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Preliminary Analysis of Advanced Reactor Spent Nuclear Fuel Storage, Transportation, and Disposal – 24319

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ABSTRACT

Due to increased interest in advanced reactor deployment and their associated potential new fuel cycles, the U.S. Department of Energy (DOE) Spent Fuel and Waste Science and Technology (SFWST) program has begun to evaluate the possible implications of long-term management and final disposition of the spent nuclear fuel (SNF) generated. Safely managing and dispositioning this SNF, along with any other associated radioactive waste, is the primary focus of this initial preliminary assessment.

This paper summarizes efforts to evaluate the characteristics and packaging options for three types of advanced reactor SNF forms: (1) tristructural isotropic (TRISO), (2) metallic, and (3) irradiated fuel salt presented in the report titled "Storage, Transportation, and Disposal of Advanced Reactor Spent Nuclear Fuel and High-Level Waste". TRISO and metallic SNF and their associated waste streams were emphasized because of the near-term anticipated demonstrations of X-energy's Xe-100 and TerraPower and GE Hitachi's Natrium advanced reactors. Preliminary information on spent fuel salts discharged from molten-salt reactors (MSRs) was also examined to provide a baseline for future efforts. All calculations and assumptions were based on publicly available information.

This paper identifies several different reactors that produce either TRISO or metallic SNF as well as a few of the reactor and fuel characteristics used for the assessments. Based on these characteristics, calculations were performed to determine the applicability of packaging SNF into existing canister designs. The evaluations included geometric (e.g., dimension, volume) and mass/weight considerations, known operational approaches and loading procedures, physical and chemical considerations/conditions for storage environments, and as-loaded radiation, thermal, and criticality analyses to identify constraints on storage, transportation, and disposal.

Gaps in publicly available data pertaining to reactor operation and/or fuel composition provide increased uncertainty in some evaluations. Additionally, uncertainty in packaging and SNF management operations provide additional uncertainty. However, preliminary conclusions can still be assessed through this work and are presented in this paper.

INTRODUCTION

In 2018, the U.S. Congress passed the Nuclear Energy Innovation Capabilities Act of 2017 (Public Law 115-248), enabling both private and public institutions to engage in civilian research and development (R&D) regarding advanced nuclear energy technologies in order to expand the theoretical and practical understanding of nuclear physics, chemistry, and materials science. With the fiscal year 2020 budget,

Congress launched the Advanced Reactor Demonstration Program (ARDP) [1], which recognized the following award categories:

- Advanced Reactor Demonstrations: Expected to lead to a fully functional advanced nuclear reactor within 7 years of the award.
- Risk Reduction for Future Demonstrations: Expected to support up to five additional teams resolving technical, operational, and regulatory challenges so as to prepare for future demonstration opportunities.
- Advanced Reactor Concepts 2020 (ARC 20): Expected to support innovative and diverse designs with the potential to commercialize in the mid-2030s.

The ARDP awardees vary in terms of reactor concept, technological maturity, and anticipated scope. Even still, they represent only a fraction of the many different advanced reactor concepts/designs being actively pursued around the world. Figure 1 shows the various ARDP award winners in each category.

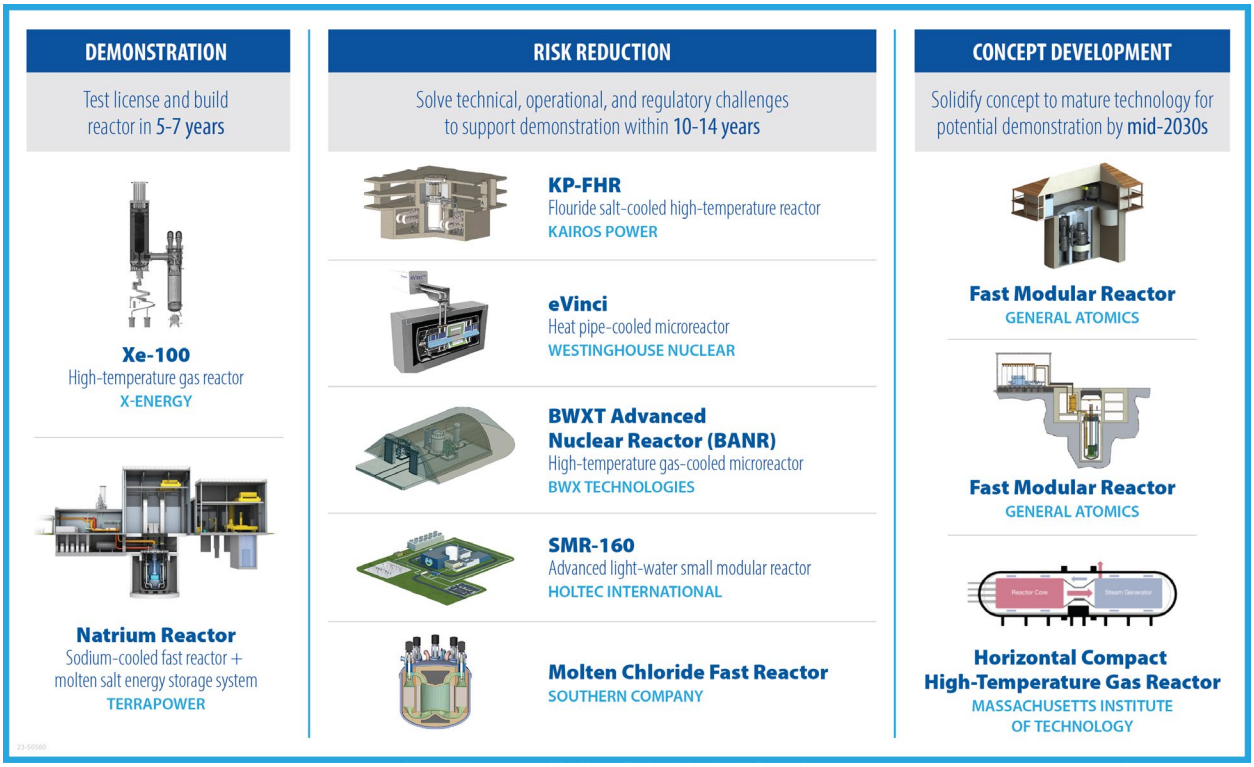

Figure 1. ARDP award winners (Images from Office of Nuclear Energy INFOGRAPHIC: Advanced Reactor Deployment).

While some definitions of advanced reactors may encompass small modular light-water reactors (LWRs) which employ uranium oxide fuel, the scope of the present work emphasizes alternative reactor technologies and fuels. Excluding small modular LWRs from the scope of this assessment still leaves a number of other advanced reactor concepts for consideration: high-temperature gas-cooled reactors (HTGRs), sodium fast reactors (SFRs), lead-cooled fast reactors, and molten-salt reactors (MSRs). And while the various concepts within each category may significantly differ with each other, the majority use one of the following forms of fuel:

- Tristructural isotropic (TRISO)
- Metallic

- Fuel salt.

The present paper is based on the work covered in "Storage, Transportation, and Disposal of Advanced Reactor Spent Nuclear Fuel and High-Level Waste" [2], which utilized publicly available information to ascertain the general characteristics of SNF and other waste products generated by advanced reactors. In regard to each fuel type, the report also indicates potential differences that could affect the back-end management of SNF and other waste streams. For the preliminary assessment, the following characteristics were either calculated or estimated: SNF volume and mass, radiation levels over time, thermal conditions over time, potential radionuclide source terms, chemical interactions and evolutions, disposal inventories, and waste form lifetimes.

Based on these estimated characteristics, calculations were performed to determine the applicability of existing canister designs. The resulting evaluations included:

- Geometric (e.g., dimension, volume) and mass/weight considerations
- Known operational approaches and loading procedures
- Physical and chemical considerations and conditions for the storage environments
- As-loaded radiation, thermal, and criticality constraints and associated analyses.

In addition, select modifications for supporting extended dry storage and transportation were evaluated based on criticality concerns and potential requirements for repackaging the waste. Material compatibility was also of concern, as advanced reactor fuel types differ substantially from the LWR SNF already contained in dry storage systems.

ASSESSMENT METHODS

To assess the characteristics and packaging options for advanced reactor SNF, the properties and geometries of different fuel types were assumed based on publicly available information. These SNF properties were then considered in scoping analyses (structural, criticality, thermal, and dose rate) for advanced reactor SNF contained in both large- and small-diameter canisters. As packaging plans for advanced reactor SNF remain yet to be determined, existing canisters or canister designs were used in the analyses. The large-diameter canister had the external dimensions of a large multi-purpose canister (MPC) for commercial SNF. The small-diameter canisters, called Department of Energy Standard Canisters (DOESCs)[1] come in four different sizes and were designed to be compatible with the wide variety of Department of Energy (DOE)-managed SNF stored across the DOE complex. Table 1 gives the various dimensions of the canisters.

Table 1. Dimensions of the SNF canisters.

Canister Name	Outer Diameter (cm)	Length (cm)	Shell Thickness (cm)	Inner Cavity Height (cm)
10 × 18 DOESC	45.7	304.8	0.95	256.5
10 × 24 DOESC	61.0	304.8	1.27	256.5
15 × 18 DOESC	45.7	457.2	0.95	408.9
15 × 24 DOESC	61.0	457.2	1.27	408.9
General 37	189.2	459.7	1.27	429.3

[1] The nomenclature for DOESCs is based on length × diameter. For example, the 10-ft-long, 18-in.-diameter DOESC is called the 10 × 18 DOESC.

The same modeled canisters were applied to each different fuel type, though the canister internals/baskets could vary. Additionally, multiple DOESCs were analyzed inside of the larger canisters. This internal multi-canister configuration is proposed for storing and transporting DOE-managed SNF for the Road-Ready Capability Demonstration Project [3].

TRISO FUEL

Coated particle fuel was invented in the United Kingdom as part of designing the experimental Dragon HTGR. Early gas-cooled reactors used coated particles featuring one or two layers of pyrocarbon surrounding the spherical fuel kernels. However, the early carbon-layer coating systems were abandoned due to insufficient retention of fission products, leading to development of the current three-layer coating system known as TRISO [4]. Since then, TRISO fuel has been used in many different reactors, including the Thorium High-Temperature Reactor (THTR)-300 (Germany), the Fort Saint Vrain Nuclear Power Plant (United States), the High Temperature Reactor (HTR)-10 (China), and the High-Temperature Test Reactor (Japan) [5,6].

TRISO particles are typically 750–1000 μm in diameter and are embedded into larger carbon matrices such as cylindrical compacts or pebble compacts. Each particle consists of a fuel kernel, usually composed of uranium oxide or uranium oxycarbide, surrounded by a buffer layer of porous carbon. Each particle also includes inner and outer pyrolytic carbon layers, as well as a silicon carbide (SiC) layer between the pyrolytic carbon layers to serve as the primary structural layer and pressure boundary. The inner pyrolytic carbon layer protects the fuel kernel from chloride interactions, while the SiC layer guards against deleterious interactions with fission products. The outer pyrolytic carbon layer provides bonding with the graphite matrix, affording further protection for the SiC layer.

DOE has a long history of sponsoring R&D to support the implementation of TRISO-based fuels in Generation-IV HTGRs [1,7,8]. Two important DOE-sponsored R&D activities relevant to this activity are the Advanced Gas Reactor (AGR) Fuel Development and Qualification Program (AGR Fuel Program) and the ARDP. Table 2 identifies a few different TRISO-based reactor concepts and their associated power outputs and fuel forms, though this is not a comprehensive list of all TRISO-based advanced reactor technologies [2]. For a more comprehensive list of the various advanced reactors currently in development, see reports published by the Nuclear Energy Agency (NEA) [9,10] and the International Atomic Energy Agency [11].

Table 2. Summary of different TRISO-based advanced reactor technologies.

Commercial Vendor	Reactor (Reactor Technology)	Reactor Power Information (MWt / MWe)	Fuel Form
X-energy, LLC	Xe-100 (modular HTGR)	200 / 82.5	TRISO particles embedded in graphite pebbles
Kairos Power	KP-X Fluoride Salt-Cooled High-Temperature Reactor (modular pebble-bed MSR)	320 / 140	TRISO particles embedded in graphite pebbles
Westinghouse Electric, LLC	eVinci (heat-pipe-cooled microreactor)	14 / 5	TRISO particles embedded in compacts within a graphite block

Commercial Vendor	Reactor (Reactor Technology)	Reactor Power Information (MWt / MWe)	Fuel Form
Ultra Safe Nuclear Corporation	Micro-Modular Reactor (HTGR microreactor)	15 / 5	TRISO particles embedded in fully ceramic micro-compacts within a graphite block
BWX Technologies	BWXT Advanced Nuclear Reactor (HTGR microreactor)	50 / 17	TRISO particles embedded in compacts within a graphite block

Characteristics were generated and analyses performed on X-energy's Xe-100 and Kairos Power's KP-X reactors. Additional analyses performed on TRISO fuel in prismatic blocks are covered in "Storage, Transportation, and Disposal of Advanced Reactor Spent Nuclear Fuel and High-Level Waste" [2].

Characteristics and Properties of TRISO SNF

The United States has experience in storing and transporting TRISO-based SNF from Fort Saint Vrain. However, TRISO-fueled advanced reactors have distinct fuel and reactor characteristics different from those of Fort Saint Vrain. Newer advanced reactors are using uranium carbide or uranium oxide fuels enriched to high-assay low-enriched uranium (HALEU)[2] levels. In contrast, Fort Saint Vrain used a uranium thorium carbide fuel enriched to high-enriched uranium levels. Additionally, the expected burnup for advanced reactor fuels significantly exceeds that of Fort Saint Vrain SNF. Table 3 provides the estimated burnup, enrichment, and specific power for X-energy's Xe-100 and Kairos Power's KP-X reactors [2]. Some of the specific power values were calculated based on the final burnup and the assumed residence time in the core. They may not be accurate for all fuels, and do not account for (axial and radial) power variations across the core. However, they provide initial average values to help enable later calculations (e.g., depletion). The rest of this section identifies particular characteristics significant to TRISO SNF behavior in storage, transportation, or disposal conditions.

Table 3. Assumed burnup, enrichment, and specific power of TRISO fuel in different reactors.

Reactor	Burnup (GWd/MTU)	Enrichment (wt.%)	Specific Power (MW/MTU)
X-energy Xe-100	165	15.5	129.6
Kairos Power KP-X	193	19.55	335

The post-irradiation condition (integrity) of TRISO particles is expected to be an important consideration when evaluating the rates and impacts of failed or damaged SNF on storage, transportation, and disposal. TRISO particles are designed to each act as its own pressure vessel and can contain fission products generated in the fuel kernel. In general, the potential in-reactor failure mechanisms of TRISO particles are functions of temperature, thermal gradient, burnup, irradiation fluence, and particle defects [12]. Of all the known mechanisms leading to fission product release, failures are dominated by manufacturing defects, despite being limited to about one for every 100,000 particles [3,13]. TRISO SNF will have a number of defective particles stemming from manufacturing or obtained during normal in-reactor performance; however, these defects are not expected to be on a large enough scale to impact SNF management practices.

The melting temperatures of the TRISO particle fuel kernel and coating layers are a key parameter in predicting particle failure. Of the various types of coating layers, SiC has the lowest melting temperature

[2] HALEU levels of enrichment are generally 10–20% U^{235}.

(i.e., 3103 K). For the fuel kernels, the melting temperature of UO_2 has been established as a function of burnup (i.e., 3113.15 K at zero burnup) [14]. There is substantial variability in the melting temperatures of the constituents of the uranium oxycarbide kernel, and the presumably conservative value of 2093 K was selected as the decomposition temperature for body-centered tetragonal uranium carbide [15]. High-temperature tests conducted on unirradiated TRISO fuel particles suggest limited particle failure due to kernel melting at temperatures exceeding 2093 K [16,17]. The incidence of TRISO particle failure was evaluated under irradiated conditions, at temperatures approaching 2090 K, with few to no TRISO particle failures [18,19,20]. Because storage and transportation temperatures are expected to be well below 2090 K, the thermal stability of the particle components means that failures due to thermal effects during storage and transportation are not expected.

While the SiC layer on the TRISO particles act as a pressure boundary, fission product diffusion through the SiC layer is an observed phenomenon. However, most of the escaped fission products are expected to still be caught in the matrix material. Furthermore, integral assessment data regarding TRISO fuel performance codes are extremely limited relative to the large body of assessment data available on LWRs.

Technical Conclusions for TRISO SNF

Preliminary structural, criticality, dose rate, and thermal analyses were performed on the estimated pebble-bed fuel discharged from both an Xe-100 and KP-X reactor in a large-diameter (~2 m) canister, as well as in DOESCs that were 0.46 or 0.61 m (18 or 24 in.) in diameter and 3.0 or 4.6 m (10 or 15 ft) long. This section presents several technical conclusions taken from the analysis covered in “Storage, Transportation, and Disposal of Advanced Reactor Spent Nuclear Fuel and High-Level Waste” [2].

- No significant issues related to the structural performance of dry storage systems or transportation packages are expected due to their lower overall masses relative to those used for traditional LWR SNF thanks to the lower material densities and potentially lessened shielding design needs.
- Single DOESCs loaded with pebble-bed TRISO SNF maintain the subcriticality of their contents under both dry storage and transportation conditions; however, the diameter of the canister may need to be limited. Additionally, volume displacement or neutron absorbers may be used in the large-diameter General 37 canisters in order to maintain subcriticality.
- No significant issues related to shielding and radiation dose control are expected due to the lower dose rates of TRISO SNF for the examined configurations relative to LWR SNF. Depending on the fuel type, configuration, irradiation history, and cooling time, it may be possible to reduce the amount of neutron-shielding material required to meet dose rate limits, though additional shielding may be needed near the package trunnions as a result of the uniform axial burnup of TRISO SNF contents relative to LWR SNF.
- No thermal challenges are anticipated in the storing or transporting of TRISO SNF within the existing LWR dry storage systems or package-size envelope. Thermal modeling of a hypothetical canister-based system indicates the maximum temperatures for TRISO SNF pebbles (Xe-100) to be very low (151°C) compared to what might be expected for LWR SNF contents (near 375°C).
- The structural stability of TRISO SNF under dry storage conditions was modeled, with particular focus on the thermal creep performance of the irradiated structural pressure boundary layer (i.e., SiC layer) of the TRISO particles. Preliminary results indicate that the TRISO SNF particle temperatures and pressures associated with dry storage are not high enough to impact the structural integrity of the SiC layer. The probability of failure is not expected to change over the course of the initial 20 years of dry storage.

Preliminary analyses suggest that dry storage and transportation of TRISO SNF from advanced reactors can be safely performed per the current regulatory frameworks for dry storage (10 CFR Part 72) and transportation (10 CFR Part 71). Development of a database of SNF material properties and a basis for potential in-reactor and post-discharge failure rates remains an important area of R&D work for supporting system and package designs and understanding age-related degradation phenomena that could impact content configuration assumptions.

Based on the desirable characteristics of the TRISO SNF form, the following preliminary general conclusions and potential R&D areas regarding TRISO waste form disposal were identified:

- All disposal concepts are amenable to TRISO SNF, thanks to the low corrosion rates of TRISO particle coatings under both dry and wet conditions [21, 22]. Disposal of TRISO-based SNF was included in the disposal plan for Yucca Mountain. The safety analysis for Yucca Mountain concluded that the introduction of water would not significantly increase the reactivity of graphite in TRISO SNF [23]. This should be confirmed based on the design of new advanced reactor TRISO SNF and packaging options (e.g., disposal with or without graphite blocks), as well as on the geology and expected chemical environment (i.e., reducing or oxidizing) of any proposed repository.
- TRISO SNF is likely to meet long-term standards for potential disposal options, without the need for additional engineered barrier system components. However, additional research is needed to study the long-term performance and stability of TRISO SNF particles and the graphite-based compacts and block elements. Post-irradiation examination results from TRISO SNF particles and the in-reactor performance of the SiC pressure boundary will provide a basis for additional necessary research into compliance with these standards, and confirm that additional barrier system components are unnecessary.

The primary challenge regarding TRISO SNF is the volume of waste generated. It is estimated that TRISO-fueled reactors will discharge 10–25 times more SNF than do LWRs per unit of power produced. This is primarily because the graphite moderator is included as part of the SNF form. Chemical or mechanical recovery of the graphite has been examined, but the level of contamination of the recovered graphite is unclear. If the activity is too high, disposition of the graphite as low-level waste will be infeasible.

METALLIC FUEL

Metallic fuel is one of the oldest nuclear fuel forms, having been used in Experimental Breeder Reactor (EBR)-I as far back as 1951. Since that time, better-behaving materials and improved operating procedures have reduced the large tolerances on dimensions and the fuel impurities associated with early metallic fuels. Historically, metallic fuel consisted of many pins assembled using gridded spacers—much like fuel rods grouped into assemblies in traditional LWRs. In a fast-neutron spectrum reactor, these pins generally consisted of a metallic fuel slug surrounded by sodium inside a metallic cladding.

New metallic fuel designs focus on limiting the fuel-cladding chemical interaction (FCCI) and eliminating the sodium bond. FCCI occurs when the slug interdiffuses with the cladding, thus limiting the useful lifetime of the fuel. As reactor vendors are anticipating higher burnups, the FCCI must be reduced if fuel failures are to be avoided. The sodium utilized in some existing designs causes material interaction issues during long-term storage and disposal. The alternatives to the sodium are to either use another thermally conductive liquid or gas to replace the sodium, or to fabricate the fuel slug with a tight fit to the cladding and provide an annulus in the middle of the slug to accommodate fuel swelling. These new designs and fabrication methods are the subject of current research [24].

Table 4 identifies a few reactor concepts that use metallic fuel as well as their associated outputs and fuel forms; however, it is not a comprehensive list of every anticipated advanced reactor technology expected to involve metallic fuel. For a more comprehensive list of advanced reactors under development, see reports published by the NEA [9,10] and the International Atomic Energy Agency [11].

Table 4. Summary of metallic-based advanced reactor technologies.

Commercial Vendor	Reactor (Reactor Technology)	Reactor Power Information (MWt / MWe)	Fuel Form
TerraPower	Natrium Type 1 (liquid-metal cooled fast reactor)	840 / 345	Metallic with sodium
Oklo	Aurora (liquid-metal-cooled fast reactor)	4 /1.5	Metallic with sodium
Advanced Reactor Concepts	ARC-100 (liquid-metal cooled fast reactor)	286 / 100	Metallic with sodium-
TerraPower	Natrium Type 1b (liquid-metal cooled fast reactor)	840 / 345	Annular metallic

Characteristics and Properties Metallic SNF

This section focuses on the characteristics and properties associated with metallic SNF generated by advanced reactors. Metallic fuel, which can be categorized as either with internal sodium or without sodium, is proposed for use in various reactor types such as sodium- and lead-cooled reactors as well as thermal and research reactors. The Natrium reactor, developed by TerraPower in partnership with GE Hitachi Nuclear Energy, is a representative example of a metallic-fueled reactor [25]. Initial plans for the Natrium demonstration reactor (Natrium Demo) are based on the use of metallic fuel with sodium (Type 1), but this fuel will eventually be replaced by an annular fuel without sodium (Type 1b). (As of April 2023, this fuel remains in the design and development phase.) Publicly available information from a TerraPower presentation [25–27] offers insights into some design details of the Natrium Demo, though certain specifics remain proprietary information.

The initial design for both the Natrium Demo and the Natrium Commercial Series I plant is assumed to operate at 345-MWe. Future plans include the Natrium Ultra or Natrium (U), a GWe-scale Commercial Series II+ plant [25]. Out of these concepts, only the Demo plant is considered near term, as its projected online date is near 2030—despite initial plans having aimed for operation to begin in 2027.

The Demo plant will use sodium-containing uranium alloyed with 10 wt% zirconium (U-10Zr) as the driver fuel and ferritic-martensitic HT9 as the cladding material [26]. HT9 is a low-void-swelling alloy composed of Fe-12Cr-1Mo-0.6WV. Extensive research and testing have been conducted on this fuel type, including its use in driver and test assemblies in EBR-II, as well as in test assemblies in the Fast Flux Test Facility (FFTF). The HT9/U-10Zr database was significantly expanded through irradiation experiments conducted in both EBR-II and FFTF [28].

The Demo plant will be a testing ground for advanced fuel options. Current plans are for the advanced fuel to feature annular, enriched U slugs bonded (without sodium) to HT9 cladding, with helium filling the inner region of the fuel pin [25]. A thin layer between the fuel and cladding will inhibit or reduce the FCCI rate. This advanced-fuel configuration is likely to be employed in the Commercial Series I plant.

Kim et al. [29] conducted detailed analyses of the waste masses, volumes, isotopic activities, decay heat, and activities pertaining to Natrium fuel, cladding, and assembly hardware post-discharge. As long as the assumed core parameters are close to the Natrium parameters, these results are directly relevant to TerraPower's plans for SNF dry storage and disposal. The planned pathway for Natrium SNF involves storing for 1.5 years in the reactor vessel (assumed to be in sodium), followed by 10 years in the spent fuel pool. After pool storage the SNF can be transferred to dry cask storage, before ultimate disposition via direct disposal in a deep repository or deep borehole, without reprocessing. This same procedure is assumed for Demo driver fuel (sodium-containing U-10Zr) and Demo lead test assemblies (non-sodium-containing U-metal fuel).

Several potential transportation, storage, and disposal issues arise from the presence of sodium, which could be reactive and ignitable under certain situations. To address these issues, it may be desirable to treat the fuel pins prior to transport, storage, and disposal. One possible treatment method, electrometallurgical treatment (EMT) [30] generates a uranium product and two waste forms: a ceramic waste form and a metallic waste form. The EMT process is being used on sodium-bonded EBR-II and FFTF fuels at Idaho National Laboratory[31].

Technical Conclusions for Metallic SNF

This section provides preliminary analyses of the technical constraints related to the transport and dry storage of metallic SNF from advanced reactors, with a particular focus on two representative fuel types (i.e., Type 1 and Type 1b) to be generated by TerraPower's Natrium SFR, a 345-MWe reactor being jointly developed by Terrapower and GE Hitachi. Type 1 will use sodium-containing U-10Zr driver-fuel clad in HT9. Type 1b utilizes an annular uranium metal clad in HT9 that uses internal helium with no sodium. These preliminary analyses support the various conclusions listed below.

Most design details pertaining to the Natrium Demo are proprietary, but publicly available information [25–27] indicates that it will use HALEU; that the hexagonal fuel assemblies will be ~470 cm long, with a maximum cross-sectional dimension of ~18.5 cm; and that the target burnup for Type 1b fuel will be ~150 GWd/MTU.

- The fuel assembly, which includes an upper handling fixture (assumed to be 31 cm long) and a lower nosepiece (assumed to be 33 cm long) [29], is too tall to fit in most existing commercial storage or transport canisters. However, the profile of the assembly would fit inside a typical pressurized water reactor (PWR) square basket (22.7 cm × 22.7 cm).
- The storage and transport canisters would need to be recertified for HALEU fuel. Canisters would also have to be relicensed for HT9 cladding as opposed to Zr-alloys. TerraPower [25] has acknowledged the need to recertify storage and transport casks for Natrium fuel assemblies.

To perform storage, transport, and disposal analyses, the Natrium core, assembly, and fuel pin parameters assumed by Kim [29] were employed. Kim based his assumptions on a revised PRISM/MOD-B design in order to develop a consistent set of parameters that would give about 150 GWd/MTU fuel burnup. Kim demonstrated that the Natrium SNF, 10 years after reactor discharge, would result in lower fuel mass, fuel volume, decay heat, activity, and radiotoxicity (per GWe-year) relative to a conventional PWR. These same parameters were used in the present work to perform criticality, dose rate, and thermal analyses.

- Criticality analyses were performed for metallic SNF in a large-diameter commercial canister and a smaller DOESC. Both wet and dry storage configurations were analyzed, and for the configurations examined, the multiplication factor did not exceed the upper critical limit for intact SNF. However, if the basket fails and the fuel assemblies are reconfigured, the upper critical limit

will be exceeded. Including neutron absorbers or utilizing burnup credit could also be effective in reducing the effective multiplication factor.

- Fuel failure was analyzed for a single fuel pin operated at the core's average power level. One potential cladding degradation mechanism is FCCI, which produces an inner-cladding brittle zone from which a softer cladding zone extends outward. To reduce the FCCI to an acceptable level during storage, the peak cladding temperature should be set at 400°C (i.e., the Nuclear Regulatory Commission's limit for LWR SNF) until further investigation has been performed. The temperature limits for other canister materials would be considerably lower.

Storage/transport experience and lessons learned were reviewed in terms of metallic fuel irradiated in EBR-II, FFTF, and the N-Reactor, as well as for the sodium-bonded blanket pins irradiated in the Fermi-1 reactor. The aforementioned experience indicates that:

- Residual coolant sodium on the cladding outer surface can be removed to an acceptable level, and a canister containing metallic fuel assemblies can likely be dried to an acceptable level via hot-vacuum drying.
- DOE has deemed sodium-bonded SNF unsuitable for disposal in the previous repository, dictating that it must first be treated prior to disposal [30].
- Experience from EBR-II included a method of processing—namely, EMT—in which the sodium is converted into a salt (mixed with zeolite and glass frit to create a ceramic waste form), noble fission products are embedded within an Fe-alloy ingot (metal waste form), and 99% pure uranium is separated to form a second metal product. The only EMT facility in the United States is located at the Idaho National Laboratory site. It has limited throughput and the process is relatively expensive. Conversion of irradiated salt into the chemical waste form has been demonstrated on a small scale for radioactive sodium; however, it has not been demonstrated for larger-scale conversion [31].

MOLTEN SALT REACTOR FUEL SALT

MSRs, a category of Generation-IV reactors, are expected to generated used fuel salt over the next decade. This section aims to enhance the current understanding and facilitate informed decision making regarding the management of waste generated by MSRs, with a particular focus on addressing the challenges associated with the storage, transportation, and disposal of MSR fuel-salt-derived waste, including wastes containing fission products and actinides. One key difference from solid fuel is the anticipation that used fuel salt could nominally be reused in future reactors without a need for intervening reprocessing or refabrication

In acknowledging the complexity and uncertainty surrounding MSR waste management, this section emphasizes the need for ongoing engagement and further research that carry a higher degree of reliability. It also underscores the significance of understanding core components, sorbed fission products, salt waste, and process waste from each MSR facility. Table 5 identifies a few different advanced reactor concepts utilizing fuel salts and their associated power outputs and fuel forms The report "Storage, Transportation, and Disposal of Advanced Reactor Spent Nuclear Fuel and High-Level Waste" [2] provides more details by identifying anticipated MSR developments; characterizing MSR waste streams; discussing the limitations of storage, transportation, and disposal packages; outlining knowledge gaps; highlighting regulatory considerations; and presenting the findings of the preliminary assessment. For a more comprehensive list of advanced reactors under development, see reports published by the NEA [9,10] and the International Atomic Energy Agency [11].

Table 5. Summary of fuel salt-based advanced reactor technologies.

Commercial Vendor	Reactor (Reactor Technology)	Reactor Power Information (MWt / MWe)	Fuel Form
TerraPower	Terra Power Molten Chloride Fast Reactor (salt-fueled fast reactor)	840 / 1200	HALEU chloride fuel salt
Terrestrial	Integral MSR (salt-fueled, graphite-moderated thermal reactor)	440 /195	Low-enriched uranium fluoride fuel salt
TerraPower	Molten Chloride Reactor Experiment	Research reactor	High-enriched uranium chloride fuel salt

Characteristics and Properties of Irradiated Fuel Salts

MSR systems possess the distinct characteristic of a molten fuel form, which allows for the release of fission gases, aerosols, and volatile species (e.g., CsI, MoF_6, iodine, and tritium). The details of capturing and managing these radioactive materials depend on the specific design considerations involved. However, the work in "Storage, Transportation, and Disposal of Advanced Reactor Spent Nuclear Fuel and High-Level Waste" [2] specifically focuses on the characteristics of the fuel salts. Two primary characteristics discussed here are actinide inventory and fission product inventory.

Actinide inventory: The reactor design, mean residence time in the core, and separation factors associated with fuel treatment all affect the discharge isotopic content. The actinide content heavily depends upon spectrum, with faster spectra enabling the fission of fissionable actinides, thus altering the neutron capture probabilities and actinide content. Actinides tend to be more of a factor for disposal than for storage/transportation. The exception is ^{244}Cm, a shielding-relevant isotope that spontaneously emits neutrons and is roughly proportional to burnup to the fourth power in LWR SNF. For thorium systems (which are not considered in the analyses performed for this assessment), ^{232}U is also an actinide of concern for shielding purposes.

Fission product inventory: At an order-of-magnitude level, fission product inventory is a function of power produced. Fission product generation also depends on the spectrum and the isotope undergoing fission. Some common activation products are much less abundant in used molten fuel salt than would be anticipated based on LWR power production equivalence. However, for order-of-magnitude feasibility calculations and gap analyses, it is sufficient to treat fissions as being identical regardless of parent isotope, enrichment, or spectrum.

Technical Conclusions for Irradiated Fuel Salts

Preliminary analyses of the technical constraints related to the transport and dry storage of MSR SNF centered around once-through U-Pu fuel cycles. To ensure a sufficient level of design maturity, these fuel cycles correspond to MSR concepts surveyed in U.S. Nuclear Regulatory Commission literature [32].

- Due to its water-soluble nature, MSR salt waste will likely need to be converted into a more stable waste form. Several waste forms were surveyed, and technology readiness level ranges are listed in the preliminary assessment for each waste form, based upon the available knowledge.

- A number of additional waste types, including radioisotopes such as ^{137}Cs produced from trapped fission gases, will arise from scrubbers and filters. These waste streams are noted, but work performed in this assessment only focuses on used fuel salt that is either unfit for continued use or remains at system shutdown.
- Both waste form and potential processing techniques to achieve a stable waste form configuration are major knowledge gaps that must be addressed to adequately assess spent salt management practices.
- Material control and accountability is a challenge for MSR SNF.

Scoping calculations were performed to examine how closely prototypic MSR waste approaches the design envelope of existing PWR SNF canisters. The evaluated parameters included waste mass, radiation shielding, criticality, and decay heat rejection.

- Canister capacity was found to mostly be dependent on fission product inventory, with existing canisters being capable of handling fission products generated by 750 GWd of thermal energy throughout the fuel's lifetime. The analysis focused on the more tractable components of canister design. Some problem isotopes for shielding, such as ^{244}Cm, ^{60}Co, and ^{208}Tl, were not evaluated as a result of insufficient information on the reactor design.
- Thermal and shielding scoping analyses were performed by assuming Molten Salt Reactor Experiment (MSRE)-type waste salt with a burnup of 105.7 GWd/MTU (much higher than the actual MSRE). At 5 years after usage, the assumed spent fuel salt produces 5 kW per metric ton of initial uranium. Given the dilution of uranium in the salt this amounted to 7 kW per canister, whereas a large commercial canister of similar size containing PWR SNF can reject over 20 kW. Increasing the uranium concentration in the salt would also increase the associated decay heat.
- A likely mode of MSR operation is to add U as needed to maintain criticality under near-optimal conditions, meaning that criticality may be of no particular concern if a burnup credit formalism or fissile nuclide concentration determination is established. However, some salts from thermal reactors may be more reactive in the presence of a moderator. Moderator exclusion may need to be applied based on the soluble nature of the salt. Non-salt waste forms were not evaluated for criticality and are presumed to be more manageable due to certainty about their geometry and presumed lack of water solubility.
- The gaps encountered are numerous. Most pertain to waste characteristics stemming from a lack of knowledge about which MSR concepts will proceed, the extent to which they will proceed, and detailed conceptual design information on each MSR type. Fundamentally, there is a lack of definitive information on the isotopic content in salt waste streams. This stems from both the early stage of development as well as caution about making premature statements of functionality.
- It remains to be determined at what point in a waste management system (storage, transportation, or disposal) the conversion from salt into one or more physiochemically stable waste forms should be undertaken.
- The preliminary assessment does not assess the level of effort necessary to maintain moderator exclusion from fuel salt being transported. The extent of radiolytic off-gassing from solidified salt and its mitigation requires assessment beyond what is presented herein. Nor does the preliminary assessment evaluate the means and costs of fissile material recovery from any research or prototype reactor.
- Expected challenges going forward include the complexity of preparing to handle many types of wastes that may need to be stabilized prior to transportation. This is compounded by the fact that many of these waste forms are not well-defined. Water solubility, potential corrosion, and off gas

must all be considered prior to transporting halide wastes. Several MSR concepts include quasi-decadal primary-system refurbishment, which will also need to be characterized and dealt with.

CONCLUSIONS

Due to increased interest in advanced reactor deployment for potential new fuel cycles, the SFWST program has begun to evaluate the possible implications of long-term management and final disposition of the SNF generated by potential advanced reactors. Safely managing and dispositioning this SNF, along with any other associated radioactive waste, is the primary focus of this initial preliminary assessment.

This paper summarizes the SFWST program's efforts to evaluate the characteristics and packaging options for three types of advanced reactor SNF forms: (1) TRISO, (2) metallic, and (3) fuel salt. TRISO and metallic SNF and their associated waste streams were emphasized because of the near-term anticipated demonstrations of the Xe-100 and Natrium advanced reactors. Preliminary information on spent fuel salts discharged from MSRs was also examined to provide a baseline for future efforts. All calculations and assumptions were based on publicly available information.

This paper identifies several different reactors that produce either TRISO or metallic SNF, as well as a few of the reactor and fuel characteristics used for the assessments. Based on these characteristics, calculations were performed to determine the applicability of existing canister designs. The evaluations included geometric (e.g., dimension, volume) and mass/weight considerations, known operational approaches and loading procedures, physical and chemical considerations/conditions for storage environments, and as-loaded radiation, thermal, and criticality analyses to identify constraints on storage, transportation, and disposal.

Gaps in publicly available data pertaining to reactor operation and/or fuel composition provide increased uncertainty in certain evaluations. Additionally, uncertainty in packaging and SNF management operations provide additional uncertainty. However, preliminary conclusions can still be assessed through this work and are presented in this paper. Additional work is being performed in 2024 to further identify knowledge and data gaps for managing advanced reactor SNF.

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