118/722%20KBS-1%20Eng%203.pdf
- https://www.skb.se/publikation/1678116/722%20KBS-1%20Eng%202.pdf

SKB 2010. Design, production and initial state of the canister. Svensk Kärnbränslehantering AB: Sweden, Stockholm. Technical Report TR-10-14.

Verhoef, E.V., Neeft, E.A.C., Deissmann, G., Filby, A., Wiegers, R.B., Kers, D.A. (2016). Waste families in OPERA. OPERA-PG-COV023. March, 2016

X.Zhang and G.Pan (2020). Collapse of thick-walled subsea pipelines with imperfections subjected to external pressure. Ocean Engineering, Volume 213 (2020), https://doi.org/10.1016/j.oceaneng.2020.107705

split

A-Insinöörit Civil Oy
Bertel Jungin aukio 9
FI-02600 Espoo
Tel.: +358 207 911 888
www.ains.fi/en