Nuclear Waste Attributes of SMRs Scheduled for Near-Term Deployment

Nuclear Fuel Cycle and Supply Chain

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SUMMARY

The purpose of this study is to evaluate the nuclear waste attributes of Small Modular Reactors (SMRs) scheduled for deployment within this decade using available data and established nuclear waste metrics, with the results compared to a reference large Pressurized Water Reactor (PWR).

The current fleet of commercial nuclear reactors in the U.S. is composed of 92 large Light Water Reactors (LWR) with an average electricity generating capacity of over 1,000 MWe each. These large LWRs built on-site in massive construction projects have been the mainstay of the industry for the last 50 years. However, new construction soon is expected to include several designs of smaller reactors primarily fabricated in factories and installed in the field in modules. Some of these SMRs will also be LWRs, while some will use other coolants such as liquid metals, molten salts or gases, and different types of fuels. The technologies and economics of SMRs have been the focus of many studies, but there has been only minimal information published on the amount of nuclear waste different types of SMRs are expected to generate and no reports focused on near-term-deployable designs.

In this study, the nuclear waste attributes of three small reactors scheduled for near-term-deployment, VOYGR™ (from NuScale Power), Natrium™ [ab] (from TerraPower), and Xe-100 (from X-energy), were assessed by comparing nuclear waste metrics with those of a reference large Pressurized Water Reactor (PWR). These reactors were selected for this study for several reasons:

- First, they represent a range of reactor and fuel technologies, allowing comparison of the waste performance of these different technologies. VOYGR™ is a PWR design using the same type of ceramic fuel as found in larger LWRs, Natrium is liquid metal cooled and uses a metal alloy fuel, and Xe-100 is a helium cooled reactor using pebbles containing TRi-structural ISOtropic (TRISO) particle fuel. Collectively, these three technologies cover a range of proposed SMRs.
- Second, they represent comparatively mature designs that have been selected for DOE support for near-term deployment and have some expectation of commercial viability. Each design has been developed to improve performance, providing for a meaningful comparison versus existing commercial reactors.
- Third, they are all active designs deployable in the near-term. Sites for first units of each design have been announced, licensing activities are underway, and all three are scheduled to be operational by the end of the decade. An understanding of the waste attributes of these designs is needed to inform on any significant differences that may impact future nuclear waste management.

The reactor concepts and key design parameters are shown in Table E-1. For a consistent comparison, the reactor lifetime and a capacity factor of each reactor is assumed to be 60 years and 90%[c], respectively. The information on reactor design and performance parameters of the three small reactors were obtained from open literature. Missing data needed for evaluating waste metrics were calculated in this work or obtained from a reactor with a similar power rating and design features. A draft of the completed report was provided to the vendors of the three SMRs to ensure the open literature information was not misinterpreted.

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[a] A TerraPower and GE Hitachi technology

[b] An accepted definition of SMRs is for reactors up to 300 MWe, while Natrium is rated at 345 MWe. However, it is included here because it is the most mature design in its technology class and is closer in size to SMRs than to current reactors.

[c] This is conservative, as current reactors have been averaging 92% and most new reactors are being designed for even higher values.

The nuclear waste metrics used in this study address "front-end" wastes generated during the fuel manufacturing process, "back-end" wastes arising from the spent nuclear fuel (SNF), and end-of-life wastes from decommissioning of the reactors when they are retired[d]. The specific metrics chosen for this analysis are derived from metrics developed through an extensive process during a comprehensive assessment of nuclear fuel cycles completed in 2014 (DOE 2014). That comprehensive fuel cycle assessment was performed by national lab and industry experts and included input from and independent review by experts from national labs, academia, government, industry, and the public. This current study also notes the authors' assessment of why differences in the calculated values may or may not be significant, based on experience from the work completed in the earlier fuel cycle evaluation (DOE 2014).

Table E-1 Comparison of reactor design parameters used in this study

	Ref. PWR	VOYGR™	Natrium	Xe-100
Power, MWe	1175	77	345	80
Thermal efficiency	34%	31%	41%	40%
Burnup, GWd/t	50.0	49.5	146	169
Uranium enrichment, %	4.5	4.95 a)	16.5	15.5
Fuel form	UO_2	UO_2	U-Zr metal w/o sodium-bond	TRISO(UOC) /Pebble
Reactor lifetime and capacity factor (years/%)	60 / 90	60 / 90	60 / 90	60 / 90

a) Public information indicates "<5%", so conservatively used 4.95%

- Front-end wastes are represented by the depleted uranium (DU) mass generated during the uranium enrichment process. All designs use the same once-through fuel cycle as current LWRs, and the DU generated has no useful application in any current once-through cycle[e]. However, DU would be an asset for a recycle fuel cycle.
- Back-end wastes are represented by five types of metrics associated with waste handling and geologic disposal: discharged SNF mass and volume, SNF activity from 10 to 100,000 years after discharge, decay heat of SNF at 10 and 100 years, and radiotoxicity of SNF at 10,000 and 100,000 years.
- Decommissioning waste metrics are represented by low-level radioactive waste (LLW) volume. Due to different disposal requirements for different classes of LLW (near surface vs. geological repository), the decommissioning LLW volume is divided into the combined volume of Class A, B, and C LLW and a separate metric for Greater Than Class C (GTCC) LLW.

Results

The calculated waste metrics are summarized in Table E-2. All metrics are normalized per GWe-year of electricity generation. Values in parentheses indicate the ratio of a waste metric to that of the reference PWR.

Front-end waste

DU mass is proportional to enrichment and inversely proportional to burn-up and thermal efficiency.

- Compared to the reference PWR, VOYGR™ generates 23% more DU mass due to relatively higher fuel uranium enrichment, lower burnup, and lower thermal efficiency.

[d] Operational wastes other than spent fuel were not addressed due to insufficient information.

[e] The "breed and burn" once-through fuel cycle concept could use some DU as an input but requires significant advancements in fuel cladding technology before it can be realized.

- Even though the uranium enrichment for Natrium and Xe-100 fuels is a factor of 3 – 4 higher than that of the reference PWR, the normalized DU mass of Natrium is only 17% higher and Xe-100 is 3% lower than the reference PWR because burnup increases by factors of 2.9 and 3.4 respectively and thermal efficiency is higher by ~20%.

Back-end waste

Back-end metric values are inversely proportional to burnup and thermal efficiency and affected by reactor-specific design features, such as neutron spectrum and fuel type. Compared to the reference PWR:

- VOYGR™ generates 1.1 times the SNF mass and 1.1 times the SNF volume of the reference large PWR due to relatively lower burnup and thermal efficiency. VOYGR™ SNF has slightly higher activity, decay heat and radiotoxicity.

Table E-2 Comparison of nuclear waste metric values calculated in this study

	Ref. PWR	VOYGR™ a)	Natrium a)	Xe-100 a)
DU mass, t/GWe-year	179	220 (1.23)	209 (1.17)	174 (0.97)
SNF mass, t/GWe-year	21.7	23.9 (1.10)	6.10 (0.28)	5.41 (0.25)
SNF volume, m^3/GWe-year	9.58	10.4 (1.08)	5.56 (0.58)	118 (12.3)
SNF activity (Ci/GWe-year) compared to Ref PWR @ 10^1, 10^2, 10^3, 10^4, 10^5 years		(1.07, 1.08, 1.04, 1.05, 1.08)	(0.63, 0.71, 0.63, 1.40, 1.17)	(0.79, 0.80, 0.45, 0.38, 0.58)
SNF decay heat, kW/GWe-year				
@ 10 years	40.6	42.2 (1.04)	24.5 (0.60)	32.2 (0.79)
@ 100 years	9.76	10.3 (1.05)	4.65 (0.48)	6.36 (0.65)
SNF radiotoxicity, x10^8 Sv/GWe-year				
@ 10,000 years	1.21	1.27 (1.06)	1.78 (1.47)	0.413 (0.34)
@ 100,000 years	0.0860	0.0912 (1.06)	0.127 (1.48)	0.0406 (0.47)
Decommissioning LLW volume				
Classes A, B, and C, m^3/GWe-year	645.3	573 (0.9)	N/A b)	N/A b)
GTCC, m^3/GWe-year	0.13	0.72 (5.7)	0.0 – 0.55 (0.0 – 4.4)	0.0 – 24.5 (0.0-193.1)

a) Values in parentheses indicate the ratio of a waste metric to that of the reference PWR.
b) Not available because the open information is insufficient to calculate the LLW volume.

- Natrium generates 72% less SNF mass, 42% less SNF volume and 100-year decay heat is reduced by 52% due to much higher burnup and higher thermal efficiency. Initially, activity is ~40% lower, but long-term activity is 20-40% higher due to a higher plutonium content in the SNF. The plutonium content also impacts long-term SNF radiotoxicity, which is 47% higher at 10,000 years.
- The Xe-100 SNF mass and 100-year decay heat are lower by 75% and 35% respectively, again due to much higher burnup and higher thermal efficiency. Normalized activity is initially ~20% lower and continues to drop as the fission products decay. The SNF radiotoxicity is reduced by 66% at 10,000 years as plutonium and minor actinides are minimized. However, the SNF volume is higher by a factor of 12.3 due to the fuel design which includes large amounts of graphite moderator and non-fuel matrix/coating materials.

Decommissioning waste

Decommissioning Class A, B, and C LLW consist of building materials activated by neutrons or contaminated by radioactive isotopes. GTCC LLW consists of reactor components located near the active core and activated above Class C levels. For PWRs, < 1% of decommissioning LLW is GTCC.

- The decommissioning volume Class A, B, and C LLW for VOYGR™ is 10% smaller than that of the reference PWR. Owing to a lack of detailed design information on reactor buildings, the decommissioning volume of Class A, B, and C LLW of Natrium and Xe-100 was not calculated, but the waste arising from disposal of coolants was assessed and found to be minimal.
- Compared to the reference PWR, the normalized GTCC volume for VOYGR™ is a factor of six larger. Natrium includes radial neutron reflectors and Xe-100 includes radial graphite blocks that protect other core structures from activation. These designs do not generate appreciable GTCC LLW if the reflector assemblies and graphite blocks are periodically replaced before they are activated to the GTCC level. However, compared to the reference PWR, Natrium and Xe-100 generate a factor of 4 and 193 more GTCC volume respectively when the reflector assemblies and graphite blocks reside in the core for the reactor lifetimes.

It must be noted that except for the SNF volume, other SNF waste metrics are driven by fundamental physics while the decommissioning waste is highly dependent on decommissioning technologies used. Therefore, there is a large uncertainty in the calculated values of the decommissioning waste values given the time available (in decades) for technology enhancement.

The waste attributes of the SMRs studied show both some similarities to the reference LWR and some potentially significant differences. Front-end waste attributes from SMR fuel production range from equivalent to 1.4 times the LWR reference. Back-end waste attributes for spent fuel disposition vary from large reductions to small to moderate increases in heavy metal mass (factors of 0.2 to 1.2), activity (factors of 0.3 to 1.5) and radiotoxicity (factors of 0.5 to 1.5). These differences have limited impact on long-term repository isolation. SMR designs can vary significantly (factors of 0.6 to 12.3) in volume (and thus heat generation density), however these differences are readily amenable to design optimization for handling, storage, transportation, and disposal technologies. Waste attributes from decommissioning can vary greatly depending on design and decommissioning technology choices and might dictate optimization of those choices.

Given the analysis results in this study and assuming appropriate waste management system design and operational optimization, there appear to be no major challenges to the management of SMR wastes compared to the reference LWR wastes. The results of this study are only applicable to a once-through fuel cycle. Any of these reactors, including the reference LWR, could be used with fuel recycle, resulting in reductions in most waste attributes as indicated in the E&S study (DOE 2014).

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ACRONYMS

ARDP – Advanced Reactor Demonstration Program

BWR – Boiling Water Reactor

DOE – Department of Energy

DOE-NE – Department of Energy - Office of Nuclear Energy

DU – Depleted Uranium

E&S – Evaluation & Screening Study

EBR-II – Experimental Breeder Reactor II

EOL – End of Life

FFTF – Fast Flux Test Facility

GTCC – Greater Than Class C Low Level Waste

HALEU – High Assay Low Enriched Uranium

HLW – High Level Waste

HM – Heavy Metal

iHM – Initial Heavy Metal

LLW – Low Level Waste

NRC – Nuclear Regulatory Commission

NPM – Nuclear Power Module

NSSS – Nuclear Steam Supply System

PWR – Pressurized Water Reactor

RCRA – Resource Conservation and Recovery Act

RU – recycled uranium

RTh – recycled thorium

RXB – Reactor building

SFR – Sodium Fast Reactor

SMR – Small Modular Reactor

SNF – Spent Nuclear Fuel

SONG – San Onofre Nuclear Generating Station

SPF – Sodium Process Facility

NUCLEAR WASTE ATTRIBUTES OF SMRS SCHEDULED FOR NEAR-TEAM DEPLOYMENT

1. Introduction

Nuclear energy has been a steady source of ~20% of US electricity generation since the 1990s and has also been the primary source of clean firm power. Due to an increased need for clean firm power to meet mid-century climate goals, the nuclear share of total generation is now expected to grow and the first steps in deploying new reactors is underway.

The current fleet of commercial nuclear reactors in the U.S. is composed of 92 large Light Water Reactors (LWR) with an average electricity generating capacity of over 1,000 MWe each. Roughly 2/3rd of these LWRs are Pressurized Water Reactors (PWR) and 1/3rd are Boiling Water Reactors (BWR). These large LWRs were all built on-site in massive construction projects and have been the mainstay of the industry. However, new construction soon is expected to include several designs of smaller reactors primarily fabricated in factories and installed in the field in modules. Some of these Small Modular Reactors (SMRs) will also be LWRs, while some will use other coolants such as liquid metals, molten salts or gases, and different types of fuels.

The technologies and economics of SMRs have been the focus of many studies, but there has been only minimal information published on the amount of nuclear waste different types of SMRs are expected to generate and no reports focused on near-term-deployable designs. It is important to understand how these new reactors may impact the rate of nuclear waste production and associated nuclear waste management practices going forward.

In this study, the nuclear waste attributes of SMRs scheduled for deployment this decade were assessed using established nuclear waste metrics and the results compared to those of a reference large Pressurized Water Reactor (PWR). The metrics were selected to inform on the following questions:

1. How will the rate of waste production per unit of electricity generation change as different types of SMRs are deployed?
2. Will the radiological characteristics of the waste change enough to warrant more detailed analysis with respect to long-term waste disposal planning?
3. What attributes of the waste will need to be considered as part of near-term waste handling, shipping and storage?

Three SMRs are assessed in this study, VOYGR™, Natrium™ [f], and Xe-100. They were selected from the dozens of SMR concepts for several reasons:

- First, they represent a range of reactor and fuel technologies, allowing comparison of the waste performance of these different technologies. VOYGR™ is a PWR design using the same type of ceramic fuel as found in larger LWRs, Natrium is liquid metal cooled and uses a metal alloy fuel, and Xe-100 is a helium cooled reactor using pebbles containing TRi-structural ISOtropic (TRISO) particle fuel. Collectively, these three technologies cover a wide range of proposed SMRs.

[f] An accepted definition of SMRs is for reactors up to 300 MWe, while Natrium is rated at 345 MWe. However, it is included here because it is the most mature design in its technology class and is closer in size to SMRs than to current reactors.

- Second, they represent comparatively mature designs that have been selected for DOE support for near-term deployment due to perceived strengths and have some expectation of commercial viability. Each design has been developed to improve performance, providing for a meaningful comparison versus existing reactors.
- Third, they are all active designs deployable in the near-term. Sites for first units of each design have been announced, licensing activities are underway, and all three are scheduled to be operational by the end of the decade. An understanding of the waste attributes of these designs can inform on any significant differences that may impact future nuclear waste management.

Conducting a credible assessment of the nuclear wastes generated from nuclear power plants requires wide-ranging information. The information includes the fuel cycle type (once-through or recycling), reactor power level, thermal efficiency, design reactor lifetime and capacity factor, fuel type, the chemical form of the coolant, fuel burnup level, post-irradiation cooling time of discharged fuels, etc. Dimensions and materials of all reactor components and buildings, neutron spectrum, flux distribution, and the quantity and lifetime of structural materials near the active core are also needed to evaluate the activation levels and identify the nuclear waste classifications. Because reactor concepts are evolving as they are optimized, the nuclear waste attributes should be assessed using the latest design information, emphasizing those designs moving forward toward demonstration or deployment in the near term.

It is noted that the three SMRs selected for this study adopt specific design features for improving reactor performance and achieving desired objectives. For instance, VOYGR™ is designed as an integral PWR for design simplification, operability enhancement, and risk reduction (NuScale 2020). VOYGR™ lacks several reactor components (primary coolant pump, concrete containment building, etc.) required in a typical PWR. Those reactor-specific design features should be accounted for in the evaluation of nuclear waste metrics because the nuclear waste metrics are affected by them. Natrium and Xe-100 use non-light-water coolants (sodium and helium) and specific fuel forms (metallic-alloy and TRISO/pebble fuels) to increase thermal efficiency and achieve higher burnup and inherent safety features. Both advanced non-light water reactors (non-LWRs) have thermal efficiencies of 40% or higher and a factor of 3–4 higher discharge burnup than the reference PWR.

The information on reactor design and performance parameters of three reactors and the reference PWR were obtained from open literature. However, if some data were missing but needed for evaluating waste metrics, the data were calculated in this work or obtained from a reactor with similar power rating and design features. A draft of the completed report was provided to the vendors of the three SMRs to ensure the open literature information was not misinterpreted.

The remainder of this report is organized as follows. Chapter 2 provides background on the source, selection, and method of calculation of the waste metrics. Chapter 3 describes the reference large PWR and the three SMRs assessed in the report and provides the design information and data used to calculate the metrics for each. Chapter 4 presents the front-end and back-end metric values calculated for each reactor, along with a comparison of performance of the three SMRs versus the reference PWR, including the authors' assessment of why differences may or may not be significant. Chapter 5 provides metric values and associated assessment of waste from decommissioning with a focus on Greater the Class C Low Level Waste (GTCC LLW). Chapter 6 summarizes the key findings. The appendix provides details on how wastes projected to be generated during decommissioning of retired reactors were calculated in this study.

2. Waste Metrics

To examine the waste attributes of specific reactor technologies, a set of relevant metrics must be selected for analysis and comparison. There are a wide range of potential waste metrics, and we will not use them all here. Some common metrics are fundamental characteristics of the waste materials, and some are useful 'derived metrics' important to various aspects of waste management. The USDOE conducted an extensive "Nuclear Fuel Cycle Evaluation and Screening" (E&S) (DOE 2014) that developed technology neutral metrics for a broad range of potential nuclear fuel cycles represented by 40 different 'evaluation groups'. These metrics were developed with extensive input from industry, academia, government, the public and past practices. Metric development included a multi-day public workshop, and all aspects of the E&S report were subjected to an independent review. Five of the 25 metrics used in the E&S were related to waste management. For that broad technology neutral study, the metrics evolved to the most fundamental waste characteristics and data – while deferring more specific 'derived metrics' for later technology specific analyses, of which this current work is an example. The fundamental waste characteristic metrics used in the E&S were:

- *Mass of SNF+HLW disposed per energy generated*
- *Activity of SNF+HLW (@100 years) per energy generated*
- *Activity of SNF+HLW (@100,000 years) per energy generated*
- *Mass of DU+RU+RTh disposed per energy generated*
- *Volume of LLW per energy generated*

The waste stream activity data in the second and third metrics was given at two times because 100 years is relevant for operations and 100,000 years is relevant to long-term isolation. The activity was calculated for each significant radioisotope, such that any commonly used "derived metrics" such as heat generation, dose rate, radiotoxicity, criticality, etc., could be calculated for any time period as desired.

The four reactors in this study fall into just two of the forty E&S evaluation groups: "*once-through fuel cycles using enriched uranium fuel in thermal critical reactors*" (reference PWR and VOYGR™) and "*high burnup in thermal or fast critical reactors*" (Natrium and Xe-100). As these are all once-through uranium fuel cycles, metrics related to recycle or thorium (HLW, RU, RTh) are not relevant. (Note that any of these reactors, including the reference LWR, could be used with fuel recycle, resulting in reductions in most waste attributes as indicated in the E&S study (DOE 2014)).

For this study, the relevant metrics from the E&S are used, with the addition of derived values for decay heat and radiotoxicity, and more detailed discrimination of activity and LLW categories. LLW during reactor operation is not included due to a) a lack of comparable/consistent data for any of the reactors and b) historic PWR data that shows very large (order of magnitude) variations in LLW generation, even for similar reactors, due to differing operational practice and priorities, regulations, technology advancement, and the commercial cost of LLW disposal.

The final metrics used for this study and their rationale are given below. All values are normalized to the same unit of electricity generated:

<u>Front-end metric</u>

- DU mass (t/GWe-yr) – The depleted byproduct of uranium enrichment. This metric is relevant to DU management/disposal, and directly related to DU disposal cost.

<u>Back-end metrics</u>

- SNF mass (t/GWe-yr) – The initial heavy metal mass of fuel discharged from the reactor and destined for geologic disposal. This metric is relevant to discharged fuel handling, storage, transportation, and final disposal, but not directly relevant to repository cost as U.S. practice has

been to charge disposal cost by unit electricity generated. (Note: There has been a statutory mass limit on the first U.S. geologic repository, but since that limit is fully subscribed by legacy fuel that limit is not applicable to future reactors.)

- SNF activity at five points between 10 to 100,000 years after discharge (Ci/GWe-yr) – A measure of the radioactivity of the fuel at various time after discharge, calculated by reactor and fuel specific irradiation conditions and isotopic content. This metric is relevant to post discharge handling, packaging, storage and transportation operations in the 10 – 100 year time frame, and repository design and performance in the 1000 – 100,000 year time frame, but is only indirectly related to repository cost.
- SNF decay heat at 10 and 100 years after discharge (kW/GWe-yr) – A derived metric from the SNF activity details. This metric is relevant to SNF handling, packaging, storage, transportation and repository design and initial emplacement, with relevance to operational costs.
- SNF ingestion radiotoxicity at 10,000 and 100,000 years after discharge (Sv/GWe-yr) – A derived metric from the SNF activity details. This metric is relevant to the long-term isolation performance of the final repository (differing times may be relevant to different repository designs and the migration of specific isotopes).
- SNF volume (m^3/GWe-yr) – The volume of the discharged fuel including structural components. This metric is relevant to SNF handling, packaging, storage, transportation and repository design, but not currently a cost driver for disposal.

<u>Decommissioning metrics</u>

- Volume of Class A, B, and C LLW (m^3/GWe-yr) – Estimated volume of waste from decommissioning a reactor that is suitable for shallow land burial. This metric is relevant to the cost of LLW disposal, which is currently a commercially available service in the U.S.
- Volume of GTCC LLW (m^3/GWe-yr) – Estimated volume of 'Greater Than Class C' waste from decommissioning a reactor that is not suitable for LLW shallow land burial. This metric is relevant to the cost of GTCC disposal, which is not currently a commercially available service in the U.S.

In this study, the front-end and back-end metrics are addressed quantitatively, with results presented in Chapter 4. Details of the reactor design and operation, as well as the decommissioning practice are required for a complete assessment of LLW volumes. Due to limitations in publicly available data and the extensive design and operational details required for calculation, the decommissioning metrics assessment is limited to only the two PWRs for the volume of Class A, B, and C LLW metric and to all the reactors, but with only bounding values for the advanced reactors, for the volume of GTCC LLW metric. The decommissioning assessment is augmented with information on primary drivers for waste quantities. The decommissioning metrics assessment is presented in Chapter 5.

3. Reactor Design Parameters

The reactors considered in this study are: VOYGR™, Natrium, and Xe-100 as compared to a reference large PWR. All four reactors employ a once-through enriched uranium fuel cycle and the per-unit power ratings of the three SMRs are much lower than the reference PWR.

In July 2022, the U.S. Nuclear Regulatory Commission (NRC) directed the issuance of a final rule that certifies NuScale's earlier 50 MWe SMR design for use in the United States (NuclearNewswire 2022) and the NRC website indicates there is an updated design application for the current 77 MWe VOYGR™ design deployable in groups of up to 12 reactors (924 MWe total plant size) by the 4th quarter of 2022. NuScale is working with the Carbon Free Power Project, a wholly owned Utah Associated Municipal Power Systems subsidiary, to deploy the first VOYGR™ power plant in the U.S. by 2029 (Reyes 2022, NuScale 2022c). The VOYGR™ design (NuScale 2022) is the reactor used in our analysis, but the building design used for decommissioning LLW is from the earlier certification (any building changes are not known).

In October 2020, DOE announced awards under the Advanced Reactor Demonstration Program (ARDP), which include two demonstration projects to build TerraPower's Natrium reactor and X-energy's Xe-100 reactor, which are to be operational by 2028 (DOE-NE 2020). NuScale had been selected for SMR development support in an earlier program for promising designs and has since evolved their design to the current VOYGR™ design.

TerraPower's Natrium reactor is rated at 345 MWe, which by accepted definitions makes it slightly too large to be an SMR. However, it is included here as the most mature design in a major technical class of SMRs. TerraPower has proposed three Natrium fuel concepts using the same reactor depending on the progress of fuel development (Neider 2021a). The demonstration project Natrium reactor concept will use a conventional sodium-bond U-Zr metallic fuel, followed by the Natrium commercial plant concept using sodium-free U-Zr metallic fuel. The design burnup of the Natrium commercial plant is 150GWd/t with 16.5% enriched uranium. Finally, TerraPower has the vision later this century to increase the burnup further to achieve a breed-and-burn mode operation with natural and depleted uranium reloads. In this work, the waste metrics were calculated using the Natrium commercial plant concept, as it is most representative of the average expected near-term performance through 2050. In addition to the ARDP demonstration unit, TerraPower and PacifiCorp recently announced a joint study to evaluate the feasibility of deploying up to five additional Natrium plants by 2035 (TerraPower 2022).

X-energy's Xe-100 is an 80 MWe pebble-bed gas cooled reactor, which can be scaled for deployment in a 4-pack 320 MWe power plant. The reactor bed includes 220,000 graphite pebbles containing TRISO particle fuel and continuous on-line refueling. The core is top-loaded and irradiated pebbles removed from the bottom, resulting in a slow downward flow of pebbles through the core. Each pebble discharged is assayed to determine if it is spent. Discharged pebbles with sufficient fissile content are reinserted at the top of the core, with the average pebble estimated to pass through the core ~6 times before being spent. A spent pebble is removed from the system for management as SNF and replaced in the system with a fresh pebble.

An 1175-MWe PWR used for the evaluation of decommissioning volume and costs by NRC (Smith et al., 1978; Konzek et al., 1995) was selected as the reference large PWR for comparison purposes. It is representative of the current LWR fleet, which was mostly constructed between 1970-1990. The AP1000 under construction in Vogtle, GA (NRC 2020b) was also considered. However, it was passed over due to a lack of information that is needed for evaluation of the waste characteristics. The reference PWR data were obtained from the nuclear waste evaluation studies performed by Smith et al. (1978), Konzek et al. (1995), and Mancini et al. (1994).

The primary design parameters of the four reactors are provided in Table 3-1. For consistent comparison, a reactor lifetime of 60 years and a capacity factor of 90% are assumed for all four reactors. The 90%

capacity factor may be conservative, as current reactors are averaging ~92% and most new reactors are designed to achieve similar or higher values. For reactors with fixed refueling periods, a conservative capacity factor assumption may result in lower calculated burnup due to fewer effective full power days.

Table 3-1 Primary reactor design parameters

	Ref. PWR	VOYGR™	Natrium	Xe-100
Power, MW - thermal - electric	 3500 1175	 250 77	 840 345	 200 80
Thermal efficiency	34%	31%	41%	40%
Capacity factor	90%	90%	90%	90%
Reactor lifetime, years	60	60	60	60
Fuel form	UO_2	UO_2	U-Zr w/o sodium-bond	TRISO(UOC) /Pebble
Burnup, GWd/t	50.0	49.5	146 [b)]	169
Uranium enrichment, %	4.5	4.95 [a)]	16.5	15.5
Number of assemblies/pebbles	193	37	186 [b)]	223,800
Charge U, kg/assembly/pebble	426	255	114.3 [b)]	7.0E-03
Assembly or pebble volume, m^3	0.188	0.110	0.104	1.13E-04
Assembly or pebble volume-to-mass ratio, m^3/t-initial heavy metal (HM)	0.441	0.433	0.912	21.8

a) Public information indicates "<5%", so conservatively used 4.95%
b) Due to the lack of information, data were obtained from a PRISM/Mod-B design that was revised to have the discharge burnup close to the Natrium design burnup of ~150 GWd/t.

The NuScale VOYGR™ building design parameters were obtained from the NuScale Standard Plant Design Certification Application for the previous NuScale 50 MWe design (NuScale 2020) by assuming that the design parameters needed for decommissioning waste evaluations, such as dimensions and masses of nuclear power module and reactor building, retain the NuScale concept used for the plant certification application. Updated physics calculations were performed as part of this study using the target burnup of the VOYGR™ design.

Design parameters of Natrium and Xe-100 were obtained from the open literature (Neider 2021a; Neider 2021b; Mulder and Boyes 2020; Mulder 2021a; Mulder 2021b; Mulder 2021c). Several design parameters needed for waste evaluation are protected as proprietary information. The missing data were obtained from a similar reactor concept or calculated in the present study. For instance, the core configuration, the number of driver fuel assemblies, and charge uranium mass per assembly of the Natrium reactor are determined using the PRISM/Mod-B reactor (Triplett et al. 2010) that was revised to approximate the Natrium's design burnup of 150 GWd/t with the heavy metal (HM) mass of 21.5 t in the core (Neider 2021a). It is noted that the Natrium design combines features from the previous GEH PRISM and TerraPower Traveling Wave designs (NRC 2022a).

Figure 3-1 shows the core configuration used for waste assessment of the Natrium reactor. The active core consists of 186 driver assemblies with U-Zr metallic fuels surrounded by radial reflectors and shielding assemblies. The assembly dimensions and cycle length were determined to reproduce the HM loading in the core and discharge burnup introduced by Neider (2021a). Thus, waste characteristics assessed in this work sufficiently represent the Natrium reactor.

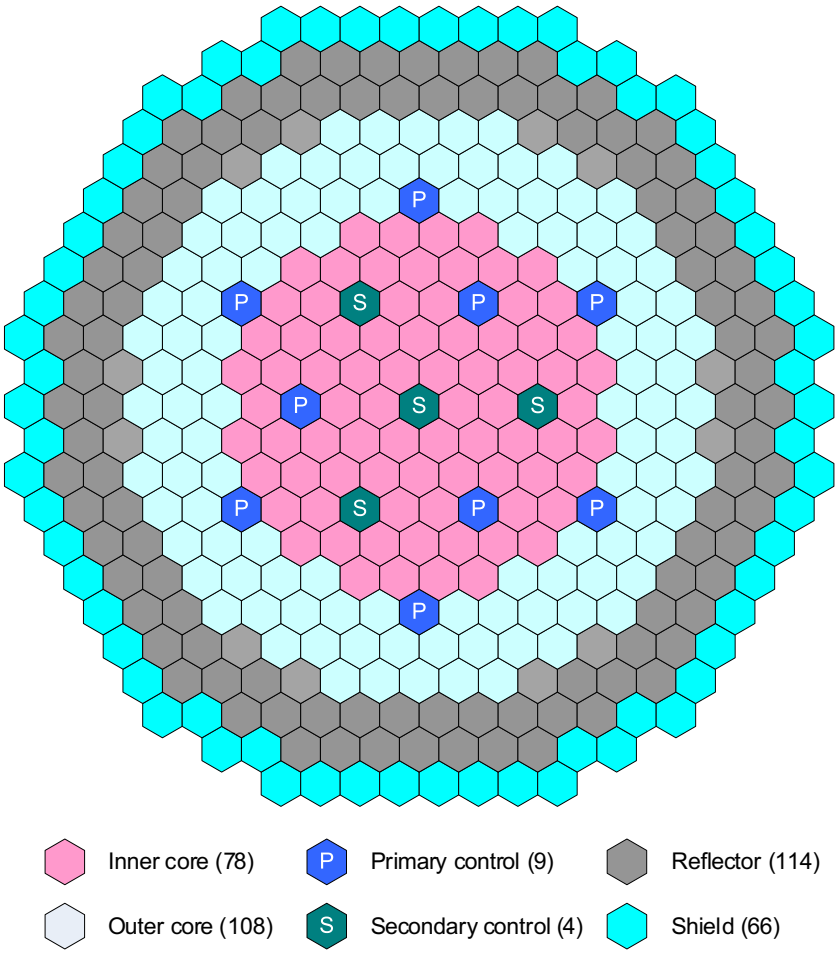

Figure 3-1 840 MWt core configuration used for Natrium waste assessment

4. Nuclear Waste Metric Results

Nuclear energy systems generate several types of radioactive waste which vary significantly in risk to humans and the environment, and this is reflected in how they are managed. In a once-through fuel cycle, spent nuclear fuel (SNF) presents the greatest risk due to being highly radioactive for very long time periods. For this reason, SNF disposal is generally expected to require permanent geologic isolation in deep repositories. Other wastes such as depleted uranium and low-level waste classes A, B, and C can be safely disposed in properly designed near-surface disposal facilities. GTCC LLW is currently assumed to be destined for geological disposal, though there is potential for alternative disposal options (NRC 2022b).

Nuclear waste metrics of the three SMR systems were calculated by grouping nuclear wastes into front-end waste associated with making fresh fuel and back-end waste associated with SNF. A preliminary assessment of decommissioning waste is also provided. To facilitate comparison across reactors of different sizes, all metrics are normalized per unit of electricity produced (waste per GWe-yr). The waste values for the SMRs are also compared with those of the large reference PWR, where values in parenthesis in the tables indicates the ratio of a waste metric to that of the reference PWR

4.1 Front-End Waste

Compared to the reference PWR, VOYGR™ and Natrium generate 1.2 times the DU mass, while Xe-100 generates comparable DU mass.

The single metric used to represent the front-end-fuel cycle waste is the DU mass generated from uranium enrichment for making reactor fuels. While there are some limited non-nuclear uses for DU, there is no practical use of for the quantities of DU produced in a once-through fuel cycle at this time[g], and therefore it is considered a waste.

DU can be disposed in specified near-surface disposal if it is converted to chemically stable uranium oxide compounds, such as U_3O_8 or UO_2, which are similar to the chemical form of natural uranium (NRC 2015). The DU masses are calculated by assuming a tail uranium enrichment of 0.25%[h], and the results are compared in Table 4-1. The values in parentheses in the table show the ratio of the SMR value compared to the reference PWR value.

Table 4-1 Comparison of depleted uranium masses

	Ref. PWR	VOYGR™ a)	Natrium a)	Xe-100 a)
Power (thermal/electric), MW	3500/1175	250/ 77	840/345	200/80
Thermal efficiency	34%	31%	41%	40%
Uranium enrichment, %	4.5	4.95	16.5	15.5
Burnup, GWd/t	50.0	49.5	146	169
DU mass, t/GWe-year	179	220 (1.23)	209 (1.17)	174 (0.97)

a) Values in parentheses indicate the ratio of a waste metric to that of the reference PWR.

[g] In a "breed and burn" once-through fuel cycle, DU can be used in fuel reloads. However, this application requires attainment of very high fuel burnup which is beyond the approved neutron flux limits of all current fuel cladding materials.

[h] A lower tail enrichment would reduce DU for all reactors proportionally.

The DU mass is proportional to the uranium enrichment, and inversely proportional to the burnup and thermal efficiency. Compared to the reference PWR, VOYGR™ fuel requires 23% more DU mass due to a combination of higher uranium enrichment, lower burnup, and lower thermal efficiency.

The uranium enrichment of Natrium fuel is a factor of 3.7 higher than that of the reference PWR, but the normalized DU mass increases by only 17% because the fuel burnup also increases by a factor of three and the thermal efficiency is higher than the reference PWR. Xe-100 fuel results in the lowest normalized DU mass due to a further increase in burnup with slightly less enrichment than in Natrium fuel.

> ***Significance of these results****: DU management does not entail significant safety issues but must be done correctly to safeguard the environment. DU disposal cost is directly proportional to the metric. However, DU disposal cost is a small (~0.1%) part of the cost of the nuclear energy system.*

4.2 Back-End Waste

4.2.1 SNF Mass and SNF Volume

The three near-term deployable SMRs and the reference PWR all use once-through fuel cycle. SNF is typically stored in interim storage and is ultimately sent intact to a geologic repository without further processing. The SNF mass (mass of initial heavy metal without assembly materials) and SNF volume (enclosed volume of fuel including assembly materials) are calculated by

$$M_{DF} = \frac{365}{B \times \eta}$$

$$V_{SNF} = f_{mass}^{volume} \times M_{DF}$$

where

M_{DF} = SNF fuel mass (t/GWe-year),

V_{SNF} = SNF assembly/pebble volume[i] (m^3/GWe-year),

B = average discharge burnup (GWd/t-initial HM),

η = thermal efficiency (%), and

f_{mass}^{volume} = ratio of assembly or pebble volume-to-initial HM mass (m^3/t-initial HM.

- *Compared to the reference PWR, VOYGR™ generates 1.1 times the SNF mass and volume due to relatively lower burnup and thermal efficiency. VOYGR™ SNF has slightly higher (~5% more) decay heat and radiotoxicity.*
- *Compared to the reference PWR, the higher burnup and higher thermal efficiency of both Natrium and Xe-100 reactors reduce the SNF mass by 3/4^{ths} and 100-year decay heat by 1/2 to 2/3^{rds}.*
- *However, Natrium and Xe-100 have different waste attributes for radiotoxicity and SNF volume. For Xe-100, the radiotoxicity is reduced by 2/3^{rds}, but the radiotoxicity of Natrium's SNF increases by a factor of 1.5 due to a higher plutonium content. The SNF volume is reduced by 2/5^{ths} for Natrium but increased by a factor of 12 for Xe-100 because pebble fuel contains a large volume of graphite and non-fuel matrix materials.*

The normalized SNF mass and SNF volume are compared in Table 4-2. Both metrics are inversely proportional to the average burnup and thermal efficiency.

Compared to the reference PWR, VOYGR™ generates 20% more SNF mass due to lower discharge burnup and thermal efficiency. In contrast, the non-LWR advanced reactors (Natrium and Xe-100) generate less SNF mass by a factor of 4 due to higher burnup and thermal efficiency. Less mass equates to less waste to be disposed per unit of electricity produced.

[i] Pebble volume is based on optimally stacked pebbles defined as individual pebble volume divided by the sphere packing density

Table 4-2 Comparison of SNF mass and SNF volume

	Ref. PWR	VOYGR™ a)	Natrium a)	Xe-100 a)
Power, MWe	1175	77	345	80
SNF mass, t/GWe-year	21.7	23.9 (1.23)	6.01 (0.28)	5.41 (0.25)
SNF volume, m^3/GWe-year	9.6	10.4 (1.08)	5.56 (0.58)	118 (12.3)

a) Values in parentheses indicate the ratio of a waste metric to that of the reference PWR.

The SNF volume is generally proportional to the SNF mass, but the ratio varies by fuel type. Even though the SNF mass may be small, the SNF volume may be larger if the assembly or fuel pebble requires extra space for structural materials or other purposes (such as the gas plenum in the Natrium fuel and graphite matrix in XE-100). Compared to the reference PWR, VOYGR™ generates 8% more SNF volume. It is noted that the Natrium assembly is taller than the active fuel length to accommodate the gas plenum in each fuel, and its assembly volume-to-HM mass ratio is a factor of two larger than in the reference PWR. As a result, Natrium's SNF volume is only 42% smaller than that of the PWR, even though its SNF mass is reduced by a factor of 3.6. Xe-100 generates less SNF mass by a factor of 4, but the SNF volume is a factor of 12.3 larger than that of the PWR due to the high graphite volume in the pebble. Volume is important for waste handling and transport, with current SNF transport casks optimized for PWR fuels. Given the differences in fuel size, shape and heat generation, new optimized cask designs are likely for the SNF of each SMR. SNF volume is typically not the constraining parameter for repository design. However, if a large number of Xe-100 reactors are built, a portion of the geologic repository could be designed specifically to optimize disposal of Xe-100 SNF with its higher volume but lower decay heat and radiotoxicity.

Significance of these results*: SNF mass and/or volume are often used as a generic measures of disposal requirements, but they are only indirectly correlated with health, safety, environmental issues, costs, or long-term isolation performance. Much larger or smaller volumes are likely to drive different optimized solutions for handling, packaging, storage, transportation, and repository design. While mass and volume are important to container and repository design optimization, actual constraints are more directly influenced by heat generation.*

4.2.2 SNF Activity

SNF Activity was calculated over several timeframes to provide general trend information as well as the isotopic data needed for calculation of decay heat and radiotoxicity. Figure 4-1 and Table 4-3 present the results of these calculations. The normalized activity of the reference PWR SNF and the smaller VOYGR™ PWR SNF are nearly identical, as they use similar technologies and are irradiated to similar burnup. In the figure, the VOYGR™ line hides most of the reference PWR line. The table reveals the VOYGR™ SNF has slightly higher activity due to slightly lower thermal efficiency resulting in less electricity generation for the same amount of burnup.

The other two SMRs initially have lower normalized activity due to higher thermal efficiency. However, the values diverge between 1,000 and 10,000 years as the activity of most fission products fade and the actinide content begins to dominate. Most nuclear fuels are discharged with the fissile content is sufficiently depleted that the k-effective value of the batch or pebble drops below 1.0 and the average k-effective of the core drops near 1.0. However, in a fast reactor like Natrium, the fissile content is not depleted as quickly due to breeding of fertile ^{238}U into fissile ^{239}Pu. One result is higher fissile content, including more ^{239}Pu in the SNF. The higher activity at 10,000 years is primarily due to this Pu content, while the difference is smaller at 100,000 years as the Pu decays away. In contrast, the high burnup and softer neutron spectrum of the Xe-100 fuel results in more of the Pu being consumed in situ than in the PWRs, again with the difference reduced at 100,000 years due to Pu decay in the SNF of the PWRs.

SNF activity is not a direct driver of storage, transportation or disposal requirements, but is a general indicator of differences that may appear in more specific requirements for shielding, decay heat management, and long-term radiotoxicity.

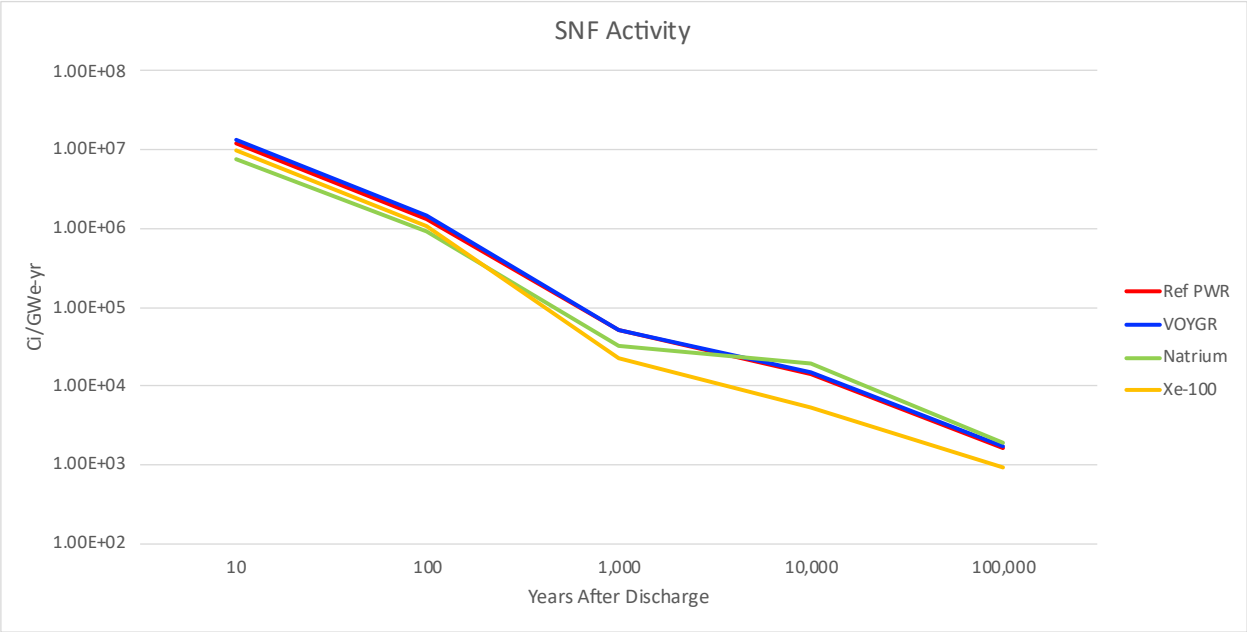

Figure 4-1 SNF Activity from 10 to 100,000 years after fuel discharge

Table 4-3 SNF Activity from 10 to 100,000 years after fuel discharge

Activity (Ci/GWe-yr)	Ref. PWR	VOYGR™ a)	Natrium a)	Xe-100 a)
Activity at 10 years	1.23E+07	1.31E+07 (1.07)	7.74E+06 (0.63)	9.67E+06 (0.79)
Activity at 100 years	1.32E+06	1.43E+06 (1.08)	9.34E+05 (0.71)	1.06E+06 (0.80)
Activity at 1,000 years	5.07E+04	5.28E+04 (1.04)	3.18E+04 (0.63)	2.29E+04 (0.45)
Activity at 10,000 years	1.42E+04	1.49E+04 (1.05)	1.99E+04 (1.40)	5.41E+03 (0.38)
Activity at 100,000 years	1.61E+03	1.73E+03 (1.08)	1.89E+03 (1.17)	9.30E+02 (0.58)

a) Values in parentheses indicate the ratio of a waste metric to that of the reference PWR.

Significance of these results: *Activity is a fundamental SNF metric that can provide an overall view of waste hazard. It is the starting point for specific derived waste attributes important to safety such as dose and radiotoxicity, and attributes important to handling and disposal such as decay heat. The activity variations (~ +/- 50%) between reactors are modest and are similar to variations between individual assemblies in a reactor. Any relevant differences would be more clearly seen in the decay heat and radiotoxicity metrics (below).*

4.2.3 SNF Decay Heat and Radiotoxicity

Decay heat and long-term radiotoxicity are important parameters derived from activity data that drive SNF handling and geologic disposal. The decay heat of SNF were calculated at 10 and 100 years after discharge and the radiotoxicity at 10,000 and 100,000 years after discharge. The decay heat and radioactivity were computed using ORIGEN 2.2 for one metric ton of SNF. The radioactivity was converted to radiotoxicity using the effective dose coefficients for ingested particulates of the

International Commission on Radiological Protection (ICRP) (Eckerman et al., 2012). Results are normalized to the unit of electricity generation (GWe-year) and compared in Table 4-4.

Table 4-4 Comparison of SNF decay heat and radiotoxicity

	Ref. PWR	VOYGR™ a)	Natrium a)	Xe-100 a)
Power, MWe	1175	77	345	80
Decay heat at 10 years, kW/GWe-year	40.6	42.2 (1.04)	24.5 (0.60)	32.2 (0.79)
Decay heat at 100 years, kW/GWe-year	9.76	10.3 (1.05)	4.65 (0.48)	6.36 (0.65)
Radiotoxicity at 10,000 years, x10^8 Sv/GWe-year	1.21	1.27 (1.06)	1.78 (1.47)	0.413 (0.34)
Radiotoxicity at 100,000 years, x10^8 Sv/GWe-year	0.0860	0.0912 (1.06)	0.127 (1.48)	0.0406 (0.47)

a) Values in parentheses indicate the ratio of a waste metric to that of the reference PWR.

Fission products are dominant contributors to decay heat at 10 and 100 years. Compared to the reference PWR, Natrium and Xe-100 SNF have more fission products per unit mass due to higher discharge burnup. However, Natrium and Xe-100 generate less SNF mass per unit of electricity generation. Comparing to the reference PWR, the decay heats of Natrium and Xe-100 SNF are 20–50% lower, and the decay heat of VOYGR™ SNF is comparable.

Pu isotopes are dominant contributors to radiotoxicity at 10,000 and 100,000 years. Compared to the reference PWR, Natrium SNF has 47% higher normalized radiotoxicity due to a higher Pu content. For thermal reactors (VOYGR™ and Xe-100), the normalized radiotoxicity depends on the SNF mass. Xe-100 SNF has 66% lower radiotoxicity at 10,000 years, while VOYGR™ SNF has 6% higher radiotoxicity. The Xe-100 difference dissipate somewhat by 100,000 years to 53% as the Pu decays.

Significance of these results: *The variation in decay heat is modest and not directly significant. However, the decay heat combined with the large volume variation can result in the volumetric heat generation varying over an order of magnitude – which will drive different design optimization for packaging, transport, storage and probably repository design, with unknown safety and cost impact. The variation in radiotoxicity values is moderate and important to long-term isolation safety, but it is not large when compared to the typical 'order of magnitude' repository performance analyses dominated by long-term material degradation, radionuclide mobilization and geologic transport uncertainties. However, for any given repository, there may be specific isotopes that dominate the long-term isolation performance, and the different fuel forms may provide significantly different repository performance and isolation. These factors combined might offer significant differences between the reactor/fuel concepts for any specific future geologic repository, but those differences cannot be quantified at this time.*

5. Decommissioning Wastes

When a reactor is retired and decommissioned, the fuel is removed, the coolant drained, equipment removed, and piping and structural materials broken down (sized) for disposal. These parts are then disposed[j] in different ways depending on their characteristics. The previous chapter focused on disposition of the SNF, which is the primary waste of concern and contains most of the activity. This chapter takes an initial look at the other wastes.

Decommissioning waste includes classes A, B, C, and GTCC LLW as defined in 10 CFR 61.55. Classes A, B, and C LLW are suitable for near surface disposal. GTCC LLW is currently assumed to be destined for geological disposal, though there is potential for alternative disposal options (NRC 2022b).

Several decommissioning approaches have been applied to commercial reactors, resulting in different amounts of waste, mainly depending on the decay times from the reactor shutdown to the start of decommissioning (Konzek 1995). These range from immediately dismantlement to placing the reactor in a safe storage status and delaying dismantlement until much of the contamination has had time to decay.

The nuclear waste volumes recovered from decommissioned PWRs in previous studies on decommissioning wastes were thoroughly reviewed to identify reactor components that contribute to the decommissioning waste. Large variation in waste amounts were noted. The United States has had decommissioning experience with multiple PWRs, including Maine Yankee (825 MWe), Rancho Seco (918 MWe), Haddam Neck (582 MWe), San Onofre (456 MWe) and Trojan (1130 MWe). Although the size of these reactors only varied by about a factor of two, the recovered LLW volume from the decommissioning processes ranged over more than an order of magnitude from 8,200 to 109,000 m^3 (McGrath and Reid, 2014; Lee et al., 2017).

Nuclear waste volumes recovered from decommissioned U.S. PWRs are compared in Figure 5-1. The decommissioning nuclear waste volumes of Maine Yankee and Rancho Seco nuclear power plants are broken down into LLW classes A, B & C, and GTCC (McGrath and Reid 2014), while other data are the total decommissioning LLW volumes (Lee et al. 2017). The breakdown shows that most decommissioning nuclear waste volume is Class A with only ~1% Classes B or C waste volume, and the GTCC volume is about 0.1%.

It is noted that the decommissioning LLW volume shows a factor of ~13 difference between the Haddam Neck power plant and the Trojan power plant, even though the power rating of the Haddam Neck power plant is lower. The primary reason for the difference is the decontamination approach for reactor buildings. For instance, as part of decommissioning, reactor buildings at Rancho Seco were decontaminated, while there was little decontamination of reactor buildings at Maine Yankee (McGrath and Reid 2014). Nuclear waste volumes are reduced by a factor of 5 to 7 through decontamination in a large PWR (Smith et al. 1978, Lee et al. 2017).

[j] In theory, the final core of a reactor could be moved to become the initial core of another identical or compatible replacement reactor, and coolant from one reactor could be used in another reactor employing the same technology.

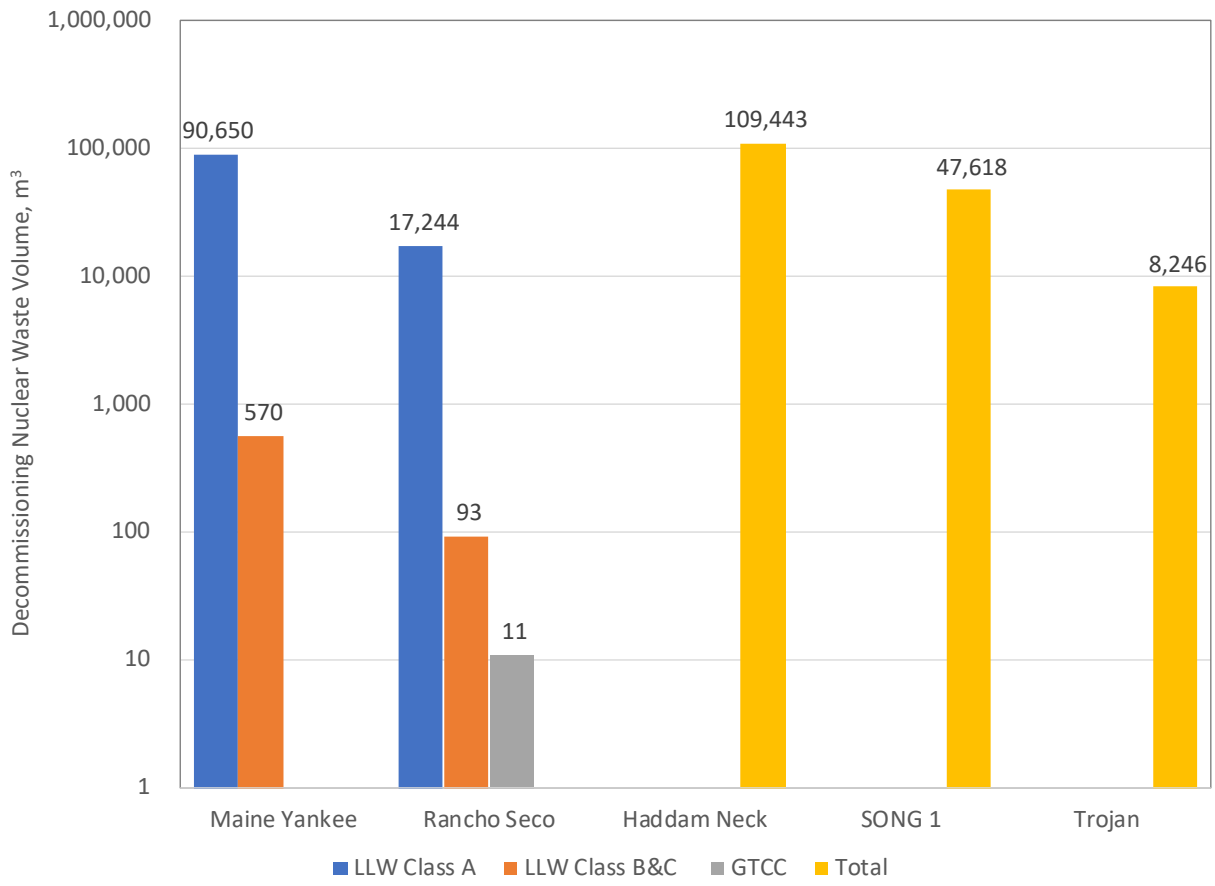

Figure 5-1 Comparison of actual decommissioning nuclear waste volumes

5.1 Decommissioning Volume of Class A, B, and C LLW

Classes A, B, and C LLW are generated through activation and contamination. Activation occurs from interactions with neutrons leaking from the active core. Except for the core supporting structures close to the active core, most activated reactor components are classified as Class A, B, or C, which include the reactor pressure vessel and internals, the concrete structure surrounding the reactor pressure vessel, etc. Contamination is caused by radioactive isotopes in the primary coolant, airborne radioactive isotopes, and radioactive effluents released during reactor operation (Smith et al. 1978). Activated coolant or corrosion products, fission products, and actinides released from fuels are the radioactive isotopes in the primary coolant. The radioactive isotopes travel and contaminate the surface of reactor coolant systems. The radioactive airborne isotopes and effluents contaminate the containment building and various buildings distributed on the reactor site.

- *The decommissioning volume of Class A, B, and C LLW is affected by reactor-specific design features as well as the decommissioning approach.*
- *The integral design features of VOYGR™ eliminates several reactor components used in a typical PWR but requires a reactor building with a large pool for co-locating 4–12 nuclear power modules. The net result is a normalized decommissioning volume of Class A, B, and C LLW that is slightly less (factor of 0.9) than the reference PWR.*
- *Due to the lack of detailed design information on reactor buildings, Class A, B, and C volumes for Natrium and Xe-100 were not calculated.*

The decommissioning volumes of the Class A, B, and C LLW of the reference PWR and VOYGR™ were evaluated without considering decontamination prior to decommissioning. There was insufficient publicly available detailed building dimensional information for Natrium and Xe-100 for similar calculations to be performed on their designs.

Data for calculation of the reference PWR waste was obtained from Smith et al. (1978) and Konzek et al. (1995). For VOYGR™, the metal volume in the reactor building and internals was calculated as part of this study by converting the total mass of a single Nuclear Power Module (NPM) of 700 t (NuScale 2022a) to a volume using an assumed average metal density of 7.8 g/cm^3. The concrete volume of the reactor building was calculated using the 12-module reactor building, which is 350 ft long, 150 ft wide, and 86 ft tall with 6-ft-thick concrete elements (NuScale 2020). Then, the reactor building volume was divided by 12 to determine the volume per single NPM. Due to a lack of information, the LLW volume of other buildings was prorated by assuming that the LLW volume from other buildings is proportional to the power rating in PWRs. Additional details of these calculations are provided in Appendix A, Section A-1.

The resulting LLW volumes are compared in Table 5-1. For comparison purposes, the decommissioning LLW volume of the reference PWR was divided into three parts (containment building/internals, fuel building, and other buildings), while the VOYGR™ LLW volume was split into two parts (reactor building/internals and other buildings). It is noted that in terms of its role, the VOYGR™ reactor building is equivalent to the containment building/internals and fuel building of the reference PWR. The normalized decommissioning Class A, B, and C LLW volume of the reference PWR is 645 m^3/GWe-year, while it is 573 m^3/GWe-year for VOYGR™.

Table 5-1 Comparison of decommissioning volumes of Class A, B, and C LLW

	Ref. PWR		VOYGR™	
	Metal	Concrete	Metal	Concrete
Power, MWe	1175		77	
Containment building and internals, m^3	344	30,013	90	1,782
Fuel building, m^3	19	2,770		
Other buildings, m^3	360	7,438	24	487
Sum, m^3	723	40,221	113	2,269
Total Net volume, m^3 Per electricity generation unit, m^3/GWe-yr	 40,944 645		 2,383 573 (0.9) [a)]	

a) Value in parentheses indicates the ratio of a waste metric to that of the reference PWR.

Significance of these results*: The Class A, B, and C waste volumes are primarily cost issues rather than safety or environmental issues. The calculated results for the two reactors are essentially the same when compared to the wide range of actual values historically observed from decommissioning of similar reactors. These results provide no differentiation between reactor types.*

5.2 Decommissioning Volume of GTCC LLW

GTCC LLW falls into three categories: activated metals resulting from operations, process wastes such as resin and filters in decontamination systems, and activated materials recovered through reactor decommissioning (Mancini et al. 1994). The activated materials recovered during a decommissioning process include the permanent structure near the active core, such as the core supporting structures and

biological shields. In the present work, the activated materials recovered from decommissioning are compared because they are the largest source of the GTCC volume.

The activation levels of the core supporting structures near the active core differ in different reactor types due to reactor-specific design features. VOYGR™ has a core supporting structure and activation level similar to the reference PWR because VOYGR™ adopts the PWR technology. In Natrium and Xe-100, the active core is surrounded by reflector assemblies or graphite blocks to protect the core internal structure from neutron irradiation damage. As a result, it is expected that the activation level of both Natrium and Xe-100 is lower for core supporting structures than that of the reference PWR but higher for the reflector assemblies or graphite blocks.

The calculation of GTCC LLW volumes is provided in Appendix A, Section A-2, with the results compared in Table 5-2 Comparison of GTCC LLW volumesand explanations provided here.

For VOYGR™, the GTCC volumes were calculated by assuming that reactor components of the reference PWR that are activated to GTCC are also activated to GTCC in VOYGR™. The net GTCC volume for each VOYGR™ module is a factor of two smaller than for the PWR due to the smaller active core height and diameter. However, when normalized to the unit of electricity generation, the GTCC volume is a factor of six larger than that of the PWR.

- *VOYGR™ core supporting structures, such as the core baffle, grid plates, barrel, etc., are activated up to the GTCC level. Compared to a large PWR, VOYGR™ generates less net GTCC volume, but the normalized GTCC volume per unit of electricity generation is larger by a factor of six.*
- *Natrium and Xe-100 are designed using reflector assemblies (Natrium) or graphite blocks (Xe-100) to protect core supporting structures. These reflector and graphite blocks could be activated up to the GTCC level depending on their residence time in the core.*
- *Natrium or Xe-100 do not generate GTCC LLW when the reflector assemblies and graphite blocks are discharged before they are activated to the GTCC level.*
- *However, compared to the reference PWR, Natrium generates a factor of 4 more and Xe-100 193 more GTCC volume when the reflector assemblies and graphite blocks reside in the core for reactor lifetimes.*

For Natrium the radial reflector assemblies are the reactor components expected to become GTCC. The GTCC volume varies depending on the residence time of the reflector assemblies in the core. Analyses provided in the appendix indicate the Natrium reflector assemblies will be activated to GTCC levels if they remain in the core for more than 30 years. The Natrium reactor will not generate appreciable GTCC LLW if the reflector assemblies are replaced before they are activated to the GTCC level. However, compared to the reference PWR, Natrium generates a factor of 4 more GTCC volume when the reflector assemblies reside in the core for the full 60-year reactor life. Replacement of the reflector assemblies reduces the GTCC volume to be disposed of in a geological repository. However, it increases the Class B or C LLW volume as a trade-off.

For Xe-100, the graphite blocks are the reactor components expected to become GTCC. Analyses indicated the primary activation involves nitrogen impurities within the graphite, which activate to generate C-14. The level of these impurities depends on the source of natural graphite used, varying between 10 and 100 ppm. The residence time for the graphite blocks to be activated to GTCC levels was calculated at both 10 and 100 ppm with the result that GTCC activation levels are reached in ~17 years for the low impurity level but in only 3 years at the high impurity level. Xe-100 does not generate appreciable GTCC LLW when the graphite blocks are replaced before they are activated to the GTCC level. However, compared to the reference PWR, Xe-100 generates a factor 193 more GTCC volume when graphite blocks reside in the core for reactor lifetimes. The replacement schedule of the graphite blocks depends on the reactor operation, waste management strategies, and nitrogen impurity levels.

Frequently replacing reflector assemblies or graphite blocks reduces the GTCC volume to be disposed but increases the Class B or C LLW volume as a trade-off. The frequency of replacement and resulting Class B or C LLW can be reduced by using low impurity graphite. Alternatively, very low impurity synthetic graphite could be used without significant activation if synthetic graphite becomes qualified for nuclear applications.

Table 5-2 Comparison of GTCC LLW volumes

	PWR	VOYGR™	Natrium	Xe-100
Baffle (shroud), m^3	[a] 1.6–2.0	2.1		
Barrel, m^3	3.5–5.8	0.6		
Grid plates, etc., m^3	0.9–2.6	0.3		
Reflector assemblies, m^3	-	-	[b] 0.0–10.3	-
Radial and axial graphite blocks, m^3	-		-	[c] 0.0–106.0
Total volume Net volume, m^3 Per unit of electricity generation, m^3/GWe-yr	 6.3–9.8 [d] 0.13	 3.0 0.72 [e] (5.7)	 0.0–10.3 0.0–0.55 [e] (0–4.4)	 0.0–106.0 0.0–24.5 [e] (0–193)

a) Range from calculated GTCC volumes in Smith et al. (1978), Konzek et al. (1995), and Mancini et al. (1994).
b) Variation is dependent on the residence time of reflector assemblies in the core.
c) Variation is dependent on the residence time of graphite blocks in the core and graphite purity.
d) Compared GTCC volumes to the net GTCC volume calculated by Konzek et al. (1995).
e) Values in parentheses indicate the ratio of a waste metric to that of the reference PWR.

Significance of these results: *In general, GTCC waste has been a very small fraction (1% or less) of the total LLW waste, and not a major factor, but there is currently no commercially available GTCC disposal capacity in the U.S. to allow quantification of GTCC disposal costs. The potential for substantial increase in GTCC waste for each SMR could become a significant issue if future GTCC disposal is difficult and/or expensive. However, it appears that there are reactor design choices, material choices and operational alternatives that would offer significant GTCC reductions if that becomes important, particularly for the non-LWR designs.*

5.3 Coolant wastes

Reactor primary coolants become contaminated when trace elements (corrosion products, etc.) in the coolant are activated and by the coolant itself becoming activated. Any cladding breaches will also contribute small amounts of fission products. While coolant is constantly filtered to remove contaminants, the coolant will have some level of reactivity when the reactor is retired. If the concentration of radioactive material in the coolant is in a stable chemical form (e.g., water) and below the Federal limits for the release of effluents, it may be released offsite. Otherwise, the liquid waste is solidified (by mixing with concrete or a similar solidifying or absorbing material) as a chemically stable form and disposed of as solidified LLW (Minns et al. 2000).

- *Natrium's coolant sodium can be solidified as a chemically stable form and disposed of as LLW as has been done for prior sodium cooled reactors. The estimated coolant sodium volume using the PRISM/Mod-B reactor is less than 1% of the total decommissioning LLW volume of the reference PWR.*
- *In this work, Xe-100's helium was not counted as the nuclear waste to be disposed of because its activation level is low.*

Coolant disposal during decommissioning of LWRs has presented no significant challenges and the amount of waste produced has been small (< 1% of total LLW). VOYGR™ coolant is expected to be addressed in a similar manner and was not analyzed further.

The coolant sodium of Natrium is expected to be solidified as a chemically stable form and disposed of as LLW. Based on past experience with sodium cooled test reactors, the Natrium coolant sodium may be contaminated by radioactive corrosion products and activated by neutrons. Natural sodium consists of 100% Na-23, and reactor-grade sodium contains impurities of 10–300 ppm of K, Ca, Cl, and Br. Na-23 is activated to Na-22 (half-life of 2.6 years) and Na-24 (half-life of 14.96 hours) through Na-23(n,2n)Na-22 and Na-23(n,γ)Na-24 reactions, and impurities are also activated. Figure 5-2 shows the activity levels at the end of life (EOL) and during the 20-yr period of post-irradiation cooling. Na-24 is a dominant contributor to the total activity of coolant sodium at EOL, but owing to its short half-life, the whole activity level of the coolant sodium decreases by 4-5 orders of magnitude within a month. In addition, the operation experience in EBR-II and FFTF shows that the activation level of primary coolant by corrosion products and fission products is low because a primary coolant purification system is used to control the levels of these contaminants (Brehm et al. 1987).

Due to the chemical properties of sodium, the discharged coolant sodium is required to undergo a process of conversion to a waste form acceptable for disposal. The experience of sodium disposal in the United States is briefly summarized here. The primary coolant sodium (290,000 liters) from the Fermi-I reactor and both primary (330,000 L) and secondary (50,000 L) coolant sodium from the EBR-II reactor have been stored at Idaho National Laboratory. In accordance with the requirements of the State of Idaho and the Resource Conservation and Recovery Act (RCRA), the sodium was required to be converted into a waste form acceptable for disposal. Also, DOE mandated transforming sodium into a stable condition for land disposal. To comply with the requirements and mandate, the Sodium Process Facility (SPF) was constructed to process the sodium into sodium hydroxide and eventually sodium carbonate, a non-RCRA-regulated substance (Michelbacher et al. 1997). The primary coolant sodium of Fermi-I and the secondary coolant sodium of EBR-II were processed in the SPF and disposed of in the late 1990s (Benedict et al. 2001). Considering the low activity after post-irradiation cooling for a few years and the processing experience of coolant sodium, Natrium coolant sodium is expected to be suitable for near surface disposal.

Due to a lack of design information on Natrium, the disposal waste volume of the solidified sodium was estimated using PRISM/Mod-B. The total sodium inventory of PRISM/Mod-B is ~890 t, consisting of 775 t of primary coolant inventory and an additional 15% sodium assumed for the secondary system (Triplett et al. 2010). Using the sodium carbonate density of 2.5 g/cm^3, the disposal waste volume of sodium coolant is ~360 m^3, which is less than 1% of the total decommissioning LLW volume of the reference PWR.

Xe-100's coolant is pressurized helium, which is also contaminated by radioactive isotopes or activated by neutrons. Radioactive isotopes in the primary coolant include corrosion products, fission products, and impurities such as H_2, CO, CO_2, CH_4, N_2, and O_2. The impurity level is limited to below 3 ppm to reduce oxidation of core graphite structures (Sakaba et al. 2004). The gaseous coolant was not considered radioactive waste to be disposed due to the low activity level achieved by removing radioactive isotopes during reactor operation.

In summary, none of the coolants are expected to generate more than 1% of the total decommissioning LLW volume. The sodium coolant of the Natrium reactor will need to be chemically stabilized before disposal, which can be achieved using proved methods successfully employed previously during decommissioning of two sodium-cooled experimental fast reactors.

Significance of these results: There appears to be adequate disposal pathways, and no significant safety or environmental disposal issues with any of the coolants used in these reactors.

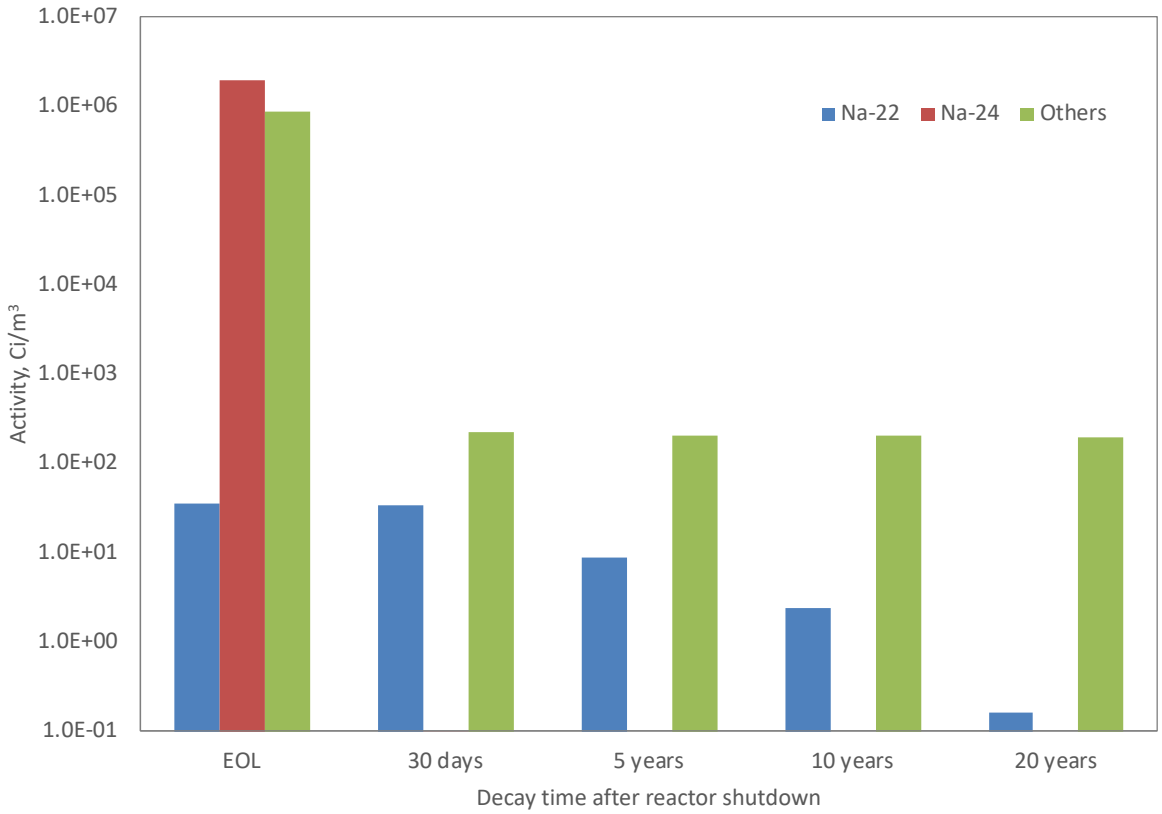

Figure 5-2 Coolant sodium activation and decay

6. Conclusions

The technologies and economics of SMRs have been the focus of many recent studies, but there has been only minimal information published on the attributes of nuclear waste that different types of SMRs are expected to generate and no reports focused on near-term-deployable designs. This study was developed to help fill this gap.

In this study, the nuclear waste attributes of SMRs scheduled for deployment this decade were assessed using established nuclear waste metrics and the results compared to those of a reference large Pressurized Water Reactor (PWR). The metrics were selected to inform on the following questions:

1. How will the rate of waste production per unit of electricity generation change as different types of SMRs are deployed?
2. Will the radiological characteristics of the waste change enough to warrant more detailed analysis with respect to long-term waste disposal planning?
3. What attributes of the waste will need to be considered as part of near-term waste handling, shipping, and storage?

Three SMRs are assessed in this study, VOYGR™, Natrium, and Xe-100. These were selected because they represent a range of reactor and fuel technologies, they have been designed to provide improved performance, and they are all active designs scheduled to be deployed by the end of the decade. Each of the three have received partial government support for demonstration because of perceived strengths. An understanding of the waste attributes of these designs can inform on any significant differences that may impact future nuclear waste management.

The results of the analysis are summarized in Table 6-1 Comparison of nuclear waste metric values calculated in this study. Analysis of these results support several findings:

- Many metrics showed only small changes versus the reference reactor. This was especially true for the PWR SMR (VOYGR™), where masses and volumes of wastes were slightly higher, due primarily to the smaller core size having moderately higher enrichment and slightly lower burnup.
- The only notable difference in metric values for VOYGR™ is a projected higher rate of decommissioning GTCC LLW generation, attributed to the reflector blocks added to the core to reduce neutron leakage. GTCC LLW volume is ~0.1% of overall LLW volume and the VOYGR™ GTCC volume is only ~5% of the VOYGR™ SNF volume, so this finding is not significant.
- Most of the back-end waste metrics for the two non-PWR reactors had notably lower values than the reference PWR, including large reductions in SNF mass, SNF activity for most timeframes, SNF decay heat and for Xe-100, long-term radiotoxicity. However, there were two areas where this pattern did not hold:
    - The long-term activity and long-term radiotoxicity of Natrium SNF were both somewhat higher than the reference, due to a higher normalized plutonium content. Whether this is significant depends on the design of the future geologic repository.
    - The normalized SNF volume of Xe-100 was significantly higher. SNF volume is usually only a constraint for transportation and not for disposal, and the Xe-100 SNF is in a robust waste form, but if large numbers of Xe-100-type reactors are placed in operation then it may be advisable to design the disposal approach for a portion of the repository to be optimized for this type of SNF.
- The decommissioning volume of GTCC LLW was bounded for the two non-PWRs, with some important findings:

Table 6-1 Comparison of nuclear waste metric values calculated in this study

	Ref. PWR	VOYGR™ a)	Natrium a)	Xe-100 a)
DU mass, t/GWe-year	179	220 (1.23)	209 (1.17)	174 (0.97)
SNF mass, t/GWe-year	21.7	23.9 (1.10)	6.10 (0.28)	5.41 (0.25)
SNF volume, m^3/GWe-year	9.58	10.4 (1.08)	5.56 (0.58)	118 (12.3)
SNF activity (Ci/GWe-year) compared to Ref PWR @ 10^1, 10^2, 10^3, 10^4, 10^5 years		(1.07, 1.08, 1.04, 1.05, 1.08)	(0.63, 0.71, 0.63, 1.40, 1.17)	(0.79, 0.80, 0.45, 0.38, 0.58)
SNF decay heat, kW/GWe-year				
@ 10 years	40.6	42.2 (1.04)	24.5 (0.60)	32.2 (0.79)
@ 100 years	9.76	10.3 (1.05)	4.65 (0.48)	6.36 (0.65)
SNF radiotoxicity, x10^8 Sv/GWe-year				
@ 10,000 years	1.21	1.27 (1.06)	1.78 (1.47)	0.413 (0.34)
@ 100,000 years	0.0860	0.0912 (1.06)	0.127 (1.48)	0.0406 (0.47)
Decommissioning LLW volume				
Classes A, B, and C, m^3/GWe-year	645.3	573 (0.9)	N/A b)	N/A b)
GTCC, m^3/GWe-year	0.13	0.72 (5.7)	0.0 – 0.55 (0.0 – 4.4)	0.0 – 24.5 (0.0-193.1)

a) Values in parentheses indicate the ratio of a waste metric to that of the reference PWR.
b) Not available because the open information is insufficient to calculate the LLW volume.

- Both reactors have the potential to generate very little GTCC, due to the use of core internal features than reduce activation of most materials near the core. These include neutron reflectors for Natrium and graphite blocks for Xe-100. However, to achieve this result, these features would need to be replaced periodically during the life of the reactor.
- Calculations indicated the Natrium reflector may only need to be replaced once, while the number of replacement cycles for the Xe-100 were higher and highly dependent on the amount of nitrogen in the graphite used.
- If these features were not replaced during reactor operation, they would be activated to GTCC levels, resulting in more GTCC generation than the reference PWR, especially for the Xe-100. However, if this becomes an issue there are alternative sources of graphite with little to no nitrogen content that could be nuclear qualified.

The waste attributes of the SMRs studied show both some similarities to the reference LWR and some potentially significant differences. Front-end waste attributes from SMR fuel production range from equivalent to 1.2 times the LWR reference. Back-end waste attributes for spent fuel disposition vary from large reductions to small to moderate increases in heavy metal mass (factors of 0.2 to 1.2), activity (factors of 0.3 to 1.5) and radiotoxicity (factors of 0.5 to 1.5). These differences have limited impact on long-term repository isolation. SMR designs can vary significantly (factors of 0.6 to 12.3) in volume (and thus heat generation density), however these differences are readily amenable to design optimization for handling, storage, transportation, and disposal technologies. Waste attributes from decommissioning can vary greatly depending on design and decommissioning technology choices.

Given the analysis results in this study and assuming appropriate waste management system design and operational optimization, there appear to be no major challenges to the management of SMR wastes compared to the reference LWR wastes. The results of this study are only applicable to a once-through fuel cycle. Any of these reactors, including the reference LWR, could be used with fuel recycle, resulting in reductions in most waste attributes as indicated in the E&S study (DOE 2014).

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Appendix A

Decommissioning Nuclear Waste Volumes

Appendix A

Decommissioning Nuclear Waste Volumes

A-1. Decommissioning Volume of Class A, B, and C LLW

Class A, B, and C LLW are generated through activation and contamination. Activation occurs from interactions with neutrons leaking from the active core. Except for the core supporting structures close to the active core, most activated reactor components are classified as Class A, B, and C LLW, which include the reactor pressure vessel and internals, the concrete structure surrounding the reactor pressure vessel, etc. Contamination is caused by radioactive isotopes in the primary coolant, airborne radioactive isotopes, and radioactive effluents released during reactor operation (Smith et al. 1978). Activated corrosion isotopes, fission products, and actinides released from fuels are the radioactive isotopes in the primary coolant. The radioactive isotopes travel and contaminate the surface of reactor coolant systems. The radioactive airborne isotopes and effluents contaminate the containment building and various buildings distributed on the reactor site.

Smith et al. (1978) and Konzek et al. (1995) estimated the decommissioning waste volumes of the reference PWR. It is noted that the reference PWR design and operation data were mainly obtained from the 1175-MWe Trojan power plant. The resulting decommissioning waste volume after decontamination of concrete building surfaces was 6,600–8,800 m^3, which is close to the actual decommissioning waste volume of the Trojan power plant (8,346 m^3, Lee et al. 2017).

Breakdowns of the estimated LLW volumes are provided in Table A-1. The contaminated waste volume is the dominant contributor (93–95%) to the total decommissioning LLW volume. In particular, the concretes contaminated by radioactive airborne isotopes and effluent are the primary source (86–88%) of the total decommissioning LLW volume. This compares well with the work of Smith et al. (1978) and Konzek et al. (1995) which estimated the nuclear waste volumes of a large PWR for decommissioning cost assessments. Their studies show that the contaminated concrete volume of various reactor buildings by radioactive isotopes (such as airborne isotopes and effluents) is the dominant contributor (more than 80%) to the total LLW volume recovered from a decommissioning process.

Table A-1 Breakdown of decommissioning waste from a large reference PWR

Component		Breakdown, %	
		Smith et al. (1978)	Konzek et al. (1995)
Activated by neutrons	Metal	1.0	0.8
	Concrete	5.6	4.2
Contaminated by primary coolant	Metal	3.8	2.8
	Concrete	-	-
Contaminated by airborne isotopes and effluent	Metal	3.8	4.6
	Concrete	85.8	87.6
Total volume, m^3		6,569	8,767

For a credible comparison of nuclear waste volumes, detailed design data on all reactor components, arrangement and dimensions of buildings, and activation levels of the reactor components and buildings are needed, and those data vary depending on the design features of reactor types. For instance, even though VOYGR™ adopts PWR technologies (i.e., fuel form, coolant material, coolant pressure, neutron spectrum, reactivity control, etc., are similar to those of a typical PWR), the reactor configuration is

different from the reference PWR. Figure A-1 and Figure A-2 depict the containment building and general plant arrangement of the reference PWR (Smith et al. 1978). The Nuclear Steam Supply Systems (NSSS) are in a concrete containment building, and some are surrounded by biological shields. Other buildings, such as the fuel building, auxiliary building, and control room, are arranged outside the containment building. Figure A-3 shows the VOYGR™ nuclear power module (NPM) and the arrangement of NPMs in a reactor building. VOYGR™ is designed as an integral PWR, such that the active core and steam generator are integrated inside a reactor vessel without a coolant pump. The reactor vessel is surrounded by a steel containment vessel, not a concrete containment building. Multiple NPMs (4–12) are submerged in a single pool in the reactor building, where fuels are temporarily stored. Concerning the differences in reactor configuration from the reference PWR, the VOYGR™ decommissioning LLW volume is decreased because there is no containment building but increased due to the large pool in the reactor building.

The decommissioning volumes of Class A, B, and C LLW of the reference PWR and VOYGR™ were evaluated without considering decontamination. The resulting LLW volumes are compared in Table A-2. For comparison purposes, the decommissioning LLW volume of the reference PWR was divided into three parts (containment building/internals, fuel building, and other buildings), while the VOYGR™ LLW volume was split into two parts (reactor building/internals and other buildings). It is noted that in terms of its role, the VOYGR™ reactor building is equivalent to the containment building/internals and fuel building of the reference PWR.

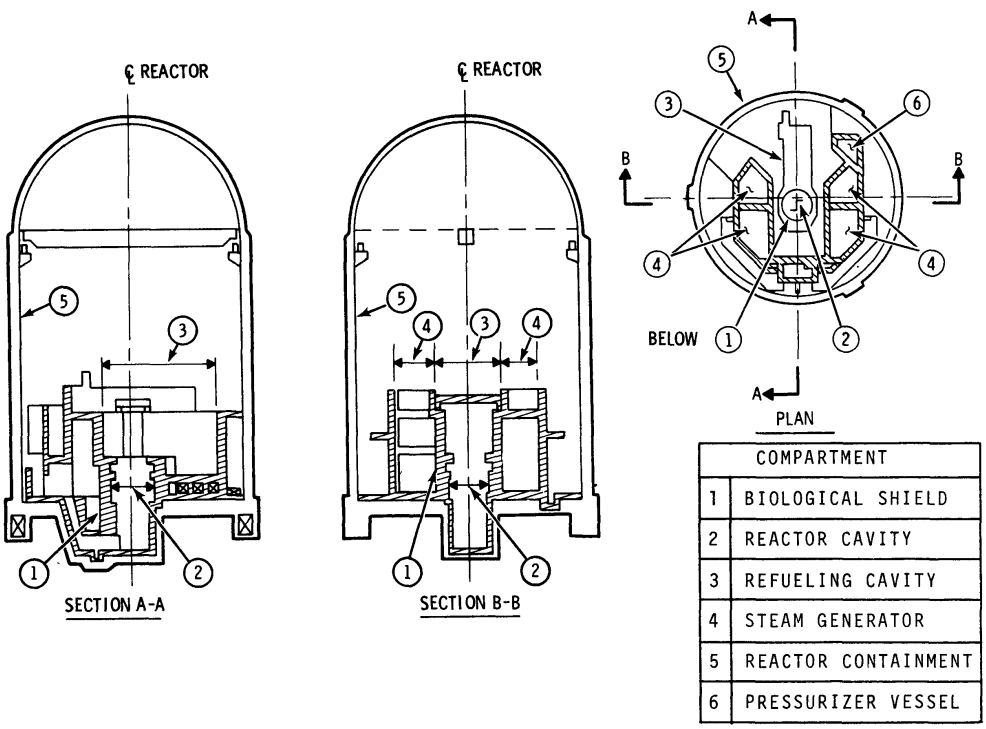

Figure A-1 Containment building of reference PWR

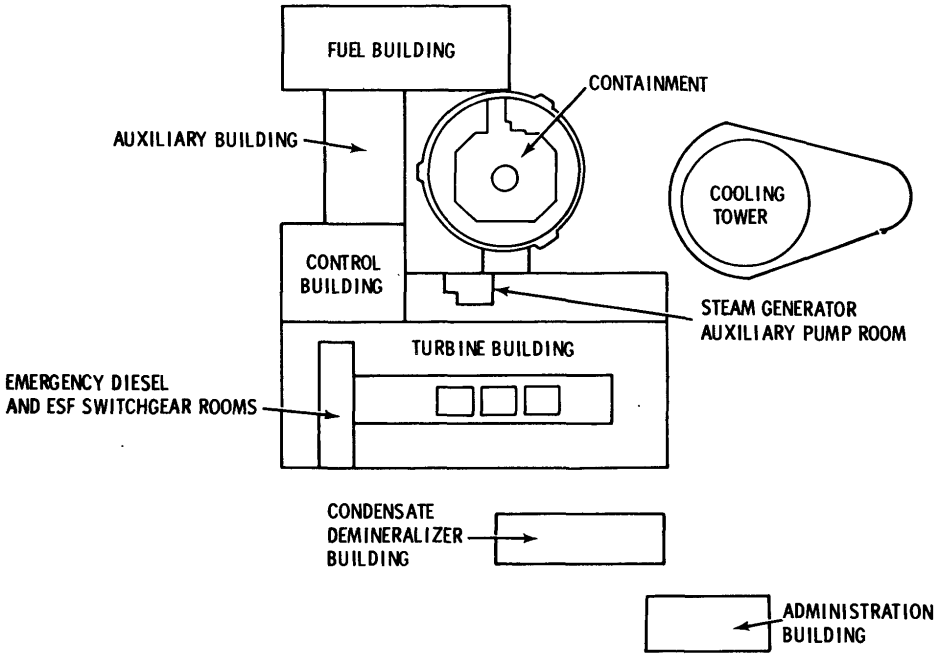

Figure A-2 General plant arrangement of reference PWR

Figure A-3 VOYGR™ nuclear power module (left) and arrangement in reactor building (right)

Table A-2 Comparison of decommissioning volumes of Class A, B, and C LLW

	Ref. PWR		VOYGR™	
	Metal	Concrete	Metal	Concrete
Power, MWe	1175		77	
Containment building and internals, m^3	344	30,013	90	1,782
Fuel building, m^3	19	2,770		
Other buildings, m^3	360	7,438	15	317
Sum, m^3	723	40,221	104	2,098
Total Net volume, m^3 Per electricity generation unit, m^3/GWe-yr	40,944 645		2,202 573 (0.9)	

Figure A-4 shows the reactor building of Natrium (NRC 2022). The reactor building has a similar function to the containment building of the reference PWR, but the arrangement of NSSS components is different because Natrium does not have a pressurizer and the steam generator is located outside of the reactor building, connected through the secondary coolant loop. To enhance the inherent safety feature, Natrium adopts a pool-type primary configuration. The active core, primary coolant pump, and intermediate heat exchanger are located in a large sodium pool surrounded by a reactor vessel and an additional guide vessel. Thus, Natrium has a large reactor vessel filled with the primary sodium coolant, but the primary coolant does not contaminate the reactor components located outside of the reactor vessel because the primary coolant does not flow outside of the reactor vessel.

Figure A-5 shows the plant arrangement of Xe-100 (Mulder 2021c). The reactor building contains the reactor module, which consists of the active core and steam generator. Various buildings (the turbine building, control room, etc.) are arranged near the reactor building.

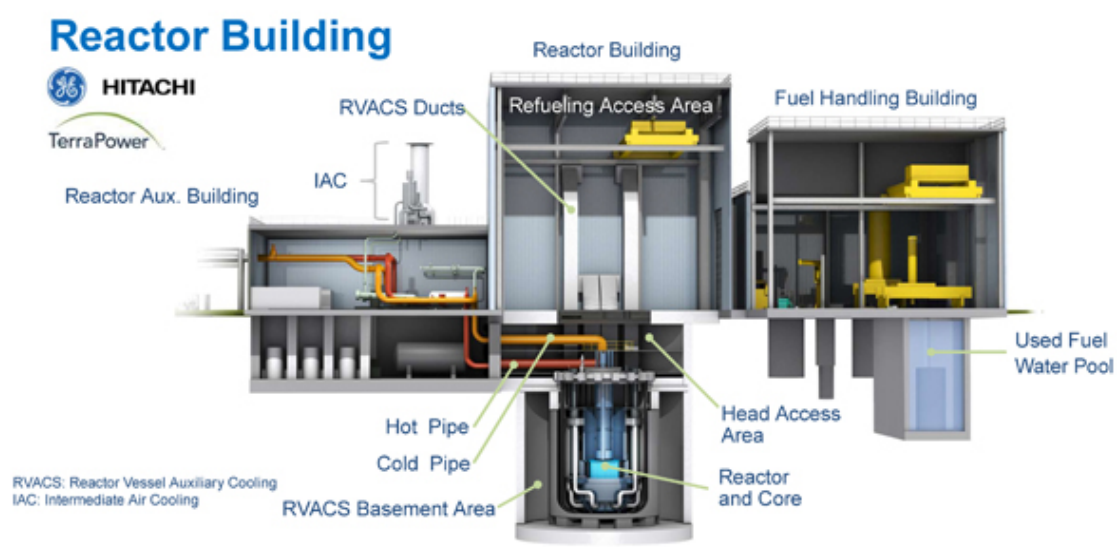

Figure A-4 General plant arrangement of Natrium

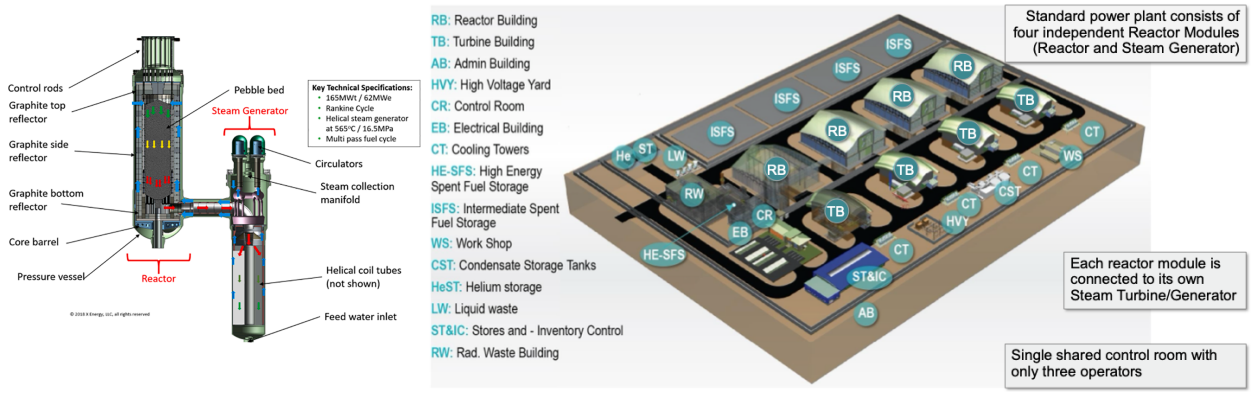

Figure A-5 Reactor module (left) and general plant arrangement of Xe-100

The data sources and results of the decommissioning volume calculations of Class A, B, and C LLW of the reference PWR and VOYGR™ are provided in the sub-sections below. Due to a lack of dimensional information on buildings and reactor components, the decommissioning volumes of Class A, B, and C LLW for Natrium and Xe-100 were not calculated in this work.

A-1.1 Reference PWR

The NRC has studied the evaluation of the decommissioning waste volumes of a large PWR power station at the end of its operating life. The purpose of the studies was to assess the available technology, safety considerations, and probable decommissioning costs (Smith et al. 1978, Konzek et al. 1995). The studies were performed using a 3500 MWt/1175 MWe PWR with detailed descriptions of the structures and equipment housed therein.

The detailed information on activation levels, masses, and volumes of all reactor components recovered from a reactor decommissioning process has been provided by Smith et at. (1978), including the burial volumes of the recovered materials. The mass and volume data from selected concrete structures in the reference PWR are provided in Table A-3. The design values indicate the mass or volume of the structures before decommissioning. Thus, the difference between the design and recovered values shows the reduction of mass or volume through the decontamination process. In those studies, a reduction of concrete volume by a factor of 5 was assumed. The decommissioning experience shows that the reduction factor is 5–7. The burial volume is about a factor of 2 larger than the recovered volume because the burial volume includes packing and containers.

The calculated decommissioning volumes of Class A, B, and C LLW are provided in Table A-4. The volumes are divided into steel and concrete for the containment building, fuel building, auxiliary building, and other reactor components. The decommissioning LLW volumes with and without decontamination of the concrete buildings are compared. Because the concrete volume is the dominant contributor to the total decommissioning LLW volume, the decommissioning LLW volume can be reduced by prior decontamination of concrete buildings.

Table A-3 Mass and volume data for selected concrete structures of reference PWR

Component	Mass, t		Volume, m^3		
	Design	Recovered	Design	Recovered	Burial
Refueling cavity liner – concrete parts	1,107.0	275.8	460	114.6	232.0
Pressurizer enclosure – concrete parts	980.0	244.9	410	102.5	195.7
Steam generator – concrete parts	1,980.0	480.8	807	196.0	384.2
CRD missile shield	99.0	17.2	41	7.1	14.5
Base liner – concrete parts	7,396.0	521.6	3078	217.1	463.9
Cask loading pit	646.0	129.2	269.0	53.8	N/A
Spent fuel pool – concrete parts	3,660.0	732	1,586.0	317.2	N/A
Reinforced vaults	2,341.0	468.2	1,015.0	203.0	N/A
Aux. building – concrete parts	17,592.0	3,570.4	7,623.0	1547.1	N/A

Table A-4 Decommissioning Class A, B, and C LLW volume of reference PWR

Component	w/o decontamination		w/ decontamination	
	Steel	Concrete	Steel	Concrete
Containment building and internals, m^3	344	30,313	344	6,003
Fuel building, m^3	19	2,770	19	554
Auxiliary building, m^3	273	7,438	273	1,488
Other components, m^3	87	-	87	-
Total, m^3	723	40,521	723	8,045
	41,244		8,768	

A-1.2 VOYGR™

The decommissioning volume of Class A, B, and C LLW of VOYGR™ was divided into three parts: NPM, reactor building, and other. Figure A-6 shows an overview of the VOYGR™ NPM. The active core, steam generator, and pressurizer are arranged inside the reactor vessel, and the reactor vessel and control rod drive mechanism are surrounded by a steel containment vessel. The NPM has a mass of ~700 t (NuScale 2022a), equivalent to a volume of about 89.7 m^3, assuming an average steel density of 7.8 $g.cm^3$.

The reactor building (RXB) and the arrangement of NPMs are shown in Figure A-3. Twelve NPMs are co-located in the reactor building pool and there is a concrete wall between NPMs. The function of the RXB is similar to the combined functions of the containment building and fuel building of a typical PWR. The RXB is assumed to be contaminated by radioactive isotopes, and it is classified as LLW. The concrete volume of the reactor building was calculated using the 12-module reactor building, which is 350 ft long, 150 ft wide, and 86 ft tall with 6-ft-thick concrete elements (NuScale 2020). The total concrete volume of the RXB was calculated using the data for all 12 NPMs, and it was divided by 12 to calculate the average value for a single NPM. The data and resulting concrete volumes are provided in Table A-5.

Figure A-7 shows the conceptual site layout of VOYGR™, which is introduced in the NuScale Standard Plant Design Certification Application (NuScale 2020). However, detailed information on the dimensions, materials, distances from the reactor building, etc., is not available. Thus, it was not possible to calculate the volumes of auxiliary buildings and the other components. However, by assuming that the

sizes of the auxiliary buildings and the other components are generally proportional to the reactor power rating for typical PWRs, they were approximated using the reference PWR data shown in Table A-2.

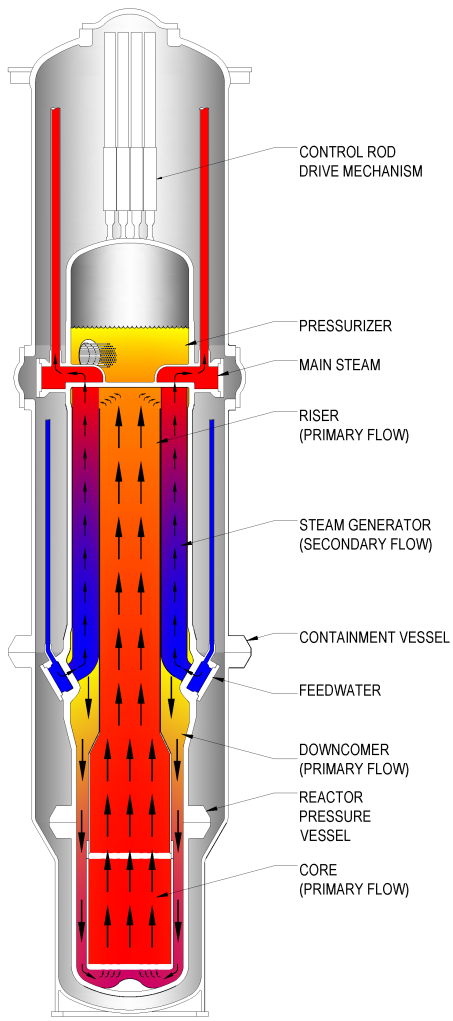

Figure A-6 VOYGR™ nuclear power module

Table A-5 VOYGR™ reactor building dimensions and volumes

Components	Symbol	Value
Number of nuclear power modules (NPMs)	N_{NPM}	12
Reactor building (RXB) side wall thickness, m	T_{RXB}	1.52
RXB length, m	L_{RXB}	106.68
RXB width, m	W_{RXB}	45.72
RXB height, m	h_{RXB}	26.21
Thickness of wall between NPMs, m	T_{NPM}	1.52
Width of wall between NPMs, m	W_{NPM}	6.0
Height of wall between NPMs, m	h_{NPM}	24.69
RXB floor volume per NPM (= $T_{RXB}\ L_{RXB} W_{RXB} / N_{NPM}$), m3	V_{RXB_f}	619.4

Volume of building walls = $\{(h_{RXB} - T_{RXB})L_{RXB}W_{RXB} - (h_{RXB} - T_{RXB})(L_{RXB} - 2T_{RXB})(W_{RXB} - 2T_{RXB})\}/N_{NPM}$, m3	V_{RXB_w}	936.6
Volume of wall between NPMs $=T_{NPM}\ h_{NPM}W_{NPM}$, m3	V_{NPM}	225.8
Total concrete volume of RXB $=V_{RXB_f} + V_{RXB_w} + V_{NPM}$, m		1781.8

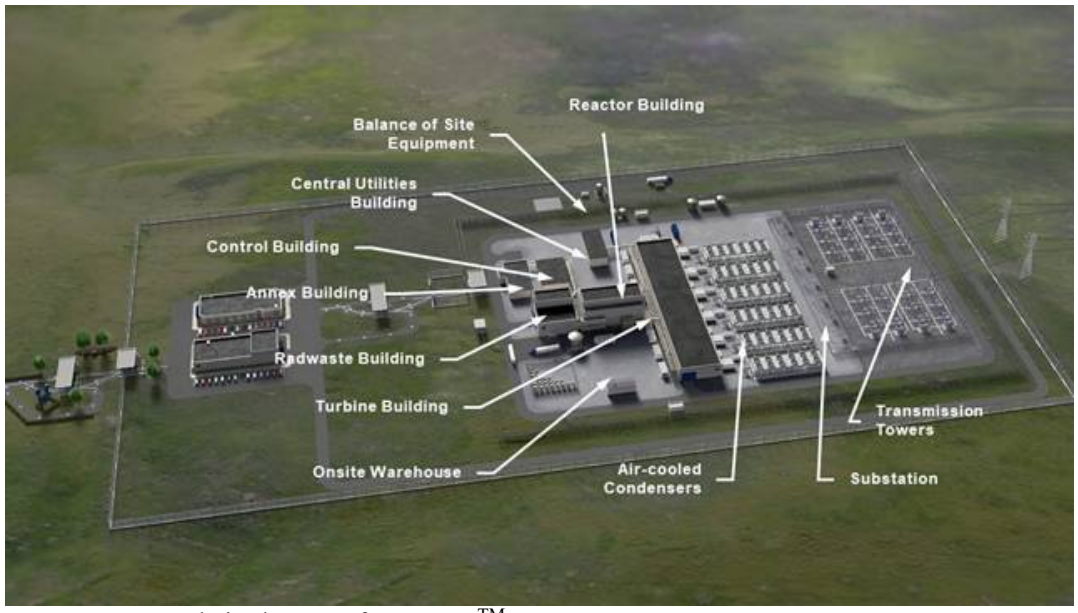

Figure A-7 Conceptual site layout of VOYGR™

A-2. Decommissioning Volume of GTCC LLW

Depending on the generation mechanisms, there are three GTCC categories: activated metals resulting from operations, process wastes such as resin and filters in decontamination systems, and activated metals recovered through reactor decommissioning (Mancini et al. 1994). The activated metals recovered during a decommissioning process include the permanent structure near the active core, such as the core supporting structures and biological shields. In the present work, the activated metals recovered from decommissioning are compared because they are the largest source of the GTCC volume.

Smith et al. (1978), Mancini et al. (1994), and Konzek et al. (1995) have evaluated the GTCC volume of ~3400 MWt PWRs. Table A-6 shows volumes and thermal neutron fluxes where reactor components are activated up to the GTCC level during a reactor operation period of 40 years with a 70% capacity factor. The thermal flux level near the active core is 10^{12}–10^{13} neutrons/cm^2-sec, and the core supporting structures, such as the core baffle, core barrel, and upper and lower grid plates, are activated up to the GTCC level. The total GTCC volume is 6.3–9.8 m^3 or 0.10–0.15 m3/GWe-year, slightly lower than the actual GTCC volume of 11 m^3 recovered from the Rancho Seco Nuclear Plant (McGrath and Reid 2014). The variation shown in Table A-6 is due to different dimensions and to the fact that the thermal shield outside the core barrel or lower supporting column was not classified as GTCC in some studies due to lower activation levels.

The activation levels of the core supporting structures near the active core differ in different reactor types due to reactor-specific design features. VOYGR™ has a core supporting structure and activation level similar to the reference PWR because VOYGR™ adopts the PWR technology. In Natrium and Xe-100, the active core is surrounded by reflector assemblies or graphite blocks to protect the core internal structure from neutron irradiation damage. As a result, it is expected that the activation level of both

Natrium and Xe-100 is lower for core supporting structures than that of the reference PWR but high for reflector assemblies or graphite blocks.

To categorize waste classifications, the activation levels of the Natrium reflector and core baffle and the Xe-100 graphite block were evaluated after simulated irradiation for a reactor lifetime of 60 years and a capacity factor of 90%. It was assumed that Natrium's reflector and core baffle and Xe-100's graphite block were made with HT9, SS-304, and A3-3 graphite, respectively. The activation level of PWR's baffle, made with SS-304, was also calculated for comparison purposes. These irradiations were conducted with the ORIGEN-2.2 code, using the Natrium neutron flux data obtained from a calculation performed with the Argonne Reactor Computation code for the 840-MWt core configuration shown in Figure 3-1. The Xe-100 neutron flux data were obtained from Mulder and Boyes (2020). The PWR neutron flux data were obtained from Smith et al. (1978). Figure A-8 shows the radial flux distribution at the core centerline of the Natrium reactor; the flux distributions of the reference PWR and Xe-100 were obtained from Mancini et al. (1994) and Mulder and Boyes (2020).

Figure A-8 shows that the peak thermal flux in the reflector assemblies and core baffle region (just outside of the shield) is 10^{11} to 10^{12} neutrons/cm^2-sec, which is about one or two orders of magnitude lower than that of the reference PWR. For Xe-100, the peak thermal neutron flux at the surface between the core outer surface and the inner surface of the radial graphite block is ~1.6 x 10^{14} neutrons/cm2-sec.

The ORIGEN cross sections of the FFTF and PWRUE cores were used in irradiation calculations for Natrium and the reference PWR, respectively. An activated graphite block contains C-14, H-3, Cl-36, and Co-60 (Li 2017). Except for C-14, the activities of these isotopes are much lower than the GTCC criteria. C-14 generated from the N-14(n,p)C-14 reaction is the dominant contributor to the graphite-block activity (Davis 1977, Poskas 2016, Li et al. 2017, Worth et al. 2021). It is noted that N-14 exists in graphite blocks as a 10–100 ppm impurity (Bush et al. 1984). Thus, the C-14 activation level in the graphite block was calculated using both the 10 ppm and 100 ppm nitrogen impurity levels and an effective N-14 (n,p) cross-section of 1.02 barns in an HTGR (Davis 1977).

The activation levels of the metals and graphite block were calculated using the activity evaluation guidance of 10 CFR 61.55. The resulting activity was normalized to the upper bounding activity level of LLW Class C and compared in Figure A-9. If the normalized activity is higher than 1.0, the metal or graphite block is classified as GTCC. The estimated GTCC volumes of four reactors are compared in Table A-6.

Table A-6 Comparison of GTCC volumes of reference PWR

Reference component	Smith et al. (1978)		Konzek et al. (1995)	Mancini et al. (1994)
	Volume, m^3	Thermal flux, n/cm^2-sec	Volume, m^3	Volume, m^3
Baffle (shroud)	1.6	9.36 x10^{12}	1.6	2.0
Core barrel	5.8	1.33 x10^{12}	3.9	3.5
Upper grid plate	0.6		0.5	0.5
Lower grid plate	0.5		0.6	0.3
Thermal shield	1.3		1.1	
Lower support column			0.4	
Total, m^3	9.8		8.1	6.3
Total per unit of electricity, m^3/GWe-yr	0.15		0.13	0.10

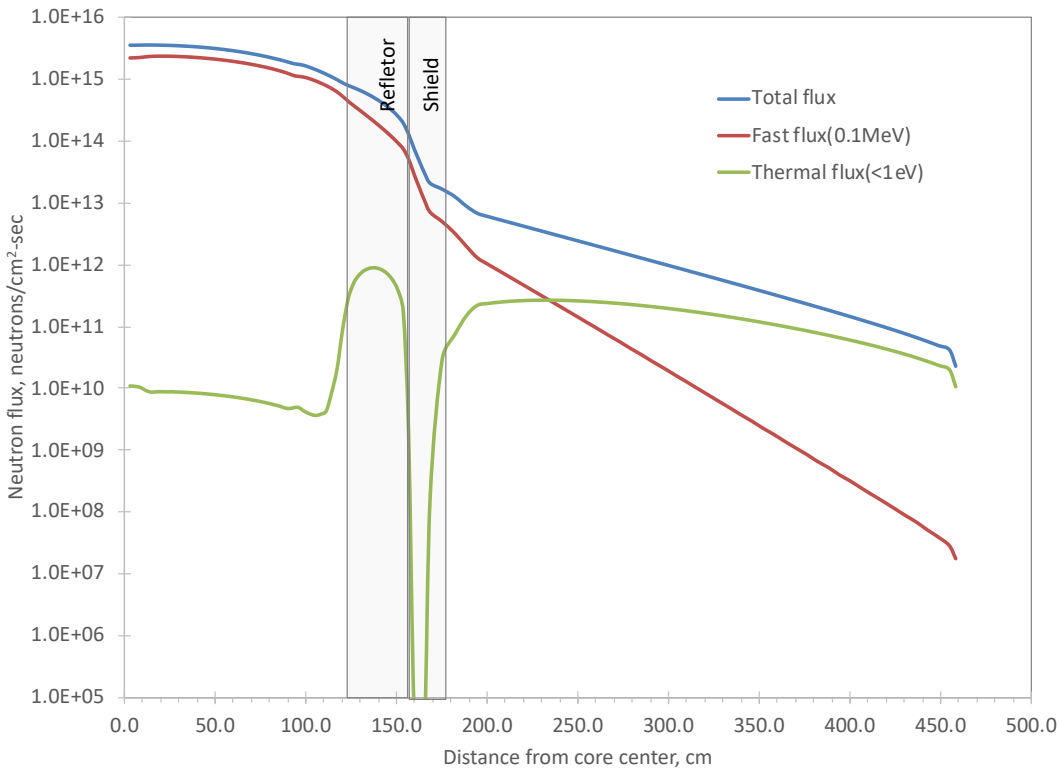

Figure A-8 Radial flux distribution in an 840-MWt SFR with HALEU metallic fuel

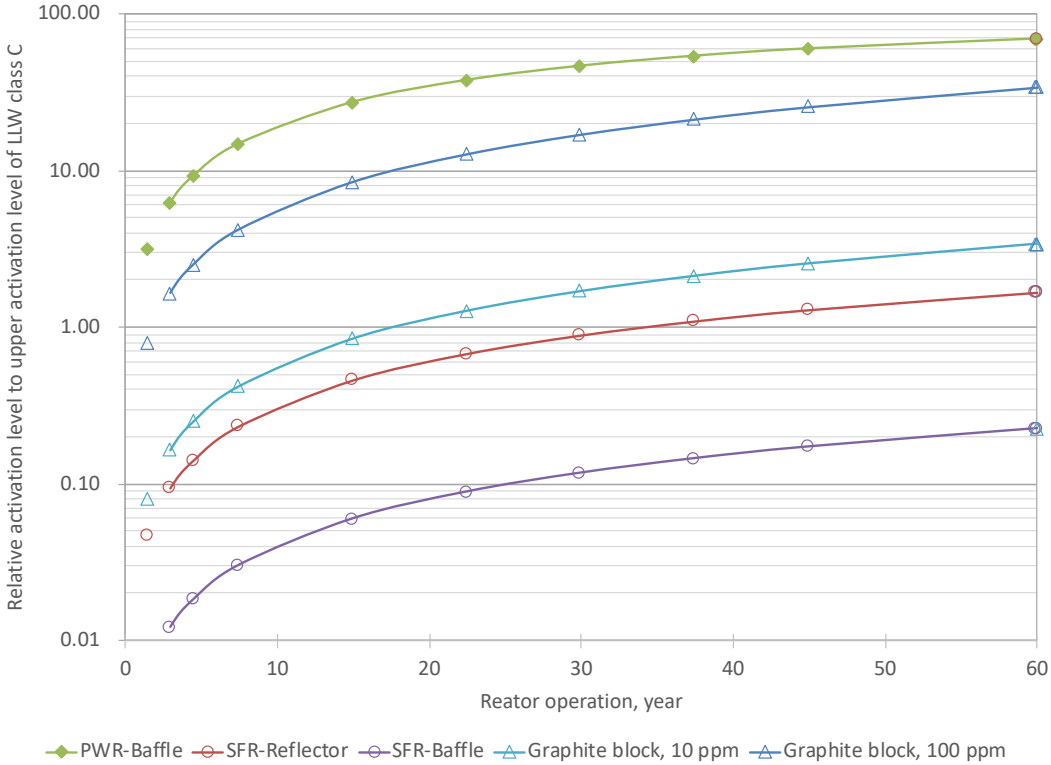

Figure A-9 Normalized activation levels of baffle and reflector in PWR and SFR

The PWR core baffle is activated to the GTCC level relatively early in the reactor life. In contrast, the Natrium baffle is not activated to the GTCC level until beyond the end of the reactor lifetime due to the low thermal flux level. The Natrium reflector becomes GTCC after 30 years. Depending on nitrogen impurity concentration, the irradiation time of the Xe-100 graphite block to reach the GTCC level is different: i.e., it takes 20 and 3 years for 10 and 100 ppm nitrogen, respectively. It is noted that both the Natrium reflector and the Xe-100 graphite block are removable. Thus, the Natrium reflector assemblies and the Xe-100 graphite block could be LLW or GTCC, depending on residence times in the core.

For VOYGR™, the GTCC volumes were calculated by assuming that reactor components of the reference PWR that are activated to GTCC (Table A-7) are also activated to GTCC in VOYGR™. For Natrium and Xe-100, the radial reflector assemblies and graphite blocks are the reactor components expected to become GTCC. The net GTCC volume for VOYGR™ is a factor of two smaller than for the PWR due to the smaller active core height and diameter. However, when normalized to the unit of electricity generation, the GTCC volume is a factor of six larger than that of the PWR.

For Natrium and Xe-100, the GTCC volume varies depending on the residence time of the reflector assemblies and graphite blocks in the core. Natrium or Xe-100 do not generate GTCC LLW when the reflector assemblies and graphite blocks are discharged before they are activated to the GTCC level. However, compared to the reference PWR, Natrium or Xe-100 generate a factor of 4 or 193 more GTCC volume respectively when the reflector assemblies and graphite blocks reside in the core for reactor lifetimes. The replacement schedule of the reflector assemblies and graphite blocks depends on the reactor operation and waste management strategies. Frequently replacing reflector assemblies or graphite blocks reduces the GTCC volume to be disposed of in a geological repository. However, it increases the LLW Class B or C waste volume as a trade-off. More details are provided in the following subsections.

Table A-7 Comparison of GTCC volumes

	PWR	VOYGR™	Natrium	Xe-100
Baffle (shroud), m^3	[a)] 1.6–2.0	2.1		
Barrel, m^3	3.5–5.8	0.6		
Grid plates, etc., m^3	0.9–2.6	0.3		
Reflector assemblies, m^3	-	-	[b)] 0.0–10.3	-
Radial and axial graphite blocks, m^3	-		-	[c)] 0.0–106.0
Total volume Net volume, m^3 Per unit of electricity generation, m^3/GWe-yr	6.3–9.8 [d)] 0.13	3.0 0.72 (5.7)	0.0–10.3 0.0–0.55 (0–4.4)	0.0–106.0 0.0–24.5 (0–193.1)

f) Range from Table A-6
g) Variation is dependent on the residence time of reflector assemblies in the core.
h) Variation is dependent on the residence time of graphite blocks in the core.
i) Calculated using the net GTCC volume calculated by Konzek et al. (1995).

A-2.1 VOYGR™

The GTCC volume of VOYGR™ was calculated using the dimensions and materials of the reactor building and components provided in the NuScale Standard Plant Design Certification Application, Part 2 - Tier 2, Revision 5 (NuScale 2020). The core supporting assembly of VOYGR™ is shown in Figure A-10. Given the assumption that the neutron flux level near the active core is similar to that of the reference PWR, the core supporting structures such as lower and upper grid plates, reflector blocks, core baffle, and barrel are activated to the GTCC level. Dimensions of the core supporting structure and calculated GTCC volumes are provided in Table A-8 and Table A-9, respectively.

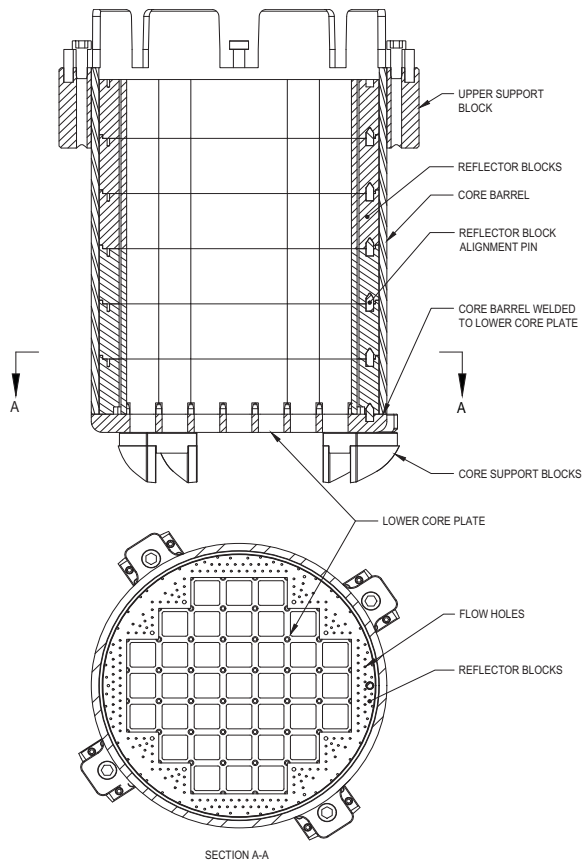

Figure A-10 VOYGR™ core supporting structure

Table A-8 Dimensions of VOYGR™ core support components

Dimension	Symbol	Value
Active core height, m	h_{cr}	2.000
Equivalent active core diameter, m	D_{cr}	1.506
Core barrel ID, m	D_{br}^{in}	1.880
Core barrel OD, m	D_{br}^{out}	1.981
Core barrel thickness, m	T_{br}	0.051
Reflector height, m	h_{ref}	2.330
Reflector/shroud radius, m	R_{ref}^{out}	0.926
Reflector thickness, m	T_{ref}	0.173
Core support plate thickness, m	T_{pl}	0.051

Table A-9 GTCC volume (m3) of VOYGR™ core support components

Components	Value
Lower and upper core support plates = $2\pi T_{pl} {R_{ref}^{out}}^2$, m^3	0.3
Reflector and baffle = $\pi h_{ref}({R_{ref}^{out}}^2 - {D_{cr}}^2)/4$, m^3	2.1
Core barrel = $\pi h_{cr}({D_{br}^{out}}^2 - {D_{br}^{in}}^2)/4$, m^3	0.6
Total, m^3	3.0

A-2.2 PRISM/Mod-B for Natrium

The GTCC volume of Natrium was approximated using the core configuration introduced in Section 2, which was developed using the 840-MWt PRISM/Mod-B reactor to reproduce the claimed core performance of Natrium (in particular, the discharge burnup of 150 GWd/t). The flux distribution and activation levels of core structures were provided above. The radial reflectors are activated to the GTCC level when they reside for more than 30 years in the core. However, the other structures (radial shield, core baffle, barrel, etc.) are not activated to the GTCC level until after the 60-year reactor lifetime. Thus, the radial reflectors are the only components that are active to the GTCC level in Natrium. Dimensions and total volume of radial reflectors are provided in Table A-10. It is noted that the radial reflectors are replaceable. Thus, the radial reflector could be either GTCC or LLW, depending on its residence time in the core.

Table A-10 Dimensions and volume of radial reflectors in 840-MWt SFR

Parameter	Symbol	Value
Radial reflector assembly length, m	L_{ref}	4.776
Radial reflector assembly pitch, m	P_{ref}	0.1614
Number of radial reflectors	N_{ref}	144
Steel volume fraction, %	F_{ref}^{steel}	83.9
Reflector steel volume (=. $\frac{\sqrt{3}}{2} L_{ref} {P_{ref}}^2 N_{ref} F_{ref}^{steel}$), m^3		10.3

A-2.3 Xe-100

The neutronics characteristics of the Xe-100 reactor were studied by Mulder and Boyes (2020). The Xe-100 dimensions and design parameters may have evolved since the development of that paper, but the design information on the latest version of Xe-100 is protected as proprietary information. Thus, the GTCC volume of Xe-100 was calculated using the dimensions and flux distributions provided previously, with the assumption that the evolution is not significant. The activation level of graphite blocks as functions of impurity level and residence time in the core were discussed above. The radial and axial graphite blocks are activated to the GTCC level within 3 to 20 years, depending on the nitrogen impurity level. However, the other core support structures are not activated to the GTCC level until the end of reactor lifetime. Dimensions and total volume of the graphite blocks are provided in Table A-11. It is noted that the graphite blocks could be either GTCC or LLW, depending on their residence time in the core.

Table A-11 Dimensions and volume of graphite blocks

Parameter	Symbol	Value
Height of radial graphite block, m	h_{RGB}	9.5
Inner diameter of radial graphite block, m	D_{RGB}^{inner}	2.4
Outer diameter of radial graphite block, m	D_{RGB}^{outer}	4.4
Outer diameter of axial graphite block, m	D_{AGB}^{outer}	2.4
Thickness of axial graphite block (bottom & top), m	T_{AGB}	1.0
Radial graphite volume (=$\pi h_{RGB}({D_{RGB}^{outer}}^2 - {D_{RGB}^{inner}}^3)/4$), m^3	V_{RGB}	101.5
Axial graphite volume (=$\pi T_{AGB} {D_{AGB}^{outer}}^2/4$), m^3	V_{RGB}	4.5
Total volume of graphite blocks (= V_{RGB}+V_{AGB}), m^3		106.0