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Original Article

Nuclear waste attributes of near-term deployable small modular reactors

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ABSTRACT

The nuclear waste attributes of near-term deployable SMRs were assessed using established nuclear waste metrics, which are the DU mass, SNF mass, volume, activity, decay heat, radiotoxicity, and decommissioning LLW volumes. Metrics normalized per unit electricity generation were compared to a reference large PWR. Three SMRs, VOYGR, Natrium, and Xe-100, were selected because they represent a range of reactor and fuel technologies and are active designs deployable by the decade's end. The SMR nuclear waste attributes show both some similarities to the PWR and some significant differences caused by reactor-specific design features. The DU mass is equivalent to or slightly higher than the PWR. Back-end waste attributes for SNF disposition vary, but the differences have a limited impact on long-term repository isolation. SMR designs can vary significantly in SNF volume (and thus heat generation density). However, these differences are amenable to design optimization for handling, storage, transportation, and disposal technologies. Nuclear waste attributes from decommissioning vary depending on design and decommissioning technology choices. Given the analysis results in this study and assuming appropriate waste management system and operational optimization, there appear to be no major challenges to managing SMR nuclear wastes compared to the reference PWR.

1. Introduction

Nuclear energy has been a steady source of ~20 % of the United States' 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 step in deploying new advanced Small Modular Reactors (SMRs) is underway. The SMR technologies and economics have been the focus of many studies, but there has been only minimal information published on the amount of nuclear waste from different types of SMRs 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 in the future.

In this study, the nuclear waste attributes of SMRs scheduled for deployment this decade were assessed using nuclear waste metrics, and the results were compared to those of a reference large Pressurized Water Reactor (PWR). This study assessed three SMRs: NuScale/VOYGR™, Natrium™,[1] and Xe-100. The three SMRs were selected because they are all active designs deployable in the near term. Sites for the 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. In addition, they represent a range of reactor and fuel technologies and comparatively mature designs that have been selected for DOE support for near-term deployment.

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. Reactor-specific design features should be accounted for when evaluating nuclear waste attributes because they affect the metrics of nuclear waste. For example, 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, while the integral core configuration of VOYGR eliminates several reactor components.

The information on reactor design and performance parameters of the three SMRs and the reference PWR were obtained from open

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[1] 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.

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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. While this work was conducted independently, results were provided to the vendors of the three SMRs to ensure the open literature information was not misinterpreted.

2. Waste metrics

A set of relevant metrics must be selected to examine the waste attributes of specific reactor technologies. There is a wide range of potential waste metrics. Some metrics are fundamental characteristics of the waste attributes, and some are useful 'derived metrics' important to various aspects of waste management. For this study, the relevant metrics from the "Nuclear Fuel Cycle Evaluation and Screening" (E&S) [1] are used with the additional derived metrics of decay heat and radiotoxicity. The U.S. Department of Energy (DOE) conducted the E&S study that developed technology-neutral metrics for a broad range of potential nuclear fuel cycles. These metrics were developed with extensive input from industry, academia, government, the public, and past practices. The final metrics used in this study and rationale are given below.

2.1. Front-end metric

- Depleted uranium (DU) mass (t/GWe-year) – While there are some limited non-nuclear uses for DU, there is no practical use for the quantities of DU produced in a once-through fuel cycle at this time,[2] and therefore it is considered a waste. This metric is relevant to DU management/disposal and directly related to DU disposal cost.

2.2. Back-end metrics

- Spent nuclear fuel (SNF) mass (t/GWe-year) –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.
- SNF activity at five points between 10 and 100,000 years after discharge (Ci/GWe-year) – 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-year) – 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-year) – 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-year) – This metric is relevant to SNF handling, packaging, storage, transportation, and repository design but not currently a cost driver for disposal.

2.3. Decommissioning metrics

- The volume of Class A, B, and C Low-level Waste (LLW) (m^3/GWe-year) from decommissioning a reactor - This metric is relevant to the LLW shallow land burial cost.
- The volume of Greater Than Class C (GTCC) LLW (m^3/GWe-year) from decommissioning a reactor - This metric is relevant to the cost of GTCC disposal which is not suitable for LLW shallow land burial.

It is noted that LLW during reactor operation is not included in this study because of a lack of comparable and consistent data for any of the reactors and 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.

3. Reactor design parameters

The primary design parameters are provided in Table I. For consistent comparison, a reactor lifetime of 60 years and a capacity factor of 90 % are assumed for all four reactors. In July 2022, the U.S. Nuclear Regulatory Commission (NRC) directed the issuance of a final rule that certifies NuScale's 50 MWe SMR design for use in the United States [2], 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 [3,4]. The VOYGR design [5] is the reactor used in our analysis, but the building design used for decommissioning LLW is from the earlier certification (any building changes are unknown).

In October 2020, DOE announced awards un der 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 [6].

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 [7]. 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

Table 1
Primary reactor design parameters.

	Ref. PWR	VOYGR	Natrium	Xe-100
Power, MWt/MWe	3,500 /1,175	250/77	840/345	200/80
Thermal efficiency	34 %	31 %	41 %	40 %
Fuel form	UO_2	UO_2	U–Zr w/o Na-bond	TRISO/ 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 (Power Reactor Innovative Small Module)/Mod-B design that was revised to have the discharge burnup close to the Natrium design burnup of ~150 GW d/t.

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

of the Natrium commercial plant is 150 GW d/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 [8].

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 are 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 with a fresh pebble.

An 1175-MWe PWR used for the evaluation of decommissioning volume and costs by NRC [9,10] was selected as the reference large PWR for comparison purposes. It is representative of the current light water reactor fleet mostly constructed between 1970 and 1990. The AP1000 under construction in Vogtle, GA [11] was also considered in this study. However, it was passed over due to a lack of information needed to evaluate the waste characteristics. The reference PWR data were obtained from the nuclear waste evaluation studies performed by Smith et al. [9], Konzek et al. [10], and Mancini et al. [12].

Design parameters of Natrium and Xe-100 were obtained from the open literature [7,13–17]. Several design parameters needed for waste evaluation are protected as proprietary information. The missing data were obtained from the same type of 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 (Power Reactor Innovative Small Module)/Mod-B reactor [18] that was revised to approximate the Natrium's design burnup of 150 GW d/t with the heavy metal (HM) mass of 21.5 t in the core [7]. It is noted that the Natrium design combines features from the previous GEH PRISM and TerraPower Traveling Wave designs [19].

4. Nuclear waste metric results

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

4.1. Front-end waste

DU can be disposed of 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 [20]. The DU masses are calculated by assuming a tail uranium enrichment of 0.25 %,[3] and the results are compared in Table II.

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.

Table 2
Comparison of DU masses.

	Ref. PWR	VOYGR	Natrium	Xe-100
Power, MWe	1,175	75	345	80
DU mass, t/GWe-year	179	220 (1.23)	209 (1.17)	174 (0.97)

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.

4.2. Back-end waste

- **SNF Mass and Volume**

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

$$M_{DF} = \frac{365}{B \times \eta} \text{ and } V_{SNF} = f_{mass}^{volume} \times M_{DF},$$

where.

M_{DF} = SNF fuel mass (t/GWe-year),
V_{SNF} = SNF assembly/pebble volume[4] (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).

The normalized SNF mass and SNF volume are compared in Table III. 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 (Light Water Reactor) advanced reactors (Natrium and Xe-100) generate less SNF mass by a factor of four due to higher burnup and thermal efficiency. Less mass equates to less waste to be disposed of per unit electricity generation.

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. The Natrium

Table 3
Comparison of SNF masses and volume.

	Ref. PWR	VOYGR	Natrium	Xe-100
Power, MWe	1,175	75	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)

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

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

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 many Xe-100 reactors are built, a portion of the geologic repository could be designed specifically to optimize the disposal of Xe-100 SNF with its higher volume but lower decay heat and radiotoxicity.

- **SNF Activity**

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. The SMR discharge fuel compositions were obtained from the reactor depletion analyses. Using the discharge fuels, the SNF activity was calculated over several timeframes to provide general trend information as well as the isotopic data needed for the calculation of decay heat and radiotoxicity. Table IV presents the results of these calculations. Compared to the PWR SNF, the normalized activity of the VOYGR SNF is slightly higher due to slightly lower thermal efficiency resulting in less electricity generation.

The other two SMRs initially have lower normalized activity due to higher thermal efficiency. However, the values diverge between 1000 and 10,000 years as the activity of most fission products fades, and the actinide content begins to dominate. Most nuclear fuels are discharged with the fissile content is sufficiently depleted. However, in a fast reactor like Natrium, the fissile content is not depleted as quickly due to the 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 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 was 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 by ORIGEN 2.2 for one metric ton of SNF using the effective one-group cross sections that were obtained from the SMR depletion analyses. The radioactivity was converted to radiotoxicity using the effective dose coefficients for ingested particulates of the International Commission on Radiological Protection (ICRP) [21]. Results are normalized to the unit electricity generation (GWe-year) and compared in Table V.

Table 4
Comparison of SNF activity after discharge.

Activity (Ci/GWe-year)	Ref. PWR	VOYGR	Natrium	Xe-100
At 10 years	1.23 E+07	(1.07)	(0.63)	(0.79)
At 100 years	1.32 E+06	(1.08)	(0.71)	(0.80)
At 1000 years	5.07 E+04	(1.04)	(0.63)	(0.45)
At 10,000 years	1.42 E+04	(1.05)	(1.40)	(0.38)
At 100,000 years	1.61 E+03	(1.08)	(1.17)	(0.58)

Table 5
Comparison of SNF decay heat and radiotoxicity.

	Ref. PWR	VOYGR	Natrium	Xe-100
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)

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 SNF mass due to higher discharge burnup. However, Natrium and Xe-100 generate less SNF mass per unit electricity generation. Compared 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 dissipates somewhat by 100,000 years to 53 % as the Pu decays.

5. Decommissioning nuclear waste

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. In this study, the decommissioning nuclear waste includes the recovered LLW after removing fuels and coolant. Decommissioning waste consists of 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 [22].

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 [10]. These range from immediate 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 variations 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 8200 to 109,000 m^3 [23,24].

Nuclear waste volumes recovered from decommissioned U.S. PWRs are compared in Fig. 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 [23], while other data are the total decommissioning LLW volumes [24]. 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, reactor buildings at

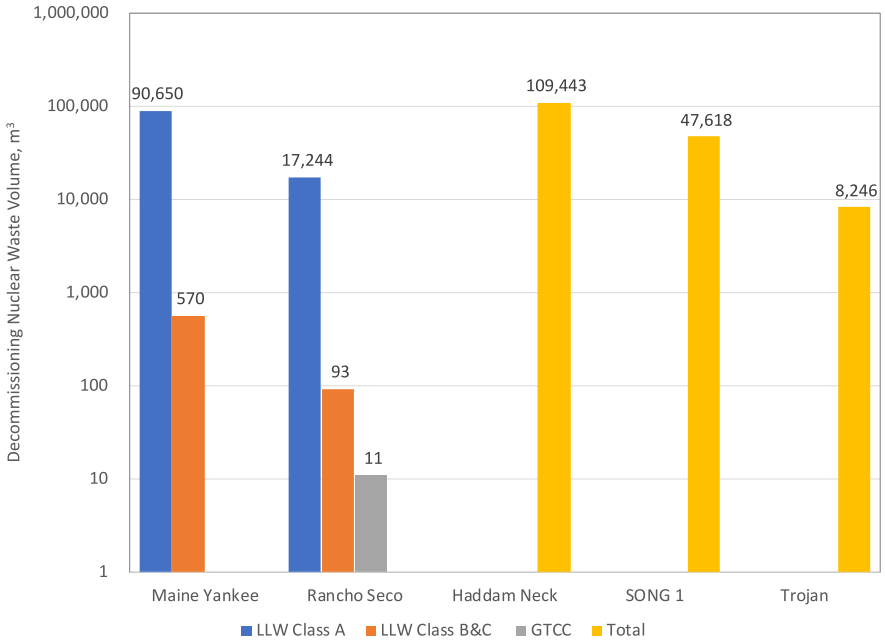

Fig. 1. Comparison of decommissioning nuclear waste volumes.

Rancho Seco were decontaminated as part of decommissioning, while there was little decontamination of reactor buildings at Maine Yankee [23]. Nuclear waste volumes are reduced by a factor of 5–7 through decontamination in a large PWR [9,24].

- **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 [9]. 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 volumes of Class A, B, and C LLW of the reference PWR and VOYGR were evaluated without considering decontamination prior to decommissioning. The information needed for Natrium and-Xe-100 was obtained from Refs of [7,13,15–17]. If data is missing but required for the decommissioning volumes, the PRISM/Mod-B data and Ref. [14] were used in this study. The detailed information used to calculate the decommissioning volumes is provided in Ref. [25].

Data for the calculation of the reference PWR waste was obtained from Smith et al. [9] and Konzek et al. [10]. 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 [26] 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 [27]. 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.

The resulting LLW volumes are compared in Table VI. 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.

- **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 [12]. The activated materials recovered during decommissioning 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

Table 6
Comparison of decommissioning Class A, B, and C LLW volume.

	Ref. PWR		VOYGR	
	Metal	Concrete	Metal	Concrete
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	40,944		2,383	
- Per electricity generation, m^3/GWe-year	645		573 (0.9)	

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, the activation level of both Natrium and Xe-100 is expected to be 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 Table VII.

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 electricity generation, the GTCC volume is a factor of six larger than that of the PWR.

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 7
Comparison of decommissioning GTCC LLW volume.

	Ref. 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	6.3–9.8	3.0	0.0–10.3	0.0–106.0
- Per unit electricity generation, m^3/GWe-year	[d] 0.13	0.72 (5.7)	0.0–0.55 (0–4.4)	0.0–24.5 (0–193)

[a] Range from calculated GTCC volumes in Smith et al. [9], Konzek et al. [10], and Mancini et al. [12].
[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. [10].

- **Coolant Wastes**

Reactor primary coolants become contaminated when trace elements (corrosion products, etc.) in the coolant are activated and the coolant itself becomes activated. Any cladding breaches will also contribute small amounts of fission products. While the 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 [28].

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 h) through Na-23 (n, 2n)Na-22 and Na-23 (n,γ)Na-24 reactions, and impurities are also activated. Fig. 2 shows the activity levels of the coolant sodium 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 (Experimental Breeder Reactor)-II and FFTF (Fast Flux Test Facility) 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 [29].

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 L) 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 [30]. 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 [31]. 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 [18]. Using the sodium carbonate density of 2.5 g/cm^3, the disposal waste volume of sodium coolant is ~360 m^3, which

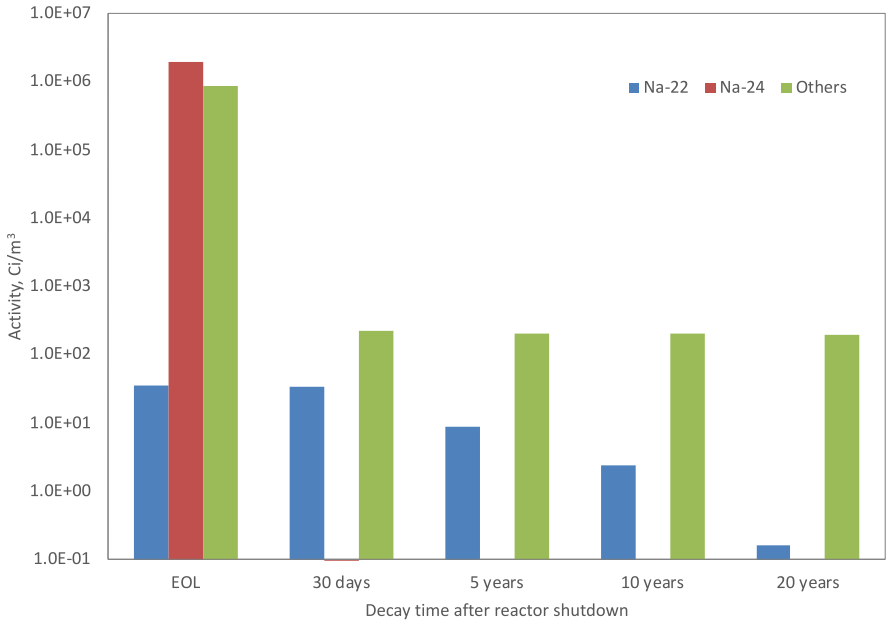

Fig. 2. Coolant sodium activation and decay.

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 the oxidation of core graphite structures [32]. The gaseous coolant was not considered radioactive waste to be disposed of 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 proven methods successfully employed previously during decommissioning of two sodium-cooled experimental fast reactors.

6. Conclusions

The nuclear waste attributes of Small Modular Reactors (SMRs) scheduled for deployment this decade were assessed using established nuclear waste metrics. The metrics include the DU mass, SNF mass, volume, activity, decay heat and radiotoxicity, and decommissioning LLW volumes. All metrics were normalized to the unit electricity generation and compared to a reference large PWR. Three SMRs, VOYGR, Natrium, and Xe-100, were selected because they represent a range of reactor and fuel technologies, have been designed to provide improved performance, and are all active designs scheduled to be deployed by the decade's end.

The nuclear waste attributes of the SMRs studied show some similarities to the reference PWR and some significant differences caused by reactor-specific design features with advanced coolants, fuel forms, and reactor configuration. Front-end waste attribute (DU mass) from SMR fuel production range from equivalent to 1.2 times the reference PWR. Back-end waste attributes for spent fuel disposition vary from large reductions to small to moderate increases in the SNF mass (factors of 0.2–1.2), activity (factors of 0.4–1.4), and radiotoxicity (factors of 0.3–1.5). These differences have a limited impact on long-term repository isolation. SMR designs can vary significantly (factors of 0.6–12.3) in the SNF 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 vary 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 managing SMR nuclear wastes compared to the reference PWR nuclear wastes. The results of this study are only applicable to a once-through fuel cycle. Any of these reactors, including the reference PWR, could be used with fuel recycle, resulting in reductions in most waste attributes, as indicated in the E&S study [1].

Declaration of competing interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

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