RESEARCH REPORT

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Waste Management of Small Modular Nuclear Reactors in Finland

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Summary

Small modular nuclear reactors (SMRs) represent advanced technology in nuclear energy aiming to produce low carbon energy at smaller unit size and enhanced passive safety in comparison to traditional nuclear power plants (NPPs). The management of spent nuclear fuel (SNF) and low- and intermediate-level waste (LILW) from SMRs is an issue that needs to be resolved as part of any deployment of SMR technology in Finland. Currently, spent nuclear fuel from NPPs in Finland is planned to be disposed in the ONKALO® deep geological repository applying the KBS-3V disposal concept. This concept should be applicable for spent fuel from SMRs using light-water reactor (LWR) technology. However, there are some differences in the waste forms, most obviously the length of the fuel assemblies, but also in the spent fuel characteristics that need to be considered in the further development of the concept for spent fuel from SMRs.

Preliminary 2D calculations were made with the continuous-energy Monte Carlo code Serpent to compare the spent fuel characteristics from two example LWR-SMRs to spent nuclear fuel from currently operating NPPs in Finland. In one example case, a NuScale Power Module™ was considered as it is one of the most advanced LWR-SMRs in the world. The other example case is an SMR planned in Finland for district heating purposes. The main differences between the SMR and NPP spent fuels are linked to lower burnups in the SMRs. Lower discharge burnups are to be studied further from the point of view of criticality safety at disposal. Otherwise, the lower average discharge burnup of these SMR fuel types, in principle, generally tends to make the handling of spent fuel assemblies less demanding with respect to the decay heat and ionizing radiation emitted from the assembly. However, rigorous calculation of the dose rates would require 3D calculations to determine the axial burnup distribution within a fuel assembly, which was outside the scope of this study.

Published studies indicate that possibly larger masses (per GWe-year) of SNF and other HLW and larger volumes (per GWe-year) of LLW will be produced in a LW-SMR compared to a large NPP. However, because of the lower decay heat in the SMR SF (due to the lower burnup), less excavated volume and, consequently, less clay-based filling material (deposition tunnel backfill) may be needed in a repository.

Depending on the number of SMR units located at sites in Finland, the amounts of spent fuel and other waste streams can be relatively small so that a centralised waste management facility and repository could be the most feasible option for processing and disposal of all the nuclear waste. Alternatively, the wastes can be disposed of locally (near SMR sites in smaller facilities) or a hybrid model, where, e.g., only SNF is disposed centrally, could be considered. These alternatives will depend strongly on the ownership structure of the SMRs deployed in Finland. Local stakeholder and public opinion will be very important as well. Other issues, such as geological suitability of the SMR sites for disposal, transport and interim storage will need to be assessed.

In terms of final disposal of SNF from LWR-SMRs, the only currently available option is the KBS-3V concept, especially considering the state of the licencing process for this concept in Finland.

Deep borehole disposal represents an intriguing, particularly in the case of local disposal for relatively small amounts of waste, but not yet fully developed alternative. The suitability of deep borehole disposal in the crystalline rock conditions prevailing in Finland will be studied in the next phase of the project.

Spent fuel from non-LWR SMRs, i.e., high-temperature-gas-cooled, fast neutron-spectrum and molten salt-type SMRs, was also discussed briefly. Challenges were identified in the pre-treatments needed for SNF from these reactors prior to disposal including lack of suitable facilities in Finland and potential proliferation issues. In some cases, e.g., reactors with graphite moderators, the disposal of the LILW waste streams was considered problematic as the current methodologies in use in Finland for disposal of LILW would not be applicable. More extensive studies would be required to specifically identify the waste streams from non-LWR SMRs and how the waste characteristics would need to be taken into account for disposal.

Confidentiality	Public

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Preface

This report has been compiled at VTT from April 2021 to January 2022 as part of the KYT 2022 Finnish Research Programme on Nuclear Waste Management. The objective of this work was to perform an initial study on the management of spent fuel and nuclear waste produced in a SMR unit in Finland. The project did not have a named steering group, but the steering took place by the KYT2022 stakeholders in the KYT2022 board meeting held 21.4.2021 and in the first project seminar that was arranged in June 2021.

The calculations concerning spent nuclear fuel characteristics were performed by Pauli Juutilainen and supervised by Silja Häkkinen and were realized in cooperation with the KYT project KÄRÄHDE. Ville Tulkki as the expert of SMR technology reviewed the report. Paula Keto worked as a project manager of the project and contributed as a waste management and engineered barrier specialist together with Timothy Schatz and Sami Naumer.

We would like to thank all KYT Board members and especially Linda Kumpula and Jaakko Luovanto from TEM, Jaakko Leino and Ville Koskinen from STUK, Annukka Laitonen from TVO, Pasi Kelokaski from Fortum and Heikki Hinkkanen from Fennovoima. In addition, we would like to thank Kalev Kallemets from Fermi Energia, Chris Parker from Deep Isolation EMEA and Juhani Hyvärinen from Lappeenranta University of Technology for valuable input.

Espoo 20.2.2022

Paula Keto, Pauli Juutilainen, Timothy Schatz, Sami Naumer & Silja Häkkinen.

Contents

1. Introduction

Background

Small modular nuclear reactors (SMR) are globally considered to be future, low carbon energy solutions, for the production of electricity and heat. SMRs are typically defined as nuclear power plants with electrical power output of less than 300 MW (megawatts) (OECD NEA 2016, STUK 2019, IAEA 2020) or heating power of < 1000 MW (Ydinturvallisuusneuvottelukunta 2019). The size of the unit can vary from something quite small (e.g., some tens of MW up to the maximum 300 MW. For comparison, the electrical power produced currently by Finnish nuclear power plants varies from 500 to 1600 MW (STUK 2019). Because of their small size, SMRs, meaning the reactor and other parts belonging to the plant, are envisioned as standardised products (Tulkki et al. 2017) that can be manufactured in a factory using serial production and transferred to the site as complete modules (STUK 2019). The small size also allows for several SMRs to be installed next to one another forming together a larger power plant unit (STUK 2019).

In Finland, SMR technology is being considered, e.g., for district heating plants currently operating with fossil fuels (coal and peat) (Tulkki et al. 2017). SMRs could be directly located in different parts of Finland close to a city or a large industrial plant requiring its own energy source. From a waste management point of view, this decentralised operation could mean that there could be several operators and small repositories handling the generated spent fuel and other radioactive waste. Alternatively, the waste management could be handled in a more centralised manner. Independent of how the management is organized, the spent nuclear fuel and other radioactive waste produced by SMRs shall remain in Finland (STUK 2019). According to current Finnish legislation, the spent fuel cannot be transported back to the country of origin (STUK 2019).

Water cooled SMRs (referred later in this report as light water reactors, LWR-SMRs) rely on the same kind of technology as the currently operating nuclear power plants in Finland (pressurised water and boiling water reactors). It has been assumed that the spent fuel management can be implemented following the same principles and methods that are in use for the NPPs and will be used in the KBS-3V type of deep geological repository in Finland (Ydinturvallisuusneuvottelukunta, Finnish Nuclear Safety Advisory Board 2019). However, it is anticipated that there will be some differences concerning the spent fuel properties and waste forms that will need to be taken into account in the design of engineered barriers, etc. Considering other operational and decommissioning waste produced in a LWR SMR unit, there may also be some differences that affect the handling of disposal of the waste. There is also, most assuredly, a significant difference in the number of spent fuel rods and volume of the waste produced annually in a single LWR-SMR in comparison to a large NPP.

There are other types of small modular reactors than LWR designs being planned for commercial use, and the fuel itself and technologies used in these systems may differ significantly from that of water-cooled reactors. According to the Finnish Nuclear Safety Advisory Board (Ydinturvallisuusneuvottelukunta 2019) in these cases, long-term international R&D would be required in order to develop safe handling, storing and disposal of the nuclear waste produced.

Safe management of the nuclear waste produced by SMRs has been identified as an important part of the development and licencing of SMRs internationally and at the EU-level (as identified in the EURAD Strategic Research Agenda SRA). Christophe Xerri, Director of the Division of Nuclear Fuel Cycle and Waste Technology at IAEA (2019) has stated that "solutions for managing spent fuel and radioactive waste arising from SMRs will be one of the most important factors to take into account when choosing a technology, along with the security of fuel supply." In response to preliminary information from the Complementary Delegated Act (CDA) of the EU Taxonomy Regulation, future investments in nuclear energy require that waste management needs to be properly funded, there needs to be operating disposal facilities for low and intermediate level waste streams and a plan for disposal of high-level waste to be in operation by 2050 and that there is notable progress in realisation of these goals (WNN, 2022b).

The work presented in this report has links to the ELSMOR EC-project (Hashymov et al. (2022) and follows national projects such as EcoSMR Ecosystem for Small Modular Reactors, VN-TEAS Possibilities and challenges linked to new nuclear energy technologies and KÄRÄHDE Spent Fuel Characterization and Source Term.

Objectives

The overall objective of this work is to study how spent fuel and other nuclear waste produced in a SMR could be managed in Finland and what aspects should be considered when planning the disposal. The following specific aims have been set for this work:

- Estimate the amount and characteristics of the spent fuel and other nuclear waste produced in a LWR SMR (both for production of electricity and district heat).
- Compare the properties and amount of the spent fuel and waste streams from NPPs currently operating in Finland.
- Study waste forms (LWR and selected non-LWR SMRs).
- Establish alternatives for organising the management of SMR waste in Finland.
- Describe needs for developing disposal concepts and engineered barriers for SMR waste management.
- Identify the most critical topics for further studies in 2022 and beyond in order for SMR waste management to progress towards safe implementation.

Limitations

The calculations presented in this report are limited to water cooled reactors. Waste forms are discussed for LWR and preliminarily for non-LWR SMRs. Licencing and possible regulatory changes are discussed only briefly within this report, since these topics will be addressed in more detail by the VN-TEAS project “Possibilities and challenges linked to new nuclear energy technologies”.

Structure of this report

The report is divided into three main parts:

- Calculations and characteristics of the spent fuel produced in LWR SMRs (chapter 2),
- Other waste streams produced in a LWR SMR (chapter 3),
- Waste management strategies, repositories, disposal concepts and engineered barriers (chapter 4-7), and
- Spent fuel and waste produced in non-LWR SMRs (chapter 8).

2. Spent fuel

2.1 Introduction

This chapter focuses on spent fuel produced in light-water reactor type SMRs. Non-LWR type spent fuel is discussed in chapter 8 (Non-LWR Spent fuel). Two LWR-SMR reactors have been selected for purposes of comparing the spent nuclear fuel (SNF) from currently operating NPPs in Finland to that of the SMRs from the waste management point of view. The objective of this chapter is to assess the applicability of the current spent fuel management concept (KBS-3, see section 5 Final disposal concepts) for the SNF produced in LWR-SMRs and identify any significant differences, for example in specific waste characteristics (e.g., residual heat production) or amount of SNF generated relative to energy produced. These comparisons were carried out through a literature review on waste streams produced in these different reactors and by calculations aiming to compare other waste characteristics presented in the following subchapters.

2.2 Comparison cases

2.2.1 Spent fuel from currently operating NPPs in Finland

Spent fuel is currently generated in Finland in nuclear power plants (NPP) located in Eurajoki (BWR-type, OL1-2 units operated by TVO) and in Loviisa (PWR (VVER)-type, LO1-2 units operated by Fortum). In addition, a third Olkiluoto unit (EPR/PWR, OL3) will start its operations in 2022.The fuel assemblies used in these reactors are presented in Figure 2-1. After reaching the end of its useful life, spent fuel is removed from the reactors and placed into interim storage pools. Spent fuel management plans in Finland indicate that selected spent fuel assemblies are eventually removed from interim storage and transported to an encapsulation facility. At this facility, sets of assemblies are placed into copper canisters containing cast iron inserts which hold the spent fuel assemblies in their proper positions in the canisters and provide additional mechanical support. When the assembly positions are filled, copper lids are placed on the canisters and sealed in place. The sealed canisters are the spent fuel waste packages for final disposal in the Finnish geological disposal facility (GDF).

The fuel assemblies used in the different NPPs in Finland differ in their horizontal and vertical dimensions and therefore the corresponding cast iron inserts and copper canisters will differ as well (see Figure 2-2). However, the copper overpack is identical for all three variants with the exception of length (Raiko, 2013).

The dimensions and characteristics of the fuel assemblies are presented in Table 2-1 and in Figure 2-3 for the OL3 assembly. According to Raiko (2013), the length of fuel rods and assemblies may grow in length during the reactor irradiation some 10 to 25 mm depending on the burnup. In the following chapter, these dimensions are compared to the fuel assemblies of a water cooled SMR unit. For a design value, the decay heat of the refence canister for a KBS 3 repository is set to 1700 W (Raiko 2013).

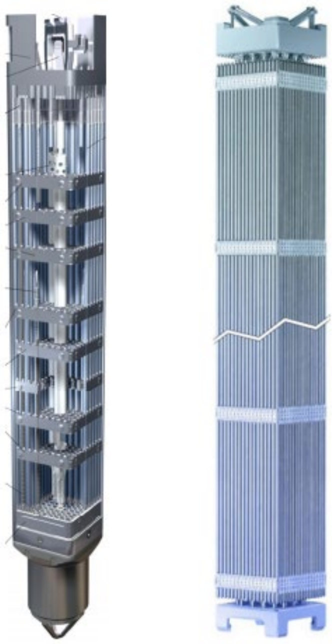
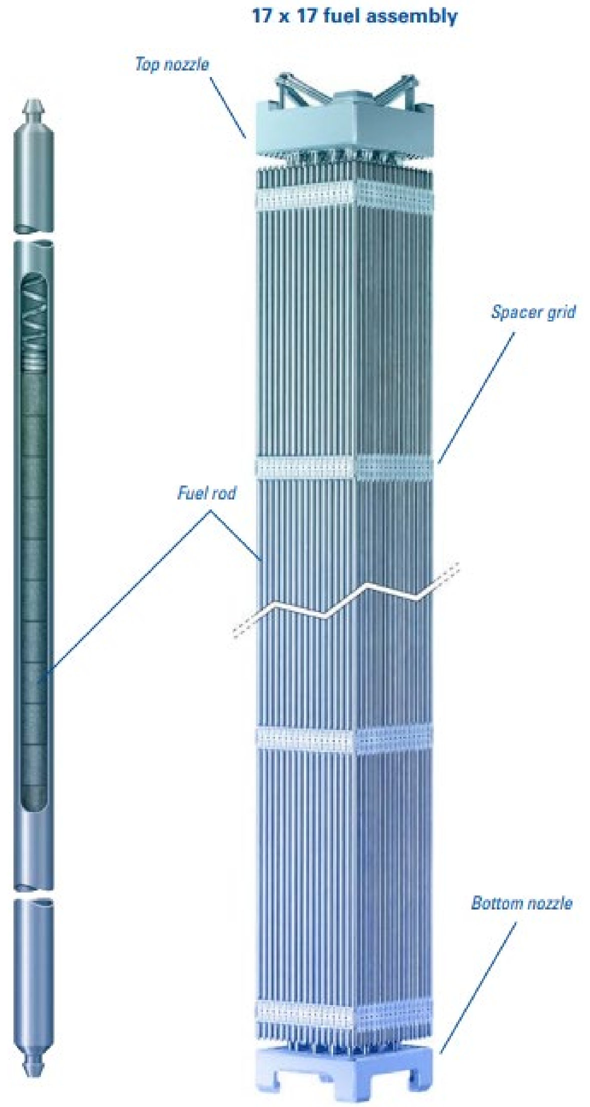


Figure 2-1. Fuel assemblies (not to scale) from Olkiluoto 1-2 (BWR) and Olkiluoto 3 (EPR/PWR) and Loviisa 1-2 (VVER-440) nuclear power plants units. Figure by Posiva (n.d.)Posiva.

Figure 2-2. Illustration of copper canisters and cast-iron inserts for the spent fuel from Loviisa LO1-2 unit (left), Olkiluoto OL1-2 unit (middle) and from Olkiluoto OL3 unit (right). From Raiko (2013).



Fuel properties

Fuel	uranium dioxide UO_2
Fuel assembly type	17 x 17 HTP
Number of fuel rods per assembly	265 pcs
Number of guide thimbles per assembly	24 pcs
Number of spacer grids per assembly	10 pcs
Length of fuel assembly	4.8 m
Weight of fuel assembly	735 kg
Width of fuel assembly	213.5 mm
Cladding material	M5™
UO_2 pellet density	10.45 g/cm^3
Fuel discharge burnup	45 MWd/kgU

Figure 2-3. Properties of OL3 fuel. From TVO (2010).

Table 2-1. Dimensions and fuel characteristics for spent fuel from the Olkiluoto and Loviisa NPPs (Raiko, 2013)

FUEL TYPE	Asea-Atom BWR	VVER-440 PWR	EPR/PWR
Assembly sectional configuration	square	hexagonal	square
Length of assembly (m)	4.127	3.217	4.865**
Sectional dimension (mm)	139*	144	215
Number of rods per assembly	63 - 96	126	265
Mass of uranium (kg)	172 - 184	120 - 126	530 - 533
Total assembly mass (kg)	292 - 331	210-214	785
Fuel channel dimension (mm)	139 (square)	144 (hexagonal)	No channel
Total length with fuel channel (m)	4.398-4.421	3.217***	4.865**
Anticipated maximum average burnup of a fuel element (MWd/kgU)	55	57	55
Estimated average burnup of all the fuel to be disposed (MWd/kgU)	39 - 40	40 - 41	45-46
Typical enrichment U-235 (%)	3.3 – 4.4	3.6 - 4.4	1.9 - 4.9
Minimum cooling time of a single fuel element (years)	20	20	20
Minimum average cooling time for encapsulation with average burnup (years)	33	30	42
Allowable average decay heat at disposal (full canisters) (W/tU)	806	950	862

*) *The top end handle of the BWR fuel assembly has some more extensive details, whose maximum sectional dimension is 151 mm.*

**) *The fuel element has leaf springs on the top tie-plate that extends the total length with some tens of millimetres. The leaf springs are planned to be removed before disposal. Geometric details of the fuel element are to be confirmed later. Control rod crown is not included in the EPR/PWR fuel elements that are to be disposed. The weight of a set of 24 absorber rods is roughly 55 kg.*

***) *Length of the control rod follower assembly is 3.200 m.*

2.2.2 Spent fuel from the NuScale reactor

There are two example cases for LWR-type SMRs discussed further in this chapter; one designed for electricity production and the other for heat production. The SMR comparison case for electricity production is the NuScale Power Module^{TM}. This SMR is one of the most advanced SMR reactors under development and it is estimated to be commercially available in 2025-2030 (https://en.wikipedia.org/wiki/NuScale_Power). The NuScale Power Module^{TM} (NPM) is described as a fully factory-fabricated SMR design that is capable of generating 60 MW of electricity using pressurized water reactor technology (NuScale, 2020c). The basic configuration of a single NPM is shown schematically in Figure 2-4.

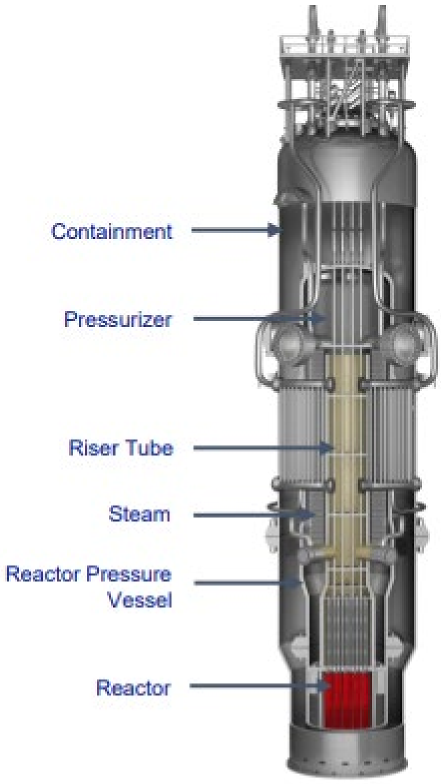


Figure 2-4. Schematic diagram of a NuScale Power Module™ which is an integrated package that in-cludes the reactor core, two interwoven helical coil steam generators, and a pressurizer within the reactor pressure vessel (NuScale, 2020c). The entire system is housed in a steel containment vessel. The dimensions of the module are approximately 25 m (height) by 5 m (width) NuScale (2020c).

Technical details of the reactor are presented by NuScale (2020b)[1]. The reactor core within each of the NPMs consists of 37 fuel assemblies and 16 control rod assemblies. The fuel assemblies are a modified Framatome HTP2™ fuel design, which is currently being used in existing PWRs. The new fuel, named NuFuelHTP2™, is different from Framatome's proven HTP™ fuel only with respect to the fuel assembly length, which is half-height. The ceramic UO_2 pellets are enriched to up to 4.95 percent and are encapsulated in M5® cladding material with an active fuel length of approximately 2 meters. The temperature coefficient of reactivity of the core is also negative. Fuel rods with gadolinium oxide (Gd_2O_3) as a burnable absorber homogeneously mixed within the pellet are used in specific locations to establish favourable radial power distribution. All rods are aligned using Framatome's HTP™ and HMP™ grid spacers. Lastly, the core is surrounded by a stainless-steel heavy neutron reflector to improve fuel utilization and prevent the radial escape of neutrons. The reflector also serves as an envelope to the core and directs the flow through it. While there is no specific design limit on cycle average burnup, the core average cycle exposure is designed such that the peak fuel rod exposure is up to 62 GWd/MTU. NPMs are designed to be refueled on a 24-month cycle (equivalent to a 12 GWd/MTU cycle).

According to NuScale (2020b), the fuel assembly is a 17x17 array of fuel rods that has been designed specifically for use with the core configuration of the NuScale reactor. The fuel assembly uses five spacer grids, 24 guide tubes, and a top and bottom nozzle, to provide the structural support for the 264 fuel rods. Each fuel assembly also includes a central instrument tube. The main NPM reactor core parameters are listed in Table 2-2.

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Mechanical and thermal-hydraulic testing of NuFuel-HTP2™ assemblies has been completed using Framatome's test facilities (NuScale, 2020b). There is a vast amount of experience regarding the performance of the fuel, cladding, and structural materials in the fuel assemblies during operations, both wet and dry storage, and eventual disposal. The fuel assembly radionuclide composition and radiotoxicity are well understood, which will facilitate transport to an interim storage facility and ultimately final disposal.

Table 2-2. Selected NuScale Core Design Parameters (NuScale, 2020b).

Parameter	Value
Core	
diameter of active core	4.94 ft (150.6 cm)
number of fuel assemblies	37
height-to-diameter ratio of active core	1.33
total cross section of active core	18.42 ft^2 (1.711 m^2)
Reflector	
height	91.75 in (233.0 cm)
width	2.5 to 12.2 in (6.4 to 31.0 cm)
Fuel Assembly	
fuel design	NuFuel-HTP2™
length	95.89 in (243.6 cm)
rods per assembly	264
fuel assembly pitch	8.466 in (21.50 cm)
fuel rod pitch	0.496 in (1.260 cm)
Fuel Rod	
peak rod exposure core design criteria for UO_2 rods	62 GWd/MTU
Gd_2O_3 concentration	up to 8%
cladding outside diameter	0.374 in (0.950 cm)
cladding inside diameter	0.326 in (0.828 cm)
cladding thickness	0.024 in (0.061 cm)
pellet-cladding diametral gap	0.0065 in (0.0165 cm)
cladding material	M5®
fuel column length	78.74 in (200.0 cm)
overall fuel rod length	85 in (216 cm)
fuel pellet material	uranium dioxide (UO_2)
fuel pellet diameter	0.3195 in (0.8115 cm)
fuel pellet length	0.4 in (1.0 cm)
fissile enrichment	< 4.95%
Control Rod Assembly	
total length	94.37 in (239.7 cm)
neutron absorber material	B_4C and AgInCd
clad material	304 SS
number of absorber rods	24
Control Rod	
outside diameter	0.381 in (0.968 cm)
inside diameter	0.344 in (0.874 cm)

2.2.3 Spent fuel from an SMR designed for district heating

The SMR considered for heat production is described by (Leppänen, Hillberg, et al., 2021; Leppänen, Valtavirta, et al., 2021) and Komu et al. (2021). This LW-SMR is envisioned to be used in Finland for production of district heat in order to replace coal fired district power plants with low-carbon energy. The SMR heating plant is planned to consist of one or multiple 50 MW reactor modules with natural circulation and operating temperatures of approximately 120°C (Leppänen, Hillberg, et al., 2021).

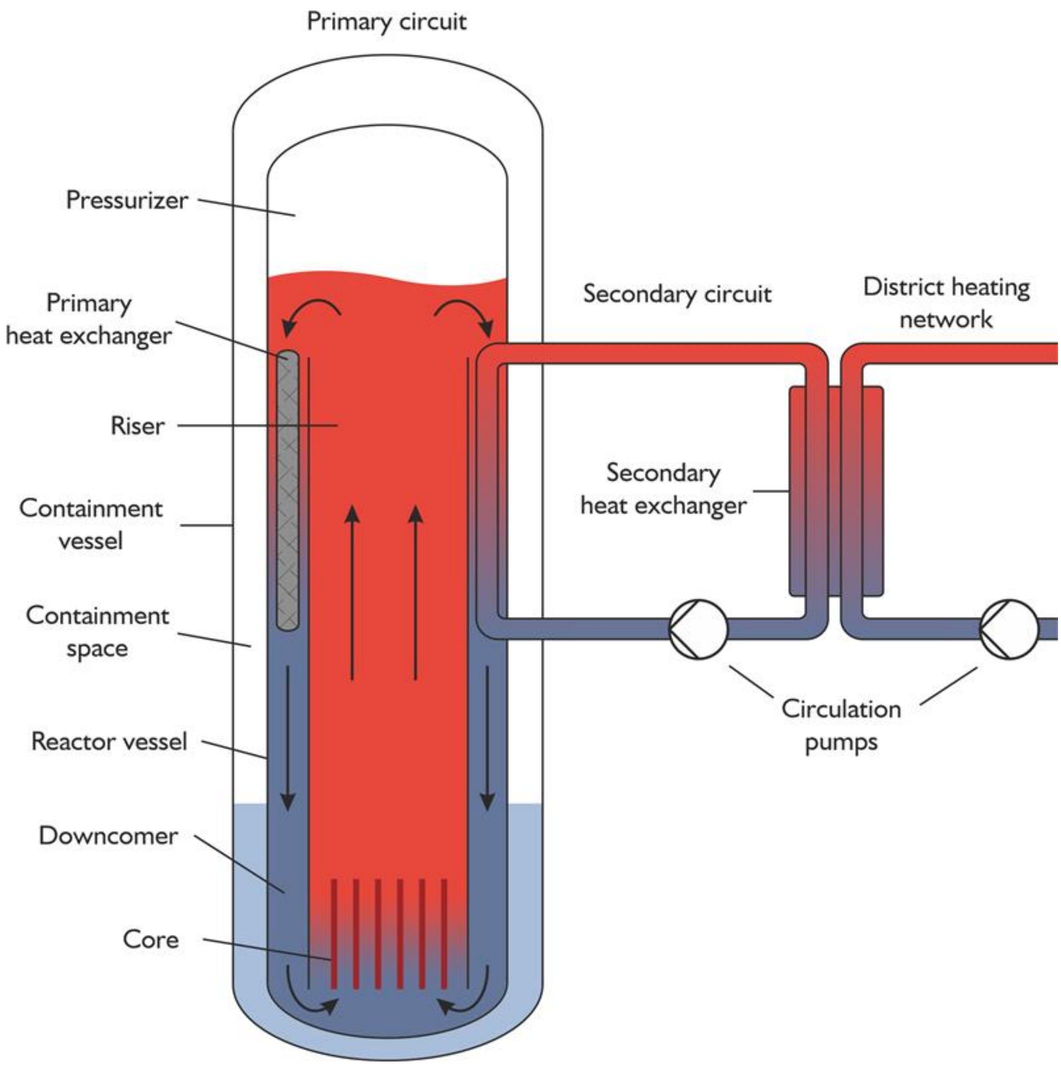

Figure 2-5. Schematic diagram of the Finnish LDR-50 LW-SMR planned to be used for district heating. Illustration copyright of VTT.

The LDR-50 LW-SMR unit for district heating has a fuel assembly design based on a Westinghouse AP1000-like 17×17 standard PWR assembly, truncated to an active length of 120 cm (Leppänen, Valtavirta, et al., 2021). The cross-section is similar to that of the EPR-PWR fuel assembly for the OL3 NPP and the NuScale SMR.

Table 2-3. An overview of the Finnish LDR-50 LW-SMR reactor parameters (Leppänen et al., 2022).

Parameter	***Value***
Fuel Assembly	
Assembly type	*17x17 PWR (AP1000)*
Number of fuel rods	*264*
number of control rod guide tubes	*24*
Number of instrumentation tubes	*1*
Fuel enrichment	*1.5 / 2.4 %*
Burnable absorber	Gd_2O_3
Control rod absorber	*Inconel-718 / AIC /* B_4C
Core	
Number of fuel assemblies	*37*
Assembly pitch	*21.504 cm*
active height	*100 cm*
Cycle length	*580 effective full-power days*
TH Boundary conditions	
Nominal power	*50* $MW_{(t)}$ */ -*
Coolant circulation	*Natural convection*
Coolant/ moderator	*Light water*
Pressure	*0.3–0.7 MPa*
Core inlet/outlet temperature	*70–95 °C / 100–155 °C*

2.3 Calculations

Calculations to characterize the spent fuel properties of the fuel types interesting with respect to the current and anticipated near-future nuclear fleet of Finland have been performed and reported within the KYT2022/KÄRÄHDE project (Juutilainen, 2021; Juutilainen & Häkkinen, 2019). For the present study, similar calculations were run for the fuel assemblies resembling the NuScale design and the district heating reactor (LDR-50). In the scope of this study, the small reactor results are compared to those of EPR due to the highly similar geometry of the three: all 17 x 17 squared assemblies with equal pattern of control rods and instrumentation tubes and highly similar radial fuel rod dimensions.

2.3.1 Computational models

Similarly to the previous calculations, the continuous-energy Monte Carlo code Serpent (Leppänen et al., 2015) (version 2.1.32) was used for the burnup calculations with various uranium enrichment and burnable absorber contents in infinite two-dimensional (2D) geometry. Practically, it is assumed that the assembly is radially surrounded by fully identical assemblies and the neutron leakage in the axial direction from the ends of the assembly is completely ignored. Such a 2D calculation provides a reasonable estimate for the spent fuel composition in the axially middle section of the assembly and if the assembly is located away from the core periphery for most of the irradiation time.

The model naturally ignores the end-effects that occur for the neutron flux in the top and bottom of the reactor core, as well as the potentially dissimilar adjacent fuel assemblies – and particularly lack of ones in the radial periphery of the core. The applied geometry model is illustrated in Figure 2-6 that shows the fuel and control rod pattern for an LDR-50 assembly and a NuScale type assembly. These examples differ from each other with respect to the number of fuel rods doped with gadolinium (Gd).

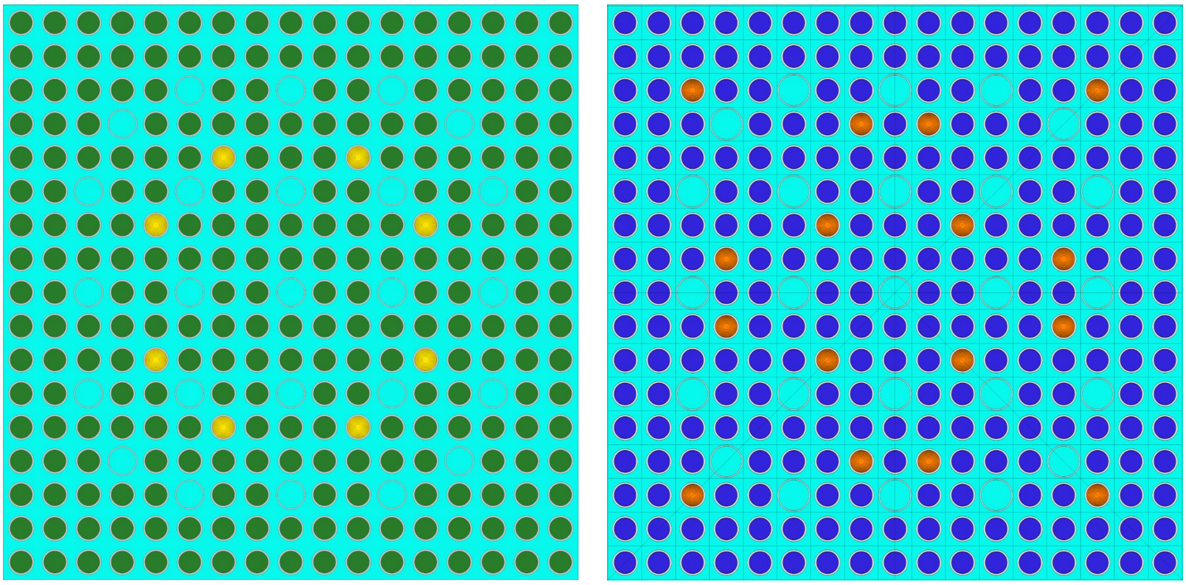

Figure 2-6. Geometry of the computational model plotted by Serpent. The left figure depicts the LDR-50 assembly containing 8 Gd-doped rods and the right one presents the NuScale model assembly with 16 Gd rods.

Calculations were run for a few cases resembling the NuScale fuel assemblies and various designs of the district heating reactor being under development at VTT. Some general irradiation parameters applied in the calculations are presented in Table 2-4. For simplicity, all parameters were kept constant, which is not true especially for the soluble boron concentration. Moreover, the current design of LDR-50 assumes relatively active use of control rods, in contrast to PWRs in general, which was also ignored in these calculations. Concerning the heating reactor, models representing both the earlier design with erbium (Er) and the latest one with gadolinium (Gd) burnable absorbers were applied. Burnable absorber is generally used to compensate for the excess reactivity of the fresh fuel in the early phase of irradiation. As the name suggests, the absorber depletes along with the irradiation and thus contributes to increasing the reactivity, when the depletion of fissile U-235 decreases it. In the former model, Er was evenly mixed into fuel. Four cases with 0.41…2.35 wt-% Er in the fuel with 3.5 wt-% enriched U-235 were calculated. In case of Gd, all modelled assemblies contained 8 Gd rods with Gd content 5–9 % and U-235 enrichment 1.5 % or 2.4 %. More accurately, the assembly with 1.5 % U-235 and 6.0 % Gd in Gd rods is irradiated over one cycle only, whereas the assemblies with 2.4 % U-235 and 5 % or 9 % Gd are irradiated over three cycles.

The assemblies of NuScale model are based on the reactor's FSAR documentation (NuScale, 2020b) as much as possible and the remaining parameters were based on iteration and educated guesses performed within EU-McSAFER project (Valtavirta et al., 2020). Concerning the 2D assembly calculations, the number and positioning of Gd rods and their Gd content were the significant topics not provided in the NuScale FSAR document. Furthermore, the documentation provides core data with U-235 enrichment for equilibrium cycles only, so the assemblies used in the first loading had to be defined by ourselves. For that purpose, assemblies of 1.5 %, 1.6 %, 2.5 % and 2.6 % enriched U-235 were constructed.

The equilibrium loading consists of assemblies with 4.05 % and 4.55 % enriched U-235, the latter of which contains some Gd. The exact parameters were defined to be 8 % of Gd in 16 rods. The U-235 enrichment was 4.55 % in both Gd and non-Gd rods. One of the assemblies uses 2.6 % U-235, since it is irradiated over one cycle only.

The average burnup of NuScale is 12 GWd/MTU per cycle, resulting 36 GWd/MTU for the three-cycle assemblies. As mentioned above, the local peak burnup can be up to 62 GWd/MTU. The calculations were run to 50 GWd/MTU for these assemblies. For those irradiated shorter times, the calculations were run up to 20 GWd/MTU (1.5 % and 1.6 % U-235) and 30 GWd/MTU (2.5 % and 2.6 % U-235). The discharge burnup for LDR is supposed to be significantly smaller: ~6 GWd/MTU for one-cycle assembly and, consequently, ~18 GWd/MTU for three-cycle assemblies. The calculations on these were run to 40 GWd/MTU.

The JEFF-3.2 cross-section data was used for transport calculations and JEFF-3.1.1 provided the fission yield and decay data. Rather dense burnup step division was applied in all calculations, mostly 1 GWd/MTU or less, and never more than 2.5 GWd/MTU. Burnup step length division in the various assemblies was slightly different, but these differences were assumed to have no impact on the results. The number of neutron histories was 10 million for EPR, 50 million for NuScale and 20 million for LDR-50 assemblies. As a result, the respective average computational standard deviations for the multiplication factor k_{eff} were roughly 16 pcm, 7 pcm and 10 pcm, which can be considered negligible compared to other approximations and uncertainties related to the study.

Table 2-4. Operating parameters applied in calculations over the whole irradiation.

	EPR	NuScale	LDR-50
Fuel temperature (K)	900	900	533
Clad and moderator temperature (K)	585	600 (clad) 557 (mod)	360
Boron concentration (ppm)	600	600	0
Specific power (W/gU)	33.6	17.03	10.7
Moderator density (g/cm^3)	0.70046	0.753264	0.96749

2.3.2 Results

As the scope of the study is to compare the spent fuel characteristics between small and large reactors, the results are mainly presented for the discharge burnup that is assumed to be close to the realistic value of each assembly. The burnup for multi-cycle assemblies was assumed to be 45 GWd/MTU for EPR, 40 GWd/MTU for NuScale and 20 GWd/MTU for LDR-50. The calculations on the EPR assemblies are reported into further detail in the VTT research report (Juutilainen, 2021).

Table 2-5. Assembly type identifiers used in following figures. Gd content was 8 % Gd_2O_3 in EPR and NuScale fuels.

ID	Assembly type	Reference discharge burnup (GWd/MTU)
f1	EPR 1.9 % U-235, no Gd	15
ol5	EPR 4.5 % U-235, 16 Gd rods	45
ol8	EPR 4.0 % U-235, no Gd	45
ns260	NuScale 2.60 % U-235, no Gd	12
ns405	NuScale 4.05 % U-235, no Gd	40
ns455	NuScale 4.55 % U-235, 16 Gd rods	40
ldr1	LDR-50 1.5 % U-235, 8 Gd rods with 6 % Gd	7
ldr2	LDR-50 2.4 % U-235, 8 Gd rods with 5 % Gd	20
ldr3	LDR-50 2.4 % U-235, 8 Gd rods with 9 % Gd	20

2.3.2.1 Reactivity after unloading

It has been concluded that the use of burnup credit (BUC) is required when considering the criticality safety of the EPR canisters of the KBS-3 concept in the final repository (Anttila, 2005). This suggests that ensuring the criticality safety for the similar fuel designs of NuScale and LDR-50 requires particular care. In this study, the multiplication factor k_{eff} was calculated following the discharge to roughly evaluate the reactivity behaviour over the millennia. The effective multiplication factor is used as the measure of criticality in the studied system. A simplified definition for keff 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. Typically, the regulatory requirements state that calculations must result in $k_{eff} < 0.95$ in all storage and transport configurations. However, the calculations presented in this section were performed in the same geometry as the ones over irradiation, i.e. no canisters and the associated dimensioning were taken into account. Hence, the results are indicative at best and most importantly, the k_{eff} will be much smaller in a realistic repository configuration than obtained in these calculations.

The assemblies were still assumed to be fully immersed in pure water with 1.0 g/cm^3 density and a temperature of 300 K was applied for the whole system in the calculations. As an exception to the discharge burnups of the study presented in Table 2-5, these calculations were run for somewhat lower burnups that compose the area of most interest in criticality safety analyses. The k_{eff} curves for the LDR-50 assemblies are presented in Figure 2-7 for discharge burnups of 7 and 13 GWd/MTU. The effect of Gd depletion can be observed in the comparison between the LDR assemblies with same U-235 enrichment but different Gd contents.

The respective curves for the NuScale type assemblies with U-235 over 4 % are plotted in Figure 2-8, For comparison, the k_{eff} of the similar EPR assemblies is presented in the same figures. Particularly, the latter plots remind that the peak k_{eff} can be found at thousands of years after the discharge. With respect to the k_{eff} values showing major supercriticality in some cases, it is worthwhile to note the above-mentioned computational simplification that the calculations were performed in the same infinite lattice geometry as the calculations over irradiation. Hence, these values do not predict the realistic k_{eff} in repository, but they demonstrate the differences between various fuel and reactor types.

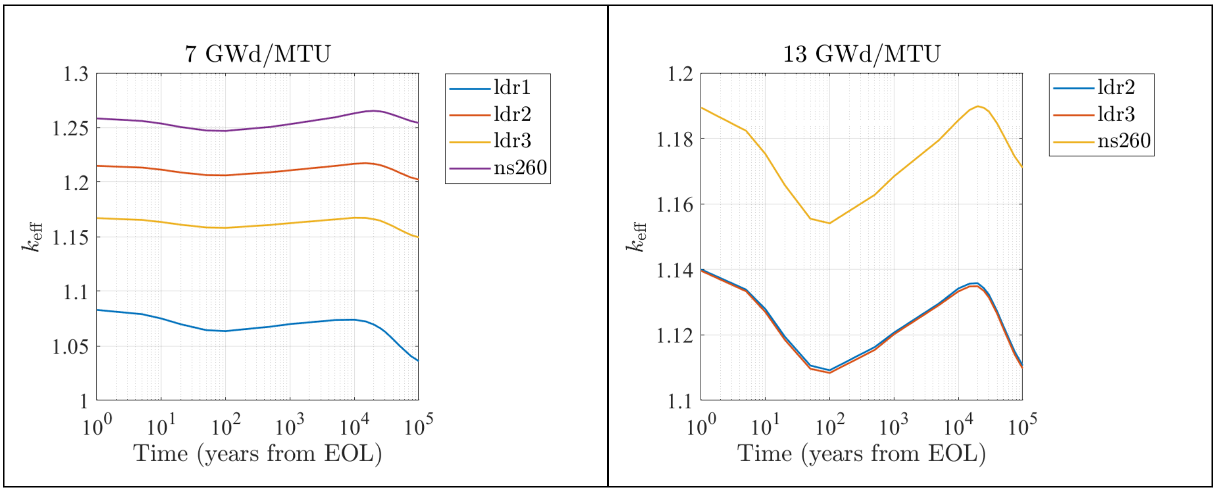

Figure 2-7. Post-irradiation multiplication factor (k_{eff}) for the LDR-50 and the 2.6 % U-235 enriched NuScale assemblies following 7 (left) and 13 (right) GWd/MTU irradiation. Note that 7 GWd/MTU approximately represents the discharge burnup for the 1.5 % enriched LDR fuel ('ldr1') so it was left out of the right figure.

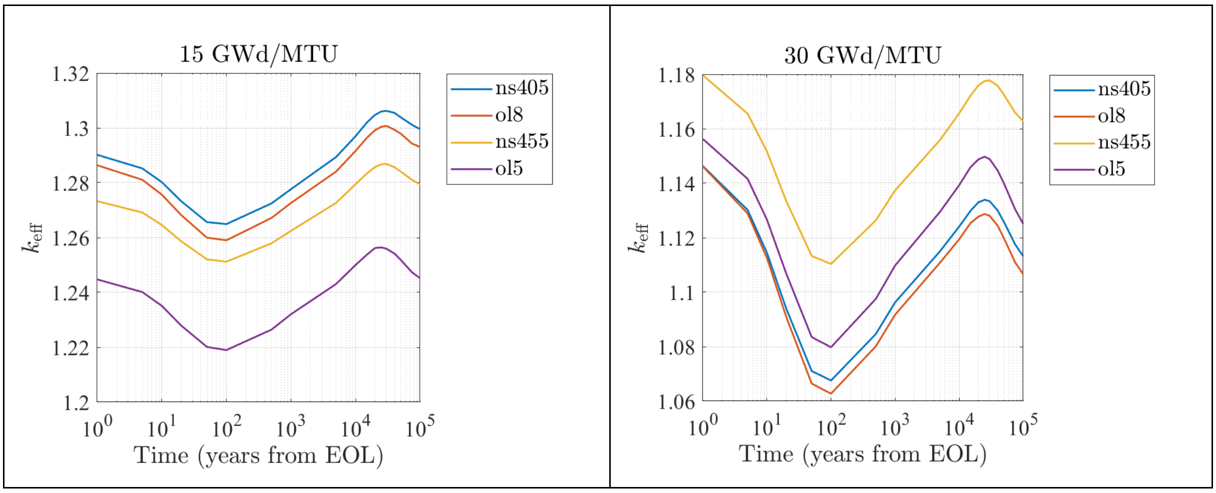

Figure 2-8. Post-irradiation k_{eff} of the NuScale model assemblies with 4.05 % and 4.55 % U-235 enrichments after and the comparable EPR assemblies ('ol8' and 'ol5') discharged at 15 GWd/MTU (left) and 30 GWd/MTU (right).

2.3.2.2 Decay heat

The decay heating power is one of the most limiting factors when the packing density of the final repository is defined. Assuming a KBS-3V type of a repository, this means that the higher the temperature, the higher the distance between two disposal canisters (and between disposal tunnels) shall be. This distance will have direct effect on the excavation costs and on the costs for backfilling the tunnels. As a design basis for the repository layout and for the canister, the outside surface temperature of the waste package (copper canister) shall be limited to 95 °C in order to maintain the performance of the bentonite buffer placed immediately around the spent fuel canister (Ikonen et al., 2018; Raiko, 2013). In order to keep the canister surface temperature at this level, the decay power of the all the fuel inside a reference canister is 1700 W (Raiko 2012). Considering the repository evolution, the peak temperatures are reached relatively shortly after the disposal, within 50–100 years after disposal (Ikonen et al., 2018). After this, the temperatures at the canister-buffer interface and in the buffer-rock interface slowly decrease to the level of natural rock temperature within 10 000 years (Raiko 2012).

The time-development of the decay heating power produced by the studied assemblies is depicted in the Figure 2-9 that is divided into two subfigures to separate the assemblies with low and high discharge burnup. The curves suggest that the three-cycle assemblies of LDR-50 are highly similar heat producers. The same is true with the two types of multi-cycle NuScale and EPR assemblies, i.e., those with U-235 enrichment above 4 %. The differences, or the lack of it, in heat production can be seen linearly in Figure 2-10, where all the assemblies are compared at 50 and 200 years after the discharge. For these figures, the assumed discharge burnup and assembly identifications are presented in Table 2-5.

Additionally, the decay heat production of the earlier LDR-50 concept with erbium as the burnable absorber was compared to that of the current Gd doped model and the results are shown in Figure 2-11. The figure suggests that the Er content and the choice between Er and Gd as burnable absorber has very limited impact. In this comparison, the discharge burnup of 20 GWd/MTU was assumed for all fuel types.

In order to provide some information on the source of the heat production, the seven most important heat producers are listed for part of the studied assemblies in Table 2-6. The listings are presented for the situations at 50 and 200 years after the discharge. Again, the underlying discharge burnup for each assembly type follows the values in Table 2-5. The contribution of various nuclides somewhat depends on the discharge burnup, but outside the printed table it can be noted that with the same or similar burnup, e.g., in comparison between the NuScale and EPR assemblies, the differences are rather minimal. This is also illustrated in Figure 2-12, in which the combined proportion of the five most heat-producing nuclides is plotted to compare the fuel assemblies and the effect of burnup for the 4.05 % enriched NuScale fuel type.

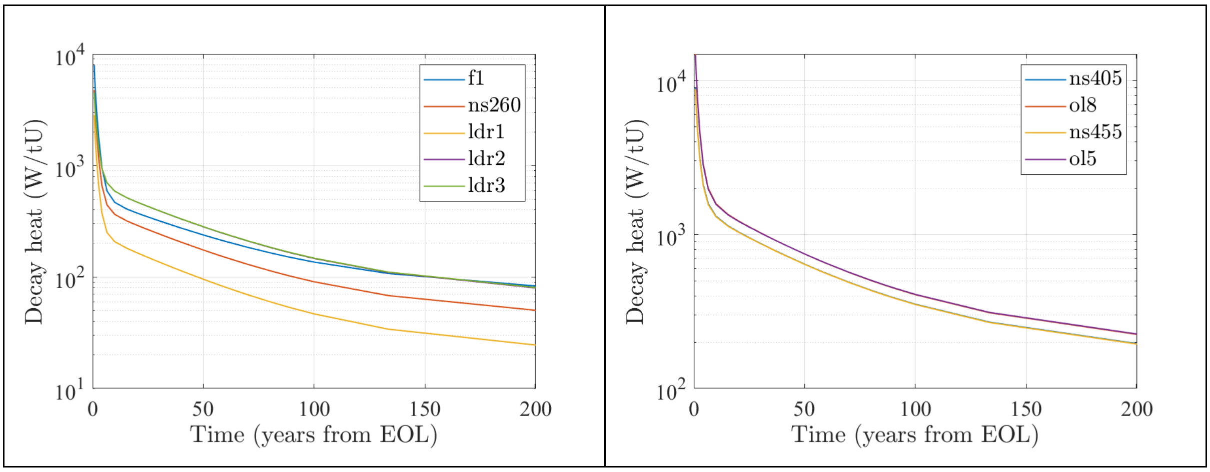

Figure 2-9. Decay heating power 3–200 years after discharge. The left figure illustrates the decay heat for the low-burnup (7–20 GWd/MTU) fuels and the right one for those with 40–45 GWd/MTU.

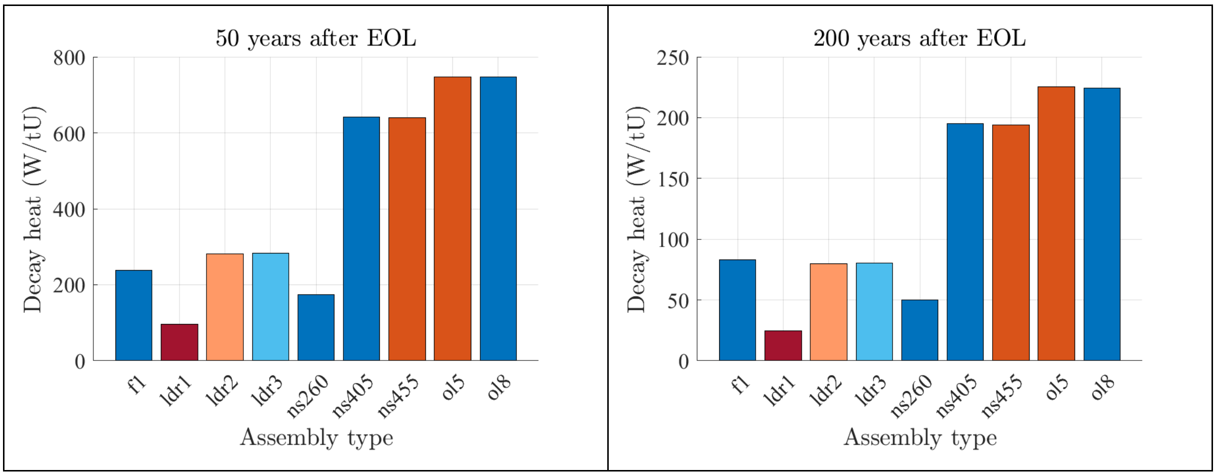

Figure 2-10. Decay heating power of all studied assemblies at 50 (left) and 200 (right) years after the discharge.

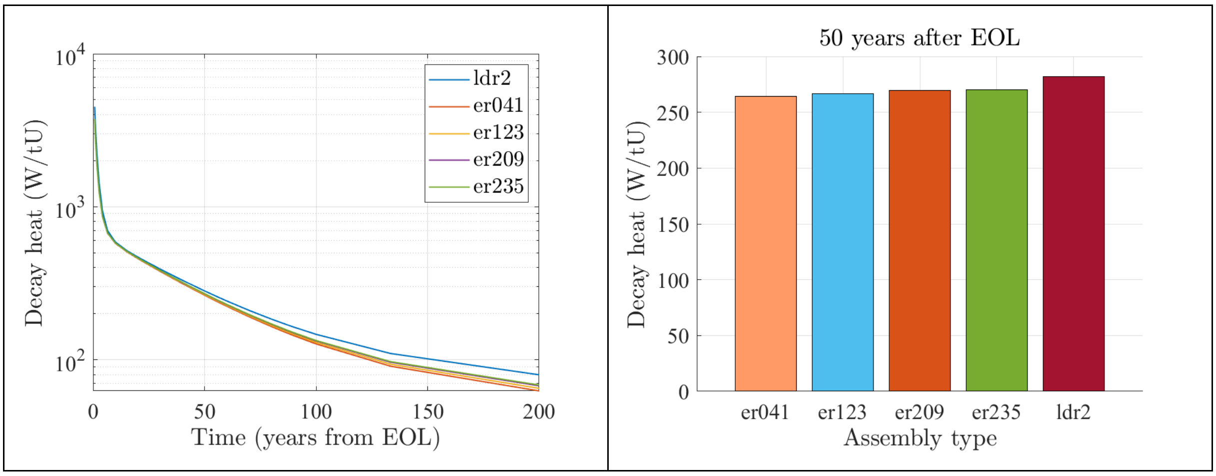

Figure 2-11. Comparison of decay heat between LDR assemblies with Er and Gd burnable absorber. The IDs 'er<d>' indicate the Er content of 0.41, 1.23, 2.09 and 2.35 wt-%.

Table 2-6. The contribution of the top-7 heat-producing nuclides (in %) 50 and 200 years after the discharge.

	LDR1		LDR2		NuSc260		NuSc405	
	Nuclide	**Heat (%)**	**Nuclide**	**Heat (%)**	**Nuclide**	**Heat (%)**	**Nuclide**	**Heat (%)**
50 years after EOL	Y90	31.2	Y90	27.3	Y90	29.0	Am241	22.8
	Ba137m	27.5	Ba137m	25.6	Ba137m	25.7	Y90	22.3
	Am241	15.5	Am241	21.0	Am241	20.9	Ba137m	22.2
	Cs137	8.3	Cs137	7.7	Cs137	7.8	Pu238	14.7
	Pu239	5.9	Pu238	6.3	Sr90	5.4	Cs137	6.7
	Sr90	5.8	Sr90	5.1	Pu239	4.3	Sr90	4.2
	Pu240	3.7	Pu240	3.4	Pu240	3.4	Pu240	2.7
200 years after EOL	Am241	52.3	Am241	64.2	Am241	63.1	Am241	64.8
	Pu239	22.7	Pu240	11.8	Pu239	14.9	Pu238	14.8
	Pu240	14.3	Pu239	10.0	Pu240	11.7	Pu240	8.7
	Ba137m	3.4	Pu238	6.8	Pu238	3.3	Pu239	5.7
	Y90	3.3	Ba137m	2.9	Ba137m	2.8	Ba137m	2.3
	Pu238	2.1	Y90	2.6	Y90	2.7	Y90	2.0
	Cs137	1.0	Cs137	0.9	Cs137	0.8	Cs137	0.7

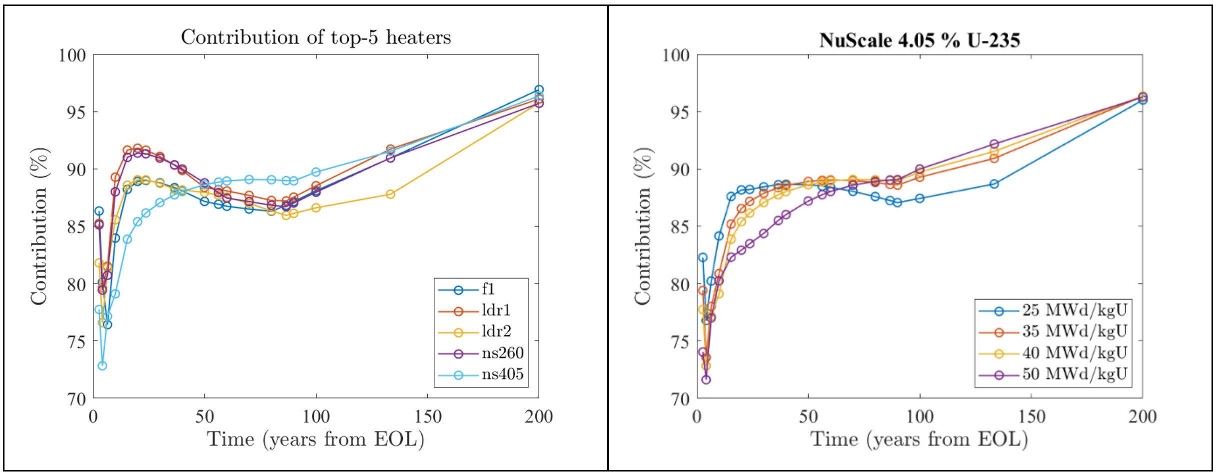

Figure 2-12. The heating power contribution of the most heat-producing nuclides in the spent fuel. Left: the contribution of the selected fuel types, following the default discharge burnup as presented in Table 2-5. Right: the impact of the discharge burnup on the top heater contribution for the NuScale fuel with 4.05 % U-235 enrichment and no Gd.

2.3.2.3 Spontaneous fission

The spontaneous fission (SF) rates illustrate the neutron emission from the spent fuel, which is an issue for the operational radiation safety when handling the spent fuel. In this case, a low SF rate is desirable. However, from the non-proliferation point of view, the higher SF rate is advantageous, as it causes increased difficulties in constructing a nuclear weapon.

The spontaneous fission rates with the default discharge burnup associated to each of the studied assembly are depicted in Figure 2-13 for both short and long-time scales, the former being up to 200 years and the latter 10 million years. In order to provide some clarity to the magnitude of differences that can easily be lost in the logarithmic plots, Figure 2-14 presents the assembly-wise SF rates with the linear scale at 50 and 1000 years after the discharge.

The main driver in SF rate is the discharge burnup, but the Gd content and the U-235 enrichment also cause an observable effect. However, the Gd content difference between 5 % and 9 % in the LDR-50 assemblies with 2.4 % U-235 yielded such a small difference to the SF rate that the curves would not be distinguishable in Figure 2-13 so the latter was left out. Furthermore, the calculations suggest that erbium design of the LDR-50 fuels would yield an SF rate half of the Gd version at 50 years after the discharge, when 20 GWd/MTU burnup is assumed for both.

The exact contribution of the main spontaneous fission source nuclides is provided in Table 2-7 at 50 and 10,000 years after the discharge. The numbers practically suggest that the discharge burnup rather strongly affects the distribution. In contrast, the data that is not included in the table indicates that the Gd content and U-235 enrichment do not affect the distribution very much with similar discharge burnup.

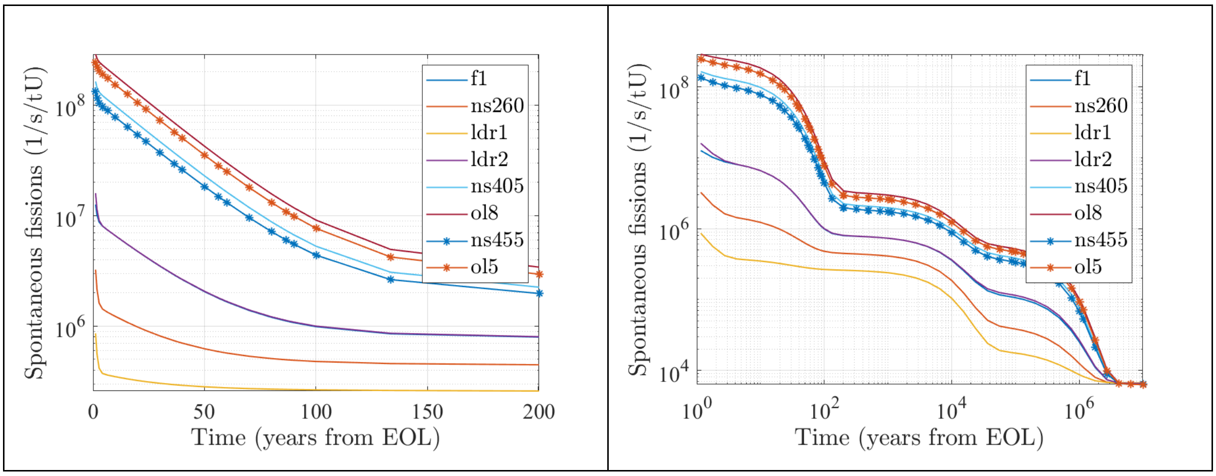

Figure 2-13. Spontaneous fission rates of the studied assemblies in short (left) and long (right) term. The SF rate curve of the LDR-50 assebly with 2.4 % U-235 and 9.0 % Gd, or 'ldr3' was left out because the SF rate is almost equal to that of 'ldr2'.

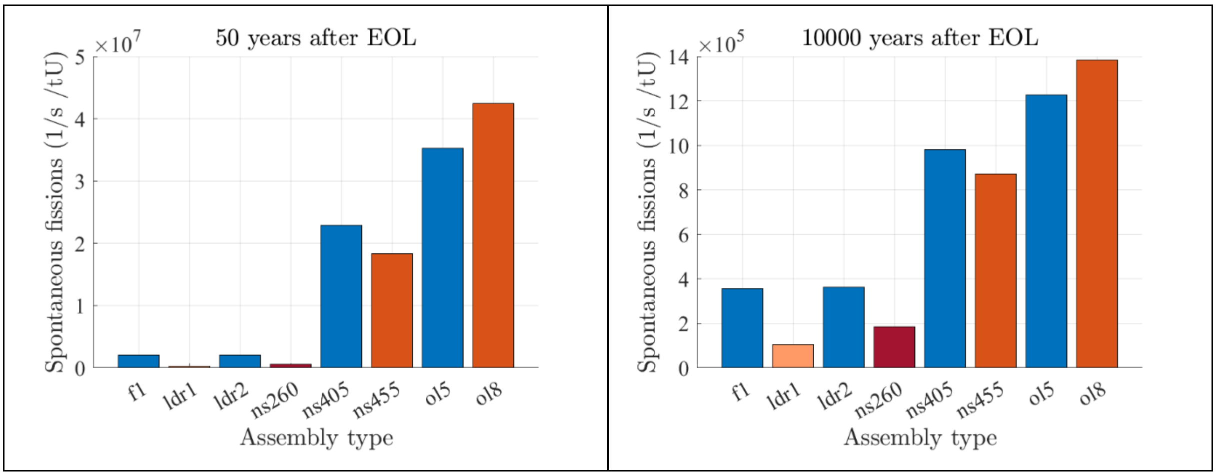

Figure 2-14. The spontaneous fission rates of the studied assemblies at 50 (left) and 10,000 (right) years after discharge. Bar colours indicate the assemblies with equal amount of Gd.

Table 2-7. Contributions (in %) of the most important spontaneous fission source nuclides at 50 and 10,000 years after discharge with the burnups presented in Table 2-5. The LDR-50 with 0.41 % erbium, covering the two right-most columns, discharge burnup of 20 GWd//MTU was assumed.

	LDR1		LDR2		NuSc260		NuSc405		LDR Er 0.41 %	
	Nuclide	SF %	Nuclide	SF %	Nuclide	SF %	Nuclide	SF %	Nuclide	SF %
50 years	Pu240	85.5	Cm244	59.5	Pu240	64.6	Cm244	89.7	Pu240	44.8
	Cm244	6.2	Pu240	31.3	Cm244	26.1	Pu240	5.1	Cm244	44.8
	Pu242	4.6	Pu242	6.3	Pu242	6.1	Cm246	2.2	Pu242	6.4
	U238	2.4	Pu238	1.8	Pu238	1.8	Pu242	2.0	Pu238	2.7
	Pu238	1.2	Cm246	0.6	U238	1.1	Pu238	0.9	U238	0.6
10,000 years	Pu240	81.3	Pu240	62.4	Pu240	76.2	Pu242	45.2	Pu240	69.5
	Pu242	12.3	Pu242	35.1	Pu242	20.2	Pu240	41.8	Pu242	27.7
	U238	6.4	U238	1.8	U238	3.5	Cm246	12.2	U238	2.5
	Pu239	0.0	Cm246	0.7	Cm246	0.1	U238	0.6	Cm246	0.3

2.3.2.4 Mobile nuclides

The studies on the long-term safety of geological disposal have identified a group of nuclides that might be capable of penetrating all the way out of the repository to biosphere. These nuclides are C-14, Cl-36, Ag-108m, Mo-93 and I-129. The first of these result from N-14 and Cl-35 impurities in the fresh fuel. A concentration of 10 ppm was assumed for these in all calculations. The concentrations of these nuclides after the discharge from reactor are depicted in Figure 2-15 and Figure 2-16 except for Ag-108m. The decay calculations depicted in these figures are based on the discharge burnups presented in Table 2-5.

The general trend can be observed that the increased discharge burnup results in higher concentration of each nuclide, but with the exception of I-129, it is not the only significant factor. Whilst the relative differences may be large, such as for Mo-93 between the low-burnup fuels and those irradiated up to 40+ GWd/MTU, the absolute mass differences are rather moderate. It is also worthwhile to note that the activation products are built up as a function of the neutron flux that is affected, among other things, by the computational assumptions, such as the boron concentration and power density.

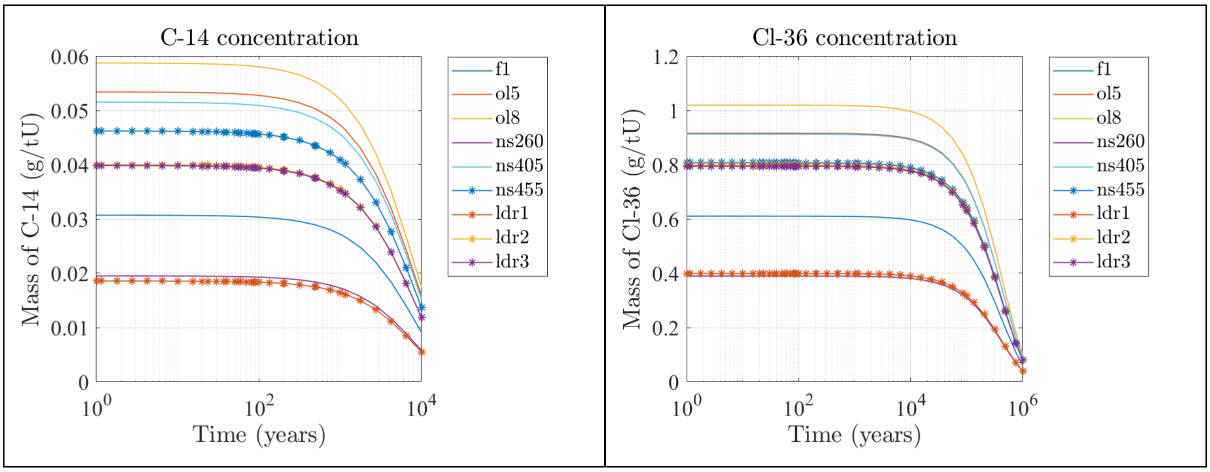

Figure 2-15. Concentrations of C-14 (left) and Cl-36 (right) of each assembly after discharge. For the LDR-50 assemblies of 2.4 % U-235, that is 'ldr2' and 'ldr3', had almost equal C-14 concentration so the difference is hardly visible in the figure. Similarly for Cl-36, almost equal concentrations were obtained for groups of a) 'ol5' and 'ns455', b) 'ns455', 'ldr2' and 'ldr3' and c) 'ldr1' and 'ns260'.

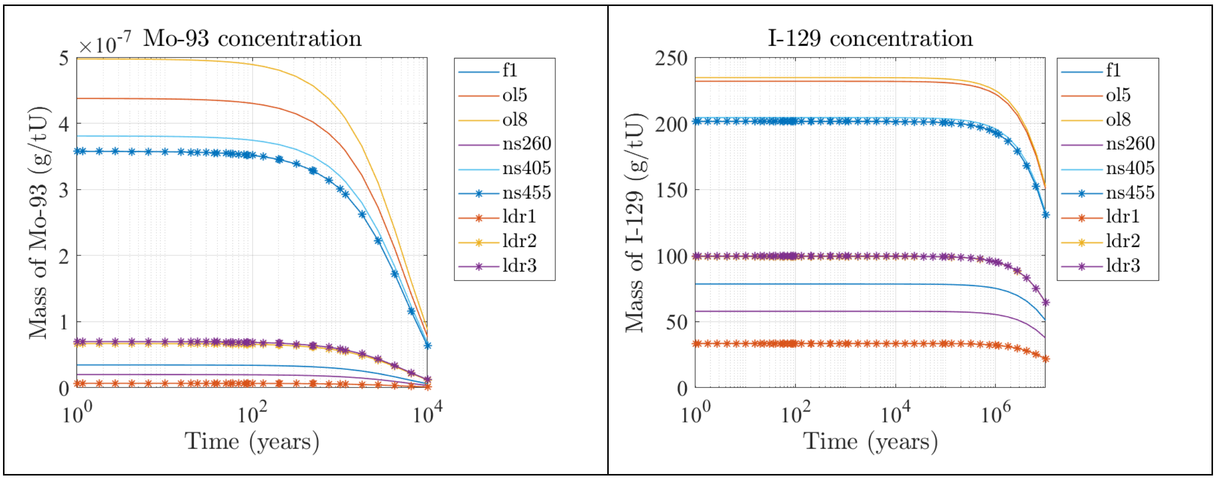

Figure 2-16. Concentrations of Mo-93 (left) and I-129 (right) in the spent fuel of each assembly type.

2.3.2.5 Uncertainties and further studies needed

The calculations performed in two-dimensional, infinite geometry do not take into account the neutron leakage that increases when the reactor core size decreases. Therefore, the approximation applied in the above-described calculations is expected to produce larger error than the similar approach for the fuel assemblies of the traditional large reactors. Full-core three-dimensional calculations would yield the most accurate results. However, such a full-core calculation is not only a question of computational cost, but also a search for the equilibrium core. The concept means that the compositions of loaded and unloaded fuel are equal for successive cycles and the cycles are approximately equally long. In addition, the reshuffling scheme for the assemblies irradiated over multiple cycles needs to be known.

The effect of the infinite lattice and 2D approximation can be assessed to some extent with the help of the so-called colorset calculation, where a small group of assemblies with different characteristics are modelled. Infinite lattice with non-reflecting boundary conditions in the axial ends of the fuel assemblies would also provide an estimate of the end-effects and provide some data on the axial burnup distribution.

The impact of various approximations related to the operating history, such as the constant and equal power density for all assembly types per reactor, constant boron concentration in the coolant and the constant fuel temperature, is also lost in the study. However, the error can be assumed rather small, particularly for the long-term safety analyses (Häkkinen, 2021). Additionally, the current design of LDR-50 requires more active use of control rods compared to typical current pressurized-water reactors because boron is not planned to be used for reactivity control. The presence of a control rod affects the nearby fuel rods, which is a feature to be taken into account in the future 3D calculations.

2.4 Preliminary conclusions concerning LWR-SMR fuels

Considering fuel form related factors that will have a direct influence on final disposal, the following comparisons can be made based on the information presented in section 2.1:

- Reactor design:
  - The NuScale SMR and the LDR-50 LW-SMR are based on EPR type reactors similar to the OL3 NPP located in Olkiluoto. LO1-2 are VVER-440 PWRs and OL1-2 are Asea-Atom BWRs.
- Number of fuel assemblies:
  - The NuScale SMR reactor core contains 37 fuel assemblies.
  - The LDR-50 LW-SMR reactor core similarly contains 37 fuel assemblies.
  - The LO 1-2 and OL 1-2 reactor cores contain 349 and 500 fuel assemblies, respectively. The OL3 unit has 241 fuel assemblies (TVO, n.d.-a, n.d.-b).
- Number of fuel rods per assembly:
  - There are 264 fuel rods in a NuScale SMR fuel assembly, 264 in an LDR-50 SMR fuel assembly, 63-96 in OL1-2 fuel assemblies, 126 in LO1-2 fuel assemblies and 265 in the OL3 fuel assemblies.
- Fuel assembly height:
  - The NuScale SMR fuel assembly height is ~2.4 m.
  - The LDR-50 LW-SMR fuel assembly height is ~1.2 m.
  - The LO1-2 fuel assembly heights are ~3.2 m.
  - The OL1-2 fuel assembly heights are ~4.1 m.
  - The OL3 fuel assembly height is ~4.8 m.
- Fuel assembly dimension:
  - For OL3, the NuScale SMR and for the LDR-50 LW-SMR, the fuel assembly dimension is 215 $\times$ 215 mm (square). For OL 1-2 and LO1-2 the fuel assembly dimensions are 139 $\times$ 139 mm (square) and 144 mm (hexagonal), respectively.

The fuel assemblies used in the LW-SMRs are based on 17 × 17 HTP type fuel designs and are essentially identical to the fuel assemblies used in OL3, with the exception of height. The NuScale SMR fuel assembly is roughly half the height of the OL3 fuel assembly and the LDR-50 LW-SMR more so. As such, a generic conclusion is that the OL3 canister design can be adapted relatively easily for spent fuel from the NuScale and LDR-50 LW-SMRs. The spent fuel assemblies from the NuScale SMR could fit directly into the same cast iron inserts designed for spent fuel assemblies from OL3. It may even be possible that two spent fuel assemblies from the NuScale SMR could be stacked end to end in the same copper canister designed for disposal of spent fuel assemblies from OL3 insofar as other design constraints, e.g., criticality, maximum decay heat at the canister surface, would not be exceeded.

Based on the comparison calculations presented in section 2.3, the following preliminary conclusions can be made:

- As the fuel and reactor types are very similar to those used in the current LWRs, the spent fuel features with comparable burnup are rather similar with small reactors.
- The generally lower discharge burnup characteristic for the smaller reactors facilitate the final disposal safety case at the assembly-level in terms of lower radioactivity and the consequent lower heating power.
- The question of dose rate around a spent fuel assembly was completely ignored in the present study but is one of the questions to be studied in the future.
- The concentration of mobile nuclides in the spent fuel is smaller.
- Lower average burnup especially in combination with the relatively high U-235 enrichment (the case with NuScale) contributes to higher post-irradiation reactivity, which may be a potential criticality safety concern. The question can be answered with 3D assembly burnup profile and calculations in a realistic repository configuration.

SMRs will have different fuel cycle performance compared with large LWRs because of differences in enrichment variation in the assemblies and because of neutron leakage, which will impact the achievable discharge burnup for a given average enrichment and fuel batch loading (Brown et al., 2017). Differences in fuel cycle performance will have an impact on nuclear waste management, resource utilization and economic and financial concerns. The smaller axial and radial core will yield significantly increased neutron leakage versus a large core, resulting in lower fuel utilization (Brown et al., 2017). Additionally, because the maximum enrichment is just under 5% and SMR designs may leverage varying enrichments to control power peaking, the average enrichment may be less than in a large LWR core which will also yield lower discharge fuel burnup (ibid).

The results of a set of scoping calculations performed by Brown et al. (2017) comparing the performance of a large reference LWR to a hypothetical SMR, where the SMR scoping model was guided by the NuScale SMR concept, relative to nuclear waste management criteria are provided in Table 2-8. These results imply that the nuclear waste management metrics (including both the mass of spent nuclear fuel and volume of low-level waste) are negatively impacted in the SMR design compared to the large LWR, i.e., more spent fuel and low-level waste are produced on an energy equivalent basis in the SMR versus the large LWR. It should be noted that these results were calculated assuming single-batch fuel loading cycles in the SMR; an SMR with a multi-batch fuel cycle would be expected to perform better (ibid).

Table 2-8. Estimates of nuclear waste management metrics (normalized per gigawatt-electric year) for a large reference LWR and hypothetical SMR at linear power ratings of 5.72 and 2.28 kW/ft (Brown et al., 2017).

Metric	Large LWR	SMR	
		5.72 kW/ft	2.28 kW/ft
Mass of SNF and HLW (t/GWe-year)	22.13	36.28	34.16
Activity of SNF and HLW at 100 years (MCi/GWe-year)	1.34	1.40	1.35
Activity of SNF and HLW at 100,000 years (MCi/GWe-year)	15.1	16.9	16.9
Mass of depleted uranium (t/GWe-year)	167.98	329.67	310.34
Volume of LLW (m^3/GWe-year)	399.6	470.6	462.04

3. Other nuclear waste

3.1 Introduction

According to Finnish regulations (STUK YVL Guide D.4. 103), "any waste generated inside the controlled area of a nuclear facility, including the components of a permanently closed nuclear facility, is by definition nuclear waste." Waste other than spent nuclear fuel can be divided into different categories based on its level of radioactivity as intermediate level waste (ILW), low-level waste (LLW), very low-level waste (VLLW) and waste streams that can be cleared from regulatory control. In Finland, ILW has been classified to have activity levels between 1 and 10 MBq/kg (YVL D.4, STUK Y/4/2018). Low level waste on has a maximal activity of 1 MBq/kg. Together ILW and LLW form the low- and intermediate level waste (LILW) class. VLLW refers to "nuclear waste whose average activity concentration of significant radionuclides does not exceed the value of 100 kBq per kilogram and the total activity does not exceed the values laid down in Section 6(1) of the Nuclear Energy Decree (161/1988) (total activity is higher than 1 TBq, or the alpha activity, excluding natural uranium, thorium and depleted uranium, is higher than 10 GBq " (STUK, 2018).

Typically, ILW and LLW is generated during the operation, maintenance or decommissioning of a nuclear facility. Most of the LILW are considered as short-lived waste defined according to STUK Y/4/2018 as follows: "*Short-lived waste shall refer to nuclear waste, the calculated activity concentration of which after 500 years is below the level of 100 MBq/kg in each disposed waste package, and below an average value of 10 MBq/kg in waste in one emplacement room*". LILW-SL represents only a small fraction of the total activity produced in a NPP, but more than 90% of the total volume of all radioactive waste worldwide (IAEA 2003). Relevant radionuclides in LILW-SL are ^{137}Cs and ^{90}Sr, which will decay to radiologically insignificant levels in about 300 years, after approximately ten half-lives (IAEA, 2002).

The generation of these waste types are discussed in the following three sections considering the difference between a typical NPP unit and a LWR SMR unit. Nuclear waste produced in non-LWR SMRs are discussed in Section 3.3.

3.2 Operational LILW and decommissioning waste produced in Finnish NPPs

Various sources of LILW exist in both Loviisa and Olkiluoto. One source is used ion exchange resins used for treatment of process waters flowing in the prime circuit of the reactor. Additionally, concentrates originating from evaporation of liquid wastes, and operative wastes are significant sources of LILW. Furthermore, dry LILW waste is formed in maintenance and operation. Typically, the wastes are first stored directly at the nuclear plants or storage areas in the vicinity. After this, the waste is usually treated to reduce its volume or turn it into a form that is easier to dispose. Finally, the treated waste is disposed into underground in so called VLJ-caves at a depth of 60-100 m below the ground surface in Olkiluoto and at a depth of 110 m in Loviisa (TVO, 2012). The design of the VLJ-caves is presented in more detail in Section 5.2 (Disposal of other nuclear waste).

In Olkiluoto, the total capacity of the silo for ILW is 3500 m^3 and for LLW it is 5000 m^3 (Posiva, 2020). In the end of 2019, total LILW disposed in Loviisa VLJ was ~2 500 m^3 and in ~6 600 m^3 Olkiluoto VLJ. Furthermore, a between LLW and ILW waste for Olkiluoto is presented in (Posiva, 2020). They state, that out of the 6 600 m^3, 2 100 m^3 is ILW and the rest is LLW. Additionally, 1600 m^3 and 1 700 m^3 of LILW is stored both in interim storage and at the NPPs in Loviisa and in Olkiluoto respectively. More detailed breakdown of some specific waste volume is presented next.

The total stored amount of non-solidified ion exchange resins in the end of 2019 was ~530 m^3 (12 000 GBq) in Loviisa. Furthermore, ~680 m^3 of non-solidified concentrates were in storage in the end of 2019. The liquid wastes are stored in tanks in the liquid waste storage area.

At Loviisa, both the used resins, concentrates and other liquid wastes are initially stored in storage tanks. Radioactive matter is removed from the concentrates using ion exchangers. After the storage, the liquids are transported to a solidification plant, where they are combined with cement, slag and other binders to solidify them in steel drums (Fortum, 2021). In Olkiluoto, ion exchange resins are solidified inside steel drums in bitumen (Posiva, 2018, 2020). The amount of solidified ion exchange resins produced in Loviisa in 2019 (LO 1-2 units) was ~12 m^3 (Posiva, 2020). In total, there were ~440 m^3 (~1 900 GBq) of solidified wastes at Loviisa in the end of 2019. These wastes have been stored and disposed during the entire NPP life. Out of these, 110 m^3 were still waiting for final disposal and the rest of them were already stored in the final repository. Similarly, in Olkiluoto, approximately 80 m^3 of solidified wastes were waiting for final disposal and in total ~1 900 m^3 were already disposed.

Dry, LILW waste, at both NPPs consists mainly of mixed solid waste (plastic, paper, cloth, etc.) and metal generated during the operation and maintenance of the reactor (Posiva, 2020). In Loviisa, the dry operative and maintenance waste is, if possible, compacted, and then placed in 200 l steel barrels. In Olkiluoto, such waste is stored in similar barrels, which are compacted after filling to approximately 100 l volume. In addition, some waste in Olkiluoto is stored in 4,4 m^3 concrete containers. In Loviisa, a total of 2 100 m^3 of maintenance waste is disposed and 300 m^3 is waiting for disposal in storage. In Olkiluoto the amount similar waste, in form of scrap and maintenance waste is 3 000 m^3. Out of this, 1 600 m^3 is waiting in storage and 1 400 already disposed in the VLJ.

Some of the VLLW has been emplaced in a landfill and some cleared from regulatory control. However, there is currently a plan to develop a landfill-like VLLW repository in Olkiluoto, where the waste would be disposed in the future (TVO & Afry, 2021). The design of such repository is discussed in Section 5.2.1.

Finally, LILW waste is produced during the decommissioning of the NPPs. Decommissioning wastes include large components such as pressure vessels and other reactor components. Additionally, any activated structures are treated as radioactive waste. In Olkiluoto, the plan is to build additional LLW and ILW silos to store the decommissioning waste (Posiva, 2020). In Loviisa, the estimated volume of decommissioning waste is 25 000 m^3 (Fortum, 2021), while in Olkiluoto the plan is to build four additional silos with a volume of 10 000 m^3 – 15 000 m^3 (Viitanen, 2010). TVO (2016) gives a bit more detailed description of specific waste volumes of the decommissioning of Olkiluoto 3 reactor. They state, that contaminated decommissioning waste has an approximate volume of 7 500 m^3 after it has been packaged into 200 l drums. In addition, 450 m^3 of concrete from the biological shield has to be disposed.

In order to compare waste streams of conventional NPPs and any future SMRs, some sort of crude estimate of waste produced per a unit of energy has to be made. In this report, such ratio is only defined with crude estimates of annual waste production and an average annual energy output. The nuclear reactors of Loviisa started their operations in 1977 and 1980 and therefore they have been operating, on average, for 43,5 years. The total LILW accumulated during these years, assuming nothing has been disposed or stored elsewhere is 4 100 m^3. This would result in an average annual accumulation of 90 m^3 of LILW in Loviisa. Fortum (2019) states, that the combined electricity output of both L1 and L2 was 8 TWh in the year 2019. This results in a waste/energy ratio of 12 m^3/TWh.

Similarly, OL1 and OL2 started their commercial operations in 1979 and 1982 resulting in an average operational life of 41,5 years. Again, assuming, that no LILW has been stored or disposed elsewhere, in total 8 400 m^3 has accumulated in Olkiluoto. This results in an average annual accumulation of 202 m^3. According to (TVO, 2019) OL1 and OL2 produced together

14,75 TWh of electricity. However, as the electricity output has been slowly increasing throughout the years, a value of 14 TWh is used for calculating the waste – unit of energy ratio. The resulting ratio is 14 m^3/TWh. It should be noted, that both estimates exclude any changes in future waste streams, including the decommissioning waste.

3.3 LILW and decommissioning waste produced in LWR-SMRs

In principle, it can be assumed that the intermediate level waste generated in a LWR-type SMR consists mainly of ion exchange resins used for treatment of process waters (reactor coolant), as it is in a typical NPP. Potential operators will need to plan for temporary onsite ILW storage both for liquid waste (primary coolant, wash waters, etc.) and solid waste (filters and ion exchange resins). Eventually the waste needs to be treated, solidified, packaged and disposed in an intermediate depth underground repository as is the case with the LILW generated in current Finnish NPPs. However, individual SMRs might not have sufficiently large waste streams to build an individual LILW-repository for just one reactor. This problem is discussed more in depth in Section 4, where waste management strategies especially SMRs are discussed.

Although similar in designs, some differences considering waste streams are present in a SMR. The main differences are the size of the unit and the way the coolant is circulating. For example in NuScale, that the reactor coolant is circulating naturally without any help of reactor coolant pumps (RCP) (NuScale, 2022). The primary coolant water is circulated using a chemical and volume control system (CVCS) which includes filters to maintain suitable chemical composition of the water (IAEA 2020). During normal operations, maintenance of the primary coolant water and pressurizer vent lines generate ILW. If other maintenance is needed (replacement of contaminated components, e.g., filters), it can be assumed that these parts are likely to be classified as ILW as well. The amount of waste will also depend on the number of parallel SMR units. A more detailed description of the planned waste management systems of the NuScale power plant is presented next.

NuScale (2020a)[2] provides a detailed description of a radioactive waste management system (RWMS) which is an additional aspect of an overall design for a NuScale power plant. In this design, radioactive waste management is conducted in a separate radioactive waste building, where gaseous, liquid and solid wastes are treated and prepared for off-site disposal. The radioactivity of the primary coolant in a NuScale reactor is generated by either fission products, such as failed fuel elements, or by neutron activation of other materials. The RWMS consists of a liquid-, gaseous- and solid radioactive waste processing system (LRWS, GRWS and SRWS respectively.) The design is presented briefly below.

The purpose of the LRWS is to collect, hold and process all liquid wastes generated in the operation of the NuScale power plant. The system treats the generated wastes so, that the residual water can either be reused or discharged. Liquid waste generated in operations include waters from the coolant systems and various drains and pumps. Radioactive waste waters are also generated in the SRWS in a dewatering process use for solid waste. Furthermore, wastewater will originate from decontamination processes. In these processes, contaminated waters are treated with filters, ion exchangers and reverse osmosis components. Additional components of the LRWS include pumps, tanks, valves and degasifiers (NuScale, 2020a).

According to NuScale (2020a), gases originating from the degasifiers, containing for example radioactive xenon and krypton, are treated with the GRWS. The gases go through a filtration system, where particulates and moisture are removed. The gas is also diluted in the process. From there, the gas passes through a charcoal guard bed, and charcoal detention beds. The detention beds are used to collect and retain xenon and krypton allowing for their decay prior to venting.

Both the LRWS and GRWS produce multiple, secondary solid waste streams, which are themselves treated in the SRWS. Such wastes include, e.g., spent resins, charcoal, filters, and sludges. In addition, other solid wastes originating in the NuScale power plant are treated in the SRWS. The wastes are classified either as wet or dry solid wastes, which both have their own processing methods (NuScale, 2020a).

The wet solid wastes include the resins, charcoal, cartridge filters and membranes and sludges. The spent resins and charcoals are sluiced and stored in tanks for up to two years. After this they are dewatered and stored. Furthermore, the cartridge filters are changed annually for each reactor core and processed as high activity waste (NuScale, 2020a).

The dry solid waste included filters from the ventilation systems, tools, equipment, construction materials such as metal, concrete and wood and other waste such as rags and paper. The dry waste is sorted and based on its properties either processed on site or sent for off-site treatment. Everything suitable for decontamination is decontaminated. The waste is also shredded and compacted if possible. Processed waste can be temporarily stored on site, but it is sent off-site for permanent disposal (NuScale, 2020a).

The maintenance of the NuScale reactor also differs somewhat from that of a typical NPP and it may be the case that less LLW (protective gear, etc.) is generated. This possibility remains to verified. It may also be that the small size of the reactor presents as yet unknown challenges that may affect waste volumes.

Finally, decommissioning waste of a LWR NPP should be discussed. The decommissioning and dismantling of a conventional LWR NPP produces significant volumes of waste and it is estimated that approximately 95% of the total volume of this waste falls within the LILW category (IAEA, 2002). Such waste includes decontamination solutions and materials, contaminated building materials, pieces of metal or wood, electric wires, etc. (IAEA, 2002). A significant fraction of this waste (or also from the clean-up and remediation of contaminated sites) is categorized as VLLW.

For a LWR-type SMR unit, decommissioning waste includes potentially all the integrated reactor components (reflector, etc.) and the containment vessel itself that are all likely to fall in the ILW or LLW category. Waste streams falling into the VLLW category may be produced as well, but considering the closed nature of the SMR reactors, it is preliminarily assumed that the volume of these waste streams will be smaller for a LW-SMR in comparison to a LWR NPP. This assumption needs to be confirmed. As shown in Table 2-8, Brown et al. (2017) estimate that a hypothetical SMR will generate more LLW (per gigawatt-electric year) than a large reference LWR.

According to NEA and OECD (2011) and IAEA (2007), the decommissioning of full factory-assembled reactors (i.e., SMRs) may be technically less demanding as they can be transported back to the factory in an assembled form. The dismantling and recycling of components of a decommissioned SMR NPP at a centralized factory is expected to be more efficient and less expensive compared to performing the activity on-site. This expectation is attributed to the economies of scale associated with using a centralized facility. On the other hand, Locatelli and Mancini (2010), estimate the specific decommissioning costs of four 335 MWe SMRs to be double that of one 1340 MWe LWR. It should be noted that the decommissioning of SMR reactors in a centralised factory would imply that either such a facility would need to be built in Finland or that this procedure would be performed outside Finland, potentially requiring changes to the current regulations for handling nuclear waste in Finland.

4. Management strategies

4.1 Introduction

There are several potential strategies for managing spent fuel and other nuclear waste produced in LWR-type SMRs in Finland. However, there are some basic principles outlined in the Nuclear Energy Act (1987/990) that need to be addressed:

- The use of nuclear energy, taking into account its various effects, shall be in line with the overall good of society (Nuclear Energy Act 1987/990, chapter 2, section 5),
- The use of nuclear energy must be safe and it shall not cause harm to people or damage to the environment or property (Nuclear Energy Act 1987/990, chapter 2, section 6).
- Nuclear waste generated in connection with or as a result of use of nuclear energy in Finland shall be handled, stored and permanently disposed of in Finland (Nuclear Energy Act 1987/990, chapter 2, section 6a).
- A nuclear facility shall have premises, equipment and other arrangements to ensure the safe handling and storage of nuclear material required by the facility as well any nuclear waste generated during operation and decommissioning (Nuclear Energy Act 1987/990, chapter 2, section 7h).
- The amount of nuclear waste generated in the use of nuclear energy shall be kept as small as reasonably achievable with practical measures with regard both to the activity and the amount without endangering the implementation of the general principles in accordance with sections 5, 6 and 7 (Nuclear Energy Act 1987/990, chapter 6, section 27a).
- A licence holder whose operations generate or have generated nuclear waste (party with a waste management obligation) shall be responsible for all nuclear waste management measures and their appropriate preparation, as well as for their costs (waste management obligation) (Nuclear Energy Act 1987/990, chapter 3, section 9).

4.2 Centralised waste management

Under a centralised waste management strategy, spent fuel and other nuclear waste is transported from different SMR sites from across Finland to a centralised site (or sites) in the country that features resources for spent fuel and waste handling, treatment, processing, packaging, interim storage, encapsulation and final disposal facilities. These facilities could include a near surface repository for VLLW, an intermediate depth geologic repository for LILW (including decommissioning waste) and a deep geologic repository for HLW (spent fuel). This strategy could be needed if the SMR plants are relatively limited in terms of heat or electricity production and are owned by numerous, small independent operators whose individual resources do not allow for building radioactive waste pre-disposal and disposal facilities. Centralised waste management could be carried out by a waste management organisation established by a consortium of SMR operators or through services furnished by a private provider in Finland. Ownership of all the SMRs in Finland by one or more large corporate entities may naturally lend itself to a centralised waste management scheme. However, it should be evaluated how the requirements linked to responsibilities of the producer of the waste (e.g Nuclear Nuclear Energy Act 1987/990, chapter 3, section 9) should be taken into account in the development of different centralised waste management approaches in Finland.

Under a centralised waste management strategy, the need for a viable transport network is obvious (CNA, 2018). If the SMRs are located in areas accessible by road, required infrastructure will need to be developed to access these locations to utilize current road transportation modes. However, the timeliness of emergency response to transport events in remote locations will need to be considered. For any SMRs which are not accessible by road, then transport of waste will be required by water or rail, which may necessitate establishing seaports or rail tracks.

At present in Finland, the handling and disposal of LILW from NPP operations and decommissioning is conducted on site; there is limited experience with the transport of large volumes such material. It should be also noted that even if all spent fuel and other nuclear waste would eventually be transported from the site to a centralised waste management site, some type of interim storage would be still needed at the SMR sites.

The transport of irradiated sealed SMR fuel cores could also create unique challenges (CNA, 2018). The transport of these cores could be quite different from conventional fuel transport and require the development of new transport packages. Depending on the size and burn-up of the cores, the required shielding could result in proposed transport packages that exceed weight limits for many transportation routes. The characteristics of sealed SMR cores could pose unique challenges to fuel transportation that should be evaluated early and factored into the SMR design process and site selection.

Post-irradiation fuel characteristics are required in order to support a comprehensive evaluation of SMR fuel transport challenges (CNA, 2018), not to mention final disposal. These key characteristics include fuel configuration, fission product inventory, decay heat generation, physical and chemical form, and fissionable material content. The transport assessment is driven by these characteristics and involves determining how the fuel may be transported while adhering to current regulatory requirements, how much decay time will be required at the SMR site prior to transport, whether there any pre-transport processing requirements and what fuel transport packages are feasible.

4.3 Decentralised waste management

Under a decentralised waste management strategy, all handling, treatment, processing, packaging, interim storage, encapsulation and final disposal, including a near surface repository for VLLW, an intermediate depth geologic repository for LILW (including decommissioning waste) and a deep geologic repository for HLW (spent fuel), would occur locally. This strategy does not seem particularly technically, economically or societally viable except for possibly near surface disposal of VLLW. It simply may not be the case that suitable geologic repository sites can be found near every SMR location, and it cannot be assumed that every SMR host community will be equally open to hosting geologic repositories as well. Another option would be to use borehole disposal for decentralized deposition of SNF. Constructing a single borehole should be less expensive than an entire repository as discussed in Section 5.1.3.

4.4 Hybrid waste management

Under a hybrid waste management model, some part of the management could be handled in a centralised manner and some locally. For example, encapsulation could be something that is done in a centralised facility and also the HLW disposal. This could be economically the most feasible option taking into account the higher volume of the LILW waste streams (also from decommissioning) that maybe most economic to store and dispose locally. However, in this case the SMR plant site should be suitable for intermediate storages and for VLLW and LILW repositories.

The applicability depends on the ownership base of the management company handling HLW and also on local opposition.

4.5 Currently unavailable waste management options

Current Finnish legislation (Nuclear Energy Act 1987/990) restricts the following options:

- Sending spent fuel (and other radioactive waste) to another country for final disposal, e.g., to a European Multinational Repository (ERDO, 2021).
- Decommissioning of SMR units outside Finland (even considering that the final disposal would still take place in Finland).
- Reprocessing of the spent fuel outside Finland to enable final disposal (pertains to non-LWR SMR units, see chapter 8).
- Leasing of the SMR (and fuel) itself with agreements that upon reaching the end of its service lifetime the entire plant facility is decommissioned and returned to its point of origin along with any accumulated wastes.
- Service model where the responsibility is contracted to a third party for handling, storage and/or for final disposal of the spent nuclear fuel (HLW) or other nuclear waste streams. Whether carrying the financial responsibility is enough to fill the requirement set in (Nuclear Energy Act 1987/990, chapter 3, section 9) is under discussion (Ahonen et al., 2020), but has not been defined yet.

5. Final disposal concepts

5.1 Spent fuel disposal

5.1.1 Introduction

Disposal of spent nuclear fuel (SNF) and other nuclear waste in Finland is regulated by the Nuclear Nuclear Energy Act 1987/990, Nuclear Energy Decree (161/188), Y/4/2018 STUK regulation on the safety of disposal of nuclear waste and by STUK YVL Guides D.5 Disposal of nuclear waste (13.2.2018) and YVL D.7 Release barriers of spent nuclear fuel disposal facility (13.2.2018). The requirements for transport of spent fuel and other nuclear waste is described in YVL Guide D.2 (Transport of nuclear materials and nuclear waste) and the requirements for handling and storage of SNF in YVL Guide D.3 (Handling and storage of nuclear fuel). Requirements for the processing and storage of operational and decommissioning wastes are described in Guide YVL D.4 (Predisposal management of low and intermediate level nuclear waste and decommissioning of a nuclear facility). The general principles and requirements related to nuclear safeguards activities are provided in Guide YVL D.1 (Regulatory control of nuclear safeguards).

The safety of final disposal is based on the defence in depth principle which is described in Y/4/2018 and is in line with the safety principles presented by IAEA (2012). According to STUK (2018), section 30 "*The long-term safety of disposal shall be based on long-term safety functions achieved through mutually complementary barriers so that the degradation of one or more long-term safety function or a foreseeable change in the bedrock or climate will not jeopardise the long-term safety*." The repository depth and type is selected based taking into account the waste characteristics and local geological conditions ((STUK, 2018), section 31). The basis for the selection is presented in Figure 5-1. The engineered barriers for a repository shall be selected to prevent the release of radioactive substances into the surrounding environment for a duration of time that is sufficient taking into account the half-life of the radioactive elements contained in the waste ((STUK, 2018), section 32).

The following subchapters focus on final disposal concepts applicable for final disposal of spent nuclear fuel from LWR-SMRs.

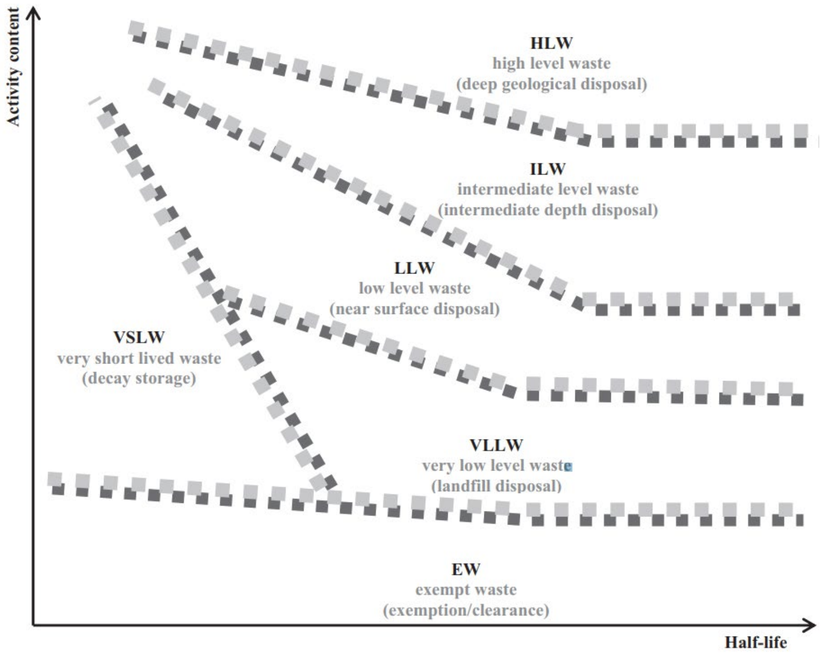

Figure 5-1. Selection of repository type and depth based on waste characteristics, according to IAEA SSG-23.

5.1.2 KBS 3V and KBS 3H concepts

The SNF-disposal solution Posiva uses for the design of the repository in Finland is based on the KBS-3 concept originally developed by The Swedish Nuclear Fuel and Waste Management Company (Svensk Kärnbränslehantering AB, SKB). This concept is based on combining natural and engineered barriers in order to ensure the long-term safety of SNF-disposal. This approach also fits the defence in depth discussed earlier. In this section the various barriers of the KBS-3 concept are presented and the distinction between KBS-3H and KBS-3V is explained. After this, the suitability of the concept for SMR-waste is discussed.

The KBS-3 concept is shown schematically in Figure 5-2. Spent fuel assemblies are encapsulated within a copper canister. The canister is placed in a deposition hole, where it is surrounded with bentonite clay. Deposition tunnels, where the disposal holes are located, are backfilled with clay-based material after a series of canisters are emplaced. After the prescribed volume of spent nuclear fuel is disposed, all access tunnels to the disposal level are filled and the entire repository is sealed. The ultimate safety target of the KBS-3 concept is long-term isolation and containment of radionuclides from the biosphere (Saanio et al., 2013). This target is achieved by means of various safety features, which ensure that all the barriers function as intended.

Figure 5-2. Engineered and natural barriers within the KBS-3V concept (Posiva, 2021a).

The repository is comprised of surface and underground facilities. The underground facilities are reached with access tunnels and include technical rooms and deposition tunnels with deposition holes. The main surface facility is the encapsulation plant, where canisters containing spent fuel assemblies are prepared for disposal. The sealed canisters are transported to the disposal level underground with an elevator (Saanio et al., 2013; SKB, 2010). Disposing the canisters underground in bedrock with predictable and favourable conditions in combination with a sufficient depth provides safe disposal (Saanio et al., 2013).

The spent nuclear fuel itself is not considered to be a barrier in the KBS-3 system, as containing and isolating any radionuclides released from the SNF is the main purpose of the system. The properties of the SNF that is disposed will impact the design of the repository system. For example, both the enrichment and burnup of the fuel in combination with the cooling time will determine the amount of residual heat that that can be released from the SNF to the surrounding environment. Temperature will affect barrier performance and temperature limits are considered in barrier design and repository layout.

The SNF is encapsulated in corrosion resistant, mechanically stable canisters. The canisters are the first barrier containing the radionuclides from the SNF and deemed to be the principal engineered barrier that all the other barriers are supporting (Raiko, 2013). The canisters consist of large cast iron inserts which support the fuel assemblies and act as the load bearing structure of the canister, and a corrosion resistant copper shell. The fuel rods are inserted into slots in the cast iron insert.

Two disposal variants are possible under the KBS-3 concept: a vertical (KBS-3V) and horizontal (KBS-3H) option. In the KBS-3V option, deposition holes are drilled vertically through the floors of the deposition tunnels. In the KBS-3H multiple canisters are emplaced end-to-end in long deposition drifts, separated by filling and distance blocks. In the KBS-3H case, the canisters are also surrounded by bentonite clay and these bundles are themselves placed into perforated titanium jackets.

When all the deposition holes in a deposition tunnel are filled, the tunnel is backfilled with clay-based material and sealed with a concrete plug. Similarly, deposition drifts would be sealed after all the canister positions are occupied. After all the deposition tunnels (or deposition drifts) are sealed, the remaining excavated openings in the repository are filled and the facility would be closed.

5.1.2.1 Applicability of KBS-3 concept for SNF from LWR-SMRs

There are no obvious reasons why a KBS-3 type repository could not be used for disposal of SNF from LW-SMRs. However, the differences in fuel assembly dimensions, fuel configuration, fission product inventory, decay heat generation, physical and chemical form and fissionable material content would need to be taken into account in the repository design. At present, all designs and technical specifications for the spent fuel repository in Finland have been chosen to accept SNF from the currently operating NPPs and the soon to be online Olkiluoto 3 plant. Acceptance of spent fuel from LWR-SMRs in a KBS-3 type repository would likely require a different repository design in several aspects. One considerable redesign would be to expand the repository for all the additional waste if necessary.

Differences in spent fuel between the currently operating NPPs in Finland and that from the NuScale and LDR-50 LW-SMRs as well as considerations related to canister dimensions are discussed in Chapter 2. It should be noted that all changes to the canister design and the fuel type would also require adjustments to the encapsulation plant.

As discussed in Chapter 2, SMRs will have different fuel cycle performance compared with large LWRs. These differences originate from enrichment variation in the assemblies and because of neutron leakage, which will impact the achievable discharge burnup for a given average enrichment and fuel batch loading. Lower burnups in the SMR will affect the fuel's temperature, radioactivity and physical makeup. These factors must be considered in final disposal (not to mention interim storage and transport). For example, lower burnup levels would lead to spent fuel with higher fissile isotope concentrations which may require fewer fuel rods per disposal canister due to criticality concerns and, therefore, less efficient packing of SMR spent fuel mass per canister. Lower discharge fuel burnup will also lead to lower decay heats which would positively impact interim storage times and assembly loading into disposal canisters relative to maximum heat load constraints per canister. The maximum decay power for the Finnish NPP SF disposal canisters is 1700, 1830 and 1370 W for the BWR, EPR/PWR and VVER-440 fuel types, respectively (Raiko 2013).

Juvankoski and Marcos (2010) present a maximum temperature of 100°C for the buffer material. In higher temperatures, the mineral contents of bentonite start undergoing chemical alterations at higher rates. This can result in undesired properties for the buffer. Ikonen et al. (2018) further limit the temperature to 95°C in order toa account for any variability of the thermal properties of surrounding materials. Heat loads can be affected by the spacing of the deposition holes (Ikonen et al. (2018). Canister spacing should be approximately 8.8, 7.4 and 10.5 m for the BWR, VVER and EPR canisters, respectively, based on thermal analysis. The spacing of the deposition tunnels can also affect the heat distribution in the repository. If the SMR fuel has lower decay heat, the spacing of canisters or tunnels can be decreased. This would decrease the cost of disposal in tunnels, where SMR is disposed.

Ultimately, in the absence of information on the fuel size, composition, characterization, etc., it is not possible to determine with reasonable accuracy the potential costs associated with activities related to the management of SNF from LWR-SMRs.

A KBS-3 -type repository could in principle be used as a centralized disposal facility for SMR SNF. The Olkiluoto repository will be accepting spent fuel waste from both the Olkiluoto and Loviisa NPPs, the latter of which is nearly 400 km away (Saanio et al., 2013). There has also been discussion about where the SNF from the planned Hanhikivi NPP will be disposed. Suggested options are to either expand the repository in Olkiluoto or build a separate one close to the Hanhikivi NPP (TEM, 2013). Similarly, it should be possible to build a centralized repository for SMR SNF or even possibly expand an existing repository to accommodate it as well. The use of a centralized repository would greatly increase the need for cross-country transport of SNF, given the potential for SMRs to be deployed at multiple locations around the country, which might face resistance from the public. The Eurajoki Municipality, where Olkiluoto is situated, has already stated some concerns regarding the disposal of additional SNF in Olkiluoto in the future (Eurajoki, 2021).

5.1.3 Borehole concepts

Research of deep borehole disposal is active in the USA. Arnold et al. (2011) present a reference design for this concept in which deep boreholes are drilled into crystalline rock. An illustration of the design is presented in Figure 5-3. Nuclear waste containers are conveyed to the bottom of the boreholes and the remaining length of the hole is filled with various backfill materials and seals. The boreholes are drilled vertically to 5000 m depth, with that between 3000 and 5000 m being reserved for waste disposal. Boreholes could be drilled in two phases so that in the first phase a narrower diameter hole is drilled to depth followed by a secondary drilling widening the hole to the target diameter. The first drilling would simultaneously act as an exploratory hole yielding information about deep bedrock geological, chemical and hydrological conditions. The boreholes are to be lined in order to provide mechanical stability and ease the passage of canisters to disposal depth to the drilled hole and therefore prevent the waste canisters of being stuck during disposal. The casing should also enable fluid flow from the borehole to the surrounding rock, as otherwise the pressure within the borehole could increase too much due to thermal expansion of fluids.

As with the KBS-3 approach, SNF is loaded into disposal canisters at the surface in an encapsulation plant. Such plants could be either site specific or handled in a centralized location. The design of Arnold et al. (2011) would load approximately 367, 1 cm diameter fuel rods into one canister.

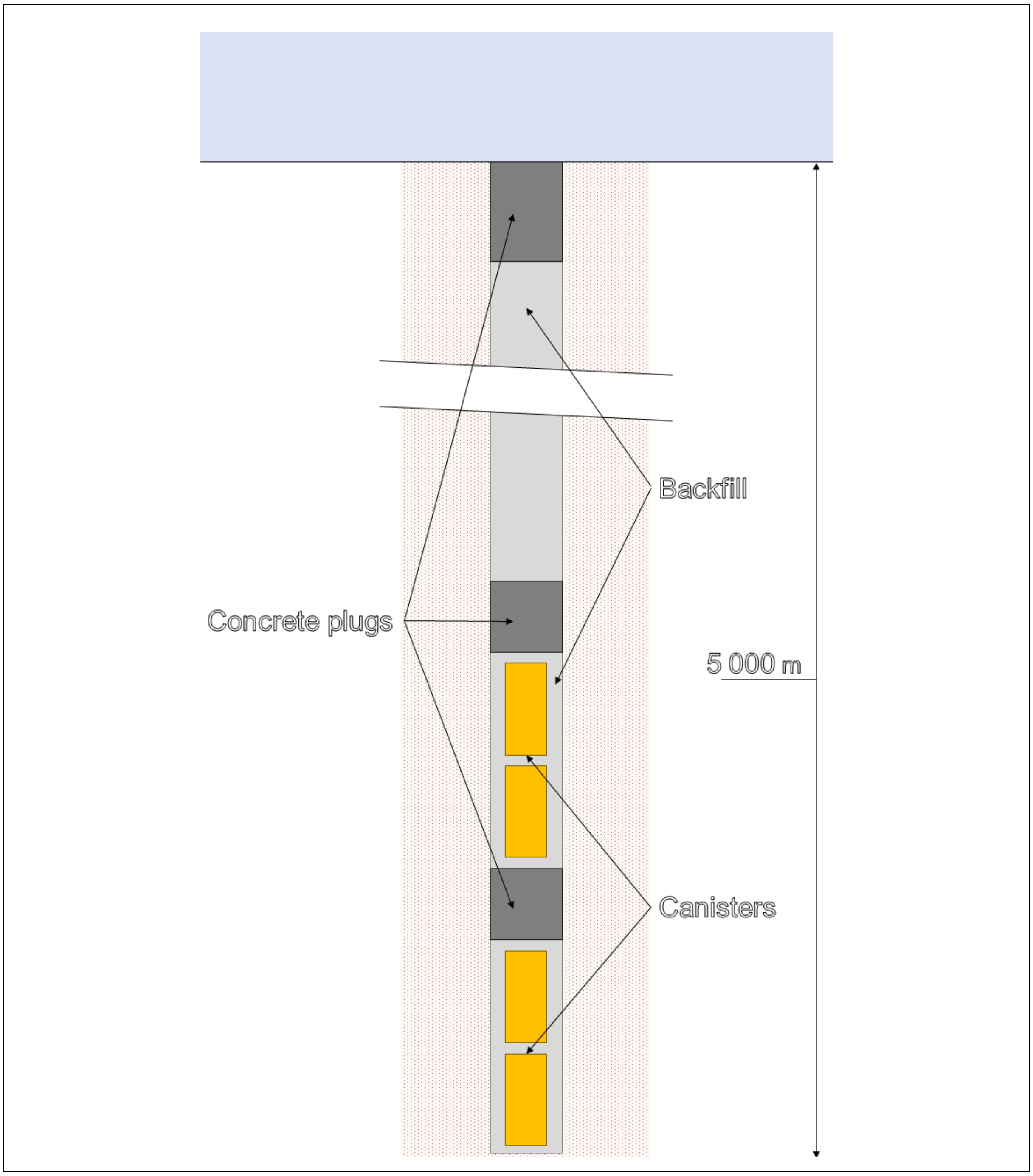

Figure 5-3. Borehole disposal concept based on the design of Arnold et al. (2011). The illustration is a simplification and not to scale.

Arnold et al. (2011) state that, unlike the KBS-3 concept, the waste canister itself does not act as a primary barrier after disposal. Its main function is to contain the nuclear waste during disposal operations. The integrity of the canister should be preserved as long as practical, but it has no long-term performance function. During disposal operations, the canisters should prevent any leakages or the possibility of nuclear criticality.

The environmental load on the canisters will be harsh already at disposal. The canisters should withstand very high hydrostatic fluid pressure, salinity levels and temperature. In the initial design described by Arnold et al. (2011) the canister would be constructed out of carbon steel. For disposal purposes, a series of canisters would be connected together. This series of canisters would be conveyed to the bottom of the borehole and a plug would be installed. Subsequently, another series would be lowered on top of the plug and this process would be continued for the entire waste disposal layer. Therefore, the first canister would have to withstand the load of all canisters in the series after it during operations.

The void between the canisters and the lined borehole walls is planned to be filled with bentonite mud (Arnold et al., 2011). The mud acts both as filling material and as a lubricant in the event waste canisters would be needed to be retrieved. Two different plugs are planned to be installed between each series of canisters. The first plug would act as a support for an overlying cement plug. The cement plug itself would act as a base for the next series of canisters and restrict fluid flow from one compartment to another during operations.

After all the canisters are disposed and the entire waste disposal layer filled, the borehole would be sealed and backfilled to the surface (Arnold et al., 2011). The borehole sealing material should have a very low permeability in order to prevent any fluid flow into or out of the borehole after closure. The borehole sealing should also withstand the mechanical load of the overlying backfill material. Backfilling would be installed as two different zones. In the lower zone, which spans from the top of the waste emplacement zone to a depth of approximately 1.5 km depth, the borehole lining would be removed. In this area a series of, 50 to 100-m-long compartments of sand, crushed rock, cement and bentonite, separated by plugs, would be installed. In the upper zone, the borehole lining would be left in place and a 100-m-long cement section would be installed followed by the same type of serial backfill compartments used in the lower zone. Finally, a concrete seal is constructed on the surface.

Deep Isolation® has developed an alternative borehole disposal method (Muller et al., 2019). A vertical borehole is drilled deep into the bedrock and, using directional drilling technology, turned horizontally to create the waste disposal section. Waste canisters are conveyed into the horizontal layer. As with the vertical borehole disposal system, the safety of the deep horizontal borehole system is proposed to stem from the isolation of the nuclear waste very deep underground.

Waste is placed within corrosion resistant canisters and any void in the canisters are filled. The filling material could also include boron to further prevent any criticality. The canisters are filled and sealed in an encapsulation plant at a centralized location or near the borehole site. The purpose of the canister is to contain nuclear waste and protect it from any mechanical loads during operations. The canisters are also designed so, that they keep their integrity as long as possible even after disposal. The canisters are lowered with a support cable with self-deploying brakes in a case of cable breakage. When the canisters reach the horizontal waste disposal zone, they are moved further using small wheels on the cable-canister interlocking device.

The boreholes are lined with carbon steel and the gap between the lining and the host rock is filled with cement. The aim of the lining is to support the drillhole during operations, so that the canisters can be effectively emplaced. The void between the canisters, and the borehole casing is filled with a buffer material. The buffer material should have a low hydraulic conductivity to prevent radionuclide transport through the borehole. However, dry and compacted bentonite cannot be installed in a deep borehole and thus liquid gels that can be solidified should be used, e.g., bitumen, silica or other commonly used grout materials. Between canister compartments, plugs are to be placed similar to the vertical borehole design (Arnold et al., 2011). From the disposal level to the surface, the borehole is sealed with plugs and backfill material. An illustration of the Deep Isolation® borehole design is presented in Figure 5-4.

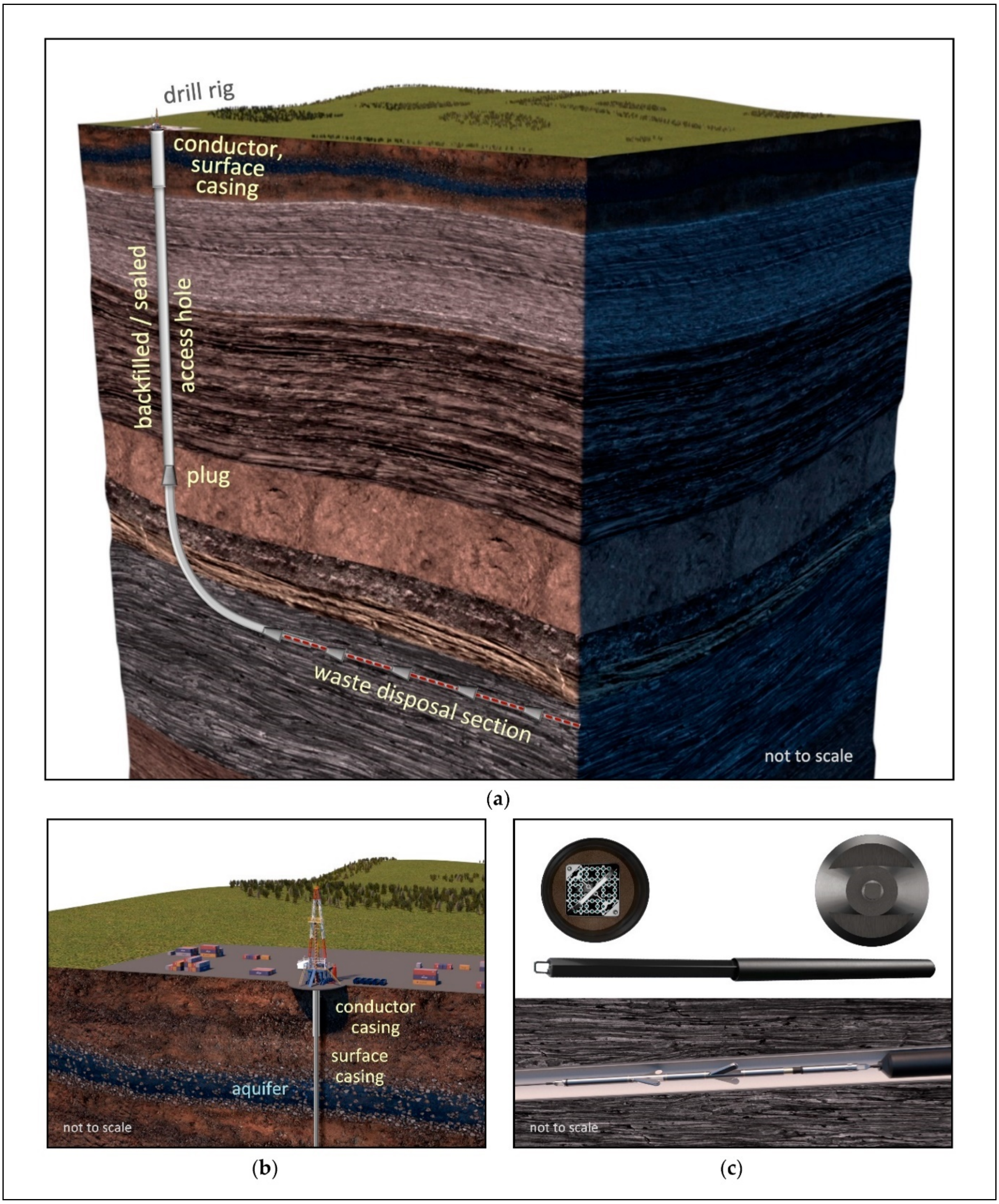

Figure 5-4. Deep Isolation® borehole disposal concept presenting an a) overview of the entire system, b) the surface facilities and c) the canister and the canister loading instrument. From Muller et al. (2019) CC-BY.

5.1.3.1 Applicability of the borehole concept for SNF from LWR-SMRs

Both borehole disposal concepts have benefits and drawbacks when compared to deep geologic repositories. The main benefit is that a constructing a single borehole is much faster and less expensive than building an entire deep geologic repository. It should be noted, that drilling such deep boreholes is already feasible with current technology (Arnold et al., 2011). The planning and construction of the deep boreholes can also be more flexibly balanced to meet the requirements of the nuclear energy industry. For example, when new nuclear power plants are being considered, locally-tailored, deep borehole disposal options can be directly included in the planning. Arnold et al. (2011) estimate the cost for construction of one borehole to range from 10 to 30 million USD. Additional costs pertaining to waste encapsulation, disposal operations and borehole closure are unknown. With a deep geologic repository, expansion of an existing facility or construction of new one would require more elaborate planning and investigation, the involvement of more stakeholders and higher costs.

Arnold et al. (2011) also state, that the conditions in very deep bedrock are also beneficial in isolating the nuclear waste from the surface. Not only is the radioactive material placed much deeper in the rock (factor of 10 deeper than a GDF), but the high salinity of groundwater at such depths limits upwards flow of water. The depth itself also provides a very thick natural barrier to prevent radioactive material from reaching the surface (Muller et al., 2019).

GDF technology and performance is much more established than that of deep borehole disposal and can accommodate much greater amounts of nuclear waste in larger waste packages. Such facilities are in advance planning stages in several countries with the one in Finland already applying for an operational license (Posiva 2021). Deep borehole disposal would still need to undergo far more scrutiny before being implemented (Arnold et al., 2011). For example, Muller et al. (2019) acknowledge that visual inspections in a deep borehole are much more complicated than in the case of a GDF.

Finally, at least in the U.S., the lack of a regulatory framework also complicates the implementation of deep borehole disposal (Muller et al., 2019). The modularity of the concept could result in a variety of applications being proposed by numerous operators for multiple deep boreholes. In the case of a centralized GDF, waste disposal is handled uniformly by a single operator and regulatory oversight should be more streamlined. Indeed, it is already the case that regulatory authorities in several countries have extensive experience in the oversight of GDFs.

When considering deep borehole disposal for spent SMR fuel, the obvious advantage is that they could be constructed close to the SMR power plants themselves. Thus, each power plant could dispose of its own waste locally, provided the site conditions are suitable. Local disposal would limit the use of extensive transport of waste forms to centralized facilities and would obviate the need for expansion of existing GDFs or construction of new repositories.

It is unknown whether SMRs would be owned and operated by one or more organizations or municipalities in Finland. It could be also possible that the management would be a combination of both (see chapter 4). This however rises an issue under the current legalization for nuclear waste management. According to the Nuclear Energy Act, Section 9 (1987), the holder of a license allowing for the usage of nuclear energy is also responsible for the waste management. Therefore, the managers of the SMRs would be responsible for the waste management themselves. Another considerable aspect is whether or not municipality-level users are willing to store spent nuclear fuel within their borders and if it is even possible. Disposal in municipalities could face public resistance and the local geology might not be suitable for safe, long-term disposal.

From a technical point of view, similar considerations have to be taken into account for disposal of SMR SNF in deep boreholes as in a KBS-3 type repository. The canister, buffer, backfill and closure barriers and site properties must be able to safely store the SNF. The feasibility of borehore disposal in Finnish geological conditions will be studied further in the next phase of the project in 2022, as part of the SMRSiMa project, in cooperation with Geological survey of Finland.

5.1.4 Other disposal concepts

In this section other concepts for SNF disposal are presented. Most of these have not been researched in depth and are more on a theoretical than implementation level. Having information on other disposal options and reasons why they are not widely adopted is however important when discussing SMR-waste disposal. The concepts discussed here include disposal in salt strata and under the sea floor. In addition, long-term above ground disposal and disposal by rock melting are discussed.

The technical readiness of the disposal methods presented in this section is lower than with the KBS-3 concept or even deep borehole disposal. In addition, some methods are not possible under current national or international legal frameworks. Therefore, these methods are clearly much less viable options for SMR-disposal at present. However, the need for spent SMR-fuel disposal lies in the future, and the options presented here could be subject to legislative change and technological maturity and could be reconsidered.

Long-term monitored storage:
Currently, most of the SNF globally is in interim storage, waiting for a suitable disposal route. In Finland, SNF is being stored in water pools directly at the nuclear power plants. At Olkiluoto, the storage building is partly underground with a separate seawater pumping station. The seawater pumping station is used to cool the stored spent fuel assemblies. Another option for interim storage is to store the fuel in dry-casks, which is applied for example in Germany and the US (IAEA & OECD, 2003). If no final disposal option is selected, this storage continues indefinitely and becomes a de facto disposal route. Presumably active monitoring and site control would remain in place. Indefinite storage is in contradiction with the IAEA safety standards for disposal of SNF (IAEA, 2011) as well as the nuclear energy act of Finland. These regulations require that SNF is isolated not only from the biosphere but also from people far into future so, and that the responsibility for SNF-management is not to future generations.

Disposal in salt strata:
Nuclear waste has been disposed in repositories mined in salt strata in Germany (IAEA, 2020b) and in the Waste Isolation Pilot Plant in the U.S. (Hansen & Christi, 2011). However, SNF disposal in salt formations has not yet been carried out. Pavelescu (2001) describes salt formations as a good option for SNF disposal, as salt has naturally very low hydrological conductivity. Salt also has good creep characteristics meaning that any unfilled excavated volumes or damage will naturally seal over time. On the other hand, salt formations also present potential hazards regarding long-term safety due to uplift of salt domes and erosion of the salt by groundwater. There are also hazards with respect to human intrusion as salt can be a resource and therefore future generations could unintentionally mine salt in the repository location. No detailed plans for a SNF disposal facility in salt strata have ever been presented.

It should be noted that Finnish bedrock consist mainly of crystalline bedrock and no suitable salt formations exist in Finland.

Deep rock melting:
A method studied earlier is disposal by rock melting (DRM) as described by Heuze (1981). In DRM, heat producing SNF is disposed deep in a rock mass. The SNF should be disposed in sufficient mass to generate enough heat to melt the surrounding rock. The disposal depth could be 2 km or deeper. The hot SNF could effectively bury itself in the melting rock to even greater depths after disposal. The nuclear fuel would be dispersed in the melting rock and cool over time. Eventually recrystallization would occur forming a permanent rock matrix around the SNF. The borehole used to place the SNF underground would be sealed and backfilled as in the deep borehole disposal concept.

Nirex (2004) states that the problem with deep rock melting, and self-burial methods is, that the final outcome is not predictable. The melting rock characteristics are affected by the local geology. Additionally, it is not obvious whether DRM disposal performs any better than having the SNF disposed in situ to similar depths. The SNF retrievability in a deep rock melting is also not possible, further limiting its usage.

5.2 Disposal of other nuclear waste

The previous section focused on options for SNF disposal. However, in addition to SNF LILW and VLLW will be generated from SMR operations as discussed in Section 3. The disposal methods for these wastes differ from disposal of SNF, as their disposal times can be considerably shorter and the impact of any release much less harmful to the biosphere. Waste disposal concepts for LILW and VLLW are already being implemented worldwide and thus mature technologies exist. In Finland, both the Loviisa and Olkiluoto NPPs produce LILW. The waste is produced during operations and maintenance of the plants. The treatment of such wastes together with waste properties is discussed in Section 3. As the LILW produced in Finland is currently disposed in the two VLJ-caves, an overview of their functions is described.

All non-exempt waste that has been treated is disposed in the caves. The designs at both plant sites are very similar. At Olkiluoto, the cave contains two large underground waste silos, built from 60–100 m depth (Posiva Oy, 2015). One silo is for intermediate waste and one for low level waste. Other infrastructure in the repository includes the access ramps to the disposal levels, hoister halls over the silos and the surveillance building on the surface level (TVO, 2017). The current capacity will entirely accommodate the waste from OL1 and OL2 and will need to be increased for forthcoming waste form OL3. Moreover, the capacity will have to be further extended for the decommissioning waste from all of the power plants to be disposed there.

OL3 waste has different properties when compared to OL1 and OL2 as discussed in Section 2. The LILW waste from OL3 will contain salts that speed up corrosion of the current waste packages and the VLJ-cave. Therefore, the increased capacity planning should consider this factor when designing any expansion. Another option is to design waste packages that would limit the release of any salts from the waste or offer increased corrosion resistance (Posiva Oy, 2015).

Liquid wastes at Loviisa are first stored in the liquid waste storage facility. It features seven 300 m^3 tanks for storage and one additional tank to capture any overflow. Over time, any solids in the liquid waste streams will settle to the bottom of the tanks and the supernatant liquid can be extracted and treated (Posiva Oy, 2015). The remaining wastes (sludges and slurries) are pumped into a solidification plant, where they are solidified as discussed in Section 3. The waste packages are thereafter transferred to the VLJ-cave for disposal.

The Loviisa VLJ cave disposal level lies at a depth of 110 m and has two underground halls for waste package disposal. A third hall was constructed later for temporary storage and waste handling purposes. Additionally, the cave has a separate room for the solidified waste and supporting infrastructure such as access ramps and shafts for an elevator and ventilation (Posiva, 2018). The NPP decommissioning waste is also planned to be disposed in the VLJ-cave, which requires an expansion of the cave. Preliminary plans include three halls: one large hall for large components and two others for dismantled materials (Fortum, 2021).

Fortum (2021) also provides plans for the closure of the VLJ-cave. The long-term safety of the repository is achieved by combining multiple barriers as in the case of the deep geologic repositories. The first barriers are the waste containers themselves. The intermediate-level waste containers are surrounded with concrete so that the entire waste disposal hall is essentially a massive concrete block. The accessways are planned to be sealed with plugs and backfilled with various materials. These designs in combination with the depth of the repository will hinder any unintentional human intrusion to the repository after closure.

Similar methods, where LILW is disposed in caverns relatively close to the surface have been planned and implemented in various countries including Sweden, the U.S. and Canada (Bergström et al., 2011). Additionally, as mentioned in the previous section, nuclear waste has been stored in old salt mines for example in Germany.

5.2.1 VLLW

TVO has started planning for a VLLW-repository at Olkiluoto (TVO & Afry, 2021). Previously, all applicable wastes have been disposed in the general landfill in Olkiluoto or treated elsewhere (TVO, 2012). The new VLLW repository is planned to be a surface repository, where the waste is disposed in a landfill-type structure. It is estimated, that the VLLW deposited in this repository will return to natural radioactivity levels in 200–300 years (TVO & Afry, 2021), and therefore the facility does not need to provide the same long-term containment and isolation performance as the LILW and SNF repositories. The design planned be used in Finland is based on those used by other VLLW repositories globally. Many such above ground repositories exist and are in current use as, for example, in Sweden (Lanaro et al., 2015).

A proposed landfill structure is presented by Keto et al. (2020). The design is comprised of multiple layers aiming to limit water flow into the repository. The foundation layer is built on top of natural barriers such as soil or bedrock, on the surface level. The lowest layer of the foundation is a mineral sealing layer, with a very low hydraulic conductivity. The layer can consist of a mixture of crushed rock and bentonite, or for example a mixture of bentonite and fly-ash (TVO & Afry, 2021). In addition, a leachate collection system is built in the foundation layer, on top of the mineral sealing layer. The leachate system is built from a combination of pipes, drainage materials, such as coarse rock, and synthetic membranes. The leachate collection layer aims to convey any leachates migrating through the repository to a catch basin. If any leachates migrate out of the leachate collection system, the mineral sealing layer slows down their transport considerably before they might reach the natural environment. The top of the foundation layer is constructed of sand or geotextile and provides a solid base to support the waste packages.

The waste packages are disposed in the middle of the repository. The surrounding areas are filled with a material that provides drainage to minimize water contact with the waste and some sorption capacity for any released waste. This material is also used to fill the remaining voids around the waste and provide mechanical support for the cover layer. A gas collection system can also be installed in the waste area. (Keto et al., 2020)

The cover layer is also comprised of multiple layers which limit and direct water flow. The layer will be covered with vegetation, which helps to hinder erosion of the cover. The top layer below the vegetation is planned to be a low-hydraulic conductivity material limiting water flow through it. Under the top layer, a drainage layer made of coarse rock is installed to direct any waters seeping through the cover layer. The layer is on top of another synthetic geomembrane, which further limits water flow. The bottom layer of the cover is made of a similar mineral sealing layer as in the foundation. (Keto et al., 2020)

5.2.2 Decommissioning waste

The methods used for final disposal of LILW and VLLW described in sections 4.2.1 and 4.2.2 can be applied for the decommissioning of LWR-SMRs, with the exception that the dismantling of the actual SMR unit may need to be performed in a factory, possibly outside Finland.

6. Interim storage and transportation

6.1 Interim storage (cooling) for spent fuel

According to NuScale (2020c), the fuel cycle of one of their modules is 24 months. During the refueling, the module to be replaced is cooled down and disconnected from the reactor systems. After this, the module is moved with a remotely operated crane to be disassembled also with remote devices. Approximately one-third of the spent fuel is removed and moved into a separate spent fuel cooling tank in the same reactor building. After the decay heat is at desirable levels, the fuel is transported either into dry-cask storage or into a interim storage facility.

Currently spent nuclear fuel in Finland continues to be stored in underwater in on-site fuel pools by the waste producers as it has been for decades. There is a robust regulatory and licensing framework for interim storage of spent fuel. The requirements for handling and storage of SNF is described in YVL guide D.3 (Handling and storage of nuclear fuel).

Given the large variety of SMR technologies and potential applications (e.g., on-grid versus off-grid), the requirements for interim storage, and potentially reprocessing, need to be further considered. It could be expected that an on-grid application, within an existing operator's facility, would be managed in a similar fashion to current interim fuel storage. However, for off-grid operators/applications, alternative arrangements may be required for the fuel to meet handling and transportation requirements for further interim storage or further processing.

Due to the large variety and emergent nature of SMR technologies, waste properties and characterization information may not be readily available to support the assessment of interim storage requirements. It may also be the case that interim storage durations (i.e., before the waste is transportable for further processing or final disposal) may be in excess of the time the SMR is operational at a site. This could result in additional post-operational security and safeguarding requirements. Finally, some of the waste for interim storage may not be in a form for which there is existing operating experience in Finland (e.g., integrated modules versus fuel assemblies). There may be a need to re-package fuel waste to meet interim storage requirements.

6.2 Storage needed for other waste streams

It could again be expected that for an on-grid application, within an existing operator's facility, low and intermediate level waste would be managed in a similar fashion to current practices. However, off-grid operators/applications may have alternative arrangements that are required for the fuel to meet handling and transportation requirements for further interim storage or further processing.

6.3 Transportation

The requirements for transport of spent fuel and other nuclear waste is described in YVL guide D.2 (Transport of nuclear materials and nuclear waste). Further studies are needed to assess the specific needs for transporting SMR SNF in Finland.

7. Non-LWR spent fuel and LILW waste streams

7.1 Introduction

The LWR-type SMR-reactors were discussed in Section 2. In addition to LWR-reactors, IAEA (2020a) presents four other main types of SMRs. These include high-temperature gas reactors (HTG-SMR), fast neutron spectrum reactors (FN-SMR), molten salt reactors (MS-SMR) and finally micro-sized reactors (M-SMR). In this section the four other types of SMRs are presented, and their main nuclear waste management related properties discussed. These reactors and their specific features regarding SNF and other nuclear waste management concerns are discussed briefly in the following subchapters.

7.2 High-temperature gas cooled SMR

In HTG-SMRs, gas is used as a coolant instead of water. According to Ingersoll and Carelli (2014) helium is used as a coolant in all recent designs of gas cooled SMRs, but other gases have previously been used as well. Operating temperatures reaching 700–800°C can be used in gas-cooled reactors. Such temperatures are much higher than temperatures used in LWR-type reactors. LWR-reactors typically operate at 300–325°C, while LWR-type SMRs may operate at even lower temperatures. Ingersoll and Carelli (2014) state that the benefit of HTGRs is that they can operate at higher efficiencies. However, such high operating temperatures must be taken into account in the reactor material designs.

Another major difference is that typically, graphite moderators are used instead of water for slowing down the velocity of neutrons in the reactors. It should be noted that graphite as a moderator generates relatively large amount of irradiated graphite waste during operation and decommissioning (Fuks et al., 2020) and no disposal routes are yet established for such waste. For example, in a 200-MeWe NPP-sized pebble bed reactor, the estimated amount of irradiated graphite is up to 17 tons annually (Fuks et al., 2020). Irradiated graphite waste typically contains significant amounts of carbon ^{14}C (half-life 5730 years) and chlorine-36 ^{36}Cl (half-life ~300 000 years) (Fuks et al., 2020) (IAEA, 2016) and sometimes also small amounts of corrosion and activation products such as ^{57}Co, ^{60}Co, ^{54}Mn, ^{63}Ni, ^{59}Ni, ^{22}Na, fission products (^{134}Cs, ^{137}Cs, ^{90}Sr, ^{152}Eu, ^{154}Eu, ^{155}Eu, ^{144}Ce) and impurities of uranium and transuranium radionuclides (^{258}Pu, ^{259}Pu, ^{241}Am, ^{243}Am) all of which pose long-term radiological risks to the environment (Fuks et al., 2020). A majority of the graphite waste falls within the LILW category. However, it should be also noted that some of the fuel-contaminated graphite may be classified as HLW (IAEA, 2016).

In addition, there are some specific features in the graphite waste that need to be taken into account in its handling, storage and disposal, including generation of Wigner thermal energy (heat production), graphite dust explosibility and the potential for releases of radioactive gas (Fuks et al., 2020). IAEA (2020a) presents multiple different variants of HTG-SMRs developed in various countries. All the designs use helium coolants and graphite moderators, but otherwise have unique design characteristics. Their thermal capacities range from 100 MW to more than 600 MW, while the electrical capacities range from 35 MW to 300 MW. With some reactors, the power output can be scaled up by adding more reactor modules. Currently, only test reactors, with lower heat and electric outputs, are operational. All of the larger reactors are at either a conceptual design phase or undergoing testing.

Two different fuel handling systems are used in the designs presented by IAEA (2020a). In the first system, fuel particles are fashioned into fuel compacts. The compacts are loaded into hexagonal graphite blocks where the hydrogen coolant can flow through vertical holes. In the second design, called the *pebble bed* design (Figure 7-1), spherical fuel elements typically tri-structural isotropic (TRISO) particles are inserted into the reactor while used elements are extracted from the bottom at the same rate. In most designs the fuel particles consist of UO_2, but some designs can use PuO_2 and MOX as fuel. The fuel enrichment varies between 8 % and 20 %, which is significantly more than the approximately 4 % in the Finnish NPPs (Rossi et al., 2009). In a SMR unit, the amount of pebbles varies significantly from ~27 000 even up to more than 450 000 pebbles in different pebble bed reactor designs (IAEA, 2020a).

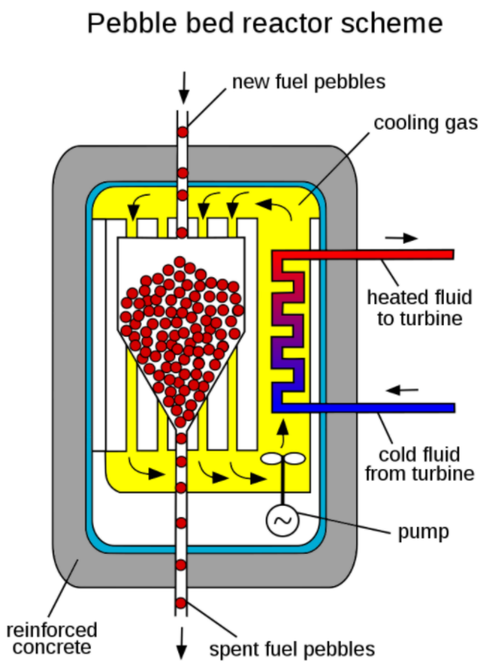

Figure 7-1. Basic principle of a pebble bed reactor (PBR) and a graphite pebble typically with diameter of 60 mm (Kühn, 2006; Picoterawatt, 2013) licensed under CC0 1.0 and CC BY-SA 3.0. In a large NPP, the amount of pebbles can be from 8000 to 25 000 (Fuks et al., 2020).

According to IAEA (2020a), nuclear waste streams, excluding SNF, from the HTG-SMRs includes liquid waste, off-gases and irradiated components, such as the reactor vessel. Most of these wastes can be treated with conventional methods already used for NPP wastes across the world. However, due to different fuel characteristics, methods used for SNF-management should be adjusted. Ingersoll and Carelli (2014) indicate that SNF generated by HTG-SMRs will produce more waste for the same output of energy when compared to fuel and reactor types in conventional large NPPs. However, the waste produces less heat per volume. IAEA (2020a) states that waste produced in a hexagonal-type SMR reactor of US design produces SNF with 50 % less decay heat when compared to LWRs. In both the KBS-3 and deep borehole disposal methods discussed in Section 5, the SNF is encapsulated in canisters. The canisters are disposed in holes of various designs and surrounded by a buffer material. The properties of the buffer material can be negatively affected by high temperatures as discussed by Juvankoski and Marcos (2010). This constraint will also need to be incorporated into designs for SMR SNF disposal in terms of heat load per canister and repository layout, as disposing waste canisters too close to each other can transfer too much heat to the buffer material in between. With the lower decay heat of HTG-SMR, such waste canisters could possibly be disposed closer to each other, which would decrease the need for excavation. On the other hand, larger waste quantities would mitigate the benefit. Therefore ways to reduce the volume of SNF produced should be sought for as stated by IAEA (2020a).

Considering pebble bed SMRs, separation of fuel kernels (TRISO particles) from the graphite pebbles might be needed to be prior to disposal, e.g., by low or high temperature acid treatments (Guittonneau et al., 2010). Such processing would result in the generation of secondary waste streams and would require facilities that do not currently exist in Finland and processing outside Finland is currently not allowed by the Finnish legislation. The fuel can be further reprocessed, stored and possibly disposed in a similar manner as spent fuel from LWR-SMRs. However, significant concept development would be needed before this type of solution could be licenced in Finland.

For the irradiated graphite produced in a pebble bed SMR, various disposal solutions have been suggested but there is currently no generally accepted approach (Fuks et al., 2020). The alternatives currently in use are 1) long-term storage in special containers filled with a binder (polymer matrix or cement), 2) conditioning for significant volume reduction (95%) by combustion or incineration creating carbon dioxide in the process or 3) recovery of the graphite (Fuks et al., 2020). Production of Wigner thermal energy (risk of overheating and fire), graphite dust explosibility and potential for releases of radioactive gas emissions must be taken into account during any processing (Fuks et al., 2020). Further information on graphite processing, waste acceptance criteria (WAC) and disposal have been discussed in IAEA (2010); (2016) and as part of the GRAPA project (The International Project on Irradiated Graphite Processing Approaches).

7.3 Fast neutron-spectrum SMR

The second class of SMRs presented by IAEA (2020a) are the FN-SMRs, also called liquid metal cooled fast reactors. Conventional NPPs use fissile isotopes of uranium and plutonium as fuel. The NPPs of Finland use nuclear fuel, where the amount of the fissile ^{235}U isotope is enriched to approximately 4 % as discussed previously. As natural uranium has less than 1 % of ^{235}U, large quantities of natural uranium are required for the enrichment process. In all reactors, some of the remaining non-fissile isotopes are converted to fissile isotopes by neutron capture. In FN-SMRs this property is optimized so, that more non-fissile fuel can be used.

In FN-SMRs a fission reaction is driven in the core by for example a fissile isotope of plutonium (Kasahara, 2014). This reaction generates fast neutrons, which are not moderated by a neutron moderator typical in other reactor designs. Therefore, the velocity of the neutrons is not slowed down, hence the name. The fast neutrons are captured by non-fissile ^{238}U isotopes, which are then transformed to fissile ^{239}Pu isotopes. More fissile material is generated than consumed in the reaction, and the reactors are also often called breeder-reactors.

The reactor core in FN-SMRs is cooled with a liquid metal (Kasahara, 2014). The heat is transferred out of the reactor core using this liquid metal, which then exchanges the heat with a secondary liquid metal cycle, and ultimately with a tertiary water-steam system. Liquid metal coolants are used as they do not act as efficient neutron moderators, allowing for the unimpeded generation of fissile isotopes. In addition, the liquid metals can be used at higher temperatures than water in conventional water-cooled reactors, resulting in increased power conversion efficiency as discussed by Ingersoll and Carelli (2014).

Various designs for FN-SMRs are presented by IAEA (2020a). All the presented designs are still either at conceptual or detailed design phases. Different metals as coolants, e.g., lead, lead-bismuth and sodium are proposed. Additionally, one design uses helium instead of liquid metal as coolant. The various designs have highly different thermal and electric outputs ranging from 8–15 $MW_{(t)}$ and 3–5 $MW_{(e)}$ to 700–950 $MW_{(t)}$ and 300–450 5 $MW_{(e)}$. The reactors use various fuel types including conventional low enriched uranium, mixed oxide pellets and uranium-zirconium alloys. Typically, depleted uranium is used as the “fertile” fuel, i.e., the non-fissile fraction of the fuel that is transformed to new fissile fuel.

The main benefit of using FN-SMRs is that less fissile material containing fuel is required. This fuel design also allows the reaction in the core to be sustained for very long times, even decades. Thus, in some cases, no shutdowns are required for refuelling (IAEA, 2020a). This also significantly decreases the amount of SNF generated per unit of energy produced (Ingersoll & Carelli, 2014). A potential drawback is that the liquid metal coolant can be very demanding on any surrounding materials both chemically and thermally and will require special handling during operations and decommissioning. Another issue that has slowed the development of breeder reactors is that the spent fuel contains higher amounts of fissile plutonium than conventional reactors (Kessler, 2012). Firess et al. (2015) state that research on the proliferation issues related to fast-SMRs has not been conducted in detail. In their calculations, one 10 $MW_{(e)}$ SMR-core with a lifetime of 30 years could contain the equivalent of 8 kg of weapons-grade plutonium. This potential will need to be considered in the waste handling for FN-SMRs, as such quantities of weapon-grade plutonium represent a large proliferation risk. It must be ensured during interim storage, transport and final disposal that SNF originating from breeder-type reactors remains under authorized handling. A current recommendation for SNF-disposal is that it should be retrievable only after the closure of a repository.

7.4 Molten salt SMR

The third class of SMRs presented by IAEA (2020a) are molten salt reactors (MS-SMRs). Two main variants exist where the molten salt functions as fuel and coolant or as coolant only. In designs using the molten salt as fuel and coolant, the fuel is dissolved in a salt coolant. The fuel salt is circulated through the core and heat exchangers which transfer the heat to a conventional steam generator. Designs using molten salt as a coolant only also employ a solid-fuel system (e.g., TRISO particles). Heat is carried from the core by the molten salt. It is possible to use MS-SMRs as breeder-type reactors, where non-fissile isotopes are converted into fissile isotopes.

According to Yamaguchi et al. (2017), fluoride salt coolants are typically used, with some variants using employing other molten salts. Operating temperatures, depending on the salt composition, range between 400°C and 800°C. Heat and electric power capacities can reach 750 $MW_{(t)}$ and 300 $MW_{(e)}$, respectively. The main difference in fuel types depends on whether the fuel is dissolved in the molten salt or not. In the fuel salt variants, the fuel is typically either low enriched uranium-, thorium-, or plutonium tetrafluoride or a mixture thereof. In designs, where the molten salt acts solely as the coolant, typically spherical fuel pellets or other fuel assemblies are used in the reactor core. One design uses an initial fissile molten salt fuel inside conventional fuel assemblies where the fuel in the molten salt consists of very low-purity plutonium in combination with uranium trichloride recycled from SNF. The enrichments in MSRs using uranium range between 2 % to slightly less than 20 %.

One benefit of MS-SMRs is that they can be used to recycle SNF from conventional NPPs. Therefore, they can reduce the total SNF capacity required to be disposed. According to IAEA (2020a) liquid fuel is also easier to recycle. This could be done for example centrally for multiple MS-SMRs. Another benefit is that the high process temperatures allow power cycles to produce electric power more efficiently. Furthermore, the high process temperatures can be achieved at low pressures, mitigating the risk of pressure induced damage (Ahonen et al., 2020). There are disadvantages to using MS-SMRs as well. It is unclear what spent fuel waste from small modular MSRs would look like; stable waste forms would need to be developed. Also, corrosion processes in MSRs are of significant concern.

7.5 Micro-sized SMR

The final category presented by IAEA (2020a) are the Micro-sized SMRs (M-SMRs). These systems encompass a wide range of designs including those presented above. In addition, a heat pipe-type design is common. M-SMRs share in common very low thermal and electrical capacities ranging from a few megawatts to less than 20 megawatts. The designs are also typically very simple due to the low thermal outputs and reliance on passive safety features. The power plants footprints themselves are also relatively small ranging from the size of a shipping container to 0.01 km^2. Most M-SMRs are in conceptual design phases with only a design called Aurora one having a combined license application accepted in 2020. However, this license application has been since denied by the US Nuclear Regulatory Commission due to information gaps in their application (WNN, 2022a).

The fully passive and portable designs of M-SMRs suggest that they can be applied in remote and urban areas with little to no monitoring and maintenance required. Some designs even fit into shipping containers and are thus well suited for usage globally (IAEA, 2020a).

The heat-pipe cooled reactor design consists of the fuel, moderator and a heat pipe used for reactor cooling. Heat pipes do not require pumps to operate, therefore they can be used for passive cooling. The heat pipes transfer heat to a secondary circuit using a heat exchanger. Fuels used in such reactors include uranium silicide and uranium oxycarbide in TRISO form while moderators are calcium-hydride and other metal hydrides.

The different reactor types produce different types of SNF. However due to the small sizes of the reactors, all SNF and waste streams are very small. One design allows for the waste to be disposed in the reactor vessel itself. Some designs have planned to use a closed fuel-cycle, where the SNF can be processed to be used again as a fuel in the reactors. (IAEA, 2020a).

Xu et al. (2020) estimated the waste streams originating from a HTG, MS and liquid metal-cooled I M-SMR proposed to be used in Canada. These reactors employed TRISO, uranium chloride and uranium nitride fuel and operated at 20 MW_e, 37.5 MW_e and 10 MW_e capacities, respectively. The fuels used by the presented SMRs have higher U enrichment and burnup than conventional LWRs.

Xu et al. (2020) state that waste originating from the steam-water loops have comparable waste streams to conventional LWRs of similar scale. Such wastes were not considered in their assessment of SMR characteristic wastes. Additionally they calculated all waste streams by assuming a 30 $MW_{(th)}$ power, so that the reactors could be compared. Other assumptions were, that all waste from possible multiple reactors would be treated in one centralized location. Additionally, the fuel used would produce similar amounts of SNF per one $MW_{(th)}$ as conventional NPPs.

In addition to the TRISO fuel, the HTGM-SMR uses helium gas as a coolant and graphite as the moderator material. The SNF produced by the fuel would be comparable to a similarly sized LWR. The helium coolant becomes radioactive from particles of activated reactor materials. Xu et al. (2020) state that it can be treated with a similar system as the off-gases at a conventional LWR where HEPA filters are used for off-gas treatment. These filters would result in an estimated 6 m^3 of ILW from the 30 year life cycle of a 30 $MW_{(th)}$ plant. Furthermore, the graphite moderator is activated during operation, which as discussed before does not have existing waste treatment methods. If the reactor is throughout the life cycle without service, moderator waste is only produced at the end. This would result in 69 m^3 of ILW graphite waste.

The MS-SMR reactor in the estimation of Xu et al. (2020) uses graphite as moderator and a chloride salt as coolant. They estimated, that 1.4 m^3 of HLW SNF in the form of fuel salt would be produced in the life time of 30 years of operation of a 30 $MW_{(th)}$ plant. In addition, the primary and secondary molten salt coolant becomes radioactive during operation resulting in 4.8 m^3 of ILW. Finally, at decommissioning, the graphite moderator would result in 6.7 m^3 of ILW.

Finally, the FN-SMR uses molten lead as a coolant and no neutron moderator. The estimated lead inventory of approximately 54 m^3 would be initially HLW and lower to ILW in approximately 3 years (Xu et al., 2020). Although they only estimated waste streams from three different reactor types, some of their method could be used in Phase 2 for waste characterization of SMRs.

Ramana and Mian (2014) present some issues associated with radioactive waste produced by breeder type SMRs, such as the FN-SMR presented by Xu et al. (2020). They state, that although breeder reactors can be used to significantly reduce the amount of SNF to be disposed, the amount of plutonium is much larger than conventional reactors. This was also discussed previously in Section 7.3. They state, that if such reactors are misused, the plutonium could be recovered for ill uses posing a proliferation risk.

7.6 Development needed for the management of spent fuel and LILW waste streams from non-LW SMRs

The large variety of SMR designs and potential applications present a challenge in assessing the requirements for long-term disposal. Current disposal concepts in Finland are designed to manage waste streams from large, LWR NPPs. In the absence of specific information on the fuel type, size, composition, characterization, etc., it is not possible to determine the potential impacts to the current disposal concept, safety case, or costs. Moreover, some waste forms may require further processing to meet waste acceptance criteria for disposal. The details of such processing are as yet unknown. Further work will be needed to assess the management of non-LW SMR wastes. This could include:

- Determine the potential forms and characteristics of the spent fuel wastes to be considered for interim storage, transportation, potential processing and/or reprocessing, and subsequent disposal.
- Resolve gaps in the regulatory framework and responsibilities in different steps of the waste management. Currently, licensed operators are responsible for management of the entire life cycle of nuclear waste in Finland, and if special processing is needed outside Finland, the responsibilities need to be defined.
- Are the fuel cycles open or closed? Currently only closed fuel cycles are possible with respect to Finnish legislation.
- Evaluate the risks linked to proliferation issues.
- Establish the characteristics of LILW waste streams (nuclide composition and concentrations, specific characteristics of the wastes, etc.) that need to be taken into account in the handling, processing, storage and final disposal.

Once the waste streams and their characteristics have been established, the development needed for final disposal begins from selecting the right disposal depth for each waste stream and development of conceptual designs for the repositories along with evolving safety case and operational and long-term safety related requirements. In general, the spent nuclear fuel may be disposed similarly to KBS-3 concept underway for spent nuclear fuel from LWR NPPs (as proposed here for SNF LW-SMRs as well), even if no ready solutions accounting for the specific characteristics of the non-LWR SNF currently exist. Applying an alternative concept for disposal of SNF in Finland would require extensive development.

Considering LILW waste streams, the disposal can be done at an intermediate depth geological facility, but the concept needs to be developed taking into account the specific characteristics of the waste from non-LW SMRs.

8. Discussion and conclusions

Posiva Oy has applied on 30.12.2021 for an operation license to conduct final disposal of spent nuclear fuel in Finland in the ONKALO® repository and it is expected that final disposal operations will start within this decade (Posiva 2021). The license covers 6500 tons of uranium from the currently operating nuclear power plant units (OL1-3, LO1-2) located in Eurajoki (Olkiluoto) and Loviisa (Hästholmen) (Posiva, 2021b). The development of the KBS-3V disposal concept and the safety case took approximately 40 years. LILW repositories are already in operation both in Olkiluoto and Loviisa.

The concept used for spent nuclear fuel disposal in Finland can likely be adapted to creating a disposal route for spent fuel from LW-SMRs, but in any case, an individual licensing process is required. Figure 8-1 provides a preliminary roadmap towards licensing a disposal facility for LW-SMR SNF. The selected management strategy (centralized, decentralised or hybrid model) and ownership structure will significantly affect this process as will the spent fuel characteristics of the LW-SMRs.

With respect to non-LW SMRs and adapting new disposal concepts (e.g., deep borehole disposal), a great deal more development work would be needed for pre-disposal management of the SNF and LILW, EBS designs and for studying the applicability of the concept in Finland (geology, site characteristics, technical feasibility, etc.).

Additionally, stakeholder, political and public opinion will be highly important factors in both the siting of SMR plants (including interim storages) and final disposal site(s) for the waste produced in the SMRs.

Considering current legislation in Finland, conflicts may arise concerning alternative waste management schemes and even assigning ownership of the waste. Furthermore, pre-disposal management or decommissioning for some SMR technologies might require that these activities take place outside Finland.

Pre-study phase:
- Define regulatory framework.
- Define responsibilities (e.g. in case of centralized waste management)
- Identification of waste streams (HLW, LILW, VLLW)
- Define interim storages needed at the site (HLW, LILW, VLLW).
- Identification of pre-disposal options.
- Identification of processes and waste streams from decommissioning.
- Identify requirements for the SMR plant site (with interim storages) and for the repository sites.
- Identify requirements for SNF transport.
- Start building up a safety case.
- Map stakeholder and public opinion.

Conceptual design phase:
- Define waste acceptance criteria (WAC)
- Define conceptual designs for the final disposal repositories.
- Initiate site screening process.
- Define conceptual designs for the interim storages.
- Development of pre-disposal management options.
- Initiate EBS design development, e.g. encapsulation and canister design.
- Evolving safety case and requirements management systems for the site and EBS.

Construction license phase:
- Site selection including Environment Impact Assessment
- Detailed repository design
- Detailed designs for interim storages.
- Detailed plans for pre-disposal management
- Detailed designs for EBS
- PSAR preliminary safety assessment report.

Operational license phase:
- Final plans for interim storages.
- Final plans for the repository and for EBS.
- Plans for closure and decommissioning.
- FSAR final safety assessment report.

Figure 8-1. Preliminary roadmap towards licensing of a disposal facility for LW-SMR waste.

The fuel form will be typically the same for LW-SMRs as for the EPR-type NPPs and the canister design can be conceivably adapted for the LW-SMR fuel. The most evident design aspect for further development of the canister is linked to the shorter length (approximately ½ of the length of the fuel assemblies in a EPR type NPP) of the fuel assemblies in the two different SMR cases considered. An attractive option would be to stack LW-SMR fuel assemblies end-to-end in existing canister designs, but, ultimately, loading arrangements will need to demonstrate compliance with repository design constraints (heat load per canister, criticality safety criteria, etc.).

2D calculations were performed with the continuous-energy Monte Carlo code Serpent made in order to compare the spent fuel characteristics from two example LW-SMRs to spent nuclear fuel from currently operating NPPs in Finland. Based on the calculations the following preliminary conclusions were made considering final disposal and engineered barriers:

- As the fuel and reactor types are very similar to those used in the current LWRs, the spent fuel features with comparable burnup are rather similar with small reactors.
- The generally lower discharge burnup characteristic for the smaller reactors facilitate the final disposal safety case at the assembly-level in terms of lower radioactivity and the consequent lower heating power.
- The question of dose rate around a spent fuel assembly was completely ignored in the present study, but is one of the questions to be studied in the future.
- The concentration of mobile nuclides in the spent fuel is smaller.
- Lower average burnup especially in combination with the relatively high U-235 enrichment (the case with NuScale) contributes to higher post-irradiation reactivity, which may be a potential criticality safety concern. The question can be answered with 3D assembly burnup profile and calculations in a realistic repository configuration.

- The conclusions from the 2D calculations require confirmation with 3D calculations. In addition, some of the input needed for the calculations were based on assumptions.

Considering the amount of SNF produced in a SMR in comparison to NPPs, the preliminary conclusion that somewhat more SNF is produced in SMRs per kW/ft (gigawatts annually) produced. However, due to lower decay heat, less space (lower excavation costs) and less material for clay based backfill materials may be needed in the repository assuming a KBS-3 type of repository.

In general, it can be assumed that the waste streams form LWR-SMRs are similar to NPPs and can thus be pre-treated and disposed in the same manner in an intermediate depth repository. For example, processing waters need to be treated from contaminants generating ion exchange resins falling into the ILW category. ILW may also be produced when replacing irradiated components from an SMR unit. Due to differences in the operating systems and in the size of the units, possibly somewhat less operational waste is generated in SMRs in comparison to SMR. In any case, the SMR plants need to prepare for handling and at least for interim storage of the LILW waste streams similarly as in a typical NPP. Considering decommissioning waste, if should be noted that some operations linked to the decommissioning of the reactors may need to be done in a factory, possibly outside Finland (possibly requiring change in legislation).

How the spent nuclear fuel and LILW waste management is organised in practice for SMRs may depend on various factors including the ownership base and size, and distribution of the plants in Finland. However, the basic principle prevails that the producer of the energy/waste is responsible for the waste. In case of small geographically widely distributed units (e.g. in the case of district heating units for smaller municipalities), central handling and disposal sites may need to be considered, at least for SNF, but possibly also for LILW, for ensuring safety and cost effectiveness of the disposal. This option requires transportations inside Finland and sufficient interim storage capacities at the site. Centralised, hybrid or distributed locations for waste disposal require local stakeholder and public acceptance, to be studied further in the next phase of the project.

Considering concepts available for disposal KBS-3V is at the most advanced stage and can be adapted for disposal for SNF from LWR SMRs. The most relevant alternative of this concept considering Finnish crystalline bedrock is the borehore disposal concept. The applicability of this concept will be also assessed as part of next phase of the project.

In general, development of the waste management non-LWR SMRs (e.g. high temperature gas cooled reactors and molten salt reactors) requires further development in the field of spent nuclear fuel pre-treatment (e.g. separation of the fuel from matrix), better identification of the LILW waste streams, development of the waste acceptance criteria for LILW and development of the final disposal concepts taking into account the special characteristics of the SNF and these waste streams. Specific nuclides (e.g. Pu) generated in some of the non-LWR reactors need to be considered also from the proliferation point of view. In principle it is possible to develop the disposal concepts based on the current ones planned for SNF from NPPs, but the concept development, testing (in different scales), building up the safety case and the licencing process will be a time-consuming process.

Recommendations for further studies include:

- Further 3D calculations for better understanding on the effect of fuel characteristics on the final disposal and on the design basis of the barriers (e.g., canister design).

- Further studies to identify aspects that may have effect on the disposal of LILW from SMR units.

- Identification of waste streams from non-LWR SMRs and pre-disposal handling and disposal of the SNF and LILW (e.g., graphite).
- Requirements for siting of SMR plants and consequent siting of intermediate storage facilities for spent fuel and other nuclear waste.
- Siting of repositories for SMR SNF, requirements and options for decentralised and centralised management.
- Public acceptability related to waste management concerning SMRs.
- Intermediate storage at the SMR plant sites.
- Transportation of SMR spent fuel (to another location).

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